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SODIUM LOOP SAFETY FACILITY EXPERIMENT P4

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ABSTRACT

SLSF experiment P4 was designed to provide an upper-limit bound on the consequences of local faults. Three of the 37 full-length FTR-type fuel pins in the test subassembly were built with 10-cm-long, sealed fuel canisters as the center sections of the fuel regions. These canisters were intentionally designed to eject molten fuel into the bundle geometry during a planned power transient to full power. Different geometries and degrees of cladding cold work were employed to provide diversity and redundancy in achieving molten fuel release of 10 to 30 grams per canister.

The fuel canisters ejected molten fuel into the bundle geometry, as planned. Each ejection was accompanied by an inlet flow deceleration and persisting flow reduction. There was no evidence of energetic fuel-coolant interactions or failure propagation through the hex duct. After about 90 s of full-power operation, scram occurred on low test subassembly flow due to gas release from fuel pins as a result of elevated inlet temperature and boiling in the blockage wake. During subsequent power operation, blockage reconfiguration and flow reduction occurred 10 minutes after 60% power was reached. The DN signal began to increase from the steady state level three minutes prior to the reconfiguration.

INTRODUCTION

Sodium Loop Safety Facility (SLSF) experiment P4 was a near-term milestone in the LMFBR Safety Program Plan to demonstrate coolability of local faults and local faults accommodation by inherent mechanisms[1]. It was the seventh, and last, in the series of SLSF large-scale in-reactor experiments and the first to simulate 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 fuel canisters built into the 37-pin test subassembly and probe consequences of fuel failure with contained power operation.

P4 OBJECTIVES

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[2].

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 SLSF P4 include: 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 would be provided on the severity of a molten fuel-coolant interaction and the resultant response of the coolant.

P4 DESCRIPTION

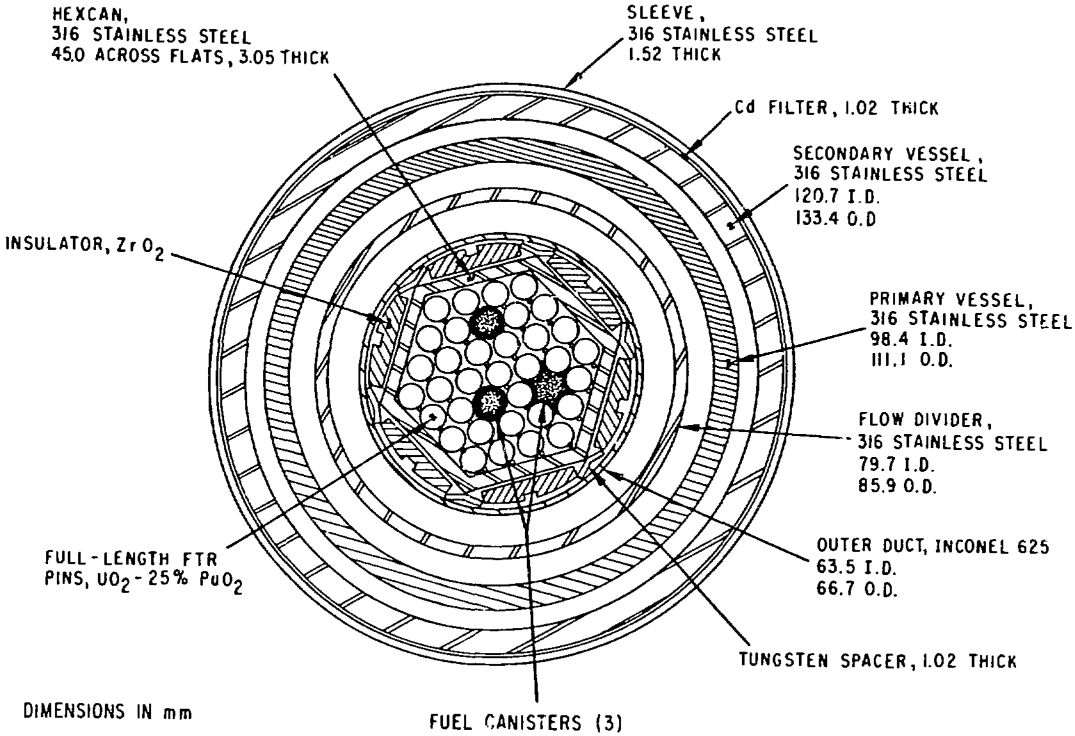
The 37-pin P4 test subassembly contained 34 full-length FTR-type fuel pins containing enriched mixed oxide fuel and three pins with 10-cm long, sealed fuel canisters as the center sections of their fueled regions (Fig. 1). Two canisters were cylindrical and one fluted geometry[3]. Different geometries 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 test train contains the test subassembly and much of the test instrumentation. The test train has two flow channels: the main flow channel through the test section and a parallel bypass channel. Sodium coolant is pumped downward in the annular flow channel between the test train and the primary vessel to the lower end of the loop. There, sodium enters the test train and flows upward through the test section and the bypass. These two flow streams merge and continue upward in the test train to the loop plenum. There it enters the helium-cooled heat exchanger and is cooled before cycling through the system again.

The SLSF loop consists of primary and secondary containment vessels, an annular linear induction pump, a tube-and-shell sodium-to-helium heat exchanger, a cadmium thermal neutron filter, removable top closure and the instrumented test train. In addition to the test train and loop instrumentation, a delayed neutron detector (DND), an on-line cover-gas syste (OLCS), and an on-line sodium-sampling system (OLSS) were operated to observe fuel failure[4,5].

A total of 24 wire wrap thermocouples were located in the test

Figure 1: P4 Loop Cross Section at Fuel Centerline



section and duplicated P3/P3A location wherever possible to take advantage of the P3/P3A steady state operating experience. The hexcan was instrumented with six thermocouples for a backup Plant Protective System (PPS) function (PF-J) which also provided data during the experiment. Seven bypass and three downcomer TC locations in P3 were retained for P4 in order to aid the overall loop heat-balance calculations.

The inlet flowsensor subassembly had multiple permanent magnets providing four sets of signals which measured the fuel bundle coolant flow rate at the entrance to the fuel bundle. These signals provided both experimental data and the input to the PPS function H low flow monitor. The subassembly also contained test section inlet thermcouples and pressure sensors. The outlet flow sensor subassembly contained an eddy current flowmeter preceded by a flow straightener, four thermocouples, and two pressure sensors. The total flowmeter subassembly, located downstream from the flow mixing region, contained two eddy current flow sensors.

Eleven pin-plenum-pressure transducers were used to detect the location of the fuel pin failure.

The data acquisition system for P4 incorporated three computers for the continuous collection of data, two computers for routing and display, and one stand-alone, high-speed, analog-to-digital computer system operated only during planned reactor transients[6].

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 to 6.3 m/s in the central coolant flow subchannels and resulted in a frictional pressure drop of 0.24 MPa (~ 35 psi) across the full-length test fuel pins. The presence of the three fuel canisters contributed about 7 KPa (1 psi) 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 $\leq$ 839 K (1050°F). Bundle inlet temperature was 695 K (792°F), the inlet temperature for the FTR rated core. Loop cover gas pressure during most of the P4 irradiation was 0.16 MPa (22.8 psia) to avoid cavitation in the OLSS pump. During and immediatedly following the power transient, the cover gas pressure was reduced to 69 kPa (10 psia) to simulate LMFBR pressure levels in the test subassembly.

PRE-TRANSIENT TESTING

Two series of pre-irradiation tests were performed. The first series, B-3, was performed in the FS&R after the loop was filled with sodium. The second series, B1, was performed after the loop was filled with sodium. A series of tests which included benchmark testing, power and flow perturbations and low power transients was performed after the loop was inserted into the ETR.

The P4 loop was filled with sodium on April 1, 1981, followed by sodium purification, sodium level-set, and loop seal-off. The objectives of the B-3 testing were to obtain pump power/flow data for comparison with predictions, to leak check the loop flange penetrations and the seals of the test-train static-pressure-sensor cables. Pressure sensor calibration was performed and the B-3 test series was conducted between April 20 and May 2, 1981. Agreement between the measured hydraulic performance and predicted characteristics was good. The measured ratio of test-section flow to loop flow was within 3% of pretest analysis predictions. Leak tightness of the loop flange penetrations and seal integrity of the pressure-sensor cables were verified by pressurizing the annulus between the primary and secondary vessels and the removable top closure to 1.83 MPa (265 psia).

The P4 loop was transferred to the ETR on May 20, 1981. Lines to the helium cooling system, the OLCS, and the OLSS were then connected.

The objective of the B-1 testing was to functionally test, demonstrate and/or charactereize, prior to ETR startup, all subsystems of the SLSF system not previously verified to the extent possible at zero reactor power. Testing included starting the secondary cooling system, balancing of the helium system loop and bypass leg impedance, repeating the pump performance and loop hydrualic characteristics mapping, and demonstrating the functioning of the plant protective system and experiment safety assurance system. Also included in B-1 was pre-operational testing of the OLCS and the OLSS[4,5]. B-1 testing began on June 24 and was completed August 3, 1981.

All power operation during the first 10-day cycle (August 8-15, 1981) and the two days (August 19-21, 1981) prior to the P4 power transient in the second 10-day cycle was limited to maximum ETR power of 40 MW to prevent premature failure of the fluted fuel canister. A 40 MW loop benchmark test was performed after three hours of steady running to provide a reference set of sensor data for comparison with post-failure data.

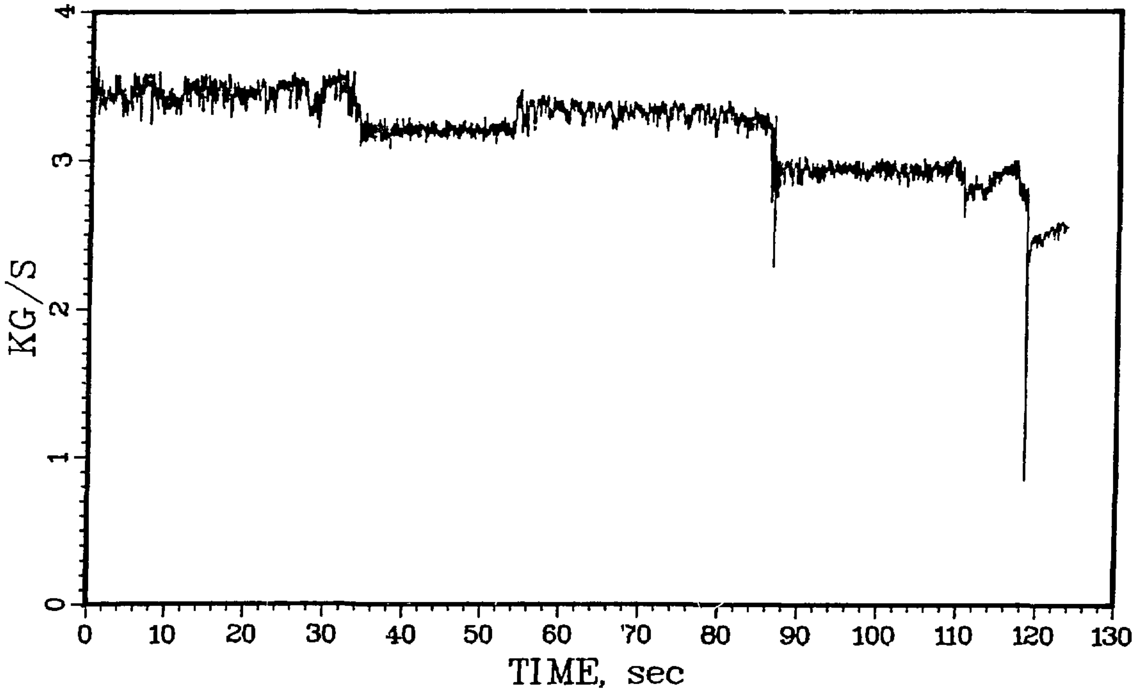
Non-adverse pressure, power, and flow perturbations were imposed on the test section prior to fuel failure. Pressure reductions were performed using the OLCS system to lower the loop plenum pressure and thus the pressure at the fuel canisters. The first and second test occurred on August 10, 1981 and August 14, 1981, respectively. Both of these tests showed no evolution of fission gas, indicating that all fuel canisters were still intact. Non-adverse power-to-flow perturbations were performed by combining 10% power decreases with 10% flow increases to give power-to-flow ratios of 1.0, 0.9, and 0.8. This data provided input to the analysis at the sampling systems. The first and second power-to-flow tests also occurred on August 10, 1981 and on August 14, 1981.

Two low power practice transients were run, the first August 11, 1981, and the second August 19, 1981. Both started from an ETR power of ~ 7 MW and terminated at 28 MW. In both cases the power transient controlled produced an acceptable increase in neutron level. In the first transient, the power level, however, only increased a factor of 3.1 rather than the 3.9 required. This was probably due to an unequilibrium decay heat level, since the starting power of 7 MW was much lower than the 35 MW to 40 MW steady state operation of the previous several days. A second low power transient was performed with a power history closer to that expected for the high power transient. This produced a higher power increase although it was slightly lower than ideal. As a result, the final power selection for initiating the high power transient was based on a maximum test section power of 342 (326+5%) or 40 MW ETR power instead of the original limit of 362 KW of 40 MW.

PLANNED POWER TRANSIENT

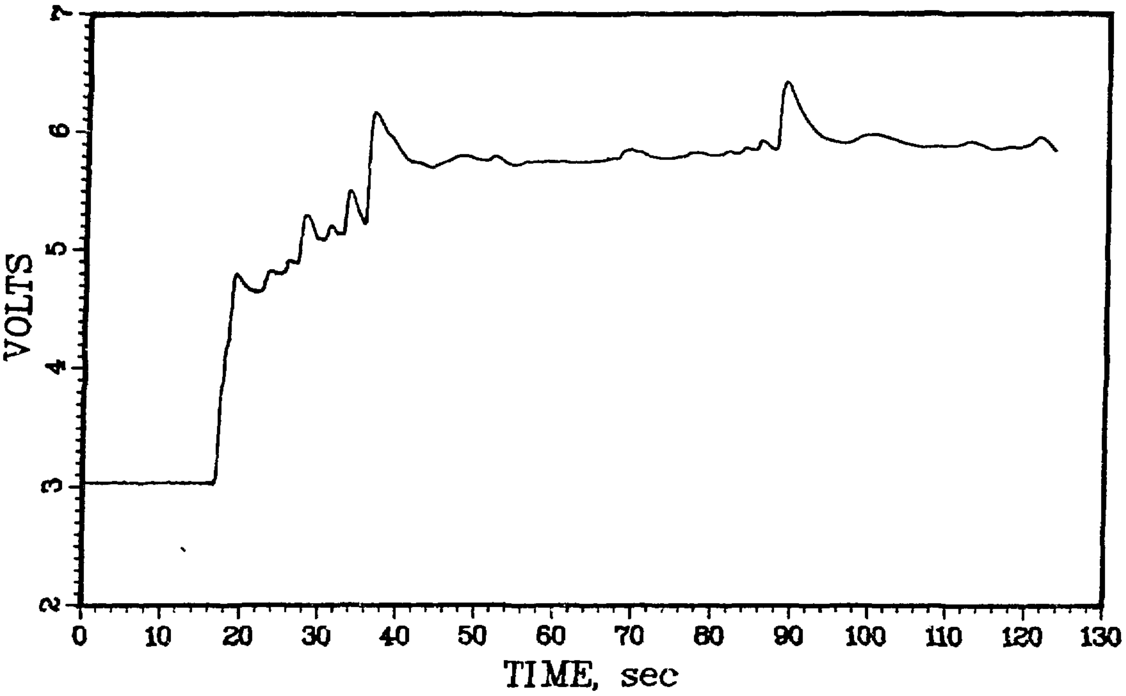
The P4 power transient was initiated at 06:44:21 MDT August 21, 1981, at a bundle power of 338 KW (40 MW ETR power). The ETR neutron level, under the control of the power transient controller, followed the planned rise to 4.375 times the initial neutron level (175 MW) in 27.75 s, held that level for 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 (Fig. 2). It was attributed to initial failure of the fluted fuel canister. [*Fluted canister cladding failure was predicted to* *occur at 16.5 s (100 MW) based on sodium boiling and cladding dryout in a nominal-thickness annular cooling channel between the canister and adjoining fuel pin. For a minimum-thickness annular cooling channel, failure was predicted at 14.5 s].* The delayed neutron level "bumped" upward following initial failure of the fluted canister as the area of fuel exposed to the sodium increased (Fig. 3). 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.

Figure 2: Test Section Inlet Flowmeter Response During Power Transient



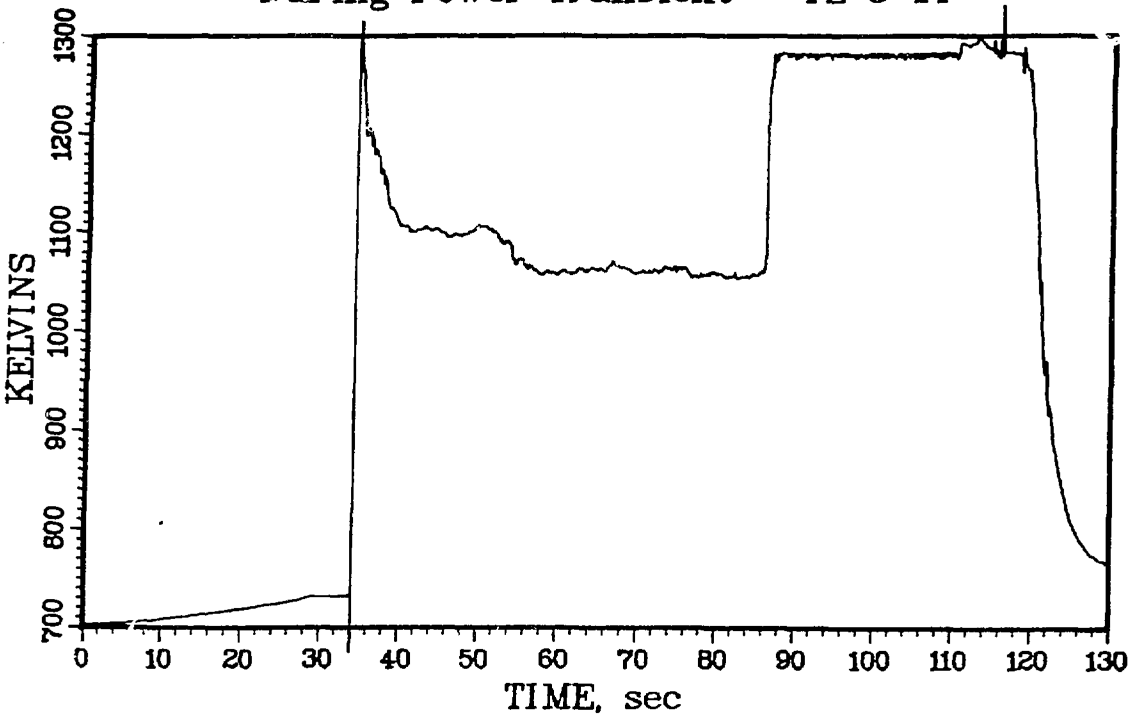
ETR reached a power level of 175 MW at 27.75 s and held at that level until 32.75 s. During this period there were flow and temperature perturbations in the test section. Significant local temperature perturbations occurred downstream of the fuel canisters. *[The 20% and 10% cold-worked cylindrical canisters were predicted to reach failure strains of ~8.8% at 24.9 s (150 MW) and 26 s (160 MW), respectively].* Although failure had not occurred yet, the "ballooning" 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.

Figure 3: Radiation Detector RD5 Response During Power Transient



Molten fuel release occurring at ~34 s (170 MW ETR power) was accompanied by perturbations in inlet and outlet flow, in local coolant temperature levels, in acoustic noice, 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 (Fig. 4) indicated a temperature jump of 600 K. The response of thermocouple TE 3-14, junction located 0.36 m below the fuel midplane, was typical of molten fuel contact and formation of a new junction. Thermocouples TE 6-1 and TE 6-2, on 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 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.

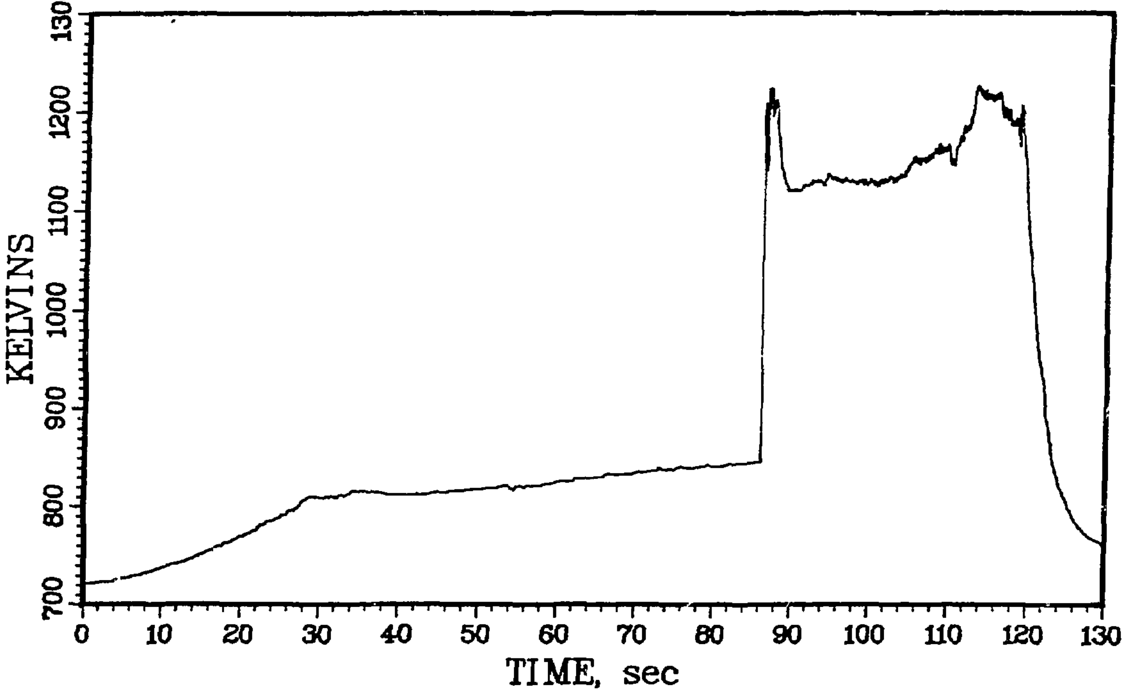
Figure 4: Temperature Near Pin 5 Fuel Canister During Power Transient – TE 3–14



The rate of average power increase in the P4 loop exceeded 2.4 MW/minute during the 27.75 s increase in ETR power from 40 MW to 175 MW. Speed of the circulators supplying helium to the SLSF heat exchanger was increased to their operating limit early in the transient. This operating procedure was predicted to result in an increase in test section inlet temperature, followed by a leveling off and recovery to the design 695 K value. At 34 s, when molten fuel release from the cylindrical fuel canister on pin 5 occurred, the inlet temperature had increased to 700 K. During subsequent power operation, the inlet temperature increased at an average rate of ~ 0.6 K/s. Bundle temperatures stabilized, following the fuel release, and then slowly increased with the increasing inlet temperature. 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 (Fig. 5) 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 twin flow perturbations, 20 ms apart, that reduced inlet flow to ~ 2 Kg/s. Flow recovered to 3 Kg/s. About 0.45 s after the twin flow perturbations, a third perturbation occurred that drove inlet flow down 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.

Figure 5: Temperature Near Pin 8 Fuel Canister During Power Transient – TE 3–15



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%). Test section inlet temperature reached 750 K at scram. Following ETR scram, the pump was transferred back to its regular power supply. Test section flow at the pump benchmark voltage also returned to its previous 2.95 Kg/s (86% of nominal) level.

BLOCKAGE RECONFIGURATION

The increase in ETR power to 40 MW during the September 3-5, 1981 operation was performed in a routine manner; no flow or temperature anomalies were observed. Subsequent preparations for a return to full power operation were completed 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 (210s prior to a flow reduction) 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 triming to reach 100 MW was in progress but its relative change was less than the DN change. Half a minute later (180s prior to the flow reduction), 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. There were temperature perturbations of up to 15K indicated by several thermocouples about 21s before the blockage reconfiguration. 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. About 5s before the reconfiguration, inlet flow dropped from 2.9 Kg/s to 2.85 Kg/s, then began a gradual decrease. The rate of inlet flow reduction increased 0.5s before the reconfiguration. This was followed by temperature increases at thermocouples on pin 8.

Basic characteristics of 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 ~0.35s prior to another perturbation in inlet flow, and reactor scram. The ~0.5s 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.

Thermocouples TE 4-8 (pin 35) and TE 4-6 (pin 32) indicated temperature jumps of 510 K and 320 K, respectively. This response for these thermocouples, which had junctions below the lower end of the fuel column, indicated molten fuel contacted their sheaths and formed new junctions. It also indicated that molten fuel moved toward hex duct faces 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.

POST-TRANSIENT TESTING

Based on the results of the IPL pressurization test on August 25, 1981 four pins were determined to be open to the sodium; pins 4, 6, 7, and 14. Four additional pins were classifed leakers; pins 3, 8, 11, and 13. Pin 19 appeared to be intact. Five other fuel pins also had pressure transducers but those either failed prior to the transient or were installed spares. Second pressurization test confirmed that pin 19 was open to sodium after the blockage reconfiguration.

GAMMA SCAN

As the P4 loop was pulled out of the reactor, a gamma scan was performed over the entire length using a collimted Ge-Li detector. Spectra were recorded at a spacing of 1-inch in the region of the blockage in the fuel zone, and at 10-inch intervals in most other regions. Spectra were recorded on magnetic tape for detailed analyis later. Peaks due to ^{95}Zr, ^{95}Nb, ^{103}Ru were seen over the whole length. These are non-volatile fission products, and presumably indicated dispersal of fuel particles. There was no indication of a large dispersal of fuel from the test section. Peaks from ^{140}Ba/La were also seen at all levels. In particular, sections of tubing leading to the OLSS and OLCS showed particularly rich spectra, including ^{140}Ba/La. This appears to indicate that fission products have been transferred to these external loops.

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. Ths 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 later 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 show a slower response time 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 a 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.

ACKNOWLEDGEMENTS

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REFERENCES

1. U.S. Department of Energy, Division of Reactor research and Technology, LMFBR Safety Program Plan, (December 1979).
2. D. K. WARINEER and D. H. CHO, "Status and Needs of Local-Faults Accommondation in LMFBR." in *Proc. ANS/ENS International Meeting on Fast Reactor Safety Technology*, Seattle, WA, (August 19-23, 1979).
3. W. A. RAGLAND, D. H. THOMPSON, and R. E. WILSON, "The Sodium Loop Safety Facility In-Pile Loop on P4 Test Train," *Proc. Fast, Thermal, and Fusion Reactor Experiments Conference*, Am. Nucl. Soc., Salt Lake City (1982), to be published.
4. T. H. BRAIN et. at., "Operation of Cover Gas System During SLSF Test," *Proc. Fast, Thermal, and Fusion Reactor Experiments Conference*, Am. Nucl. Soc., Salt Lake City (1982), to be published.
5. T. H. BRAID et. al., "DN Detection During SLSF Test," *Proc. Fast, Thermal, and Fusion reactor Experiments Conference*, Am. Nucl. Soc., Salt Lake City (1982), to be published.
6. R. D. BALDWIN et. al., "Data Acquisition for the Sodium Loop Safety Facility Experiment L4, " *Proc. Fast, Thermal, and Fusion reactor Experiments Conference*, Am. Nucl. Soc., Salt Lake City (1982), to be published.