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LMFBR Operational and Experimental In-Core
Local-Fault Experience, Primarily with Oxide Fuel Elements[1,2]

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D. K. Warinner
Mechanical Engineer,
Magnetohydrodynamics Project and Reduced Enrichment
Research and Test Reactors Program,
Argonne National Laboratory
9700 South Cass Avenue
Argonne, Illinois 60439

Member, ASME



Abstract

Case-by-case reviews of selective world experience with severe local faults, particularly fuel failure and fuel degradation, are reviewed for two sodium-cooled thermal reactors, several LMFBRs, and LMFBR-fuels experiments. The review summarizes fuel-failure frequency and illustrates the results of the most damaging LMFBR local-fault experiences of the last 20 years beginning with BR-5 and including DFR, BOR-60, BR2's MFBS-and Mol-loops experiments, Fermi, KNK, Rapsodie, EBR-II, and TREAT-D2. Local-fault accommodation is demonstrated and a need to more thoroughly investigate delayed-neutron and

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gaseous-fission-product signals is highlighted in view of uranate formation, observed blockages, and slow fuel-element failure-propagation.

## Introduction

The intent of this paper is to survey fuel-element behavior, primarily oxide endurance, (1) while operating in various liquid-metal-cooled fast-neutron breeder nuclear reactors (LMFBRs) around the globe, (2) during LMFBR experiments and fuels testing in prototypes, and (3) during experiments in thermal reactors with LMFBR-design fuels in an epithermal neutron flux (e.g., BR2/Mol 7B). This survey of experience with oxide fuels, primarily, and with carbide and metals fuels to a much lesser extent, is intended to help guide the research and development (R&D) remaining to ensure safe and clean LMFBR operation through the fulfillment of the LMFBR lines-of-assurance (LOAs) discussed in the companion paper, reference [1].

This paper focuses on fuel failure and fuel degradation as subsets of local faults. To improve the fuel design, we must not only gauge where the fuel has fallen short of the design expectation, but also note how design changes, and change from pellet to vibratory-packed (vipak) fuel, might have affected the operating performance of the fuel. Reference [1] tabulates some of these results. We cannot provide such definitive design-change results here, but we can review the operation of some reactors and experiments and leave impressions of how the fuel performed.

Although the conceptual-design-study (CDS) fuel has yet to be selected, certain weaknesses of (and less testing of and operation with) the attractive carbide fuels strongly suggest that we will drive the first-generation commercial breeders with the familiar oxide fuel. Metal fuels successfully drove Fermi and Dounreay Fast Reactor (DFR), and continue to drive EBR-II, but safety considerations have left this choice a distant runner for selection in a commercial LMFBR. Nitrides are too new and unfamiliar to designers to be serious candidates for the CDS fuel in the foreseeable future. Thus, a fuel akin to that chosen for the Clinch River Breeder-Reactor Plant (CRBRP) deserves our primary attention in a performance review, even though carbide and metallic fuels may perform better than oxide fuels with respect to such factors as breeding ratio, transient behavior, power density, and sodium bonding.

Reference [1] set the framework within which we are dealing through (1) a brief review of the LMFBR lines-of-assurance, (2) a review, history, and definition of local faults, (3) a description of the fuel designs and major reactor-design parameters in decommissioned, operating, and planned reactors, (4) a summary of what the U.S. and other nations expect for fuel-design performance, and (5) a review of failed-fuel detection. Finally, the conclusions call for a review of operational experience so that one could determine what trends appear to possibly meet the design goal and what R&D remains to ensure that the lines-of-assurance goals are met.

To recapitulate the conclusions of reference 1 and prepare for the review following, a summary of out-of-pile experiments and analyses (coupled with previously published summaries) showed that [1,2]:

- rapid FEFP has been deemed extremely unlikely, if not incredible
- slow FEFP should be (1) detectable, and (2) self-limiting
- slow blockage propagation is unlikely
- slow blockage growth appears nonmechanistic from within and highly unlikely even for external debris
- in-core planar blockages can be ruled out as a credible local fault
- molten-fuel release is very improbable, but even given a small release, resultant failure propagation or subassembly damage is unlikely
- although pin distortion and vibration, wire-wrap breakage, and other faults are possible - indeed, likely - the basic conclusions from the analyses and out-of-pile studies appear to be relatively insensitive to such perturbations.

Again, these conclusions ignored the reactor experience; indeed, Warinner and Cho [2] went on to review in-pile experience and experiments and altered these conclusions somewhat in view of the operational experience.

Whether the consequences of an in-core local-fault will always be contained within the subassembly (S/A) will remain in question until many years of prototypic operating experience have been witnessed. The conclusions, based on analyses and prototypic out-of-pile experiments, often with a critical parameter or characteristic bounding, can be challenged to be conjectures. However, the inreactor experience summarized here lends credence to these conclusions.

## In-Reactor Operating-Experience and Experiments

To illustrate the basic conclusion of S/A containment of local faults, we limit our case-by-case review to selective multi-pin experience (or experiments) with fuel failures that had appeared to have some credible path to whole-core involvement. (This excludes many valuable single-capsule tests in GETR, SILOE, AND FR-2; to include effects of adjacent pins and to illustrate fuel-element failure-propagation (FEFP) or lack thereof requires a pin bundle.) Our review includes the following reactors and experiments: Sodium Research Experiment (SRE), Hallam, BR-5, Fermi, Dounrey Fast Reactor (DFR), Transient Reactor Test (TREAT), BR 2 (Manufacture-Franco Belge-au-Bauche-Sodium (MFBS) and Mol 7 Series), KNK, Rapsodie, BOR-60, and EBR-II Run Beyond Cladding Breach (RBCB). This includes the planar blockage inserted into the DFR during decommissioning and MOL 7B, 7C/1, and 7C/2 summarized below; Mol 7C/3 results should be available this year. These studies have shown that such given blockages can be accommodated. Although we have no reason to believe that these in-pile results would not apply to a wire-wrap spacer design, to convince others of this is not without complications.[2] However, the reactor experience covers a broad range of many parameters as listed in Tables 3 and 4 of reference 1.

[2]Results from the ANL Sodium Loop Safety Facility P4 Experiment planned and designed by the author and run in August and September of 1981 should provide such information.

The reactor experience for oxide fuel, tabulated through 1976 in Table 1 from reference 3, provides a birds-eye view of the overall behavior.

However, this table appears to have underestimated the oxide irradiation considerably. Table 2 from an earlier report [4], shows that by the end of 1976, the USSR alone would have irradiated at least 61,600 pins. Also, experience such as that in PFR is not included.

These two tables illustrate a need for a comprehensive, continued tabulation that includes a screening process so that experiments, different cladding materials, differing pin diameters, etc. can be sorted to reveal the true failure rate for prototypes with a given linear power, fluence, and burnup. This will not be done here; rather a case by case approach will illustrate the behavior of the fuel elements.

Local Faults in Sodium-Cooled Thermal Reactors. The 1959 SRE fuel-failure accident is included because it presents one extreme of many conditions, was sodium cooled, and had cylindrical metal (and later oxide) fuel elements on a triangular pitch with wire-wraps spacing 7-pin clusters. Further similarity to a U.S. LMFBR prototype ceases there, however (viz, SRE was a 20 MWt sodum-graphite thermal-breeder reactor with NaK bonded uranium metal fuel 1.83 m long, 19.1 mm in diameter, cladded by 0.25 mm thick 304 SS and spaced 2.34 mm). After eight to 38 L of oil (Tetralin) leaked into the sodium and deposited on the fuel pins, the oil was "stripped" with nitrogen, only to possibly nitride the SS and zirconium and contribute to later fuel and moderator failures. Although the oil accumulated for 13 months and its effects were detected over eight months with temperature anomalies and unexpected reactor behavior persisting, the reactor was repeatedly recovered from various scrams until repeated checks of a S/A showed the elements to have lost all freedom of motion. The reactor was finally shut down after reaching 14 MWt, scramming, returning to 3-5 MWt, scramming, and so on several times during its last run of ~14 days. (During this 8-month period, several potential accidents were logged: failure of automatic scram, loss of secondary

Table 1

Statistics on Irradiated Oxide Fuel Pins in Sodium-Cooled Fast Reactors [3]

| Reactor | Number of Irradiated Pins | Number of Failed Fuel | Fuel-Failure Rate, % |
|---|---|---|---|
| BR-5,BOR-60 | 4,600 | 150 | 3.3 |
| DFR* | ~1,000 | ~50 | 5.0 |
| Rapsodie | 25,000 | 24 | 0.1 |
| Phenix | >30,000 | ~10 | 0.03 |
| EBR-II, SEFOR | 1,500 | 10 | 0.7 |
| GfK-Program | 209 | 30 | 14.5 |
| | ~62,300 | 274 | 0.44 |

*NaK-cooled

Table 2

Oxide Fuel Pins Irradiated in FBRs [4]

(Based on information available Dec., 1973)

| Country | Reactor | Irradiated Fuel | ~Failure Rate, % | Total |
|---|---|---|---|---|
| USSR | BR-5 | ~2,490 | | ~61,600 |
| | BR-10 | ~1,520 | | |
| | BOR-60 | 11,400 | <0.5 | |
| | BN-350 | ~46,200 | | |
| France | DFR | 41 | 10 | 44,650 |
| | Rapsodie-Core 1 | 4,305 | | |
| | Rapsodie-Fortissimo | ~17,300 | <0.2 | |
| | Phenix | 23,002 (>40,000)[a] | <0.01 | |
| USA | SEFOR | 648 | | ~2,450 |
| | EBR-II | ~1,800 | | |
| UK | DFR | ~1,000 | 10 | ~1,000 |
| DEBENELUX[b] | Rapsodie | 73 | | 181 |
| | DFR | 108 | 10 | |
| Other | | | | ~150 |
| | | | | ~110,000 |

[a]From IWGFR-24-3 (April 1978), not included in total

[b]DEBENELUX refers to a joint program between West Germany (Deutschland), Belgium, The Netherlands, and Luxemburg; Luxemburg has since become inactive and the program is now referred to as "DEBENE."

coolant flow, loss of auxiliary primary flow, fast periods, and seemingly runaway behavior.) The post-accident analyses strongly suggest repeated voiding of blocked channels. Yet, although the exit temperature had exceeded 760°C (nominal ~500°C) and a steel-uranium eutectic had formed, the cladding melted in but 10 of 43 assemblies; the melting was severe enough to separate the top and bottom halves of the fuel pins. Thompson [5] comments in his description of the accident, "one can postulate that there was a reactivity interaction between channels and that voiding in one led to heating and voiding in others and so on." Even with this serious accident, the reactor had two more core loadings of different designs and finally attained 45 MWt with $UO_2$ fuel before it was shut down in 1964. All involved in nuclear safety, from safety analysts to utility managers, should periodically review the nearly forgotten SRE experience to learn from this incident of malfunction and poor judgement. As of January, 1979, the SRE decommisioning was "nearing completion"[6]. Finally, the oil leak, which can be regarded "external debris," has appeared in other reactors (e.g., 27 L in Phenix, 10 L in BOR-60, and small amounts of Fluorolube in Fermi) as a potentially severe local-fault that can seriously damage the fuel. (It is interesting to note that the SRE supplied 6 MWe into the Southern California Edison Corp. grid.)

The Hallam Nuclear Power Facility (HNPF or "Hallam"), an outgrowth of the SRE and similarly designed as another sodium-cooled thermal-reactor, operated from 1962 to 1964. The reactor was troubled with many component failures; the relevance to local faults lies in the failure of, and sodium permeation of, some SS cladded graphite moderator elements (such failures encouraged the use of the vented-fuel-to-coolant concept, thought to prevent FEFP), difficulty of carbon-content control, and coolant flow maladjustment partly due to subassembly inlet sodium dioxide deposition. The $NaO_2$ deposition has recurred in other reactors (e.g., Rapsodie).

<u>Early Fast-Reactor Experience with Oxide Fuels: BR-5.</u> The Soviet BR-5 had $PuO_2$ fuel designed for 2% heavy atom burnup (b.u.) as its first charge in 1959. Significant activity registered at 2.4% b.u. and worsened beyond $^{137}Cs$-activity detection at 3.2% b.u. The reactor was not shut down until ~5% b.u. (September, 1961) when 18 of the 81 S/As were found to leak fission gas (gaseous fission products, GFP) badly. In early 1962, $UO_2$ S/As replaced the

outer 20% of the $PuO_2$ 19-pin S/As. The $PuO_2$, rearranged in the central zone, had 4.85% max. b.u. and included leakers. The reactor then operated for three years with up to 6.5% b.u. until a major increase in the GFP activity gave BR-5 its lifetime maximum xenon activity on November 1, 1964. From Ref. 7, "In seven years of continuous monitoring of the BR-5 reactor, only once did an emergency fuel-element leakage (with $PuO_2$ fuel) occur (November 1, 1964). Various signs indicated that more than ten fuel elements started leaking in 1 min. The rapidly worsening dosimetric conditions made it essential to reduce the power to 10% nominal within 6 h, and after another 2 h to shut the reactor down altogether. On recharging the packs, no melted fuel elements were found, although every fifth pack contained a leaking element. The recharging itself involved no difficulties." Twenty seven of 59 centrally located $PuO_2$ S/As were found to contain failed pins a month after discharging the fuel; 17 $PuO_2$ S/As with >5.1% b.u. had been defective for 2 1/2 years. Four months after reactor shutdown, the gas activity (primarily $^{85}Kr$) from these S/As was from 10 to 1000 times higher than the background from leak tight S/As. BR-5 was restarted in May, 1965, charged with UC fuel except for four ~6% b.u. $PuO_2$ S/As, two ~1% b.u. $PuO_2$ S/As, and two ~1% b.u. $UO_2$ S/As. A delayed neutron monitor (DNM) had been installed in 1964. Two S/As were found to contain the first failed pins in August, 1965. Figure 1 shows one reported investigation of a BR-5 19-pin bundle of $PuO_2$ at 6.1% b.u.; all pins exhibited high swelling, <u>but only the center pin failed - quite badly.</u> The cladding had cracked along the <u>entire length on opposite faces;</u> the bundle had been irradiated in BR-5 for about 5 1/2 years [8]. Although we are not aware of whether transients initiated fuel failure or degraded the fuel, this BR-5 experience with close-to-prototypic geometry demonstrates that no propagation occurred (not even to an adjacent pin in the above case) while operating 1) with failed fuel much longer than our projected refueling cycle, 2) with highly embrittled and swollen fuel, and 3) at contamination levels not permitted for a prototype LMFBR. Having irradiated $PuO_2$ to a maximum of 6.7% b.u. with $3 \times 10^{22}$ n/cm$^2$ fluence over 468 actual operating days by 1971, BR-5 was upgraded, received a third loading of $PuO_2$ intended for 10% b.u. and 10 MWt in 1972, and renamed BR-10. BR-10 ran at powers much less than 10 MWt from March 1973 through 1978; extensive structural irradiation prevented higher-power operation. Too little information on BR-10 fuel-failure has been received to include in this review. It is interesting to note that BR-5 had an estimated 38 g of $^{235}U$ in the primary circuit coolant in 1969 [7].

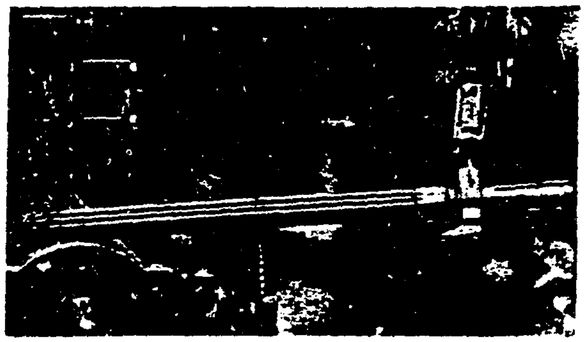


Fig. 1(a) Nineteen-pin subassembly

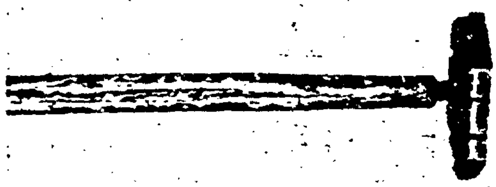


Fig. 1(b) The failed central pin

Fig. 1

BR-5 19-pin bundle of $PuO_2$ at 6.1% burnup with only central pin failed [8] (by permission of Plenum Publishing Corp.)

The Fermi Subassembly Meltdown. The 1966 Fermi incident illustrates a degree of coolability, although the geometry, fuel, and cladding were unlike any projected prototype. This, the then largest FBR (200 MWt) and sodium cooled, had an inlet blockage affecting the flow through about four core S/As (144 fuel elements per S/A, 105 core S/As total). Ironically, the inlet blockage was a zirconium plate that had been added for safety, not operational, purposes. Three years after it went critical and during a rise to power (at ~34 MWt), at least one S/A voided, the fuel melted in two S/As (dispersed radially and slumped slightly), and two adjacent S/As were slightly damaged (deformation but no melting). The reactor-power increase was stopped, but the reactor was not immediately scrammed; a post-accident analysis suggests that the fuel melted a hole in the square SS wrapper and then the S/As were cooled (at that low power) by sodium flowing through the hole and exiting the most severely damaged S/A, hardly a case of S/A-to-S/A propagation to the whole core. While being mindful of the low power at which the accident occurred, this accident suggests favorable inherent behavior even in the most severe local-fault case of an inlet blockage. Indeed, the accident was so attenuated that repairs were possible; four years later the reactor returned to full power following an intense technical and governmental investigation, only to be decommissioned in 1972. Finally, the reference subassembly inlet design (e.g., for CRBRP and FFTF) has been significantly altered so as to effectively preclude an inlet flow blockage.

The DFR Fuel Irradiation Tests. The 60-MWt DFR, unique for its combined NaK primary coolant, coolant downflow, $N_2$ cover gas, fuel-pin spacer grids, and vented-fuel-to-coolant design, went critical in 1959 and, except for jammed elements in 1965, ran remarkably trouble-free through decommissioning in 1977. It served beautifully as a fast-flux materials-test reactor; the operating experience, per se, adds optimism to local-fault issues. The $O_2$ level was held to about 6-8 ppm or less after about 1970. As early as 1964 and 1965, nucleate-boiling detectors and S/A-outlet temperature-noise monitors were installed, both intended to detect the presence of a S/A blockage (in anticipation of increasing the number of test rigs beyond 30 and therefore increasing the probability of blockage formation). The three basic pin irradiations were conducted in (1) reactor center S/As, (2) core periphery mini-S/As, and (3) single-pin or trefoil test rigs replacing driver fuel pins. The descriptions of selected DFR tests will be brief reminders of those that had operated with encapsulated mixed-oxide fuel; photographs will illustrate the results where possible. All failed pins ran for some time beyond failure because the DFR vented-fuel design precluded the use of an effective DNM.

Although failures have been relatively infrequent, some naturally failed pins (noticed during startup) were left at power for over 100 days to study the effect of $Na_3(U,Pu)O_4$ with no resultant gross deterioration; Figure 2 illustrates the development of one failure [9]. However, more serious effects may be masked because NaK hinders sodium uranate formation and the potassium reacts to form a higher density compound than sodium uranate. Thus, the pin damage might be far less than it would in a Na-cooled reactor. Perhaps the worst failure rate of a prototypic bundle in DFR was with the Mk-VIIA 60-pin S/A in which 90 percent of the fuel failed while the reactor was operating at ~60 kW/m up to 9.0 percent b.u. Although the high-failure-rate cause is unknown, this bundle differed from others by having wire-wrap spacers, 30 and 40 percent Pu, and all vibrocompacted fuel [10]. Yet, the bundle retained its original shape with no sign of overheating. Before this, the most extensive failures had been 22 failures in a 77-pin bundle; the failures were signaled by radon release into the blanket gas during the rise to full power, where the reactor remained for 30 hours without developing a hazardous situation [11].

DFR-324/2 was a trefoil irradiated for ~1 1/3 years (~57 MWd/kg) at 44.0 kW/m maximum. A "massive fracture" of "not-obvious cause" occurred at the top of the fuel column (the cold end) and was not "associated" with the breeder region [12]. The experimenters cite high fission-product content as the only unusual feature in the failure region; this suggests Cs migration to the lower-temperature regions and subsequent reaction with the fuel and axial-blanket, and/or cladding attack. Irradiation continued without propagation.

DFR-350 was a 39-pin bundle of 86.4 percent-enriched $U_{0.80}Pu_{0.20}O_{1.98}$ fuel irradiated to 52 MWd/kg at 45.0 kW/m maximum with no failures, although 18 pins with one of two cladding types were badly swollen. The 21 pins with little swelling were then included in the DFR-435 trefoil series for further irradiation at ~40 kW/m. All seven failures occurred between 64 and 90 MWd/kg (9.7 percent b.u.). The two trefoils with the highest b.u. had either two or

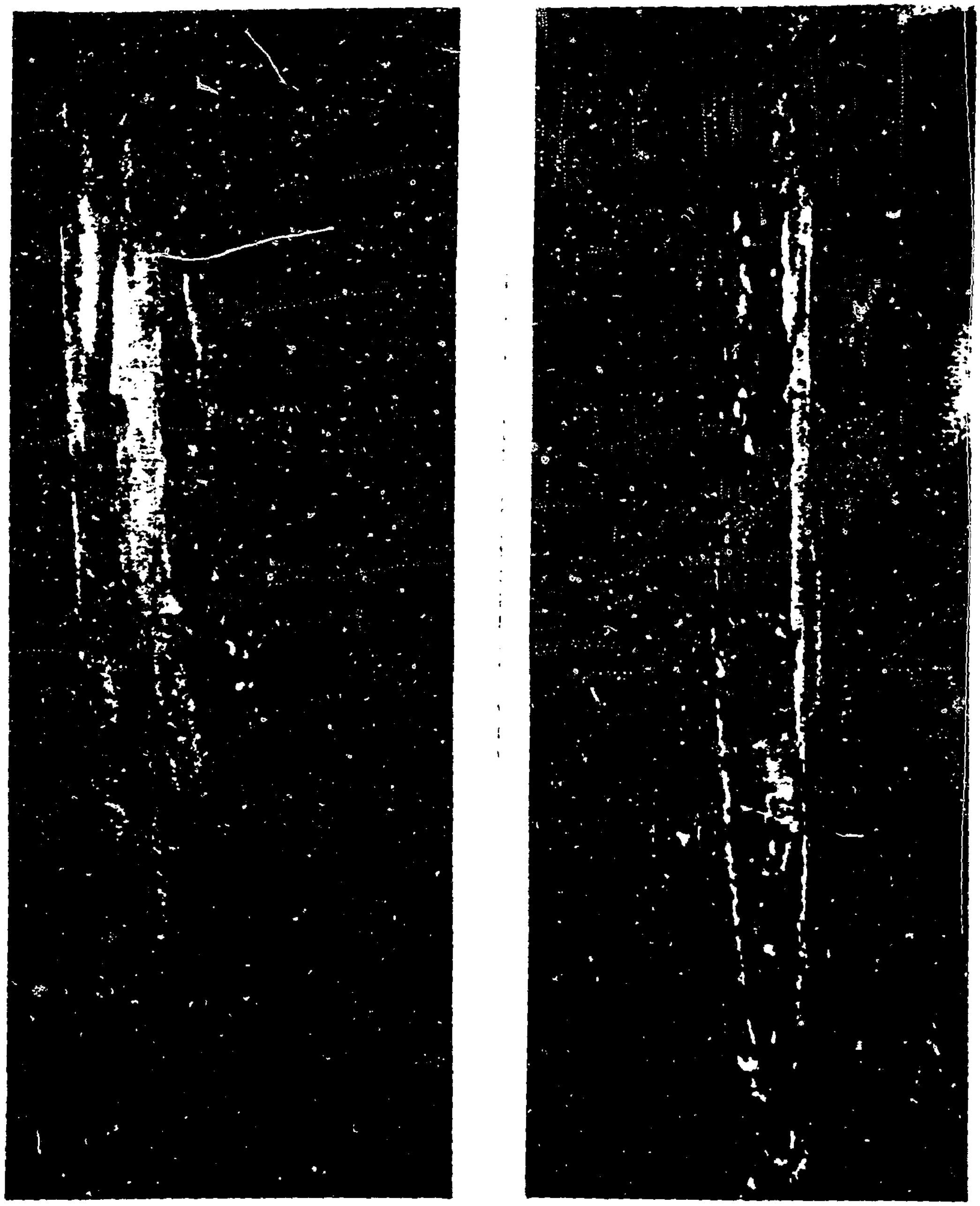


Fig. 2(a) 5.0 percent b.u.

Fig. 2(b) 6.8 percent b.u.

Fig. 2

DFR oxide pin failure showing natural development of failure [9]
(by permission of British Nuclear Energy Society)

three of the three pins failed; although the experimenters do not cite propagation positively, it appears clearly possible that an initial failure caused others (particularly given the failure locations in the number seven trefoil and the 31 shutdowns it experienced). Also, fuel was released and $Na_3(U,Pu)O_4$ was formed, but no further damage was reported. Figure 3 illustrates two of the failures; [12]; Bagley et al. postulated the upper defect to possibly result from gas entrainment which created an overheated zone. Figure 4 shows cross-sections of a failed pin; the cladding fissure and fuel-flushing zone are clearly shown as are the tiny particles separated by fissure zones [13,14].

DFR-455, a 60-pin bundle of 93 percent enriched $U_{0.70}Pu_{0.30}O_{1.95-1.98}$ irradiated from February 1973 to February 1974 at 49 kW/m, maximum, to 50 MWd/kg with failed fuel since ~5 MWd/kg [12]. Three cladding types and the spacer-grid, tie-rod design were used. Most of the seven failures have been attributed to "a single fabrication batch where some impurities causing heavy internal corrosion remained in the pins during fabrication." [15] Although "fuel particles were released ... and distributed within the assembly, ... neighboring pins were [not] damaged even [around] severe failures." [12] Again, such a severely damaged bundle remained cooled for months with no threat to the S/A wall. Figure 5 illustrates the damage.

The DFR-522, -528, -536, -539, and -540 series were the most severe DFR experiment series, entitled DFR Special Experiments, conducted, logically, during decommissioning. A series of prolonged (i.e., hours) boiling runs without and with a thin steel plate in place to simulate a 70 percent local heated-zone blockage provided a "proof test" for the mixed-oxide fuel. the CW M 316 SS cladding contained either pellets or vipak fuel with from 0 to 10 percent b.u. The power was typically around 32 kW/m, maximum. The pins, contained in trefoils, mini-S/As without blockage, and mini-S/As with a blockage, show very little damage. Those tests without a blockage showed some swelling, significant bowing in DFR-528, and only one failure in 528/1 with little or no fuel loss or melting. The local-blockage tests exhibited blanketing by trapped vapor, several failures, loss of complete sections of some fuel pins, and cladding melting; even so, no secondary blockages formed and no fuel melted. These results are remarkable because boiling times ranged from

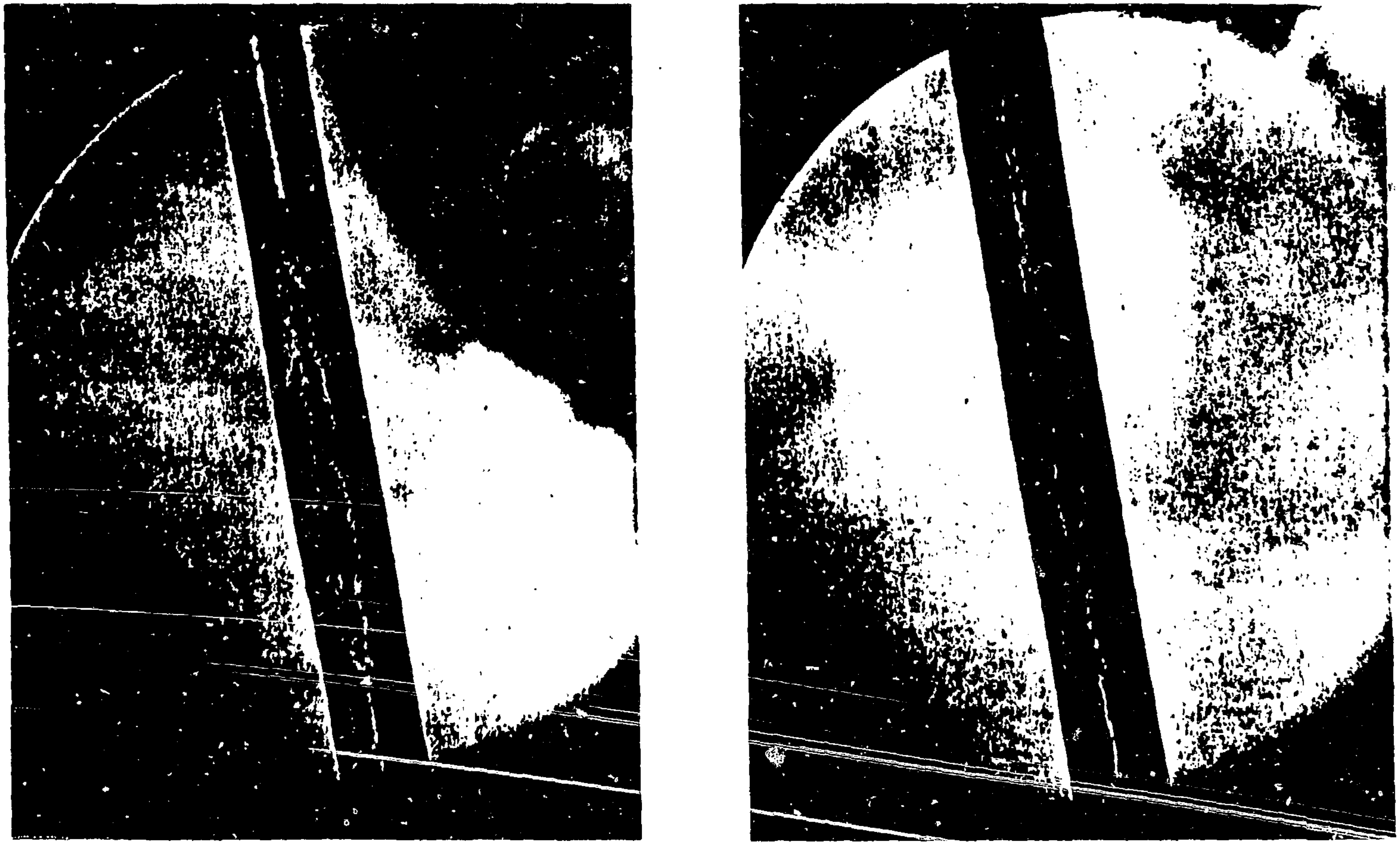
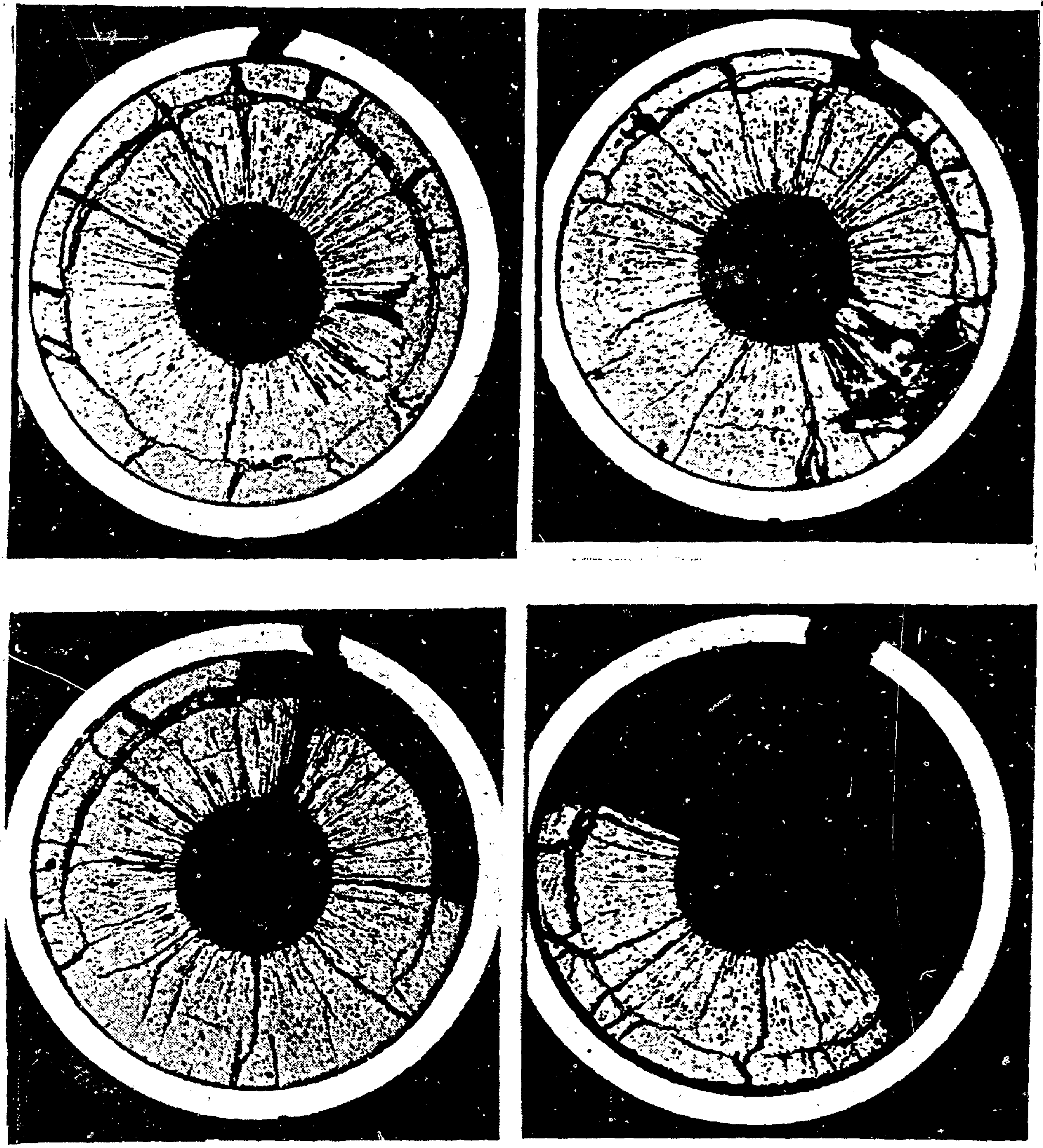


Fig. 3(a) Upper defect, Pin G 24

Fig. 3(b) Lower defect, Pin G 24

Fig. 3

Two failures in DFR-435 test [12] (by permission of IAEA)



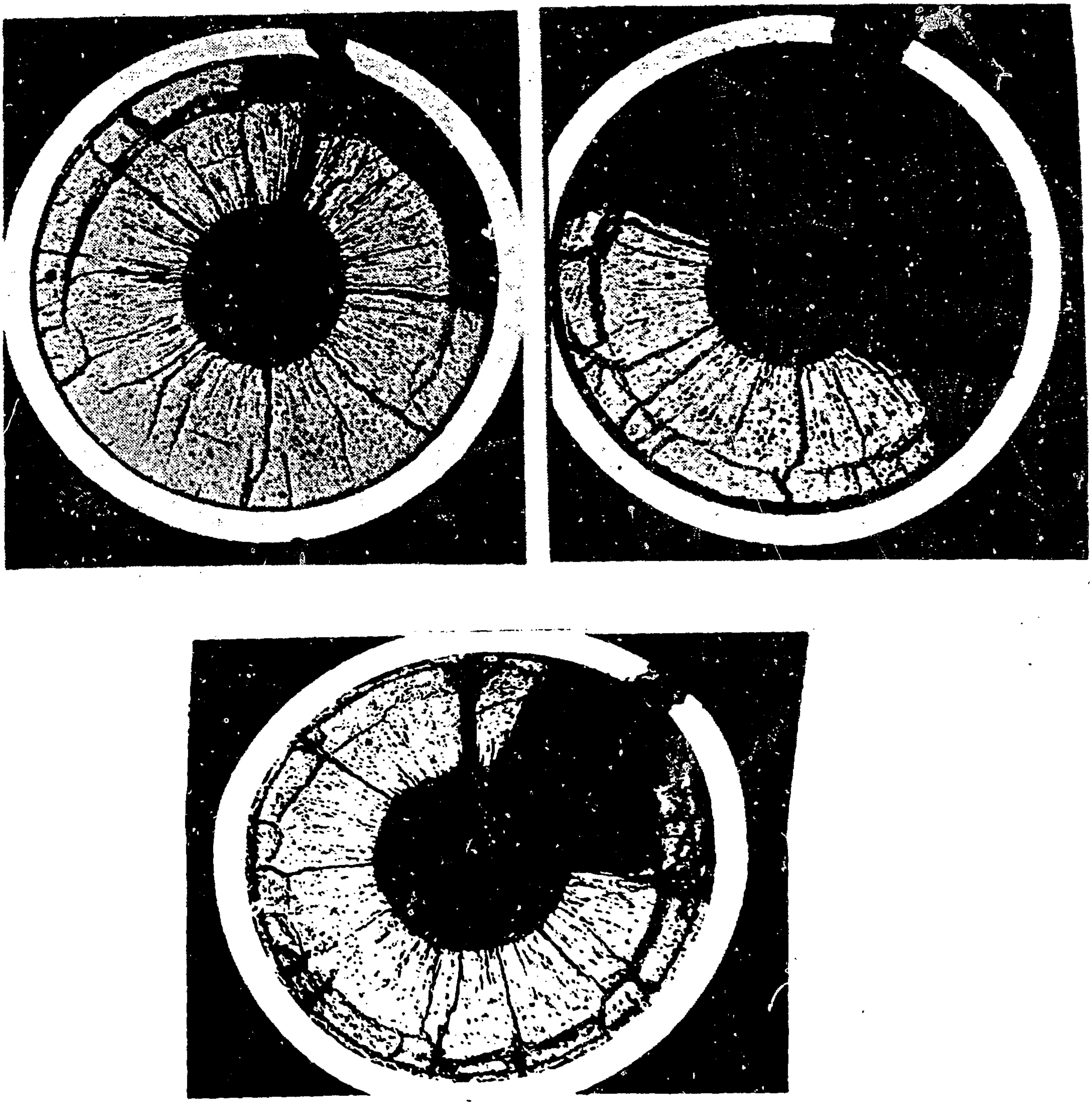


Fig. 4 Cross sections of failed pin in DFR-435 [13,14]

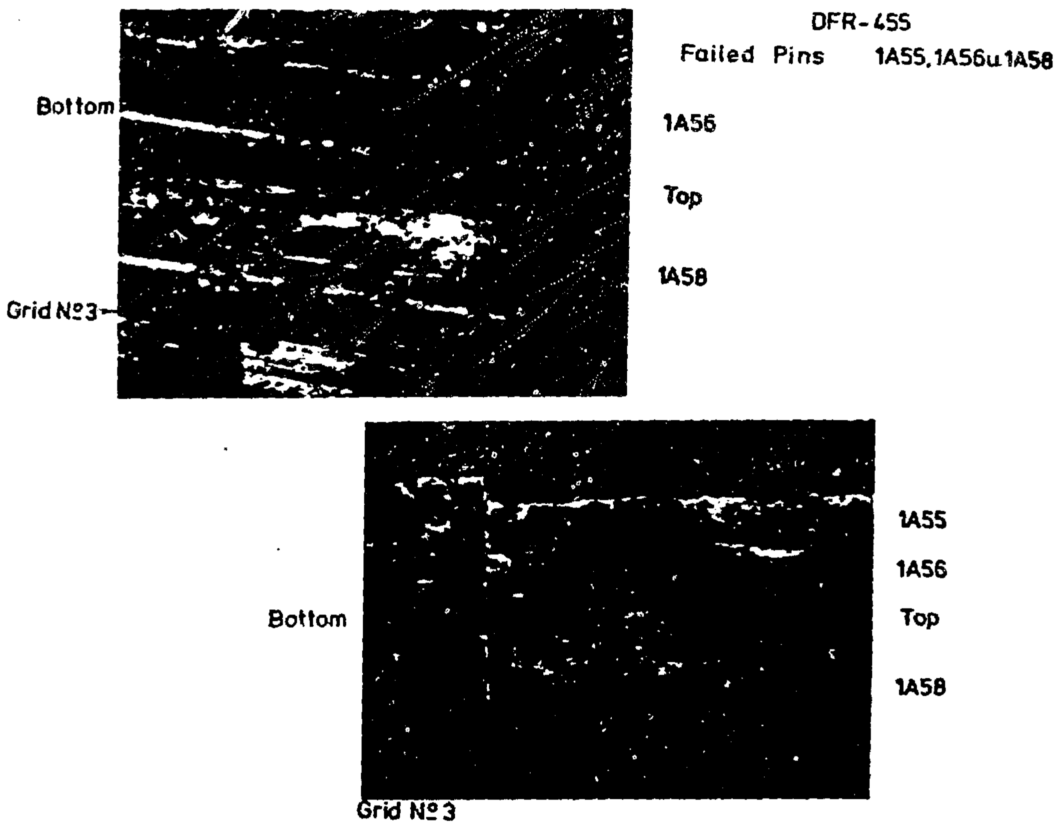



Fig. 5

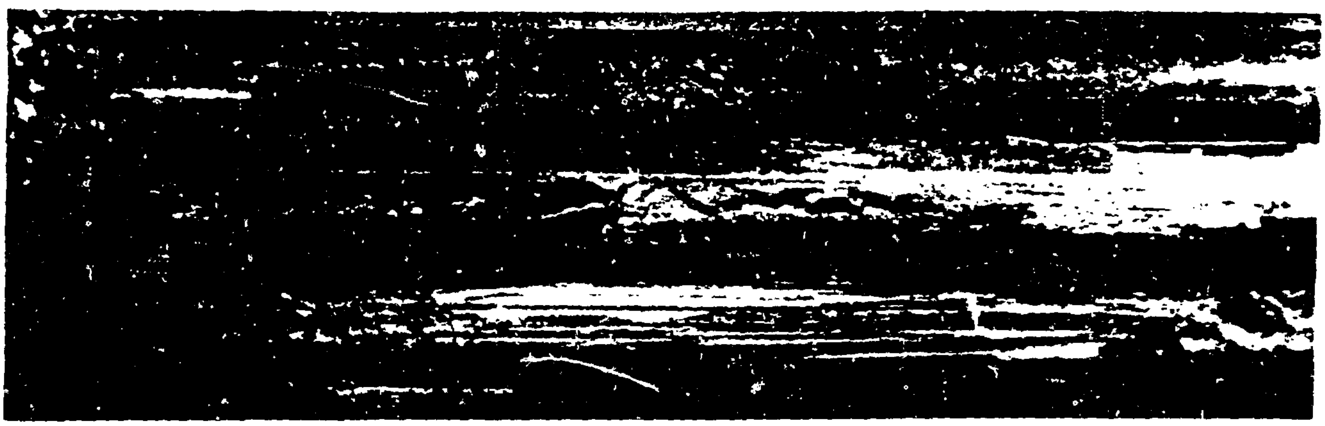
Bundle damage in DFR-455 [12] (by permission of IAEA) minutes to a day, the boiling remaining stable throughout [16-19]. The down-flowing coolant should provide a worst-case scenario for bubble attachment and vapor blanketing.

The reactors or experiments discussed so far had no effective DN detection capability, except for BR-5 after 1964 and Fermi after 1970. Although the DNM information will not always be discussed with the following tests, to recognize which reactors and/or experiments can and did provide DN data is important [1].

Experiments in the BR2. Like the Engineering Test Reactor (ETR), the BR2 in Mol, Belgium, provides an epithermal flux with a cadmium filter on the test loop [20]. Several BR2 experiments bear directly on fuel failure, FEFP, blockages, and their accommodation.

Mol 7A (MFBS-5), a $U_{0.80}Pu_{0.20}O_{1.99}$-fueled seven-pin test, ran from September 1968 to September 1969 at 59 kW/m (maximum) to 44 MWd/kg with three cladding types. One pin had severe melting of fuel and cladding (of unknown cause, probably fabrication defect [15]); no other failures were reported. This is illustrated in Figure 6 [20]. This appears even more unusual upon considering that an earlier test, MFBS-4, ran from April 1968 to November 1968 wth similar fuel (three pins) at 70 kW/m maximum to 28 MWd/kg with molten fuel, but without a failure (as in TREAT D1 and D2, discussed below). MFBS-6, a key test run from January 1970 to December 1972 at 57 kW/m maximum (and 650°C maximum cladding temperature) to 95 MWd/kg, had nine pins with pellets and five with either SOL-GEL or vipak fuel. Two SOL-GEL pins failed severely and might have propagated to at least two pins with pellets [12]. The other failed pin contained vipak fuel. Although the pin failures could be due to propagation (particularly with the extremely high fission-gas pressures present), the postulated propagation was limited and the S/A accommodated the faults.

Before Cleaning



After Cleaning

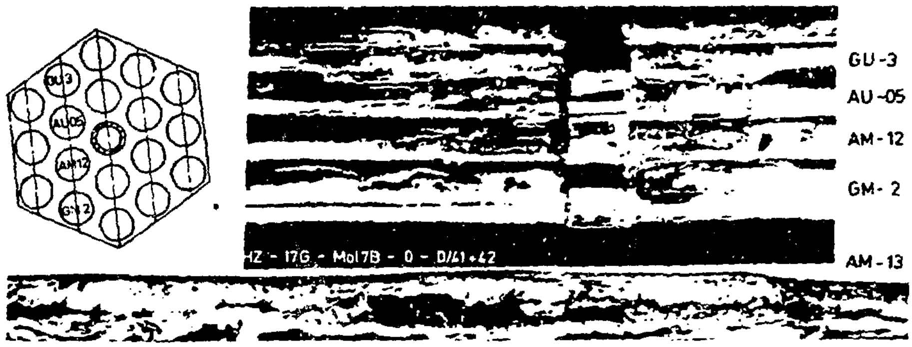
Fig. 6

BR2 Mol-7A (MFBS-5) Experiment; Fuel Pin Damage Between Spacer Grids 7 and 8. [20]

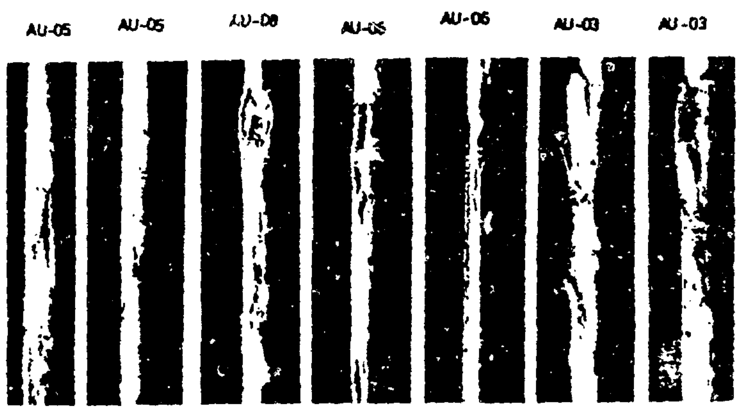
The Mol 7B fuels irradiation experiment was the next BR2 test of significant importance to the LOA-2 accommodation topic discussed in reference [1]. This test irradiated 18 pins of 70 percent enriched $U_{72.5}Pu_{27.5}O_2$ from July 1972 to February 1974 at SNR-300 hot-channel temperatures and high power (700°C and 50 kW/m, maximum). Sixteen pins contained sintered pellets; two were vipak. The first failure was detected at 38 MWd/kg (~5.3 percent b.u.) after which a simple DNM was installed. The bundle was irradiated to 81 MWd/kg (~11.3 percent b.u.). Although only one fuel failure was reported initially [15], postirradiation examination (PIE) revealed that all pins had failed and a local 38 percent blockage had formed at one axial location (Figs. 7 and 8; note the central dummy pin [12,13,21,22]), $Na_3(U,Pu)O_4$ formed and the fuel swelled extensively - effects enhanced by the 16 reactor shutdowns experienced and the ~50 ppm $O_2$ present. Weimar [21] concluded, "a bundle running beyond failure will show the first blockage in the colder outer sub-channels". The pressure drop across the S/A was not held constant, but the Na flow rate was.

One should carefully note that the above MFBS and Mol experiments were fuels irradiation tests (as were the DFR tests) and were not intended for eventual interpretation with regard to safety (i.e., local faults).

We progress from the blockage formed in Mol 7B to the midplane 24-channel 40-mm-long porous SS blockage (of 0.5 mm SS spheres) inserted into the 37-pin bundle Mol 7C series (May 1977 and March 1978; Mol 7C/3, run successfully on Oct. 29, 1980 [23,24] was identical except the blockage had 10 µm chromium-plated 93% enriched fuel spheres and Mol 7C/N will repeat this except the fuel pins will be irradiated. For 7C/1 and /2, the experimenters ran BOL 65-90 percent enriched $UO_2$ fuel at 40 kW/m maximum to 2.5 MWd/kg, halved the flow, interrupted the local cooling of the blockage (via flow through a central tube), and continued at power for 49 min (Mol 7C/1) and 6 min (Mol 7C/2). At least six pins failed (detected quickly by DND as shown in Figure 9 [26]) and secondary blockages formed in Mol 7C/1; preliminary PIE showed no $Na_3UO_4$ formation and all failures to be between the blockage and the next spacer grid downstream [26,27]. Figure 10 beautifully illustrates the bundle condition of Mol 7C/1. Mol 7C/2 was more mild. Two of the conclusions drawn were [28]:



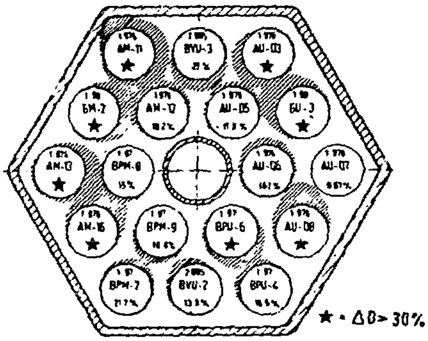

a. View of the Defect Region (Hot End)




b. Some Pins of Mol-7B with bigger Defects

Fig. 7

Bundle and Pin Damage in Mol 7B with $Na_3UO_4$ Blockage [21]



Defect Size and Orientation
(Mol-7B)

Fig. 8

Schematic of Mol 7B Bundle Showing Regions of Blockage and Failed Pins [21].

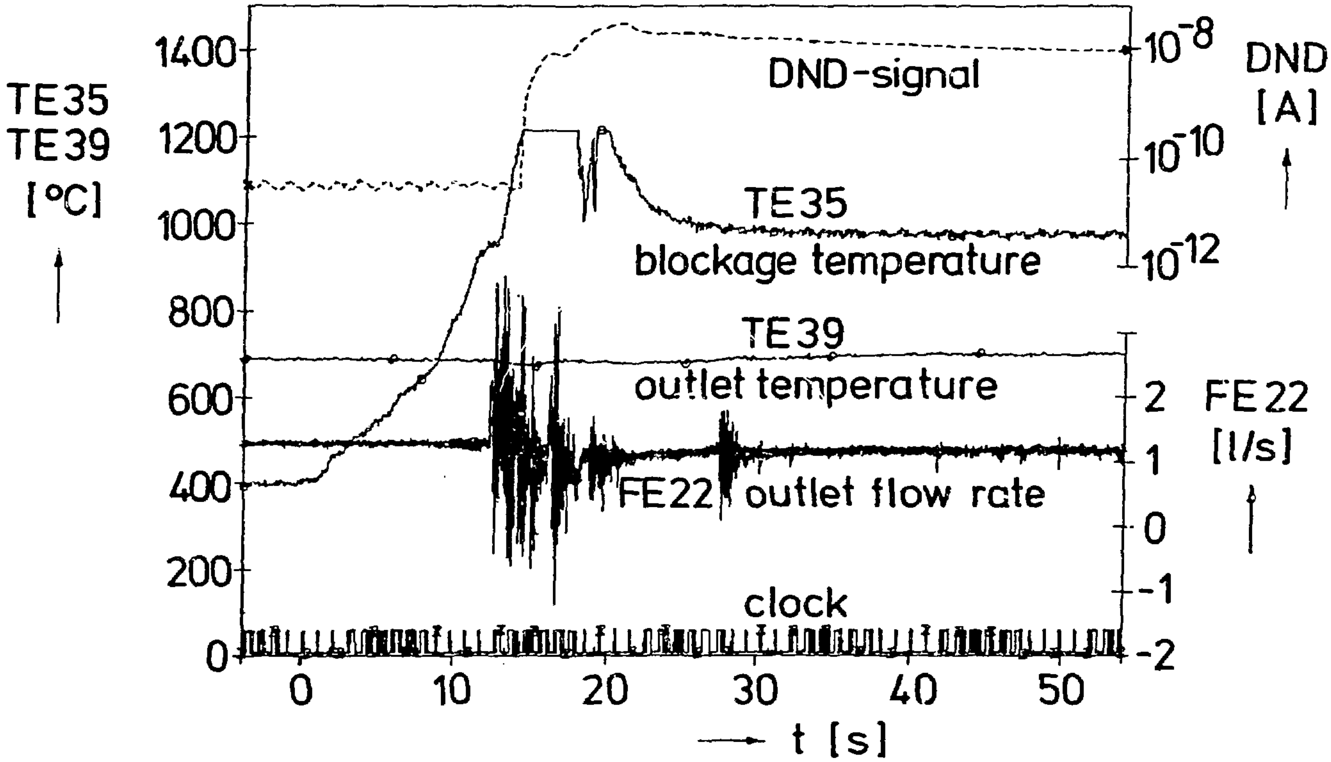



Fig. 9

Typical Signals of the In-Pile Blockage Experiment Mol-7C/1 [26] (courtesy of W. Kramer, KfK).

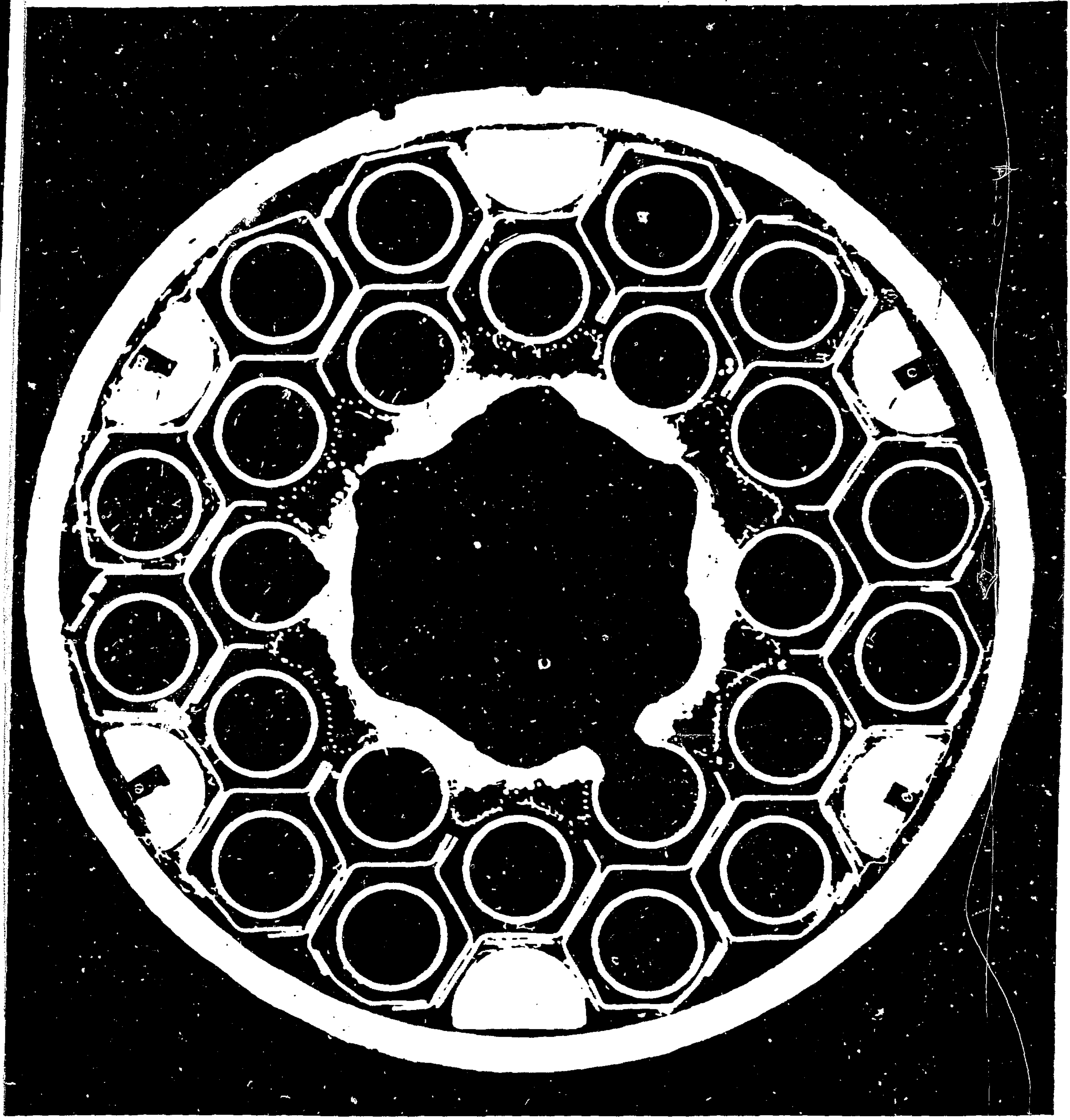


Fig. 10

PIE of Mol-7C/1: Blocked Region [26] (courtesy of W. Kramer, KfK).

1) At continued power production local cooling disturbance with enforced pin failures do not lead to a fast pin-to-pin failure propagation,

2) Even at a strong destruction of the original fuel element geometry, integral cooling is guaranteed over a long time at full reactor power.

Figure 11 illustrates the signal responses in Mol 7C/2; a striking result is shown in Figure 12 where the DN responses are nearly identical for Mol 7C/1 and /2 [27]. The Mol 7C experimenters anticipated similar responses from Mol 7C/3 and 7C/N [29,30]. Indeed, the Mol 7C/3 DN response was lower [24].

Differences between the Mol-7C bundle and a prototype U.S. LMFBR include the pin length, six dummy S.S. rods, one dummy central oversized coolant-filled S.S. tube, and low flow, besides the aforementioned items. Most non-prototypicalities appear conservative for bundle coolability. We anxiously await the Mol 7C/3 results for a blockage of highly enriched $UO_2$. The Mol 7C/3 results for a blockage of highly enriched $UO_2$ should be available after July 1981. Later, Mol 7C/N will investigate an identical blockage accident with a bundle of preirradiated fuel.

From the drama of what would appear to be bounding experiments in DFR and BR2, we retreat to experience that includes local faults otier than "age-old" failures and appears more mundane (as we hope all operating experience will be).

Miscellaneous Fuel-Failure Experience and Other Local Faults. In 1973, the KNK reactor at Karlsruhe experienced an unexplained transient partial blockage of one subassembly (~35 percent reduction of flow, again during startup), apparently with no damage [31]. (Recent KNK-II experience with fuel-failure detection and location has been addressed by Jacobi [30]).

The Rapsodie subassembly Capricorn 1b, had a fuel failure on rise to power (at 20 percnnt full power) detected by DND, acoustic, TC, and other signals. The cause is attributed to the dislocation of a templug holder; two pins failed and released small amounts of fuel (>5g) into the sodium [32].

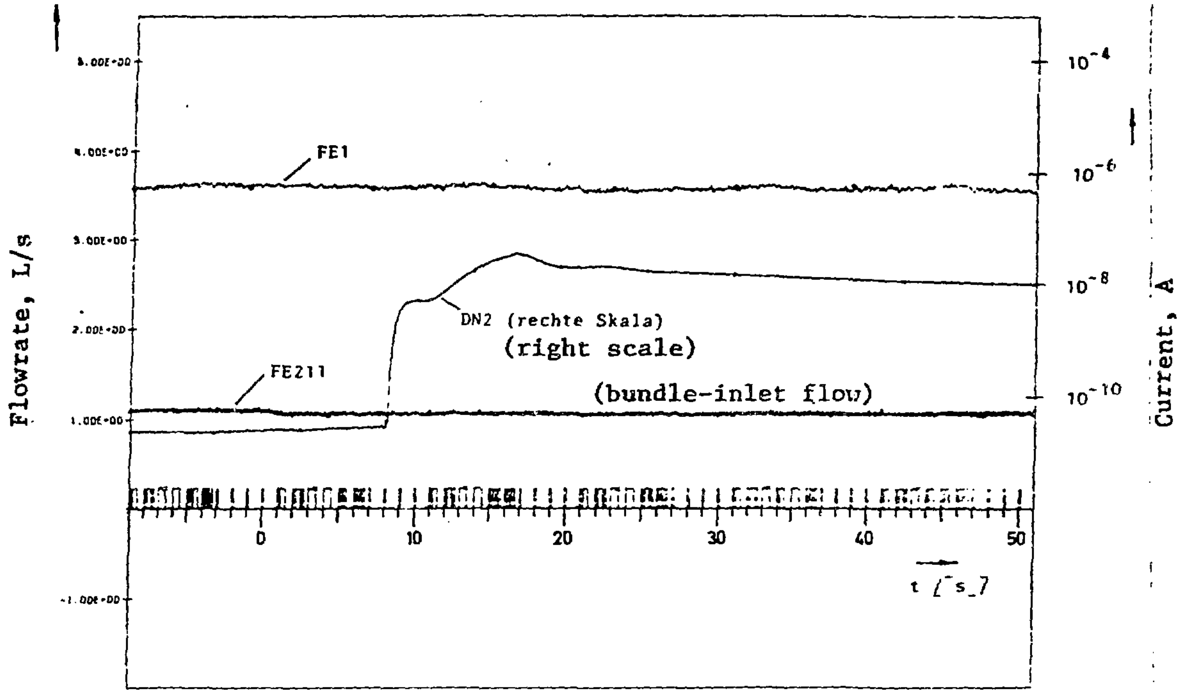



Fig 11

Trace of Signals from Total Flowmeter, Bundle-Inlet Flowmeter, and Delayed-Neutron Detector for One Minute During Mol 7C/2 Transient (courtesy of W. Kramer, KfK)

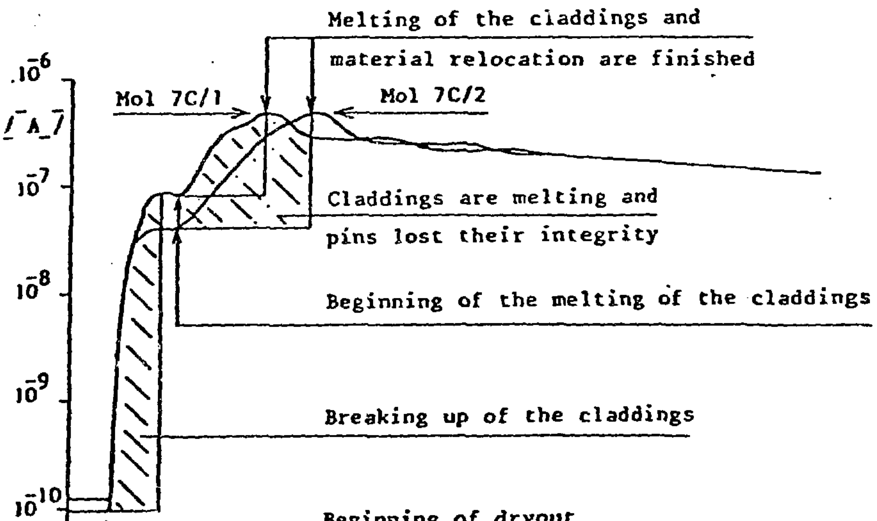



Fig. 12

Ionization Chamber Current, I, of the Logarithmic DND Channel DN-2 for Mol 7C/1 and Mol 7C/2. [27]

TREAT D1 and D2 were seven-pin tests run with fresh 20 percent (peripheral) and 26 percent (central) enriched $UO_2$ of FFTF design except for shorter fuel length (348.3 mm). The center pin in each had a short "enrichment-error" section (38.7 mm fully enriched) to provide power ~2.1 times that of the 26 percent enriched portion (i.e., 140 kW/m in D2). The 0.38 mm cladding did not fail, even though ~58 percent areal average and ~75 percent areal maximum of the overenriched fuel melted and relocated axially in D2, the more severe test with power increased 71 percent over D1 [33].

Like DFR, BOR-60 has provided more evidence of the behavior of severely failed fuel; it has operated with up to one percent failed fuel. The experience of interest to our subject is that found upon investigation of >150 37-pin S/As with >10 percent b.u. irradiated from 1970 to mid-1974. The $UO_2$-fueled pins were of annual pellet (50 kW/m) and vipak (59 kW/m) designs. Eleven S/As contained failed fuel; failures were registered in one "at 2 percent b.u., another at 7.7 percent b.u. and the remaining ... at 9 percent and higher burnup." [34] The pellet-fueled 7.7 percent b.u. failed-pin remained in the reactor for a month and "the fuel ... was not found at all." Three emergency shutdowns during the failure period and the wide longitudinal breach shown in Fig. 13 infer why all fuel was lost. The pin (at top) is difficult to see because the view is of a "cavity" or "black hole." A 37-pin S/A (A-89) with 10 percent b.u. had 20 failures with less damage to the four of 18 peripheral pins (see Fig. 18). The failure mechanism suggested is fuel swelling that exhausted cladding ductility for initial failures and subsequent failures stimulated by the presence of those failed pins (slow FEFP). Finally, an "essential fuel loss was observed in a pin from the subassembly EB-158" (b.u. not cited, although S/As were irradiated to 13 precent b.u. without pin failure) [34]. Aristarkhov et al. [35] provide more information; S/A EB-158 is cited to have 8.45 percent b.u.

EBR-II RBCB-1 had 16 10.8 percent b.u. pins mixed with 21 2.0 percent b.u. pins in a reconstituted 37-pin S/A. After five days, a high b.u. pin failed and the S/A was stored in a basket for six months. The S/A was returned for further irradiation and after five days of nearly full-power steady-state irradiation, the highly sensitive DNM exceeded the 800 cps limit with a DN spike which terminated the test. The results were minor compared to

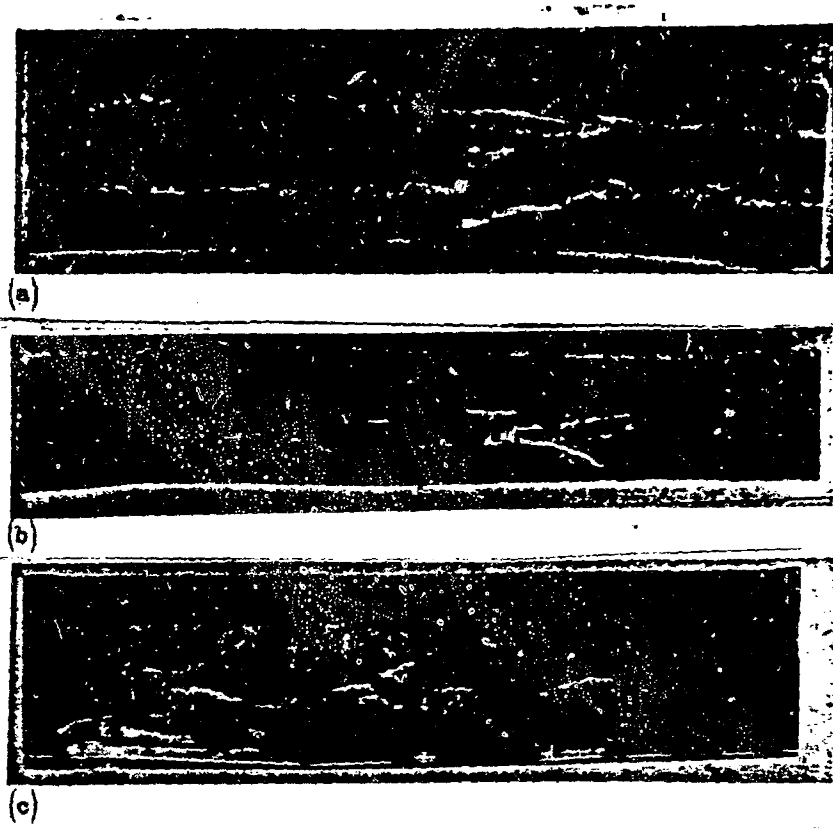



Fig. 13

BOR-60 Breached Pins with (a) All Fuel Lost and (b and c) Little or No Fuel Loss [34].

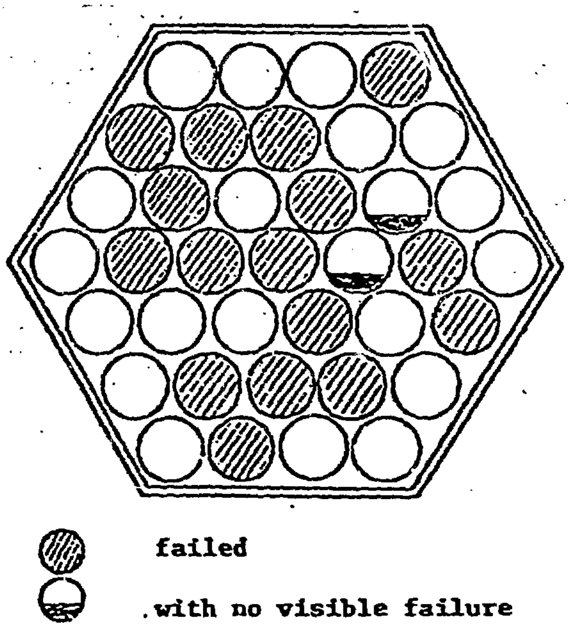



Fig. 14

Schematic of BOR-60 Bundle Showing Distribution of 20 Failed Pins [34].

those seen in the more severe foreign tests discussed above. The center of the longitudinal crack in the 10.8 percent b.u. pin faced a "cladding stain;" although this would appear to be a case of the classical self-limiting FEFP, the stain is attributed to line pin-to-pin contact (a case of wire-wrap loosening) [36]. This domestic result alarmed some, but we can see from the foreign experience that such alarm was unwarranted.

Indeed, information on bowing and deformation has been obtained at EBR-II with instrumented subassemblies; so far these cases have been of benign nature. In addition to the above foreign experience, we have a recent report [37, 38] which contains photographs of a potted cross section of a Phenix subassembly after 6.6 h.a.% burnup. One photograph, shown in Figure 15, illustrates 1) wire-wrap movement from the nominal location, 2) fuel pin contact with the subassembly wrapper, and 3) contact between two and possibly three fuel pins. Upon photographic "reversal" and shading the involved fuel pins, one can clearly see the abnormal (bowing and deformation) regions (see Fig. 16).

Prototypic fuel failure results from BR-10 (e.g., reference [39] which includes FPM and DNM traces), BN-350, and Phenix have shown the failures to be benign and readily detected in a large reactor with very little background DN signals. However, if the U.S. operates LMFBRs with failed fuel, we must better understand the meaning of DN and GFP signals. (The reader is referred to references 15, 40, and 41 for more detailed information on some GETR, EBR-II (including more recent RBCB tests), DFR, BR2, FR-2, Rapsodie, and KNK II tests.)

To satisfy the DOE LMFBR LOA-1 and -2 criteria, we are faced with the question, "can subassembly coolability be maintained under severe local-fault conditions?" Our reactor operating and in-reactor experimental evidence, more extensive than one might first venture a guess, is highly supportive of an affirmative answer and reinforces conclusions drawn from out-of-pile tests and analyses.

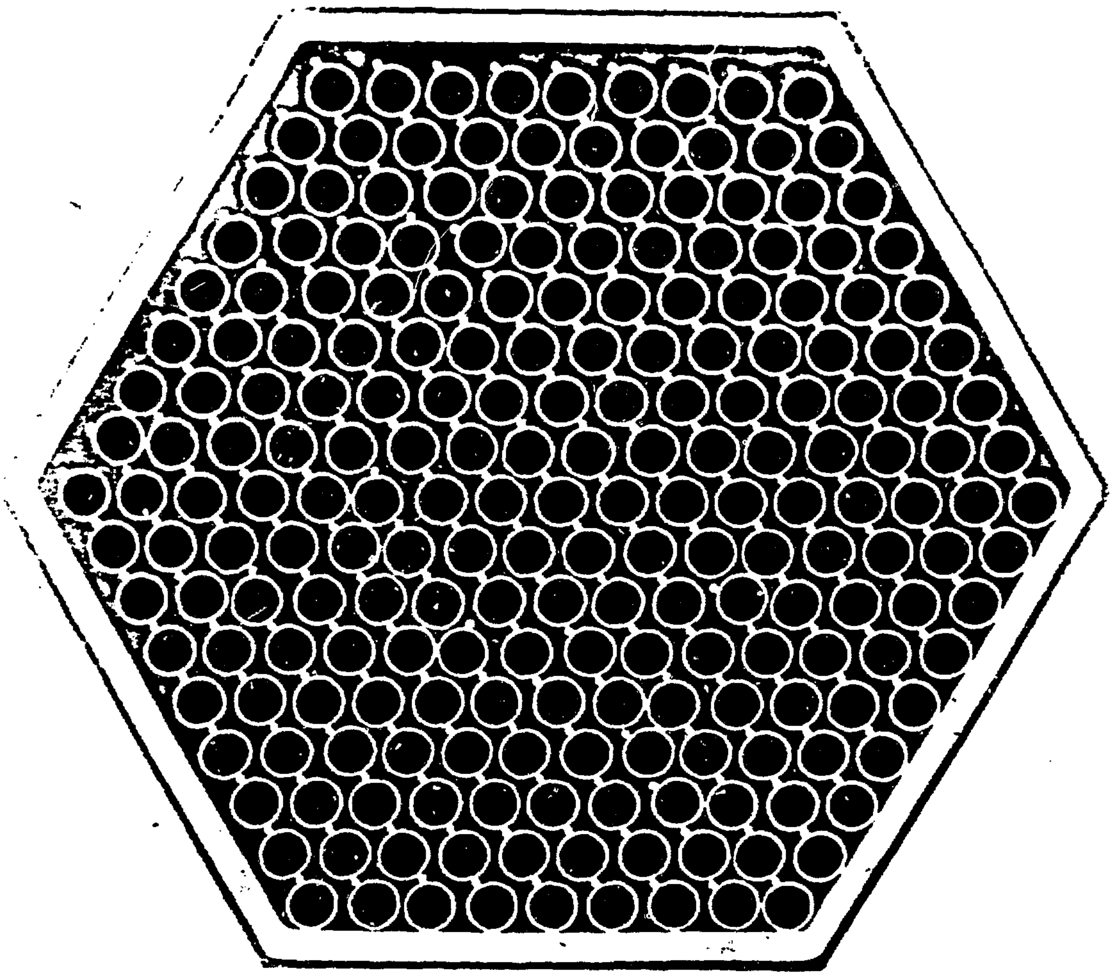


Fig. 15

Phenix Subassembly Cross Section after 6.6 at.% Burnup.

Photograph of Potted Subassembly [37,38]

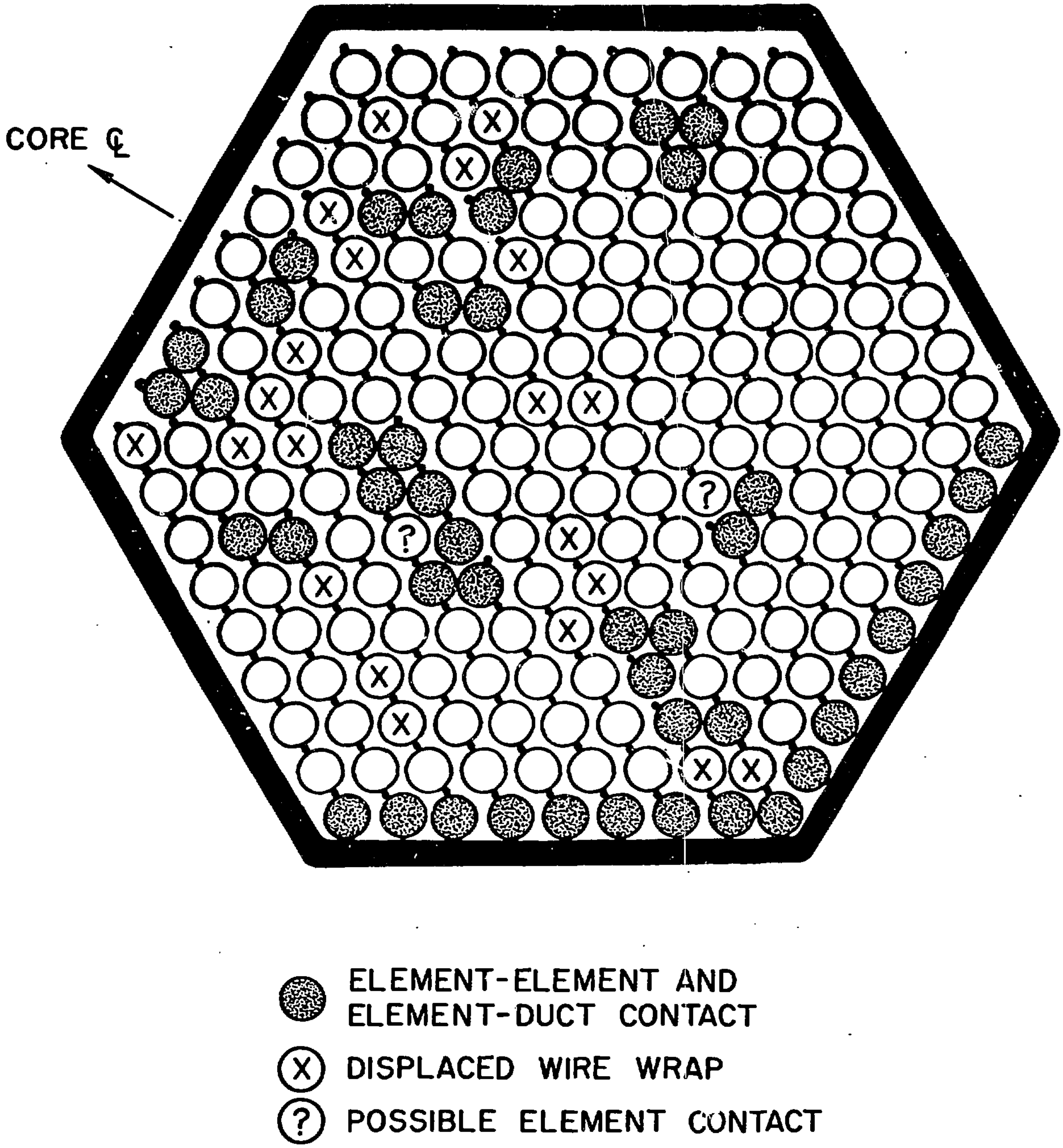



Fig. 16

Phenix Subassembly Cross Section after 6.6 at.% Burnup.

Photograph of Potted Subassembly Reversed to Illustrate Fuel-Pin Deformation.

## Conclusions

Reference [1] briefly outlined the LMFBR LOAs to show the constraints of fuel-pin design. It also reviewed local faults, summarized fuel designs and reactor parameters, and discussed the requirements of the fuel and failed-fuel detection. LMFBR operating experience and experiments are summarized here with several "worst-case" occurrences reviewed.

An immense amount of information is available on LMFBR operations and experience, but a simple tabulation of failed fuel elements does not suffice to provide the designer with proper feedback. The operating history, etc., as cited in cases here, must be taken into account as well as the cladding type, fluence, power density, and so on. In each case of extensive fuel-failure, one can attributed it to such non-prototypic factors as high $O_2$ content, high power, reconstituted bundles, vibratory-packed fuel, and high b.u. Thus, the conclusions of references 1 and 2, cited above, appear valid for prototypic operations of an LMFBR.

The fuel-failure data, incoming now at an ever-increasing rapid pace, must be screened to determine the behavior of fuel under prototypic, off-norm transient, and upset conditions. The fuel-failure rate in Phenix is reported to be less than 1 in $10^4$ fuel pins irradiated. This is indeed impressive.

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