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REVIEW OF RECENT ANL SAFETY EXPERIMENTS IN SLSF AND TREAT

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ABSTRACT

Among the recent significant in-pile experiments conducted by ANL are Sodium Loop Safety Facility (SLSF) experiment P4 in the Engineering Test Facility (ETR) and TREAT experiments F3, F4, and J1. The P4 experiment, which had three heat-generating flow blockages each installed in six coolant channels in a 37-pin bundle of FTR (Fast Test Reactor)-type fuel elements, investigated the bounding consequences of severe local faults. The principal objectives were to eject molten fuel into the bundle geometry and, during subsequent extended operation, to characterize the behavior of (and response of instrumentation to) any subsequent blockage growth; secondary objectives included characterizing the severity of any molten-fuel/coolant interaction and the response of the coolant. The F3 and F4 experiments in TREAT were phenomenological tests to study the fuel-column disruption mode in loss-of-flow accidents. The J1 experiment was the first slow period (~10 s) transient overpower experiment done in TREAT. Results of these experiments will be presented.

INTRODUCTION

The results of a large number of in-pile experiments that have been conducted by Argonne National Laboratory (ANL) have been reported at past international meetings.[1-5] Since those experiments were conducted and reported the results from several more in-pile experiments which have been conducted by ANL have become available. The experiments for which results are now available are SLSF experiment P4, and four TREAT experiments F3, F4, J1, and L01. The results from TREAT experiment L01 are reported elsewhere at this meeting [6]. SLSF experiments P4 was a 37 pin experiment designed to study local faults accommodation. TREAT experiments F3 and F4 were single-pin phenomenological experiments to study fuel break-up and dispersal in a voided channel LOF accident. TREAT experiment J1 was the first slow-period (~10 s) TOP experiment done in TREAT. It was done with a 7 pin bundle of previously irradiated fuel in a flowing sodium environment.

SLSF TEST P4

Objectives

SLSF experiment P4 was an experiment in the LMFBR Safety Program Plan to demonstrate coolability of local faults and local faults accommodation by inherent mechanisms. It was the seventh, and last, in the series of SLSF large-scale in-reactor experiments and the first to simulate something other than a whole core accident. P4 was planned to release molten fuel into the coolant stream and bundle geometry from one or more of the three canisters built into the 37-pin test subassembly and to probe the consequences of continued power operation with fuel failure.

The overall objective of P4 was to demonstrate with an in-reactor experiment that a hypothetical blockage, which bounds the consequences of credible local faults, could either be tolerated or be detected by global monitors in time to prevent significant fuel failure or blockage propagation.

The specific objectives of the SLSF P4 bounding local-faults experiment were to determine 1) the extent, if any, of fuel-failure propagation, and 2) the signals received which indicate such propagation after the release of small amounts of molten fuel (design goal of 10 to 30 grams, each canister) from one or more of the fuel canisters at the fuel's beginning-of-life. A primary goal for the P4 experiment was to provide information needed to define upper-bound signal characteristics from whole-core instruments. This would confirm experimentally that continued operation of an LMFBR is safe following the occurrence of a "small" local fault, including pin failure whose signatures fall within the defined band. Specific experimental information expected from the experiment P4 included: nature of fuel release; nature of secondary failure, if any; the amounts of molten and solid fuel released; the extent of damage to the fuel pins; the tendency of the released fuel to either be swept out or form secondary blockages; and the delayed-neutron (DN) and fission-product signals characteristic of molten-fuel release. Secondarily, information was expected on the severity of a molten fuel-coolant interaction and the resultant response of the coolant.

Test Articles

The 37-pin P4 test subassembly contained 34 full-length FTR-type fuel pins containing enriched mixed oxide fuel and three pins with 10-cm long, sealed fuel canisters as the center sections of their fueled regions. Two canisters were cylindrical and one had a fluted geometry (Fig. 1). Different geometries and degrees of cladding cold work were employed to provide diversity and redundancy in achieving molten fuel release. Total fuel content of the three canisters was ~180 grams, equivalent to the total fuel inventory of one FTR-type pin.

The P4 loop consisted of primary and secondary containment vessels, an annular linear induction pump, a sodium-to-helium heat exchanger, a cadmium neutron filter, and the instrumented test train. The test train [7] contained the test subassembly and much of the test instrumentation. The test train had two flow channels: the main flow channel through the test section and a parallel bypass channel. In addition to the test train and loop instrumentation, a delayed neutron detector (DND), an on-line cover-gas system (OLCS), and an on-line sodium-sampling system (OLSS) were operated to observe fuel failure [8,9].

Design Operating Conditions

The P4 design operating condition was a nominal bundle power of 1272 kW and test subassembly coolant flowrate of 3.38 kg/s. This operating condition simulated the nominal maximum FTR fuel pin linear power, but at a coolant flowrate that duplicated the FTR row four (minimum flow) orificing region of the core. This yielded a power-to-flow ratio about 24% higher than nominal for the FTR core-center subassemblies, typical of the hot-channel conditions. The steady-state coolant velocity was 6.0 to 6.3 m/s in the central coolant flow subchannels and resulted in a frictional pressure drop of 0.24 MPa across the full-length test fuel pins. The presence of the three fuel canisters contributed about 7 kPa to the total flow resistance of the 37-pin bundle. Loop flow was 8 kg/s to maintain the steady-state inlet temperature to the heat exchanger < 839 K. Bundle inlet temperature was 695 K the inlet temperature for the FTR rated core. Loop cover gas pressure during most of the P4 irradiation was 0.16 MPa to avoid cavitation in the OLSS pump. During and immediately following the power transient, the cover gas pressure was reduced to 69 kPa to simulate LMFBR pressure levels in the test subassembly.

Planned Power Transient

The P4 power transient was initiated on August 21, 1981, at a bundle power of 338 kW (40 MW ETR power). The ETR neutron level, under the control of the power transient controller, followed the planned rise to 4.375 times the initial neutron level (175 MW) in 27.75 s, held at that level for five seconds, ramped down to 3.9 times the initial neutron level in five seconds, and held at that level (156 MW). The first disturbance in steady temperature, pressure and flow conditions was observed at 15.2 s. It was attributed to initial failure of the fluted fuel canister. [*Fluted canister cladding failure was predicted to be caused by sodium boiling and cladding dryout in annular cooling channel between the canister and adjoining fuel pin and was predicted to occur between 14.5 s and 16.5 s into the transient*]. The delayed neutron level "bumped" upward following initial failure of the fluted canister as the area of fuel exposed to the sodium increased. There was no evidence of gross molten fuel release, molten fuel-coolant interaction (MFCI), or flow blockage. Test section flow perturbations of 10% were observed, as were temperature perturbations of up to 20K recorded on the wire wrap thermocouples. Temperature perturbations were also seen in the signals of thermocouples located downstream from the cylindrical canisters. The temperature perturbations increased with time and an increasing ETR power level.

ETR reached a power level of 175 MW at 27.75 s and held at that level until 32.75 s. During this period there were flow and temperature perturbations in the test section. Significant local temperature perturbations occurred downstream of the fuel canisters. [*The 20% and 10% cold-worked cylindrical canisters were predicted to reach failure strains of ~8.8% at 24.9 s and 26 s, respectively*]. Although failure of the cylindrical canisters had not occurred yet, the "ballooning" of the cylindrical canisters in the test bundle were probably perturbing local coolant flow. Both the increased cross-sectional area of the cylindrical canisters and the resulting tighter packing or dislocation of the pins would contribute to reduced cooling.

Molten fuel release occurred at ~34 s (170 MW) and was accompanied by perturbations in inlet and outlet flow, in local coolant temperature levels, in acoustic noise, and by a sharp increase in DN level. Inlet flow decelerated to a minimum 2.3 kg/s and returned to persist at a flow of 3.2 kg/s, or 93% of nominal. About half the test section thermocouples indicated a response to the molten fuel release; the others indicated little change. Thermocouple TE 3-14 on pin 14, located 0.36 m below the fuel midplane indicated a temperature jump of 600 K, typical of molten fuel contact and formation of a new junction. Thermocouples TE 6-1 and TE 6-2, with junctions 0.14 m above the fuel midplane on hex duct flats 1 and 2, respectively, also were hit by molten fuel and failed. Thermocouple TE 3-7, junction 0.9 m above the fuel midplane on pin 6, indicated a 140 K upward spike at 34 s, then recovery. These large local temperature perturbations were not repeated elsewhere in the test section. This indicated that the molten fuel release was from the fuel canister on pin 5 and was directed toward the outer row pins and hex duct. The minimum axial distance traveled by the molten fuel in reaching thermocouples TE 6-1 and 6-2 was ~0.1 m. The temperature at thermocouple TE 3-12 (pin 12) dropped only 50 K after the ~100 K upward spike at 34 s, indicating reduced local cooling in the area containing the released fuel. Test section flow remained steady at 3.2 kg/s until 54 s, when flow increased to 3.35 kg/s (97% of nominal). The flow increase was preceeded by a 40 K drop in temperature at thermocouple TE 3-14 (pin 14), beginning at 50 s, and accompanied by a pause in the gradual temperature increase exhibited by the test section and outlet thermocouples. ETR power remained steady at 156 MW and loop operating conditions continued to be quasi-steady state. Inlet flow began to gradually drift lower after the increase to 3.35 kg/s.

The rate of inlet flow reduction increased at 80 s and temperature offsets were observed in a number of test section thermocouples, similar to the events a few seconds prior to the first cylindrical canister failure. At 86 s, the cylindrical canister on pin 8 failed, releasing molten fuel toward the center of the bundle. Again, about half the test section thermocouples showed a jump in indicated temperature; the other half were relatively unaffected by the release. Most of the thermocouples responding to this fuel release were located on the 19 center pins and retained most of the temperature offsets that accompanied the fuel release. TE 3-15 (pin 18) indicated a 380 K temperature increase, from 845 K to 1225 K, and then a drop to 1125 K. The temperature at the junction of thermocouple TE 3-14 (pin 14) jumped from 1060 K to 1280 K and then appeared to indicate steady sodium boiling thereafter. Molten fuel release was accompanied by three flow perturbations, within 0.5s, that temporarily reduced inlet flow to 1.3 kg/s. Flow then recovered to 2.95 kg/s (86% of nominal). Two broadband noise peaks were indicated by the acoustic sensors. The DND signal peaked at 88 s. Test section exit temperature increased 40 K, from 1065 K to 1105 K. Inlet temperature at 86 seconds was 739 K.

Operation continued at a steady test section flow of 2.95 kg/s until 110 s. Thermocouples TE 3-2 (pin 3), TE 3-7 (pin 6), and TE 3-15 (pin 18) began to increase in temperature prior to a gas release from some fuel pins at 110 s. The gas released into the test section perturbed the coolant flow and cause large oscillations in the indicated flowrate at the bundle exit and total loop flowmeters. Test section temperatures increased while the released gas was being swept upward in the loop, then began to return toward earlier values as the test section flow recovered to 2.95 kg/s.

Test section flow began to drift lower at 117s and ETR scrammed at 118.4 s on a low test section flow (low flow setpoint of 80%). Following ETR scram, the pump returned to its previous 2.95 Kg/s (86% of nominal) level.

Subsequent Irradiation and Blockage Reconfiguration

Preparations were made for a return to full power operation and ETR power operation resumed late on October 1,1981. The power increase proceeded in 20 MW increments in a step-and-hold manner toward a full-power level of 156 MW.

About seven minutes after reaching 92 MW, on October 2, 1981, the DN signal from detectors around the loop sodium plenum (DND) began to increase. Some reactor power trimming to reach 100 MW was in progress but its relative change was less than the DN change. Half a minute later the DN signal from detectors in the OLSS began to increase. The increase in signal was very gradual at first, building in an exponential manner as time passed. About three minutes later, there were temperature perturbations of up to 15 K indicated by several thermocouples. This coincided with the beginning of a sharp increase in the DND signal and indicated that events were building toward a change in the blockage configuration. Fifteen seconds later the DNM signal also began to increase rapidly and the inlet flow began a gradual decrease which steepened after about 4.55. About 0.55 later, after temperature increases occurred on the thermocouples on pin 8, the blockage reconfigured.

Basic characteristics of the flow response during the blockage reconfiguration appeared similar to those for molten fuel release from the cylindrical fuel canisters. There was a flow deceleration at the inlet followed by a persisting flow reduction. Inlet flow dropped from 2.55 kg/s to 1.75 kg/s and recovered to 2.25 kg/s. Flow held at 2.25 kg/s for ~0.35 s prior to another perturbation in inlet flow, and reactor scram. The ~0.5 s between increase in the bundle flow resistance and a subsequent inlet flow perturbation that did not produce a further flow offset were also observed during the P4 power transient.

Temperature jumps observed on thermocouples TE 4-8 (pin 35) and TE 4-6 (pin 32) indicated that molten fuel moved toward hex duct flats 3 and 4. Pin 19, thought to be intact following the P4 power transient, was open to the sodium following the blockage reconfiguration. Failure of pin 19 provided additional evidence that the reconfiguration originated near the center of the bundle and moved outward into previously unblocked flow channels near hex duct flats 3 and 4.

Test section flow, at pump benchmark voltage, changed from ~86% to ~60% of nominal as a result of the blockage reconfiguration. This flow reduction corresponds to a halving of the flow area at the blockage, if it was in a single plane.

A pressurization test was run on August 25, 1981; pins 4, 6, 7, and 14 were determined to be open to the sodium; pins 3, 8, 11, and 13 were classifed leakers. Pin 19 appeared to be intact. A second pressurization test confirmed that pin 19 was open to sodium after the blockage reconfiguration.

Preliminary Observation of Molten Fuel and Blockage Behavior

In all of the fuel release and reconfiguration events, a small decrease in test section flow occurred immediately prior to the event. The inlet flow responded to the event with a pair of pulses separated by 20 ms. Some oscillation of inlet flow was observed for the next 0.4 to 0.5 s. At that time, either a second pair of pulses occurred or a single large pulse occurred. The oscillations in flow occurring between the pairs of pulses may be due to the formation of a small locally voided region in the test section or due to a characteristic frequency of the test section. The double pulse is not yet understood but may be either a characteristic of hot fuel exposure or a reflection of a pressure pulse. The magnitude of flow disturbance due to the interaction of molten fuel and sodium was not of sufficient magnitude to be judged an energetic MFCI.

A number of fuel pin failures occurred during the power transient, releasing gas into the coolant stream. These failures appear to have occurred simulanteously and are probably due to local boiling and dryout in the blockage wake. The increase in inlet sodium temperature above the maximum expected peak was a contributing factor to reaching local boiling. One additional fuel pin was found to have a cladding breach following the blockage reconfiguration. This failure may have been caused by the reconfiguration or may be the result of events unmasking a previous failure, such as the movement of fuel which was blocking an existing leak in the pin. The latter is supported by the absence of a gas release during the reconfiguration. The response of the fuel pin pressure transducers to changes in loop plenum pressure became slower with exposure to sodium. This is indicative of formation of sodium uranate within the fuel pins. Verification of sodium uranate formation will be performed during post test examination.

The DND gave sufficient advance warning to initiate a reactor shutdown prior to blockage reconfiguration. This warning was definitely sufficient for an automatic shutdown system and may have been sufficient for operator intervention. The DNM, which is more typical of planned reactor failure detection systems gave advance warning but for a shorter time than the DND. The actual use of this system in a power reactor will be highly dependent on design and the supporting software needed to diagnose the DN signal. Consideration should be given to locating the DNM as close as possible to the core exit.

During the blockage extension, only a few of the wire wrap thermocouples gave any level change prior to the event and these changes were so small as to be noticed only when compared to the DND and DNM signals. The test section exit thermocouples showed no level change prior to the event. Based on comparison of the DN signals versus the in-core thermocouple signals, the P4 experience indicates that the DN detection systems are the more promising of the two for failure detection.

TREAT EXPERIMENTS F3 AND F4

The F3 and F4 experiments were performed to determine the timing and disruption mode of fuel pins in the lead power-to-flow subassemblies of a LMFBR with a moderately positive sodium void reactivity such as CRBR during an LOF situation. The intent was to study the clad and fuel behavior, especially fuel break-up and dispersal, subsequent to clad dispersal. Experiments which use a portion of one pin in a voided capsule were known to be appropriate, from the experience of experiments F1 and F2 [4].. A window was placed in the capsule so that the fuel break-up and dispersal could be studied photographically as well as with the hodoscope. The capsule design and the configuration of instrumentation for these tests are shown in Figs. 3 and 4. Figure 3 shows the axial region around the modified pin with a 5.08 cm (2.0 in.) fuel column. A mirror next to the fuel column reflects the pin image upward. The overall axial view in Fig. 4 shows how the pin image is eventually viewed with a high speed 16 mm motion picture camera positioned on top of TREAT. The fuel used in F3 and F4 had been pre-irradiated in EBR-II to 9% at a power level of 29.5 kW/m. The fuel pins were at room temperature at the start of the TREAT transients. TREAT power was increased to levels yielding power levels of 82.0 and 216.5 kW/m, in F3 and F4, respectively, and held for the duration of the transient. (Typical lead power-to-flow CRBR subassemblies operate at ~30 kw/m nominal power.) Fuel failure occurred at integrated powers of 206 kJ/m (871 J/gm) in F3 and 191 kJ/m (809 J/gm) in F4.

Interpretation of the pin image after pin disruption is difficult because the mirror near the pin degraded very quickly. In F3 (at 2000 frames per second), the pin failure event took less than 1.5 ms. A visible clad intact pin edge was observed on one side of the pin throughout the fuel pin failure event; ejection of fuel and clad occurred on the opposite side of the pin. The time scale for mirror degradation (~1.5 ms) and the initial pin-mirror distance (25.4 mm) imply an acceleration of ~2300 g from the pin to the mirror. Therefore the pin disruption was very violent.

The photography and the hodoscope analysis were done independently. Subsequently it was found that both observations agree on the time and direction of fuel ejection from the pin following failure. The failure times observed by the photographic and the hodoscope systems differed by only 20 ms nominally for F3 and 8 ms nominally for F4.

The fuel was not molten at failure in either F3 or F4. The clad may not have been molten at failure in F4, according to heat transfer calculations using the experimental conditions, including the experimentally determined pin disruption times.

A conclusion which is related to fuel behavior modelling is that initial pin disruption may occur in the absence of molten clad or fuel. A conclusion related to the execution of the experiment is that camera framing rates must be greatly increased to observe the detail of pin disruption process.

TREAT EXPERIMENT J1

The primary purpose of the J1 experiment was to demonstrate that a slow overpower simulation to fuel pin and cladding failure conditions could be done in TREAT. The fuel bundle consisted of seven pins, 0.34 m long fuel column, which had been irradiated in EBR-II to about 7 atom percent burnup. The test was done in flowing sodium in a filtered MK-II loop capable of providing reasonably prototypic hydraulic conditions. The power transient employed a ramp with a 10 s period; for fuel failure analysis, a ramp with this period is the slowest transient that would cause cladding failure within the energy limits of TREAT and the filtered loop used in this test.

Initially, at TREAT time zero, the coolant velocity through the fuel pin bundle was 610 cm/s (940 cm^3/s) and the coolant temperature was 750 K. The transient consisted of a power plateau at 180 MW (77 kW/m) starting at 4.29 s. A burst from this power level was initiated at 6.05 s. The burst had a nominal period of 10 s and reached a peak power of 255 MW (109 kW/m), at 9.114 s. Fuel pin failure, as indicated by flowmeter and pressure transducer response, first occurred at 9.035 s, when the reactor power was 252 MW (108 kW/m peak power in the fuel) and the corresponding energy release was 1019 MJ. COBRA analysis indicated that at this time fuel-pin mid-cladding temperatures at the top of the pin were in the region of 950 to 1000 K, and fuel areal melt fractions of about 45% had been attained at the axial mid-plane. A pressure pulse of approximately 0.8 MPa was recorded by the test section inlet pressure transducer and the inlet flow rate was momentarily decreased by about 25%. The second failure event, at 9.073 s, generated a relatively slow pressure increase (0.3 MPa in 40 ms) which was recorded on both the inlet and outlet pressure transducers. The inlet coolant flow was decreased to about 44% of its original value. The reactor power shutdown criterion was satisfied during this failure event and, at 9.13 s with TREAT power and energy at 254 MW and 1044 MJ, respectively, the direction of motion of transient control rods was reversed. The TREAT power decreased quickly to about 30 MW and then remained relatively constant until 10.2 s when the reactor was scrammed. This period of constant power permitted good observation by the hodoscope of the post-transient fuel motion without excessive "overpowering" of the fuel. At no time did the coolant flow rate halt or reverse, and eventually it stabilized at about 90% of its original value.

Preliminary hodoscope analysis showed an axial expansion of the fuel beginning at 8.69 s. Fuel started to emerge from the fuel pin bundle at 9.06 s with subsequent motion upward (downstream) and out of the hodoscope field of view. At 9.29 s there was evidence of a possible net downward motion of fuel. It is estimated that at least 50 g of fuel, that is, about 11% of the total fuel inventory of the pin bundle, moved upward and out of the hodoscope field of view. The post-test neutron radiograph indicated that cladding failure occurred at the top of the axial fuel column.

In conclusion, the J1 experiment demonstrated that low-ramp-rate experiments to cladding-failure conditions can be run in TREAT. Post-test analysis indicated that at failure the fuel-sample heating rate was roughly equivalent to a 12 to 14 ¢/s TOP excursion in CRBR. Mechanistic cladding failure threshold models calculated the failure time with acceptable accuracy. Empirical models did not calculate accurate failure times, which indicated their limitations outside of their data bases. Finally, this was the first experiment with preirradiated fuel in which both a significant quantity of fuel was removed from the active core region and in which a coolable geometry was maintained during the failure sequence.

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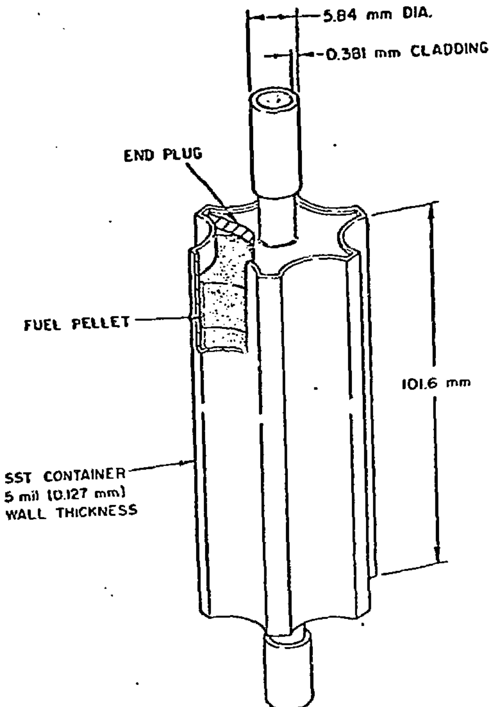
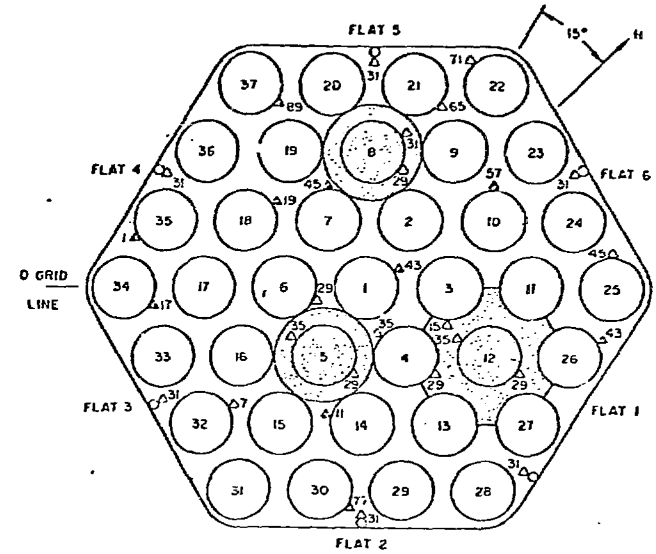


NOT TO SCALE

Figure ~~/~~ 1. Three-Dimensional View of Six-Channel Blockage Pin.



△ TC LOCATION SAME AS P3A

△ TC LOCATION FOR P4

NOTE: BLOCKAGE EXTENDS FROM 23.31 TO 27.31 in. ABOVE SPACER WIRE STARTPOINT.

Figure ~~/~~ 2. Wire Wrap Thermocouple Junction Locations (designated by the symbol, △,▲). Junction Locations, in Inches, are Measured from the Spacer Wire Startpoint Which is 7.31 in. Below the Heated Zone.

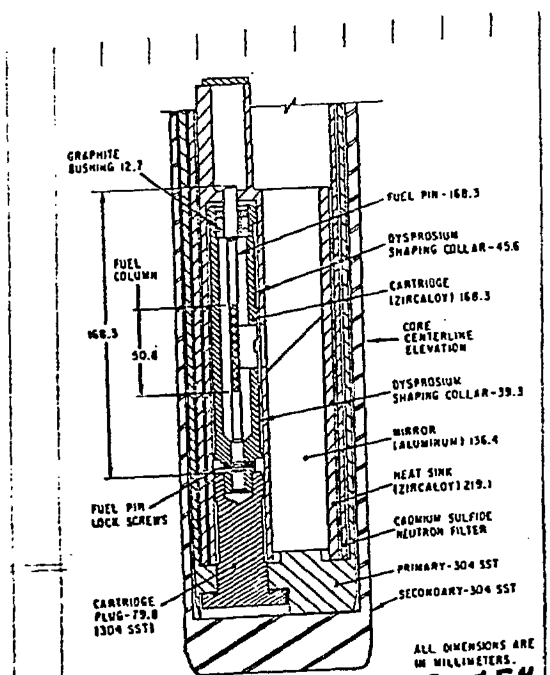


Fig. ~~14~~ 3. Fuel Region of F3 ~~through F6 Capsule~~ and F4 Capsule

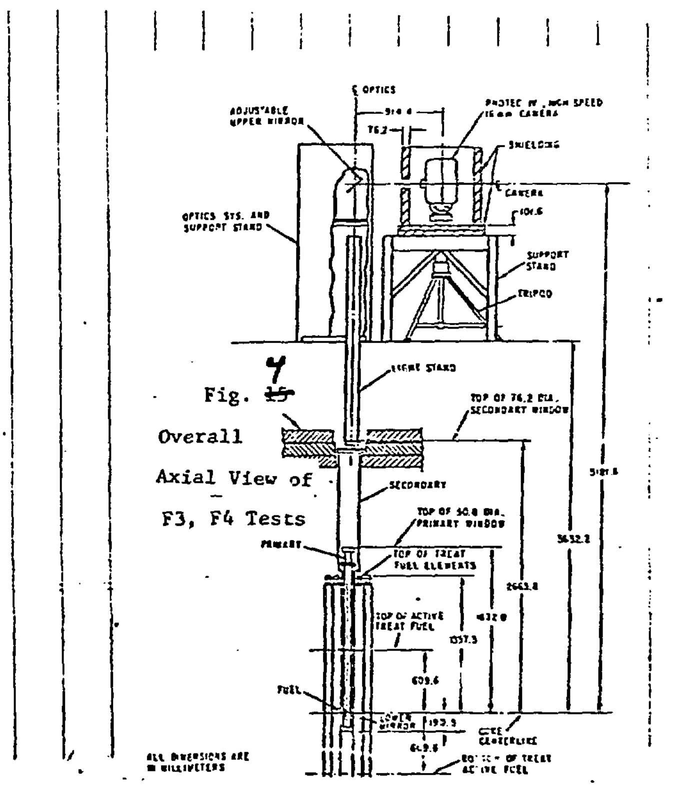


Fig. ~~15~~ 4

Overall Axial View of F3, F4 Tests