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# Final Report on the SLSF In-Reactor Local Fault Safety Experiment P4

by D. H. Thompson, J. W. Holland,  
T. H. Braid, and W. A. Ragland

September 1985

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ANL/RAS 85-9

**MASTER**

FINAL REPORT ON THE SLSF IN-REACTOR  
LOCAL FAULT SAFETY EXPERIMENT P4

by

D. H. Thompson, J. W. Holland, T. H. Braid and W. A. Ragland

with contributions by

T. T. Anderson, R. D. Baldwin, T. H. Bauer, D. J. Dever, T. H. Hughes,  
J. M. Kramer, J. J. Sienicki, and D. K. Warinner

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ACRONYMS

|        |                                            |
|--------|--------------------------------------------|
| AGHCF  | Alpha Gamma Hot Cell Facility              |
| ANL    | Argonne National Laboratory                |
| DAS    | Data Acquisition System                    |
| DN     | Delayed Neutron                            |
| DND    | Delayed Neutron Detector                   |
| DNM    | Delayed Neutron Monitor                    |
| EAS    | Experiment Assurance System                |
| EBR-II | Experiment Breeder Reactor-II              |
| ETR    | Engineering Test Reactor                   |
| FS&R   | Filling Storage & Remelt Facility          |
| FTR    | Fast Test Reactor                          |
| HEDL   | Hanford Engineering Development Laboratory |
| HFEF   | Hot Fuel Examination Facility              |
| INEL   | Idaho National Engineering Laboratory      |
| LMFBR  | Liquid Metal Fast Breeder Reactor          |
| LOF    | Loss of Flow                               |
| MFCI   | Molten Fuel Coolant Interaction            |
| NRAD   | Neutron Radiography Facility (HFEF)        |
| OLCS   | On-Line Cover Gas Sampling System          |
| OLSS   | On-Line Sodium Sampling System             |
| PF     | Protection Function                        |
| PPS    | Plant Protection System                    |
| PTC    | Power Transient Controller                 |
| PTE    | Posttest Examination                       |
| RTC    | Removable Top Closure                      |
| SLSF   | Sodium Loop Safety Facility                |

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ABSTRACT

The Sodium Loop Safety Facility (SLSF), a major facility in the U.S. fast-reactor safety program, has been used to simulate a variety of sodium-cooled fast reactor accidents. SLSF experiment P4 was conducted to investigate the behavior of a "worse-than-worst-case" local fault configuration. Objectives of this experiment were to eject molten fuel into a 37-pin bundle of full-length Fast-Test-Reactor-type fuel pins from heat-generating fuel canisters, to characterize the severity of any molten fuel-coolant interaction, and to demonstrate that any resulting blockage could either be tolerated during continued power operation or detected by global monitors to prevent fuel failure propagation. The design goal for molten fuel release was 10 to 30 g. Expulsion of molten fuel from fuel canisters caused failure of adjacent pins and a partial flow-channel blockage in the fuel bundle during full-power operation. Molten fuel and fuel debris also lodged against the inner surface of the test subassembly hex-can wall. The total fuel disruption of 310 g evaluated from posttest examination data was in excellent agreement with results from the SLSF delayed neutron detection system, but exceeded the target molten fuel release by an order of magnitude. This report contains a summary description of the SLSF in-reactor loop and support systems and the experiment operations. Results of the detailed macro- and microexamination of disrupted fuel and metal and results from the analysis of the on-line experimental data are described, as are the interpretations and conclusions drawn from the posttest evaluations.

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## SUMMARY AND CONCLUSIONS

Objectives

Sodium Loop Safety Facility (SLSF) experiment P4 was conducted in the Engineering Test Reactor (ETR) to investigate the behavior of a "worse-than-worst-case" local fault configuration. Objectives were to eject molten fuel into the 37-pin bundle of full-length, Fast-Test-Reactor (FTR)-type fuel pins from one or more of the three heat-generating fuel canisters, to characterize the severity of any molten fuel-coolant interaction, and to demonstrate that any resulting blockage could either be tolerated during continued power operation or detected by global monitors to prevent fuel failure propagation. If the consequences of the P4 "worse-than-worst-case" local fault configuration could be shown to be benign, more aggressive operating strategies for liquid metal fast breeder reactors (LMFBRs) could be considered, requirements for LMFBR core surveillance and sensitivity of failed fuel detection systems revised, and the scope of further failed-fuel testing reduced.

Experiment P4 was intended to bound the consequences of credible subassembly faults by releasing between 10 and 30 grams of molten fuel into the fuel bundle at full power. This quantity of molten fuel was an order of magnitude larger than the minimum needed as a short sleeve to cause cladding damage and fuel melting or as fuel bridge between pins to cause sodium temperature to exceed 1145 K (1600°F). The potential for further cladding and fuel pin damage was to be probed by continued steady power operation and, later, short-term variations in both power and flow yielding more severe operating conditions. Bounding operating conditions were further emphasized by selection of an undercooled operating condition (power/flow = 1.24 times that of a central FTR core subassembly), high inlet coolant temperature (695 K), and use of half-diameter wire wrap spacers on the outer row of fuel pins to reduce "cold" peripheral flow. The P4 design operating condition was a nominal bundle power of 1272 kW (1240 kW from 37 pins, plus 32 kW from the fuel canisters) at an ETR power level of 156 MW and a fuel-bundle coolant flow rate of 3.38 kg/s.

Description of Experiment System

The P4 fuel bundle consisted of 34 5.84-mm-dia., 2.37-m long,  $\text{UO}_2$ - $\text{PuO}_2$ -fueled FTR-type fuel pins, each containing 172 g of fuel, and three  $\text{UO}_2$  fueled pins, of similar design, with 10-cm-long fuel canisters located at their fuel midplanes. The pins were spaced on a 7.26-mm triangular pitch by 1.42-mm-dia. spacer wires wrapped the length of the central 19 pins on a 30.5-cm pitch. The spacing between the outer-row pins and the hex-can was 0.71 mm to improve the uniformity of the coolant distribution and the radial temperature field within the bundle. The  $\text{PuO}_2$  content of the fuel was 25 wt% with a  $^{239}\text{Pu}$  enrichment of 88%. A row-by-row grading of the fuel  $^{235}\text{U}$  enrichment, from 93% in the center pin to 58% in the outer-row pins, in conjunction with the cadmium neutron filter on the loop secondary vessel, produced a flat radial power profile across the bundle. The three fuel-canister pins contained  $\text{UO}_2$  enriched to 93%. All fuel was of stoichiometric composition to enhance sodium urano-plutonate formation in exposed fuel and failed pins.

Two fuel canisters were of cylindrical geometry and one had a fluted cross section. The fuel inventory of the cylindrical and fluted canisters was 56 g and 69 g, respectively. The outside diameter (OD) of the circular canisters was selected to produce line contact with the adjacent fuel pins in an undistorted bundle geometry; one cylindrical canister had 20% cold worked (CW) cladding, the other 10% CW cladding. The fluted canister occupied the nominal area of six coolant flow subchannels. It was spaced away from each of the six neighboring fuel pins by a 0.16-mm-dia. wire positioned vertically between each cusp region of the fluted canister and the adjacent pin. Two 0.25-mm-thick, 12.7-mm-wide stainless steel bands were fastened around the six pins that contacted the fluted canister to prevent it from "ballooning" back to a circular cross section during experiment operation.

The P4 fuel bundle was enclosed by a 3.05-mm-thick, Type 316 stainless steel hex-can measuring 45.0 mm across flats, on the inside. Between the hex-can and a 1.57-mm-thick, 66.7-mm-OD cylindrical Inconel 625 outer tube was an insulating region containing six zirconia insulator segments positioned by chevron-shaped tungsten spacers in a high-purity argon gas environment. The fuel bundle was located near the lower end of a test train that was inserted into the SLSF in-reactor loop. The P4 test train contained 100 instruments (7 flowmeters, 60 thermocouples, 29 pressure sensors, 2 sodium level detectors, and 2 acoustic sensors) grouped heavily in and around the fuel bundle.

The SLSF in-reactor loop was a test vehicle of concentric, reentrant design that consisted 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 loop and test train, the facility included a handling system, filling, storage and remelt system, plant protection system, loop and reactor control system, helium secondary coolant system, data acquisition system, and three failed-fuel detection systems. The loop was ~8.2-m long and weighed about 3200 kg, including the test train. The heat exchanger and pump were located above the in-reactor section. Both the heat exchanger and pump had a central bore through which the test train was inserted and positioned in the loop. The loop contained 100 liters of sodium and had a 30-liter gas plenum above the helium-cooled heat exchanger. The sodium inventory was initially purified to an oxide content of less than two parts per million.

In addition to the test train and loop instrumentation, a delayed neutron detector (DND), an on-line cover-gas sampling system (OLCS), and an on-line sodium sampling system (OLSS) were operated to observed fuel failure. The DND observed the DN activity of the flowing sodium as it passed through the 13-liter volume of the loop upper sodium plenum. The transient delay from the fuel bundle to the DND was ~2 s, hence the DND observed much of the short-lived DN activity. The DND consisted of 26  $\text{BF}_3$  detectors and 3 fission detectors in a Benelux moderator assembly. It was contained in a sealed box mounted under the top dome of the ETR, against the SLSF loop plenum region.

The OLSS removed a stream of sodium from the loop plenum region and transported it to a 0.5-liter sample volume where the delayed neutrons were counted by the delayed neutron monitor (DNM). The DNM contained 27  $\text{BF}_3$  detectors and 3 fission detectors in a Benelux moderator. The DNM and most of the OLSS were located under the shielding atop the ETR. The flow rate could be varied in the OLSS to adjust the transit delay from 12 to 40 s.



The OLCS continuously drew a gas stream from the loop gas plenum to a sampling chamber where the sample was viewed by two high-resolution Ge-Li gamma-ray spectrometers. Gamma rays of energies ranging from  $<80$  keV to 3 MeV could be detected. The delay time could be varied to allow the only serious background activity ( $^{23}\text{Ne}$ ,  $t_{1/2} = 39$  s) to decay, and the size of the sample and the detection sensitivity could be varied over a range of 9 decades to allow for a very wide range of failures (from pinholes to fuel melting).

### Experiment Operation

P4 irradiation testing was planned to cover five 10-day irradiation cycles and reach a fuel burnup equivalent to 30 full-power days of ETR operation at 156 MW. Experiment P4 irradiation began on August 8, 1981, and continued through the first 10-day cycle at a maximum ETR power of 40 MW, to avoid failing the fuel canisters while building up fission products in the initially fresh fuel for subsequent failure detection. After a 24-hour irradiation at 40-MW ETR power in cycle two to restore equilibrium levels of short-lived fission product activity, a planned power transient was initiated to reach full reactor power. The power transient to full power consisted of an ETR power increase from 40 to 175 MW, on an 18.8-s period, followed by a 5-s hold at 175 MW, a 5-s ramp down to 156 MW and continued steady operation. ETR power increased under the control of a preprogrammed power transient controller; helium circulator speed was controlled manually.

During the increase in reactor power, the cladding of the fluted fuel canister melted in the cusp regions next to the adjacent fuel pins between 15 and 16 s, as predicted. This produced disturbances in the temperature, pressure, flow, and acoustic noise. The DN level "bumped" upward as the area of fuel exposed to the sodium increased. Distortion in the bundle geometry caused by strain in the cylindrical canisters produced local temperature plateaus of 50 to 100 K that persisted from  $\sim 30$  s until the row-of-six, 20% CW cylindrical canister burst and expelled molten fuel toward the outer rows of pins and the hex-can at 34 s. This fuel was directed upward by the spiral wrap of the fuel pin spacer wires. The molten fuel release was accompanied by perturbations in inlet and outlet flow, local coolant temperature levels, pressure, acoustic noise, and by a sharp increase in DN level. Inlet flow decelerated to a minimum of 2.4 kg/s (70%) and returned to 3.2 kg/s, 93% of the initial steady-state flow rate. (Pretest calculations predicted a 70% minimum inlet flow reached in 2.5 ms following rupture of one cylindrical canister, based on a slug-response coolant behavior driven by the 14.4-MPa peak pressure calculated by the PLUTO code.) Several thermocouples, both in the bundle and adjacent to the hex-can wall, were hit by the molten fuel and formed new junctions. A five-fold increase in the DND signal occurred at 36 s.

Following the molten fuel release and ETR power reaching the 156-MW steady-state level, loop conditions became quasi steady state. At 80 s, a thermocouple on a fuel pin adjacent to the fluted canister failed. This was followed by a rupture of the fluted canister at 86 s and ejection of molten fuel toward the center of the bundle. The intact, 10% CW cylindrical canister was hit by some of this molten fuel and suffered a cladding meltthrough. Again, simultaneous perturbations in flow, temperature, pressure, and acoustic noise were observed. The increase in partial flow blockage reduced test section flow to 86%. Sodium boiling was indicated by two thermocouples that had been hit by molten fuel at 34 s and formed new junctions in or adjacent to the

partial flow blockage. A spike in the DND signal at 88 s was characteristic of a molten fuel release and qualitatively similar to the earlier DN response.

After the second molten fuel release, loop conditions again became quasi steady state. The test section inlet temperature had been slowly rising during the power transient and continued to increase. Protective system set-points had been selected to accommodate a 45 K overshoot in bundle inlet temperature as a result of the planned helium system operation. However, an error in the operation of the helium system yielded a maximum helium flow rate to the heat exchanger of only 1600 to 1900 kg/h. At 86 s, the fuel bundle inlet temperature had risen to 739 K, which was approximately the predicted peak bundle inlet temperature, but it then continued to rise at a rate of 0.6 K/s. The increasing inlet and system temperature reduced the performance of the pump, which was operating at a fixed voltage, and also affected the non-temperature-compensated signal from the inlet flowmeter to the plant protection system.

Slow temperature increases of 30 to 60 K were indicated by several thermocouples in the center of the bundle, beginning at 104 s. Flow remained steady until 110 s, when a large gas release occurred. This gas release resulted from a failure in the hex-can at a location 130 mm above the fuel midplane, where molten fuel released at 34 s had adhered. Fuel bundle temperatures and flow were perturbed for three to four seconds during the gas release. Then temperatures began to return toward pre-gas-release levels as the fuel bundle flow recovered to 2.95 kg/s (86%).

At 117 s, the fuel-bundle flow and inlet pressure began to drift lower. ETR scrambled at 118.4 s on a signal to the plant protection system that the time-averaged low-flow setpoint of 80.3% had been reached.

Because the fuel-bundle coolant flow rate was below the safety analysis and administrative operational limit of 90% for steady-state power operation, a restart for full-power operation was delayed until an expanded safety envelope was prepared and approved. During the extended hot standby, the fuel-bundle inlet temperature was maintained at ~680 K, coupled with intermediate operation to 40 MW ETR power, to enhance sodium ingress and egress at fuel pin failure sites and the formation of sodium urano-plutonate within the partial flow blockage. (A return to power after a significant fuel failure and an extended hot standby is certainly atypical of LMFBR operation. The return to power was consistent with the bounding nature of experiment P4 and provided the opportunity, in a heavily instrumented fuel bundle, to obtain information on effects of sodium urano-plutonate dissociation.)

ETR power operation resumed on October 1 with a P4 low-flow setpoint of 68%. At the reference pump power, the fuel-bundle flow rate of 2.95 kg/s (86%) was unchanged from that during the later stages of the power transient. ETR power was increased in 10 and 20 MW increments in a step-and-hold manner toward 156 MW. About seven minutes after reaching ~92 MW, and coincident with the trimming of reactor power at 100 MW, the DND signals began to increase. Then the DNM signals began to increase. The increase in signal was gradual at first, building in an exponential manner as time passed. Temperature perturbations of up to 15 K were indicated by several thermocouples 21 s before an abrupt blockage reconfiguration occurred and coincident with the beginning of a sharp increase in DND signal.

Many characteristics of the blockage reconfiguration appeared similar to those for molten fuel release during the power transient. There were perturbations in temperature, pressure, flow, acoustic noise, and DN signal. Two thermocouples formed new junctions, indicated by temperature spikes of 320 and 510 K. Inlet flow dropped to 1.75 kg/s followed by a partial recovery to 2.25 kg/s, then dropped to -0.35 kg/s and recovered again to 2.25 kg/s. Fuel-bundle flow steadied at 60%; this was accompanied by an ETR scram on low fuel-bundle inlet flow. The DN signal began to increase 210 s prior to scram. A gradual increase in DND signal was followed by a step increase to a plateau 5.5 times the earlier steady-state level. This was followed 20 s later by a spike to 11 times the steady-state level. Fuel material relocated during the reconfiguration was directed outward and downward by the spiral wrap of the fuel pin spacer wires.

An evaluation of all instrument responses indicated that the blockage reconfiguration originated near the center of the bundle and moved fuel material outward into previously unblocked flow channels. These flow channels were adjacent to hex-can walls across the bundle from the fluted canister. Because the coolant flow rate through the fuel bundle was below the low-flow setpoint stipulated in the extended safety envelope, low-power operation to study the sensitivity of the DN signal to sodium temperature concluded the experiment.

#### Posttest Examination and Analysis

The SLSF loop and P4 test assembly were transported to the Hot Fuel Examination Facility (HFEF) for initial disassembly following the in-reactor experimentation. Goals of the HFEF-phase of the posttest examination (PTE) were to retrieve the fuel bundle by dismantling the loop and withdrawing the test assembly, to assess the macrocondition of the fuel bundle (in particular the blockage region at the midplane) by nondestructive examination techniques, and to prepare the blockage region for shipment to the Alpha-Gamma Hot Cell Facility (AGHCF) at Argonne National Laboratory for the detailed PTE.

Full-length neutron radiographs of the fuel bundle provided evidence that a significant portion of the bundle flow area became blocked by fuel expelled from the fuel canisters and relocated from adjacent pins during the power transient. The 28-cm-long blockage region extended from 13 cm below to 15 cm above the fuel midplane; disrupted fuel contacted the inner surface of the hex-can at the fuel midplane. A 37-cm-long fuel-bundle segment, containing the entire blockage region, was cut from the P4 test section and transferred to the AGHCF for detailed PTE.

Two major uncertainties persisted after evaluation of the on-line instrument data and nondestructive examination results: namely, blockage morphology and composition, and the extent of failure in fuel pins adjacent to the UO<sub>2</sub>-filled fuel canisters.

Information was collected during the macroexamination phase of the PTE to permit assessment of the following:

- cladding disruption and relocation,
- fuel disruption and relocation in fuel canisters and fuel pins, and
- nature and extent of blockage formation.

Epoxy was used as a stabilizing agent to preserve the posttest condition and configuration of fuel-bundle components and disrupted materials during macro-examination of the blockage region. A high-speed SiC cutoff machine was used to produce a series of closely spaced transverse cuts that divided the 37-cm-long segment into 27 thin cross sections. The epoxy-filled macrosections were metallographically ground to produce smooth, flat surfaces that were photographed at 2X magnification to permit critical, visual examination.

Careful examination of the 2X macrophotographs provided information concerning the blockage morphology and extent of failure in fuel pins adjacent to the fuel canisters. Failed fuel canisters, numerous failed and disrupted mixed-oxide fuel pins, flow-channel blockages, a large centrally located region voided of fuel and metal, and an isolated hex-can failure were observed.

The three canisters were breached over their entire lengths with the 10% CW, row-of-12 cylindrical canister sustaining the least disruption at all elevations. Melting at the cusps of the fluted canister typified the nature of its initial failure. Most of the  $\text{UO}_2$  fuel filling the fluted canister near the lower end cap remained but nearly all of the fuel in the remainder of the fluted canister became disrupted and relocated so that only two tips of the canister remained at higher elevations.

The 20% CW, row-of-six cylindrical canister ejected molten fuel radially outward toward outer-row pins and two hex-can flats. Molten fuel and fuel debris were directed upward by the spiral wrap of the fuel-pin spacer wires. Molten fuel caused cladding failure and fuel disruption of adjacent pins before lodging against the hex-can and adjacent fuel pins within a short axial region centered 130 mm above the fuel midplane. Disrupted fuel and metal debris accumulated against the hex-can wall from 24 mm below to 75 mm above the fuel midplane.

Rupture of the fluted canister expelled molten fuel into the center of the bundle, causing fuel-pin failures and locally melting cladding on the intact row-of-12 cylindrical canister. Maximum damage to the canister occurred near the fuel midplane where slightly more than half of the  $\text{UO}_2$  became disrupted.

The severely damaged center of the bundle, bounded by the failed fuel canisters, contained little, if any, pin structure following rupture of the fluted canister. A significant amount of cast metal, and a lesser quantity of failed fuel, formed a cap over the voided region below. The damaged region between the fluted canister, the row-of-six cylindrical canister, and the adjacent hex-can wall was caused by the relocation of molten  $\text{UO}_2$  from the row-of-six cylindrical canister. Movement of the molten  $\text{UO}_2$ , mixed with disrupted mixed-oxide fuel, continued upward causing little cladding damage for 40 mm and then lodged against the hex-can wall, leading to local failure of the hex-can 130 mm above the fuel midplane.

PTE evidence indicated extensive formation of sodium urano-plutonate in failed pins and disrupted fuel in flow channels during the lengthy ETR shut-down following the power transient. Dissociation of the sodium urano-plutonate led to a reconfiguration of the disrupted materials during the subsequent approach-to-full-power operation. Molten fuel and metal relocated downward from the center of the bundle just below the fuel midplane toward the hex-can flats opposite the fluted canister.

The quantitative posttest axial distribution of metal and fuel within the hex-can have been evaluated from the 2X macrophotographs. The lowest elevation displaying disruption was -64 mm, where cladding failed and ~10% of the flow channel was blocked with disrupted fuel and metal from the reconfiguration event. Damage to the fuel bundle was most extensive throughout the canister region (-50 mm to +50 mm). Nearly one-third of the mixed-oxide and slightly more than 80% of the  $\text{UO}_2$  fuel were disrupted in the upper two thirds of the canister region. Likewise, nearly one-third of the initial metal structure became disrupted in this same axial region. The extent of areal flow-channel blockage was relatively constant throughout the canister region. The most extensive blockage occurred at the top of the canisters. An axial region containing little or no disruption extended from +60 mm to +100 mm, above which the effects of the earlier fuel motion and the hex-can failure are observed. The amount of disrupted fuel from this PTE evaluation was estimated at  $270 \pm 27$  g in the canister region and  $310 \pm 31$  g total.

A dense band of once-molten fuel separated the disrupted fuel and metal at the periphery of the voided region from the damaged but intact mixed-oxide fuel. The dense band contained a mixture of  $\text{UO}_2$  and mixed-oxide fuel and micron-size stainless-steel particles. The mixture contained both  $\text{UO}_2$  canister fuel and mixed-oxide fuel from failed pins, stainless steel particles of varying sizes and geometries and a significant amount of porosity. The relative proportion of fuel, steel, and porosity was measured at some 17 different locations that contained disrupted material from the different fuel-motion events. Seven pins survived the in-reactor testing without failure at any elevation; three others experienced minor cladding-melting near the bundle midplane. Fuel was disrupted in 23 pins.

Sodium urano-plutonate was found at the periphery of mixed-oxide fuel pellets in failed pins. The sodium-fuel phase was observed only in failed, mixed-oxide pins and not in any disrupted fuel structures. Conditions during the extended hot standby were suitable to promote the formation of sodium urano-plutonate in both failed pins and disrupted fuel within the bundle flow channels; however, the sodium urano-plutonate in the flow-channel material must have dissociated during the reconfiguration event because of higher temperatures there.

Posttest analyses were performed to identify cladding strain and molten fuel inventory in the cylindrical canisters as a function of time during the power transient, to identify conditions leading to the local failure of the hex-can and gas release into the bundle, to estimate the effects of entrained gas and perturbations in sodium temperature on pump performance, to study the influence of sodium urano-plutonate on temperature fields in the failed fuel pins and the partial flow blockage, to evaluate the DN data, and to extrapolate results from experiment P4 to reactor conditions.

The posttest analysis of the cylindrical canisters was performed using the FPIN2 code and the measured experiment conditions. FPIN2 results indicated that, at 34-s failure, the 20% CW cylindrical canister contained 40 grams of molten fuel at an average temperature of 3925 K, had reached a cladding plastic strain of ~12%, and had a cavity pressure of 24 MPa. For the P4 operating conditions, the FPIN2 calculations predicted the cylindrical canister cladding to fail by local necking at diametral failure strains of 12% and 24% for the 20% and 10% CW canisters, respectively. The agreement between calculated and experimentally determined failure conditions for the 20% CW canister was very good. The peak cladding strain predicted for the 10% CW canister was ~13%, hence it would not be expected to fail mechanically. This agrees with experiment observations that the 10% CW canister failed by melt-through in contact with fuel ejected from the fluted canister.

Fuel temperatures within the fluted canister during the power transient were estimated using the THTB code. At 86 s, most of the fuel within the fluted canister was molten and fuel in the adjacent pins was melted from the contact point in each cusp out to the centerline of the fuel columns.

To identify the conditions leading to local failure of the hex-can, temperature and velocity fields in the wake of the partial flow blockages were estimated using the SABRE-1 code. This information was input into a THTB model of molten fuel solidifying on the hex-can at 34 s and the hex-can wall then experiencing a further increase in local coolant temperature at 86 s, prior to failure at 110 s into the power transient. The power density of the previously molten, 93%-enriched fuel adhering to the hex-can wall was estimated to be 480 W/gm, or twice that in the fuel pins, owing to the fuel relocation into a higher-worth region. At 86 s, the temperature of sodium flowing past the fuel on the hex-can wall in the wake of the blockage was estimated to be 1235 K, or 245 K hotter than the 990 K free-stream sodium temperature. Results of the calculations indicate that local failure of the hex-can wall was not due to impingement by a jet of molten fuel, but was a result solely of the nonprototypicalities of the P4 experiment. Combined effects of both the uncooled outer surface of the hex-can and the radial-position dependency of fuel power density in the graded-enrichment fuel bundle were required to elevate the local hex-can wall temperature to melting. In the absence of these nonprototypicalities, the margin to hex-can melting was conservatively estimated to be 650 K.

Local hex-can failure release 1280 cm<sup>3</sup> (STP) of argon gas, or ~5000 cm<sup>3</sup> at loop sodium plenum conditions, into the flowing sodium. Although the characteristics of this gas release fitted worst-case conditions for cladding overheating due to gas jet impingement and for coolant temperature increase due to two-phase flow effects, no cladding failure attributed to the gas release was identified. Excellent agreement was found between the peak coolant temperature perturbations measured in P4 and the ~100 K rise projected for worst-case (large breach, low  $\Delta P$ ) two-phase flow effects.

Pump performance was sensitive to changes in sodium temperature and the presence of any entrained gas within the pump duct. Both pump model and experiment data indicated a sodium flow decrease of 5.5% per 100 K increase in sodium temperature at constant power/voltage conditions. Entrained gas produced a much greater reduction in pumping efficiency. Out-of-pile tests had shown incomplete gas deentrainment at full-flow loop operating conditions.

Following the local hex-can failure, the void fraction in the sodium entering the pump was calculated to be ~7.5%, which corresponds to an estimated 16.5% reduction in pumping efficiency. The measured time from hex-can failure to reactor scram was in excellent agreement with the calculated time for entrained gas to reach the pump. This indicates that the low-flow signal which scrambled the reactor during power transient operation resulted from entrained gas entering the pump.

The depth from the blockage surface to the isotherm for dissociation of sodium urano-plutonate was estimated at various power levels. The partial flow-channel blockage was considered to be a relatively dense fuel cooled only by sodium flow past the surface. Results of these calculations indicate that, for sodium urano-plutonate dissociation temperatures of ~1500 K, the dissociation isotherm would have moved to within one to two mm of the blockage surface when ETR power reached 100 MW. This, and the lack of sodium urano-plutonate in the disrupted materials, confirm the dissociation of sodium urano-plutonate as a factor in the reconfiguration event.

Two DN models simulating the operation of the loop and failed-fuel detection systems were used to evaluate the P4 DN data. The first model had been used to interpret the DN signals from experiment W2. Its quantitative agreement with the amount of disrupted fuel identified during the W2 PTE was within a factor of two. A second model was developed later that calculated the DN inventory directly in the simulation program. A recalculation of W2 with this second DN model, using more accurate estimates of the fission yields, produced a near-perfect agreement between the calculated and measured W2 fuel release.

During initial low-power P4 operation, the failed-fuel detection systems responded to the signal from the U-Ni fission product source. The ratio of DN count rate to reactor power remained unchanged, indicating no premature failure of the fuel canisters. Failure of the fluted canister early in the power transient produced six DN peaks corresponding to sudden fuel exposure followed by continuous exposed area. Total fuel exposure was estimated to be 17 to 20 g. Using the first DN model, spikes in the DND signals at 36 and 88 s corresponded to molten fuel releases of about 40 g each. Molten fuel releases of ~60 g (approximately the total fuel inventory of a single cylindrical canister) and 50 g, respectively, were computed using the second DN model.

At ~100 MW ETR power during the return-to-power operation, the DND signal exhibited a gradual increase, then a jump to a plateau 5.5 times the earlier steady-state level. This corresponded to an estimated fuel mass exposure of about 140 g. A subsequent spike in the DND signal corresponded to an additional 100 g fuel exposure, based on evaluations using the second DN model. Total fuel mass exposure during the blockage reconfiguration was estimated at up to ~240 g, or greater than twice that exposed during the power transient.

The evaluation of a large-plant DN response to the P4 power transient events was performed by repeating the P4 DN analysis with the large transport and recirculation times and the reduced sample concentration typical of a large plant. These calculations demonstrated that though the longer transport times reduce the contribution of the short-lived isotopes, which reduces the sharpness of the leading edge of the signal, the effect is not so great as to



reduce the effectiveness of the detection system. In general, short delay times are better, and allow earlier detection and action, but they are not critical to the operation. Had the major events in P4 occurred in a large power plant, they would have been detected unambiguously by a whole-core DN detection system.

### Conclusions

1. Test objectives for molten fuel release were exceeded and a partial flow blockage resulted. There was no evidence of fuel sweepout, which agrees with results of the 37-pin CAMEL II C5 test. Lack of sweepout in the C5 test was attributed to significant two-dimensional effects that allowed the sodium to bypass the expelled fuel without significant intermixing.
2. The measured fuel-bundle inlet coolant response to molten fuel release was in excellent agreement with pretest predictions based on slug-response coolant behavior driven by the 14.4-MPa peak pressure calculated by PLUTO. This result confirms an earlier conclusion that fuel-coolant interactions resulting in greater than 20-MPa pressurization with measurable impulses simply have not occurred.
3. The spiral wrap of the spacer wires on the fuel pins influenced the transverse and axial movement of molten fuel and fuel debris in the fuel bundle.
4. Local failure of the hex-can wall resulted from the extremes of a 480-W/g fuel power density and an uncooled hex-can outer wall, atypicalities unrelated to LMFBFR design or operation.
5. Characteristics of the P4 gas release through the failed hex-can fitted the worst-case conditions for cladding overheating due to gas jet impingement and for coolant temperature increase due to two-phase flow effects. No cladding failure attributed to the gas release was identified. Excellent agreement was found between the peak coolant temperature perturbations measured in P4 and the ~100 K rise projected for worst-case (large breach, low  $\Delta P$ ) two-phase flow effects.
6. ETR scram during power transient operation occurred, in part, due to the inadvertent flow reduction in the helium secondary cooling system and coincided with entry into the pump of sodium carrying entrained gas that was released through the hex-can failure. There was no evidence that a changing blockage configuration triggered this scram.
7. Although no data were obtained relevant to long-term flow blockage behavior, due to the ETR scram, experimental evidence supports the position that rapid pin-to-pin propagation can be ruled out for oxide fuels.
8. Dissociation of sodium urano-plutonate that formed in the partial flow-channel blockage during the extended hot standby resulted in the blockage reconfiguration during the subsequent approach to full power.



9. DN systems performed as designed and demonstrated a higher sensitivity to failed fuel than other instrumentation. Multiple fuel exposure events were easily distinguished from one another. Although lack of fundamental basic data on isotope release and transport limited detailed analysis, there was excellent quantitative agreement between the DN estimate of disrupted fuel following the blockage reconfiguration and that identified during the PTE evaluation.
10. Had the major events in P4 occurred in a large power plant, they would have been detected unambiguously by a whole-core DN detection system. Although the longer transport time in a large plant reduces the contribution of the short-lived isotopes and the sharpness of the leading edge of the DN signal, the effect is not so great as to reduce the effectiveness of the detection system. To derive maximum benefit from a DN detection system, transit delays should be kept as small as possible and DN signals should be processed "intelligently" in conjunction with power, flow, and other data relevant to the operation of the reactor.

## I. INTRODUCTION

The Sodium Loop Safety Facility (SLSF) experiment P4 was a near-term milestone in the LMFBR Safety Program Plan<sup>1</sup> to demonstrate coolability of local faults and local fault accommodation by inherent mechanisms. It was the seventh, and last, in the series of SLSF large-scale in-reactor experiments performed between 1975 and 1981. Six SLSF experiments addressed whole-core-accident issues. P4 was the only SLSF experiment to address single-subassembly local fault and fuel-pin failure propagation issues. These experiments were conducted by both Argonne National Laboratory (ANL) and Hanford Engineering Development Laboratory (HEDL); P4 was the fifth ANL-sponsored SLSF experiment.

The facility, operated by EG&G Idaho, Inc., was located in the Engineering Test Reactor (ETR) at the Idaho National Engineering Laboratory (INEL). Operation of the SLSF in the ETR provided a controlled nuclear and thermal-hydraulic environment in which experiments could be conducted for extended periods (~35 full-power days) under conditions that closely simulated steady-state operation as well as postulated transients in liquid-metal fast breeder reactor (LMFBR) systems. SLSF experiments were planned to be prototypic of LMFBR conditions, i.e., to use full-length fuel pins, to precondition the test fuel, to provide LMFBR design flow rates, and to simulate reactor-coolant-expulsion characteristics.

Local fault safety experiment 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 fuel bundle and to probe consequences of fuel failure with continued power operation. It was performed to demonstrate that a major molten fuel release and potential flow-channel blockage, which bounds the consequences of credible local faults, could either be tolerated during continued full-power operation or be detected by global monitors in time to prevent significant fuel failure or blockage propagation. Specific objectives<sup>2</sup> were to determine 1) the extent, if any, of fuel-failure propagation, 2) the behavior of molten fuel within a pin bundle, and 3) the signals received that indicate the release of molten and/or solid fuel and failure propagation. In addition to studying the consequence of molten-fuel release, a primary goal for experiment P4 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 local fault whose signature falls within the defined band. If the consequences of the P4 "worse-than-worst-case" local fault configuration could be shown to be benign, more aggressive operating strategies for LMFBRs could be considered, requirements for LMFBR core surveillance and sensitivity of failed fuel detection systems revised, and scope of further failed-fuel testing reduced.

Specific experimental information expected from SLSF P4 included:

1. the time, location, and nature of fuel release,
2. the time, location, and nature of secondary failure, if any,
3. the amount of molten and solid fuel released to the coolant,

4. the extent of damage to the remaining fuel pins,
5. the tendency of the released fuel to be swept out and/or adhere to the fuel pins, and in what distribution, i.e., the tendency of secondary blockage formation, and
6. the delayed-neutron (DN) and fission-product signals characteristic of molten-fuel release and subsequent pin failures, if any.

Secondarily, information would be provided on the severity of a molten fuel-coolant interaction (shown to be mild in out-of-reactor experiments), and the resultant response of the coolant. These P4 data add to the local fault technology base to which the EBR-II Run-Beyond-Cladding-Breach and Operational Reliability programs<sup>3</sup> and the MOL 7C experiment program<sup>4</sup> are contributing.

Experiment P4 was intended to bound the consequences of credible subassembly faults by releasing molten fuel into the 37-pin bundle of full-length Fast-Test-Reactor (FTR)-type pins at full power and failing fuel with the potential for further cladding and fuel pin damage during continued power operation. The design goal for molten fuel release was 10 to 30 grams from one or more of the three fuel canisters built into the P4 fuel bundle. Bounding operating conditions were further emphasized by selection of an undercooled operating condition (power/flow ratio = 1.24 times that of a central FTR core subassembly), high coolant inlet temperature (695 K), and use of half-diameter wire wrap spacers on the outer row of fuel pins to reduce "cold" peripheral flow. Reactor operation through five 10-day irradiation cycles to a fuel burnup equivalent to 30 full-power days was planned.

This report provides a description of the loop system, design operating conditions, test performance and observations, posttest examination results, results of the posttest analyses, and extrapolation of results to reactor conditions.

## II. LOOP SYSTEM DESCRIPTION

### A. In-Reactor Loop

The SLSF in-reactor loop was a test vehicle of concentric, reentrant design that consisted 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 an instrumented test train. These loop components were arranged with the heat exchanger located above the pump and the in-reactor section, as illustrated in Fig. 1. Physical characteristics of the loop and components are listed in Table 1. In addition to the loop and test train, the SLSF system included a handling system, filling, storage and remelt (FS&R) system, plant protection system, loop and reactor control system, helium secondary coolant system, data acquisition system (DAS), and three failed-fuel detection systems. These failed-fuel detection systems included the delayed neutron detector (DND) system, the on-line cover gas sampling (OLCS) system, and the on-line sodium sampling (OLSS) system.

Wall thicknesses of the primary and secondary in-reactor containment vessels for the SLSF loop were selected to achieve a maximum containment capability at operating conditions. Each vessel had a wall thickness of 6.35 mm (0.25 in.), which represents an optimal trade-off between the temperature-dependent strength characteristics of Type 316 stainless steel and vessel heating due to ETR gamma-energy deposition. To establish the loop design, a peak pressure (design envelope molten fuel-coolant interaction (MFCI)) of 194.2 atm was postulated for a hypothetical "worst-case" event. The design envelope MFCI could be contained by a fully-plastic primary, without touching the secondary, at primary vessel temperatures  $\leq 1090$  K (1500°F). Outside diameters of the primary and secondary containment vessels in the in-reactor region were 111 mm (4.38 in.) and 133 mm (5.25 in.), respectively. The portion of the secondary vessel that was within the ETR core had a cadmium thermal-neutron filter bonded to the external surface to harden the neutron spectrum within the loop. The 1-mm (0.040-in.)-thick cadmium filter was 1.22-m (48-in.) long, extending 0.15 m (6 in.) above and below the core, and was overlaid with a thin stainless steel outer sheath.

Above the 3.05-m (10-ft)-long in-reactor tube, the secondary containment vessel OD increased to 439 mm (17.27 in.) [thickness = 12.7 mm (0.50 in.)] to accommodate the pump stator. The loop was supported by the dome flange of the reactor vessel top head and by a support ring suspended from lateral hangers. The secondary vessel diameter increased from 439 mm (17.27 in.) to 483 mm (19.0 in.) immediately above the pump, forming a tapered shoulder that seated in the support ring.

The inner diameter of the primary containment vessel formed the outer wall of the flow annulus within the pump. The pump stator was located between the primary and secondary. The primary and secondary containment were welded to a common flange that was part of the closure with the reactor vessel. Primary tube closure consisted of a bolted flange which supported the test train from the top of the loop. A removable top closure attached to the top of the secondary connected the data acquisition system and the helium secondary cooling system to the loop.

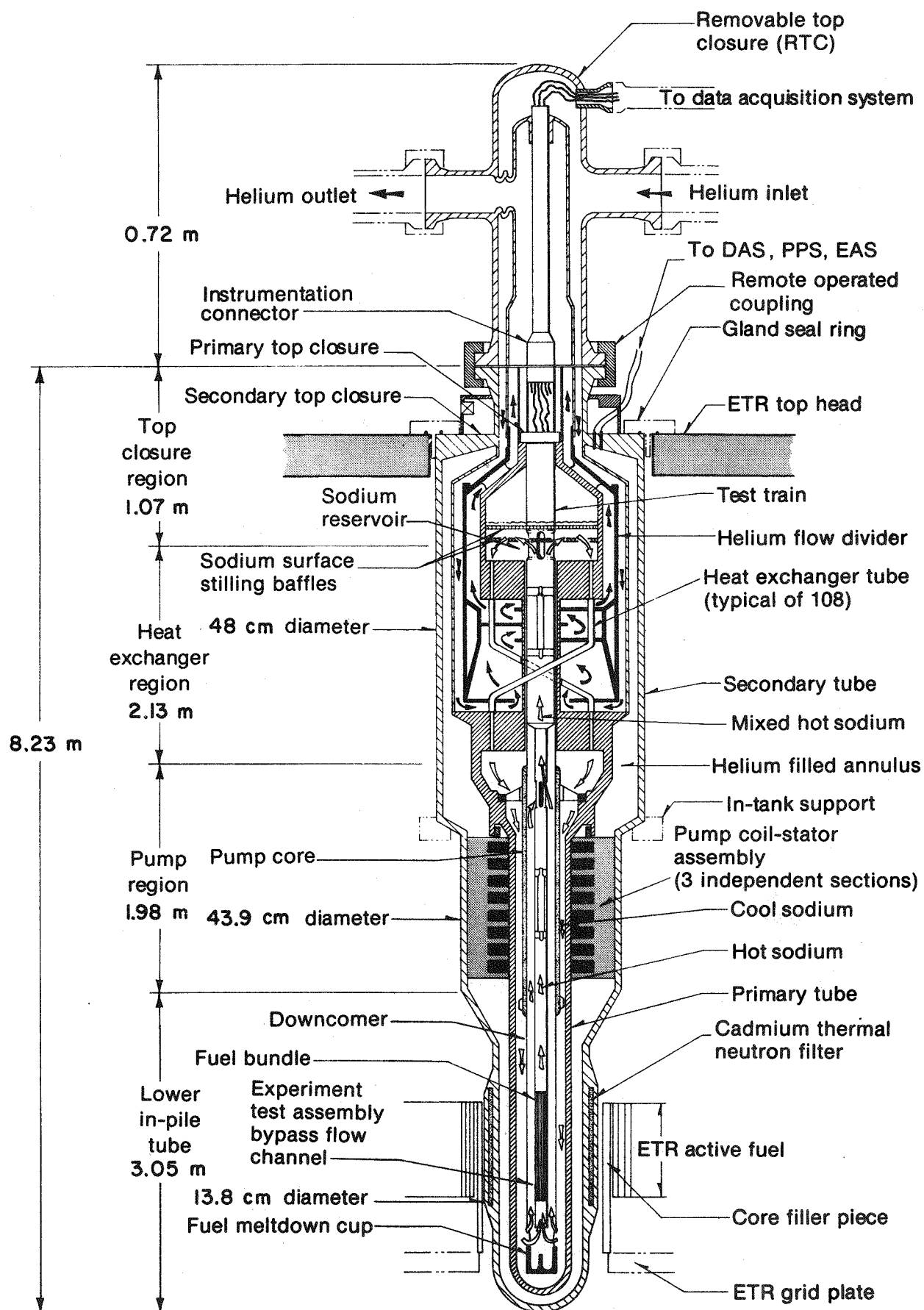


Fig. 1. SLSF In-Reactor Loop

Table 1. Summary of Physical Characteristics of the SLSF In-Reactor Loop

| Overall Loop Description                          |                                                                                                                        |
|---------------------------------------------------|------------------------------------------------------------------------------------------------------------------------|
| Type                                              | Package loop, doubly contained                                                                                         |
| Total length                                      | ~8.23 m                                                                                                                |
| Weight                                            | ~3175 kg (including test train)                                                                                        |
| Contained sodium                                  | ~85 kg                                                                                                                 |
| Containment vessel thickness<br>(in-reactor tube) | 6.35 mm                                                                                                                |
| Primary vessel, OD (in-reactor tube)              | 111.1 mm                                                                                                               |
| Secondary vessel, OD (in-reactor tube)            | 133.4 mm                                                                                                               |
| Cover gas volume                                  | ~30 liters                                                                                                             |
| Heat Exchanger                                    |                                                                                                                        |
| Type                                              | Tube and shell, Na-He counterflow,<br>Na inside tubes                                                                  |
| Overall length                                    | 2.06 m                                                                                                                 |
| Outer diameter                                    | 483 mm                                                                                                                 |
| Central tube ID                                   | 91.9 mm                                                                                                                |
| Number of tubes                                   | 108                                                                                                                    |
| Tube OD                                           | 19.1 mm                                                                                                                |
| Tube wall thickness                               | 1.2 mm                                                                                                                 |
| Capacity                                          | 1.5 MW at 840 K sodium inlet<br>temperature, 2610 kg/h helium<br>flow; 1.8 MW at full helium system<br>flow rate       |
| Pump                                              |                                                                                                                        |
| Type                                              | Annular linear induction, nine<br>pole, electrically divided into<br>three independent sections<br>operating in series |
| Physical                                          | Cylindrical                                                                                                            |
| Stator length                                     | 1.65 m                                                                                                                 |
| Outer diameter                                    | 439 mm                                                                                                                 |
| Wall thickness                                    | 12.7 mm                                                                                                                |
| Inner diameter of core                            | 88.9 mm                                                                                                                |
| Flow channel OD                                   | 167 mm                                                                                                                 |
| Flow channel area                                 | 18.91 cm <sup>2</sup>                                                                                                  |
| Design operating condition                        | 570 liters/min. at 1.03 MPa, 124-kW<br>power input; operated on test<br>stand to 950 liters/min. at 1.4<br>MPa.        |
| Test Train                                        |                                                                                                                        |
| Physical shape                                    | Cylindrical                                                                                                            |
| Length                                            | ~7.92 m                                                                                                                |
| Weight                                            | 200 kg                                                                                                                 |
| Flow divider OD (over insulating sleeve)          | 85.9 mm                                                                                                                |
| Number of FTR-type pins                           | 61                                                                                                                     |
| Meltdown-cup fuel capacity                        | 3.5 kg                                                                                                                 |
| Number of instruments                             | 100                                                                                                                    |

## B. Test Train

The P4 test train contained a 37-pin fuel-bundle assembly and most of the test instrumentation. Both the pump and heat exchanger had a central bore through which the test train was inserted and positioned in the loop. Sodium flow was downward through the heat exchanger, the pump, and the annular downcomer region to near the bottom of the loop. There sodium entered the test train and flowed upward through the parallel fuel bundle and bypass channels. These two flow streams merged midway up the test train. The heated sodium continued upward to the loop plenum and entered the heat exchanger to cycle through the loop again. The bypass channel aided in the simulation of reactor-core hydraulic response during the P4 experiment and ensured continued loop cooling in the event of a major flow blockage in the fuel bundle.

A longitudinal cross section through the test train and loop primary containment vessel at the midplane of the fuel-bundle assembly is shown in Fig. 2. Major individual components of the P4 test train are described below.

### 1. Meltdown Cup

A meltdown cup at the lower end of the test train protected the primary vessel from overheating due to hot fuel and cladding debris from an experiment or accident. It was sized to contain ~3.5 kg of molten fuel (equivalent to half the fuel in the 37 fuel pins) plus 2.4 kg of molten stainless steel. The meltdown cup consisted of a central 38-mm-dia. tungsten spike within a 85-mm-OD, 3.1-mm-thick Inconel tube and end cap having a 1.6-mm-thick tungsten inner liner insulated from the Inconel outer shell by a tantalum-sheathed boron-nitride insulator. Three thermocouples measured the sodium temperature inside the meltdown cup and one thermocouple measured the outer-wall temperature. Overall length of the Inconel tube and end cap was 298 mm.

### 2. Inlet Flow Region

Sodium entered the test train from the downcomer region of the loop immediately above the meltdown cup. The flow through the fuel bundle entered the test train through six inlet orifices, 7 mm in diameter, that were drilled horizontally through the outer tube. Immediately above these orifices were 15 4.8-mm-dia., 129-mm-long "ribbed" tubes, located in vertical passages in the flow proportioner body, through which sodium flowed from the downcomer into the bypass flow channel.

A permanent-magnet flowmeter measured the fuel-bundle coolant flow rate at the entrance to the fuel bundle. This flowmeter had a 19-mm-dia. flow channel and was 174 mm long. Two sets of permanent magnets, each with two pairs of leads, provided four inlet flow signals from this 'quad' inlet flowmeter. Four thermocouples measured the bundle inlet sodium temperature and four pressure sensors measured bundle inlet sodium pressure. Both slow response (0 to 1.38 MPa) and fast response (0 to 13.8 MPa) pressure sensors were used.

### 3. Test Section

The P4 fuel-bundle assembly consisted of 37 5.84-mm-OD, 2.37-m-long FTR-type fuel pins separated in a triangular array by 1.42-mm-dia. spacer

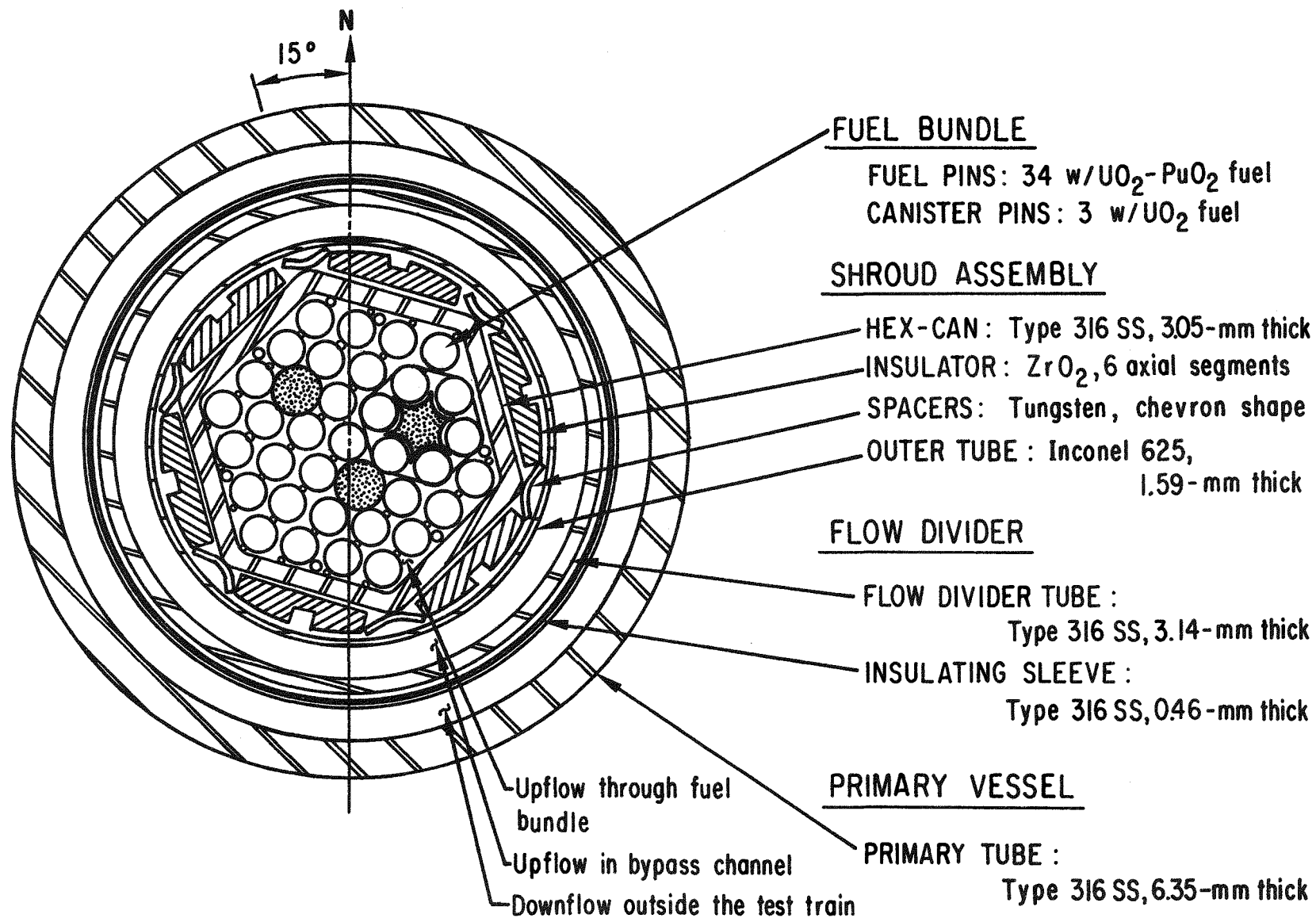


Fig. 2. Longitudinal Cross Section Through the P4 Test Train and Loop Primary Containment Vessel at the Midplane of the Fuel-Bundle Assembly



wires wrapped on a 305-mm pitch over the length of the pins. Three pins had fuel canisters at their fuel midplane. This fuel-bundle assembly was enclosed by a 3.05-mm-thick Type 316 stainless-steel hex-can, which provided support for the fuel bundle and served as the coolant flow boundary for the bundle sodium flow. Mechanical attachment of the pins to the hex-can was via a "T-bar" grid. The fuel-bundle assembly contained 24 thermocouples positioned at various radial and axial locations in the wire-wrap spacers. An additional six thermocouples were located in sheaths adjacent to the hex-can flats for both data acquisition and backup safety-protection purposes. Fourteen fuel pins had pressure transducers to monitor internal pressure.

An insulator region exterior to the hex-can provided a thermal barrier between the fuel bundle and the bypass. Zirconia insulators were enclosed within the argon-filled volume between the hex-can and the 1.57-mm-thick, 66.7-mm-OD cylindrical Inconel 625 outer tube that was the inner boundary of the bypass flow channel. The hex-can and the outer tube were joined at the top; a bellows at the bottom of the hex-can accommodated differential thermal expansion.

Bypass coolant flow was upward between the insulator-region outer tube and the flow divider. The 3.14-mm-thick, 84.2-mm-OD Type 316 stainless steel flow divider was the main structural member of the test train and separated the downward flowing sodium in the loop from the upward sodium flow within the bypass channel. Nine thermocouples measured bypass sodium temperature.

The cross-sectional geometry of the test train was relatively unchanged throughout the length of the test section. Starting above the upper elevation of the test-pin fuel columns, 0.46-mm-thick, 85.9-mm-OD grooved tube sections were welded to the outside of the flow divider, sealing a vacuum in the gap to minimize regenerative heat transfer from the hot upward sodium flow in the test train to the cold downward flow in the loop. Also, above the fuel column of the fuel pins and continuing upward, the various instrument leads were gathered into conduits that terminated immediately below the wire terminal and Viking connector at the top of the test train.

The P4 test train contained a fission-product recoil source to provide delayed neutrons for testing and calibrating the delayed-neutron, failed-fuel detection systems prior to fuel failure. The source consisted of a 1-mm-thick, 10-mm-wide, 100-mm-long strip of nickel-uranium alloy containing 3.5 wt.% U enriched to 93%  $^{235}\text{U}$ . One side provided 10 cm<sup>2</sup> of area from which delayed neutron precursors were liberated into the sodium stream during reactor operation. The other side was plated with nickel to stop fission recoil. This source was mounted in a "picture frame" welded to the Inconel 625 inner surface of the bypass flow channel at the fuel midplane. The unplated surface of the source was exposed to the bypass channel flow.

#### 4. Bundle Outlet

From the upper end of the hex-can and outer-tube assembly, the fuel-bundle coolant flow stream continued upward through a 0.67-m-long bundle outlet instrumentation region before joining the bypass sodium. A total of 13 instruments were located in this region. Three thermocouples and four pressure sensors, two fast response and two slow response, monitored the

bundle-outlet sodium conditions. Three thermocouples protruded through the flow divider to sense the downcomer sodium temperature at the exit of the pump; pump discharge pressure was also sensed at this elevation. One acoustic detector was located in the bypass channel. The bundle-exit sodium flow rate was measured by an eddy-current probe-type flowmeter. The flowmeter was preceded by a flow straightener that developed a turbulent flow profile in the annular channel past the sensing region of the flowmeter probe. Overall length of the flowmeter and flow straightener was 380 mm.

#### 5. Combined-Flow Section

This region of the test train contained two combined-flow flowmeters, four thermocouples, two pressure sensors and one acoustic detector in the 3.4-m length of insulated flow divider between the elevation where the fuel bundle and bypass flow streams combine and the flow diverter. The vacuum-insulation sleeve on the flow divider was interrupted near the top of the pump for a 54-mm-long labyrinth seal to minimize recirculation of sodium flow from the pump discharge directly back to the pump inlet along the outside of the test train. Within the 77.9-mm-ID of the test train flow divider were a variety of conduits containing instrument leads and mounting hardware for the sensors.

Two eddy-current probe-type flowmeters measured the combined fuel bundle and bypass sodium flow rate. They were mounted nose to nose within a common annular channel having a flow straightener at the inlet end. The mixed-mean sodium temperature was measured at three locations: at the flow straightener, at an elevation 250 mm above the location of fuel bundle and bypass channel flow combination, and at the exit of the upper flowmeter.

A 51-mm-dia. by 0.74-m-long filter, formed from No. 14 square mesh (0.51-mm thick) wire cloth, was located between the total-flow flowmeters and the flow diverter to remove fuel or cladding debris  $\geq 1.3$ -mm dia. from the coolant. The flow diverter directed the total flow out of the test train and into the sodium plenum volume that surrounded the test train above the upper tube sheet of the heat exchanger.

#### 6. Plenum and Connector Section

The instrument-lead conduits penetrated the bulkhead that formed the top of the flow divider and terminated above the conduit seal plug, which sealed the test train to the loop by means of a metal "K"-seal gasket ring. This section of the test train contained four plenum-gas pressure sensors (0 to 0.69 MPa), two gas-plenum thermocouples, two "J-probe"-type sodium level sensors, and one thermocouple to monitor the temperature of the Viking instrument connector. It formed part of the 30-liter gas plenum volume above the operating sodium level. The 178-mm-dia., 38-mm-thick conduit seal plug, in addition to containing conduit penetrations, also contained penetrations for gas- and sodium-system service.

The Viking electrical connector formed the top of the test train. It contained 211 pins and mated with its counterpart in the removable top closure (RTC) to connect the test train instrument leads to the DAS.

### C. Fuel-Bundle Assembly

The P4 fuel-bundle assembly consisted of 34 FTR-type fuel pins and three fuel pins with fuel canisters located at the midplane of the fuel pellet columns. A cross section of the P4 bundle assembly through the fuel canister region is shown in Fig. 4.

The 34 FTR-type fuel pins were 2.37-m long and contained a 0.91-m-long  $\text{UO}_2\text{-PuO}_2$  fuel pellet column and upper and lower Inconel 600 reflector rods within a 0.38-mm-thick, 5.84-mm-OD, Type 316 stainless steel cladding. The  $\text{PuO}_2$  content of the fuel was 25 wt.% with a  $^{239}\text{Pu} + ^{241}\text{Pu}$  enrichment of 88%. A row-by-row grading of the fuel  $^{235}\text{U}$  enrichment was used, in conjunction with the cadmium neutron filter on the loop secondary vessel, to produce a flat radial power profile across the bundle. The center pin had a  $^{235}\text{U}$  enrichment of 92.5%; row by row out from the center pin the  $^{235}\text{U}$  enrichments were 88.7%, 78.8% and 58.3%, respectively. One exception - Pin No. 4 containing fuel with a  $^{235}\text{U}$  enrichment of 92.5% had a row-of-six location to serve as a hot "target" pin between two fuel canisters. The P4 FTR-type fuel pins are described further in Table 2. Fourteen of the central 19 pins had pressure transducers in lieu of upper end caps; a variety of fill-gas tags were used.

The three fuel canister pins had the same nominal length, diameter, cladding thickness and fuel-pellet diameter as the other pins, except at the fuel canisters, but contained 92.8% enriched  $\text{UO}_2$  fuel pellets in both the pin and the 100-mm-long fueled region of the canisters. Two canisters were of cylindrical geometry and one had a fluted cross section; the fluted canister is shown in Fig. 3. The 8.69-mm-dia. of each circular canister was selected to produce line contact with the adjacent fuel pins in an undistorted bundle geometry. The fluted canister occupied the nominal area of six coolant flow subchannels. It was spaced away from each of the six neighboring fuel pins by a 0.16-mm-dia. wire positioned vertically between each cusp region of the fluted canister and the adjacent pin.<sup>5</sup> Two 0.25-mm-thick, 12.7-mm-wide stainless steel bands were fastened around the six pins that contacted the fluted canister to prevent it from "ballooning" back to a circular cross section during experiment operation. The fuel canister pins are described further in Table 3.

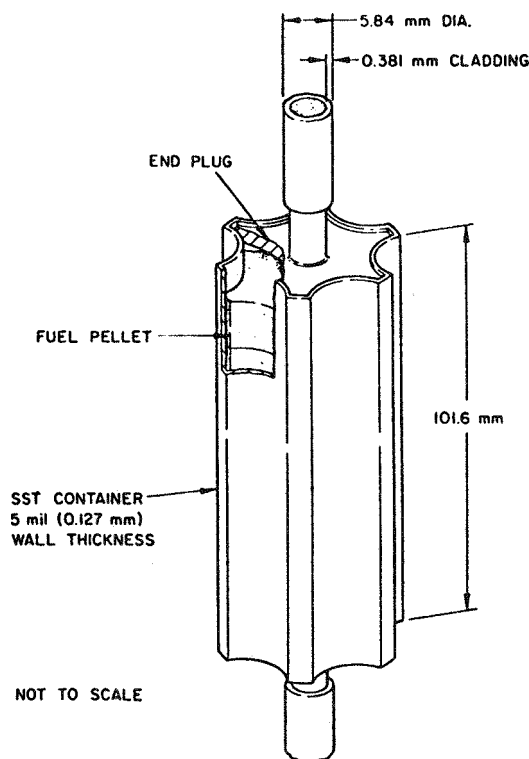


Fig. 3. Fluted Fuel Canister

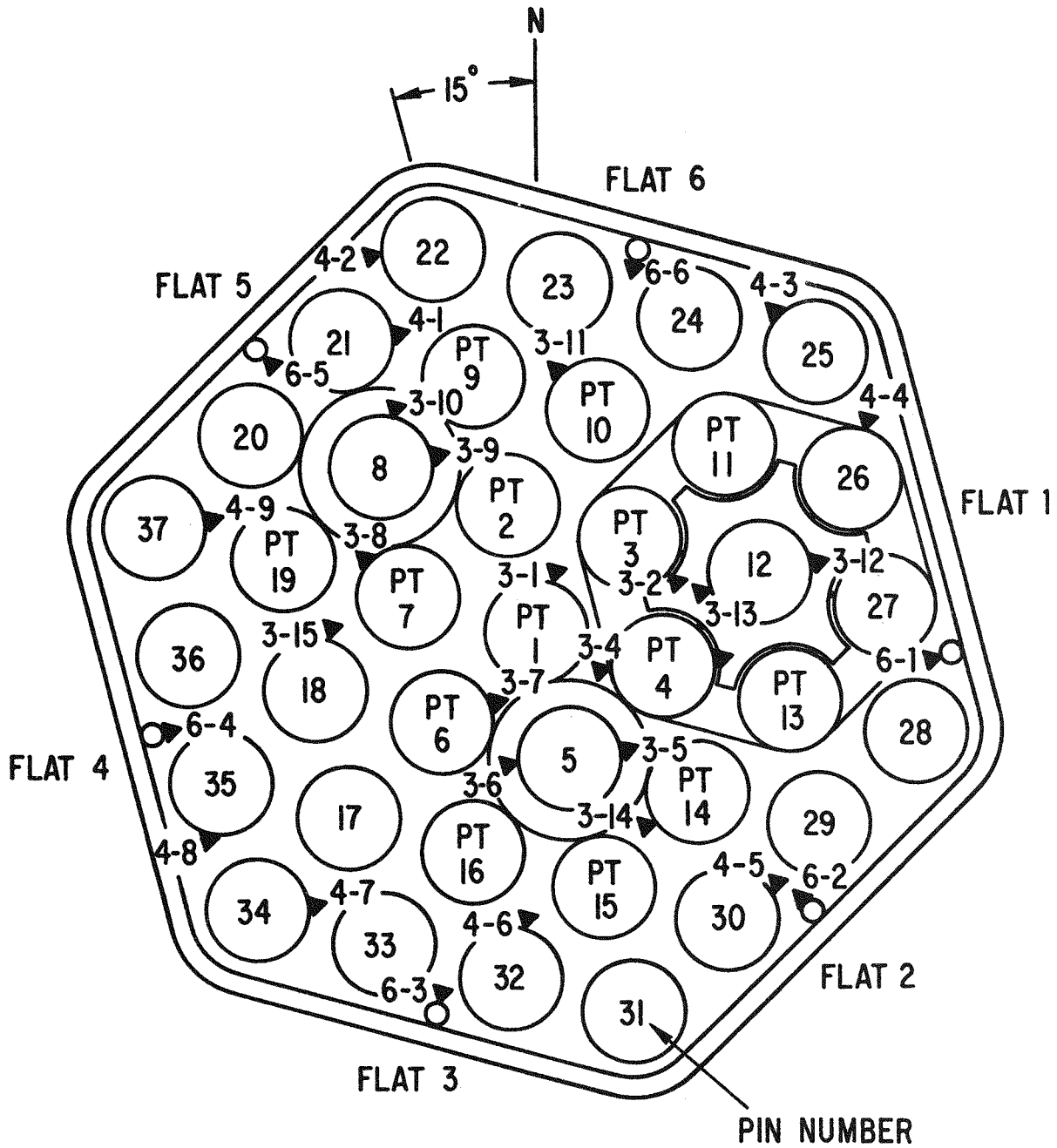


Fig. 4. Cross Section of the P4 Fuel-Bundle Assembly Through the Fuel Canister Region

Table 2. Parameters for FTR-Type SLSF P4 Fuel Pins

| Parameter                                                  | Center and<br>"Target" Pins                                                                                               | Row-of-Six<br>Pins | Row-of-12<br>Pins | Row-of-18<br>Pins |
|------------------------------------------------------------|---------------------------------------------------------------------------------------------------------------------------|--------------------|-------------------|-------------------|
| Fuel                                                       | UO <sub>2</sub> -PuO <sub>2</sub>                                                                                         |                    |                   |                   |
| No. pins                                                   | 2                                                                                                                         | 4                  | 10                | 18                |
| No. with pressure transducers<br>in lieu of upper end caps | 2                                                                                                                         | 4                  | 8                 | 0                 |
| Uranium enrichment ( <sup>235</sup> U/U), wt.%             | 92.49                                                                                                                     | 88.66              | 78.81             | 58.34             |
| Fuel composition (Pu/Pu+U)                                 | 0.245                                                                                                                     | 0.245              | 0.245             | 0.250             |
| Fuel stoichiometry (O/M)                                   | 1.992                                                                                                                     | 1.998              | 1.992             | 1.996             |
| Plutonium enrichment, wt.%                                 | 88.0 ± 0.5 <sup>239</sup> Pu + <sup>241</sup> Pu in Pu<br>0.5 max <sup>238</sup> Pu<br>11.0 ± 1.0 <sup>240</sup> Pu in Pu |                    |                   |                   |
| Fuel pellet diameter, mm (in.)                             | 4.94 (0.1945 ± 0.0015)                                                                                                    |                    |                   |                   |
| Fuel pellet geometry                                       | Dished ends                                                                                                               |                    |                   |                   |
| Fuel pellet density, % TD                                  | 90.4 ± 3.5 (~9.85 g/cm <sup>3</sup> )                                                                                     |                    |                   |                   |
| Fuel smeared density, % TD                                 | 85.5 ± 2.0                                                                                                                |                    |                   |                   |
| Fuel column length, mm (in.)                               | 914.4 (36.0 ± 0.15)                                                                                                       |                    |                   |                   |
| Fuel column weight, g                                      | 171 to 173                                                                                                                |                    |                   |                   |
| Natural UO <sub>2</sub> column - total length, mm (in.)    | 40.6 (1.6 ± 0.08)                                                                                                         |                    |                   |                   |
| Cladding material                                          | 20% CW 316 SS                                                                                                             |                    |                   |                   |
| Cladding OD, mm (in.)                                      | 5.84 (0.230 ± 0.0005)                                                                                                     |                    |                   |                   |
| Cladding ID, mm (in.)                                      | 5.08 (0.200 ± 0.0005)                                                                                                     |                    |                   |                   |
| Pin plenum volume<br>(nominal), cm <sup>3</sup>            | 19.18                                                                                                                     |                    |                   |                   |
| Plenum gas (prior to activating gas<br>tag capsule)        | Helium                                                                                                                    |                    |                   |                   |
| Gas-tag capsule gas                                        | Helium + 1.00 ± 0.25 cm <sup>3</sup> Xenon for<br>inner 19 pins                                                           |                    |                   |                   |
| Gas pressure (room temperature), kPa (atm.)                | 304 (3)                                                                                                                   |                    |                   |                   |

Table 3. Characteristics of P4 Fuel-Canister Pins

|                                                |                                                    |                                                                             |                                |
|------------------------------------------------|----------------------------------------------------|-----------------------------------------------------------------------------|--------------------------------|
|                                                | Pin No. 5                                          | Pin No. 8                                                                   | Pin No. 12                     |
| Fuel Pin Length, m (in)                        | 2.37 (93.4)                                        |                                                                             |                                |
| Upper, Lower Sections                          |                                                    |                                                                             |                                |
| Fuel                                           | UO <sub>2</sub>                                    |                                                                             |                                |
| Uranium enrichment ( <sup>235</sup> U/U), wt.% | 92.8                                               |                                                                             |                                |
| Fuel stoichiometry (O/M)                       | 2.000                                              |                                                                             |                                |
| Fuel pellet diameter, mm (in)                  | 4.95 (0.195)                                       |                                                                             |                                |
| Fuel pellet density, % TD (g/cm <sup>2</sup> ) | 87.9 (9.52)                                        |                                                                             |                                |
| Fuel smeared density, % TD                     | 83.4                                               |                                                                             |                                |
| Cladding material                              | Type 316 stainless steel; 20% CW                   |                                                                             |                                |
| Cladding dimensions, mm (in)                   | 5.84 (2.300) OD; 5.08 (1.999) ID                   |                                                                             |                                |
| Upper Section                                  |                                                    |                                                                             |                                |
| Fuel column length, mm (in)                    | 369 (14.51)                                        | 372 (14.66)                                                                 | 369 (14.54)                    |
| Fuel column weight, g                          | 66.9                                               | 67.4                                                                        | 66.9                           |
| Fill/tag gas                                   | 47.8 mol % He<br>52.2 mol % Xe                     | 50.4 mol % He<br>49.6 mol % Kr                                              | 46.5 mol % He<br>53.5 mol % Xe |
| Fill gas pressure,* kPa (psia)                 | 390 (56.7)                                         | 390 (56.7)                                                                  | 390 (56.7)                     |
| Lower Section                                  |                                                    |                                                                             |                                |
| Fuel column length, mm (in)                    | 372 (14.65)                                        | 371 (14.62)                                                                 | 372 (14.63)                    |
| Fuel column weight, g                          | 68.0                                               | 67.7                                                                        | 67.5                           |
| Fill gas (No tag)                              | He                                                 | He                                                                          | He                             |
| Fill gas pressure,* kPa (psia)                 | 390 (56.7)                                         | 390 (56.7)                                                                  | 390 (56.7)                     |
| Canister Section                               |                                                    |                                                                             |                                |
| Canister geometry                              | cylinder                                           | cylinder                                                                    | fluted                         |
| Cladding material                              | 20% CW 304 SS                                      | 10% CW 304 SS                                                               | annealed 304 SS                |
| Cladding OD, mm (in)                           | 8.69 (0.342)                                       | 8.69 (0.342)                                                                | ---                            |
| Cladding thickness, mm (in)                    | 0.13 (0.005)                                       | 0.13 (0.005)                                                                | 0.13 (0.005)                   |
| Fuel                                           | UO <sub>2</sub>                                    | UO <sub>2</sub>                                                             | UO <sub>2</sub>                |
| Uranium enrichment, ( <sup>235</sup> U/U) wt%  | 92.8                                               | 92.8                                                                        | 92.8                           |
| Fuel stoichiometry (O/M)                       | 2.000                                              | 2.000                                                                       | 2.001                          |
| Fuel pellet diameter, mm (in)                  | 8.369 (0.3295)                                     | 8.291 (0.3264)                                                              |                                |
| Fuel pellet density, % TD (g/cm <sup>3</sup> ) | 94.85 (10.27)                                      | 94.96 (10.28)                                                               | 94.7 (10.25)                   |
| Fuel smeared density, % TD                     | 94.55                                              | 94.64                                                                       |                                |
| Fuel column length, mm (in)                    | 99.82 (3.930)                                      | 99.64 (3.923)                                                               | 99.48 (3.877)                  |
| Fuel column weight, g                          | 56.6                                               | 55.5                                                                        | 68.7                           |
| Fill/tag gas                                   | Kr<br>95% <sup>78</sup> Kr,<br>5% <sup>80</sup> Kr | Kr<br>76% <sup>82</sup> Kr,<br>20% <sup>80</sup> Kr,<br>4% <sup>83</sup> Kr | 85 mol % He,<br>15 mol % Xe    |
| Fill gas pressure*, kPa (psia)                 | 239 (34.7)                                         | 239 (34.7)                                                                  | 239 (34.7)                     |

\* at room temperature

The P4 fuel pins were spaced on a 7.26-mm (pin center to pin center) triangular pitch with 1.42-mm-dia. spacer wires wrapped the length of the central 19 pins on a 305-mm pitch. Many of these spacer wires contained single- or dual-junction thermocouples. The spacing between the outer-row pins and the hex-can was 0.71 mm, half the distance between pins in the bundle, to decrease flow streaming at the edge of the bundle and to improve the uniformity of the coolant distribution and the radial temperature field within the bundle. Sleeves were placed over the 0.71-mm-dia. single-junction thermocouples or solid spacer wires on the outer-row pins, at appropriate intervals, to maintain the proper pin spacing and support within the bundle. Where the fuel canisters prevented the continuous wrapping of a spacer wire over the entire length of a fuel pin, the wire wraps were terminated adjacent to the end caps of the canisters, both top and bottom, and held in place by a "garter" arrangement.

The P4 fuel bundle was enclosed by a 3.05-mm-thick Type 316 stainless-steel hex-can measuring 45.0 mm across flats on the inside. Between the hex-can and a 1.59-mm-thick, 66.7-mm-OD cylindrical Inconel 625 outer tube was an insulating region containing six zirconia insulator segments positioned by chevron-shaped tungsten spacers. An argon gas atmosphere was maintained in the insulator volume. The pressure of the high-purity argon gas at design operating conditions was 520 kPa (75 psia), 69 kPa (10 psia) above the bundle-inlet sodium pressure, for hex-can failure detection purposes. Bundle outlet and total flowmeters would sense gas release into the sodium coolant from a hex-can failure. A summary of the P4 fuel-bundle-assembly geometry is given in Table 4.

#### D. Instrumentation

Basic instrumentation in the SLSF test trains included flowmeters, thermocouples, pressure sensors, acoustic detectors, and sodium level sensors. Similarity in instrumenting of the test trains was maintained where permitted by objectives of the experiments. The instrumentation in each of the 37-pin SLSF test trains is summarized in Table 5. SLSF experiments P3A and P3 simulated whole-core unprotected loss-of-flow (LOF) accidents involving all the pins in the bundle in a predictable sequence of events. Because the LOF event sequence would be influenced by significant radial heat loss from the fuel bundle, the P3A and P3 insulator regions were heavily instrumented to monitor integrity of the thermal barrier.

Fuel-pin-plenum pressure sensors on pins in the center of the P4 fuel bundle were necessary to identify individual fuel-pin failure and possible failure propagation. Fuel-bundle low-flow protection was required for P4 postfailure power operation. Protective function (PF) H, which monitored filtered signals from the "quad" inlet flowmeter, was added to the SLSF plant protection system (PPS) to preclude continued full-power operation at low bundle flow. The six hex-can inner-wall thermocouples for PF-J monitored sodium coolant temperatures above the midplane of the fuel bundle to prevent excessive primary vessel temperatures. PF-J could be used as either a backup or as an alternate to the PF-H low-flow function.

Table 4. P4 Fuel-Bundle-Assembly Parameters

|                                                      |                                                                 |
|------------------------------------------------------|-----------------------------------------------------------------|
| Number of fuel pins                                  |                                                                 |
| - full-length FTR type                               | 34                                                              |
| - full-length w/fuel canister                        | 3                                                               |
| Fuel-pin cladding OD, mm (in)                        | 5.84 (0.230)                                                    |
| Fuel-pin pitch, mm (in)                              | 7.26 (0.286)                                                    |
| Wire wrap spacer diameter, mm (in)                   |                                                                 |
| - center pin                                         | 1.42 (0.056)                                                    |
| - edge pins*                                         | 0.71 (0.028)                                                    |
| Hex-can flat-to-flat dimension, mm (in)              | 45.01 (1.772)                                                   |
| Fuel bundle across-flats dimension, mm (in)          |                                                                 |
| - at fuel canister                                   | 45.80 (1.803)                                                   |
| - away from canister                                 | 45.01 (1.772)                                                   |
| Bundle flow area, cm <sup>2</sup> (in <sup>2</sup> ) |                                                                 |
| - at fuel column midplane                            | 6.039 (0.936)                                                   |
| - at fluted canister bands                           | 5.877 (0.911)                                                   |
| - away from canisters                                | 7.026 (1.089)                                                   |
| Hex-can thickness, mm (in)                           | 3.05 (0.120)                                                    |
| Zirconia insulator thickness, mm (in)                |                                                                 |
| - average                                            | 3.30 (0.130)                                                    |
| - maximum                                            | 5.61 (0.221)                                                    |
| Insulator-region fill gas                            | high purity argon<br>177 kPa (25.7 psia)<br>at room temperature |
| Outer-tube inside diameter, mm (in)                  | 63.50 (2.500)                                                   |
| Outer-tube thickness, mm (in)                        | 1.59 (0.062)                                                    |

\* sleeves at appropriate locations to maintain proper pin-to-pin spacing.



Table 5. SLSF 37-Pin Test Train Instrumentation

|                          | Experiment<br>P3A |           | Experiment<br>P3 |           | Experiment<br>P4 |           |
|--------------------------|-------------------|-----------|------------------|-----------|------------------|-----------|
|                          | spare             | active    | spare            | active    | spare            | active    |
| Sodium level sensors     |                   | (2)       |                  | (2)       |                  | (2)       |
| loop gas plenum          | 1                 | 1         | 1                | 1         | 1                | 1         |
| Acoustic sensors         |                   | (2)       |                  | (2)       |                  | (2)       |
| total flow region        |                   | 1         |                  | 1         |                  | 1         |
| bypass flow region       |                   | 1         |                  | 1         |                  | 1         |
| Pressure sensors         |                   | (18)      |                  | (18)      |                  | (29)      |
| loop gas plenum          |                   | 4         |                  | 4         |                  | 4         |
| total flow region        |                   | 2         | 1                | 1         |                  | 2         |
| pump discharge           |                   | 1         | 1                |           |                  | 1         |
| fuel bundle outlet       |                   |           |                  |           |                  |           |
| - fast response          | 2                 |           | 1                | 1         | 1                | 1         |
| - slow response          |                   | 2         |                  | 2         | 1                | 1         |
| hex-can insulator region | 1                 | 1         |                  | 2         |                  |           |
| fuel pin gas plenum      |                   | 1         |                  | 1         | 3                | 11        |
| fuel bundle inlet        |                   |           |                  |           |                  |           |
| - fast response          | 2                 |           | 1                | 1         |                  | 2         |
| - slow response          |                   | 2         |                  | 2         |                  | 2         |
| Thermocouples            |                   | (76)      |                  | (77)      |                  | (60)      |
| instrument connector     |                   | 2         |                  | 2         |                  | 1         |
| loop gas plenum          |                   | 2         |                  | 2         |                  | 2         |
| total flow region        | 1                 | 3         | 1                | 3         |                  | 4         |
| pump discharge           |                   | 2         |                  | 3         | 1                | 2         |
| fuel bundle outlet       |                   | 3         |                  | 3         |                  | 3         |
| fuel bundle              | 4                 | 20        | 4                | 20        |                  | 24        |
| hex-can                  |                   |           |                  |           |                  |           |
| - inside wall            |                   |           |                  |           |                  | 6         |
| - outside wall           |                   | 5         |                  | 5         |                  |           |
| - outside wall (hi temp) |                   | 4         |                  | 4         |                  |           |
| - insulator              | 1                 | 9         | 2                | 8         |                  |           |
| bypass flow region       | 3                 | 9         | 2                | 10        | 4                | 5         |
| fuel bundle inlet        | 1                 | 3         | 1                | 3         |                  | 4         |
| meltcup                  |                   | 4         |                  | 4         |                  | 4         |
| Flowmeters               |                   | (4)       |                  | (4)       |                  | (7)       |
| total flow               |                   | 2         |                  | 2         |                  | 2         |
| fuel bundle outlet       |                   | 1         |                  | 1         |                  | 1         |
| fuel bundle inlet        |                   | 1         |                  | 1         |                  | 4         |
|                          | <u>16</u>         | <u>86</u> | <u>15</u>        | <u>88</u> | <u>11</u>        | <u>89</u> |

Adding the fuel-pin-plenum pressure sensors and the additional instruments for PPS PF-H and PF-J monitoring during experiment P4 reduced the number of other test train instrument signals available to the DAS through the 211-pin Viking connector. All instrumentation, both thermocouples and pressure sensors, to monitor the integrity of the gas-filled insulator region in the P4 test train was eliminated and the number of bypass-channel thermocouples reduced. For P4, the safety analysis was based on a sodium-filled hex-can insulator region, and gas passage through bundle outlet and total flowmeters was relied upon to signal hex-can failure.

The thermocouples located throughout the P4 fuel bundle were of the chromel-alumel, ungrounded-junction type. Although slower in time response than grounded-junction thermocouples, the ungrounded-junction thermocouples were selected for higher accuracy and reliability during steady-state operation. Fuel-bundle-thermocouple radial locations and fuel pins with plenum pressure sensors are shown in Fig. 4. The axial distribution of thermocouples in the P4 fuel bundle is shown in Fig. 5.

The test train instrumentation was augmented by instruments located on the loop and also on the failed-fuel detection systems. Test train, loop, and certain ETR data were logged by the SLSF DAS.<sup>6</sup> Most of the data from the OLCS and OLSS failed-fuel detection systems were logged by the OLCS computer. A complete listing of the input to the SLSF data acquisition system is shown in Appendix A.

#### E. Failed-Fuel Detection Systems

The DND, OLCS, and OLSS were operated during experiment P4 to observe fuel failure. The OLCS had been operated earlier during experiment W1 and the DND operated in support of experiment W2.

To provide a source of delayed neutrons in the absence of failed fuel pins, a fission product source was mounted in the coolant stream in the loop. This source enabled the operation of the DND and DNM to be checked out and calibrated under actual conditions.

The source material was an alloy of nickel and uranium which had already been used at EBR-II and had proven to be of great value in the interpretation of the observed failure measurements. It was in the form of a flat strip 10-cm long by 1-cm wide by 0.089-mm thick. The composition, by weight, was 96.5% Ni and 3.5% U; <sup>235</sup>U enrichment was 93.2%.<sup>7</sup>

The total mass of the fission product source was 7.74 g; the total mass of <sup>235</sup>U was, therefore, 252 mg. One side of the strip was plated with pure nickel thick enough to prevent any fission fragments from leaving that surface. The strip was located on the outside of the fuel-bundle outer tube with the plated side against the outer tube, being slightly curved to fit snugly. At the axial center of the ETR core, fission products were emitted from the unplated surface into the upward-flowing sodium stream in the bypass coolant channel.

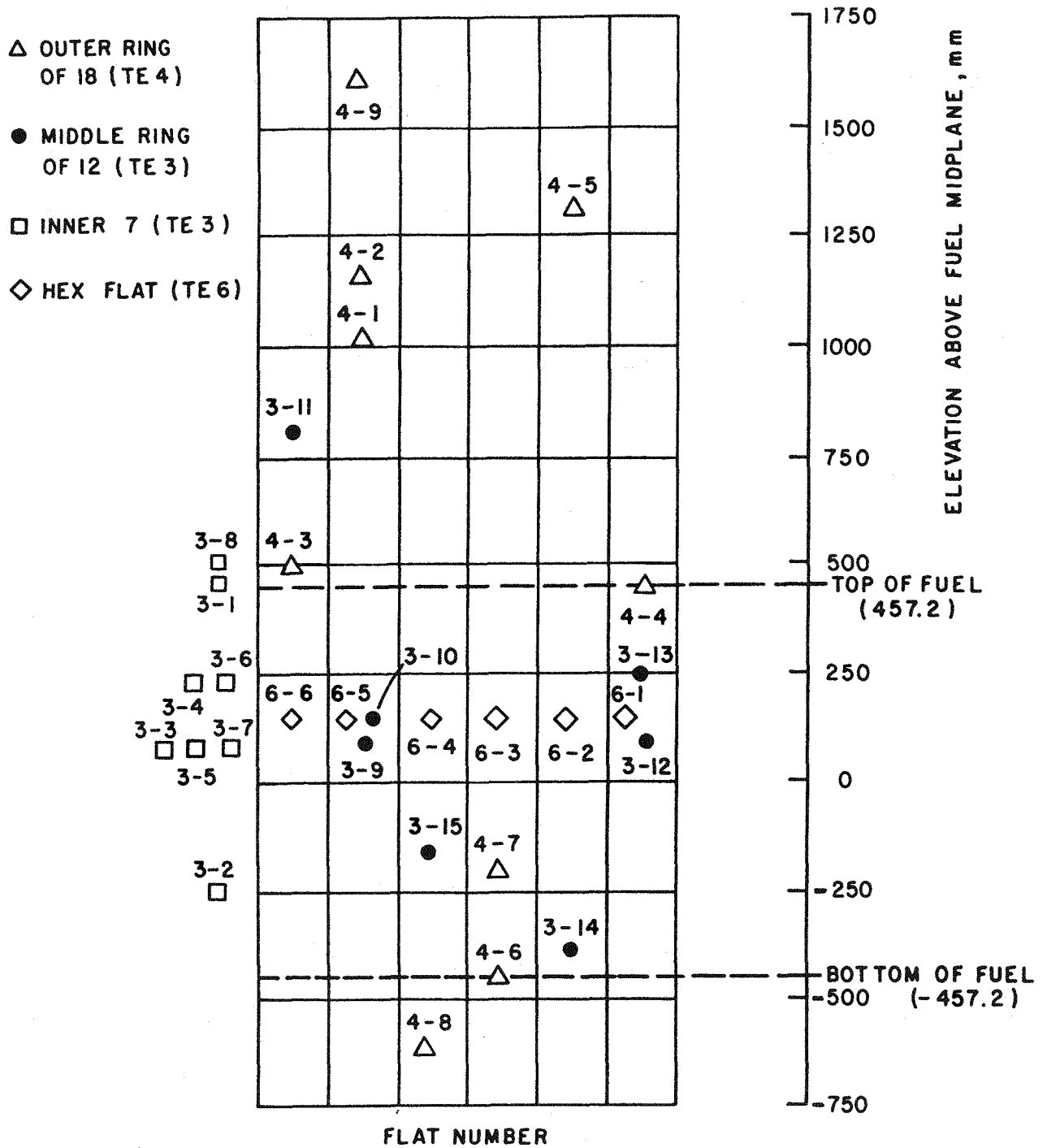


Fig. 5. Wire-Wrap Thermocouple Layout

The strip thus formed a "recoil source" having an alloy area of  $10 \text{ cm}^2$ , equivalent to  $32.5 \text{ mm}^2$  of  $^{235}\text{U}$ . Fission products were able to leave only if they came from a layer of thickness equal to the range of the fission fragments. Because the fragments move at all angles to the surface, only 1/4 of all fragments from this layer were able to escape. For calculation purposes, a fission range of seven microns was used (values of 5.5 to 6.5 microns have been measured). Therefore, the effective source was  $5(10^{-4}) \text{ g}$  of  $^{235}\text{U}$ . This value was used previously in calculating the DN count rate from the fission product source during calibration of the detection systems for SLSF experiment W2.<sup>8</sup>

### 1. On-Line Cover Gas Sampling System

The OLCS was a once-through gas-handling system that continuously sampled and conditioned a stream of argon cover gas from the loop to monitor for radioactive isotopes of Kr and Xe from failed fuel.<sup>9</sup> Figure 6 illustrates the piping system and major components of the OLCS. Two liquid-nitrogen-cooled germanium crystal spectrometers recorded spectra of gamma rays in the energy range from 80 keV to 2.7 MeV. The gamma data were logged by a mini-computer that also had OLCS system operation, control, and data analysis functions.

Purified argon was supplied to the loop at the desired pressure through insulated piping that was heated to deliver the gas at the loop-plenum temperature, to inhibit sodium aerosol formation. Gas sample flow, between 300 sccm and 2500 sccm, was provided by downstream vacuum pumps and was regulated by pressure and flow control. The gas sample stream from the loop passed through warm and cold vapor traps prior to entering the sample chambers. The flow rate, in conjunction with a set of delay lines, determined the transport time to the detectors. The range of delay times could be varied between 40 and 1500 s to permit the  $^{23}\text{Ne}$  background activity ( $t_{1/2} = 23 \text{ s}$ ) to decay. Experience during SLSF experiment W1 showed that a delay time of 5 minutes was sufficient to reduce the  $^{23}\text{Ne}$  background to an acceptable level. Therefore, the inserted delay and flow rate were chosen to produce that delay time. In addition to transport delay, the gas sample stream could be diluted with clean, dry argon in a range of ratios from 1:1 to 2000:1 to reduce the count rate.

One lead-shielded sample chamber was designed for low source strength. It contained ~10 g of activated charcoal to enhance Xe and Kr activities over the  $^{23}\text{Ne}$  background. The other chamber had a sample volume variable between  $0.5 \text{ cm}^3$  and  $25 \text{ cm}^3$ . One of a number of collimators could be selected to limit the volume of the chamber "seen" by the detector. By means of sample dilution, adjustment of sample chamber volume, and use of detector collimators, the sample chamber could span an intensity range of nine decades without detector saturation.

Because only large releases of gases were expected in P4, the flow of radioactive gases were directed to this second sample chamber. Two ionization detectors were used to observe the activity in the OLCS piping, RD3 before and RD4 after the dilution point (see Fig. 6). These detectors were interfaced to the computer which controlled the entire OLCS system, so as to activate the dilution controls when a large release of gases from the fuel reached RD3. The program also controlled the volume of the sample chamber and

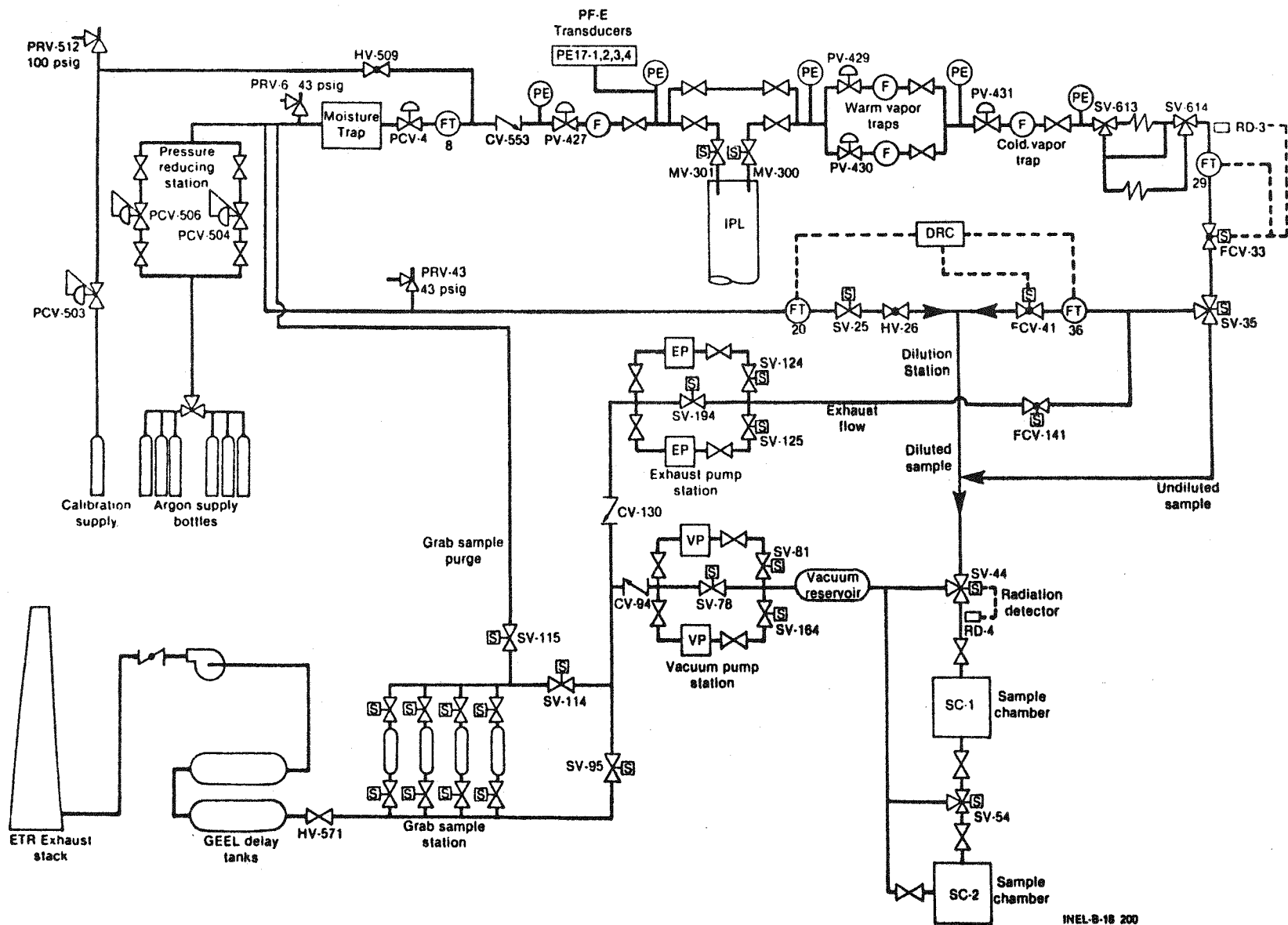


Fig. 6. Piping System and Major Components of the OLCS

the collimator to limit the count rates in the high-resolution germanium detectors, RD1 and RD2. Both gamma spectrometers could observe the sample volume simultaneously. RD2 contained hyperpure germanium diode sensitive to low-energy gamma rays; RD1 contained a closed-end coaxial GeLi crystal designed for gamma rays from 500 kev up to several Mev. From the sample chambers, the gas sample stream passed into baffled delay tanks prior to being discharged to the atmosphere through the ETR stack.

## 2. Delayed Neutron Detector

The DND monitored the DN activity of the loop sodium flowing through the 13-liter sodium-reservoir volume above the heat exchanger, without removing a sample from the loop. The detector assembly was contained in a housing sealed against the reactor primary coolant water environment and mounted under the ETR top dome. It was carefully positioned against the loop secondary vessel to minimize the thickness of the water interlayer. The DND detector assembly consisted of 26  $\text{BF}_3$  proportional counters and three fission detectors in a Benelux-moderated and lead-shielded assembly, as shown in Fig. 7, that covered an approximate  $90^\circ$  sector external to the loop.<sup>10</sup> Also shown in Fig. 7 is the DNM detector assembly. A 127-mm-thick lead shield was used in the DND to limit the high gamma radiation ( $^{24}\text{Na}$ ) field. The shield was necessary for two reasons. The activity of the sodium was such that the

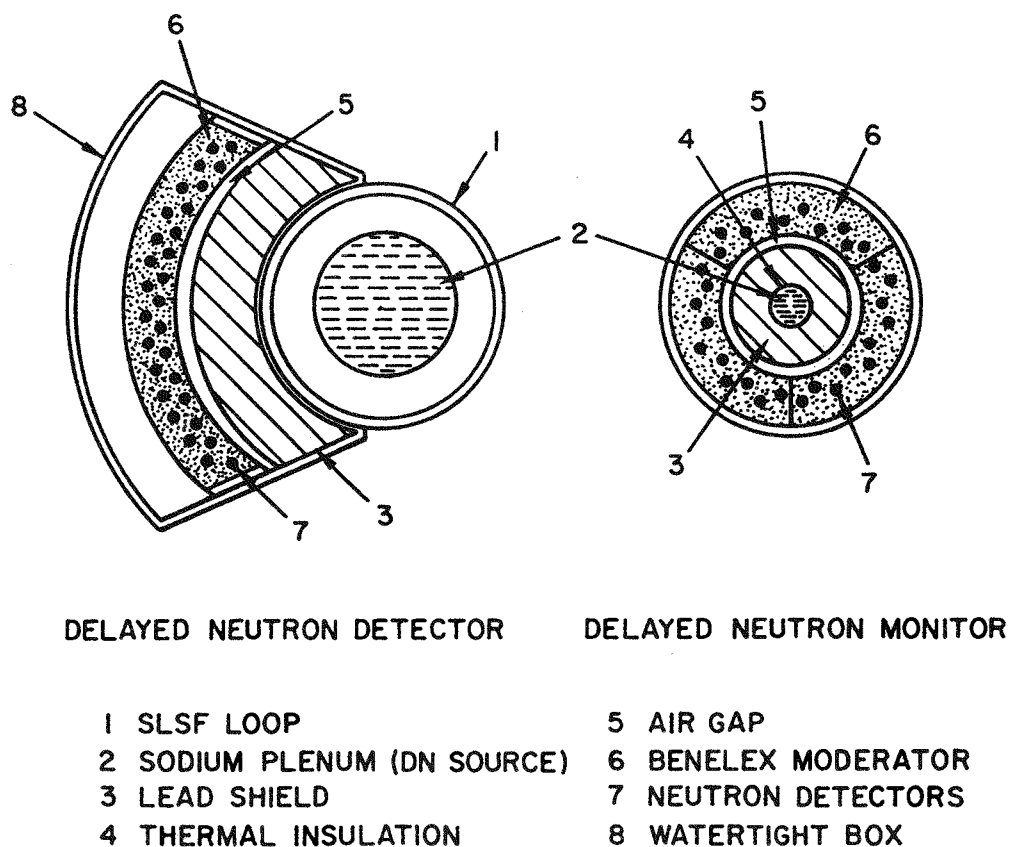


Fig. 7. DND and DNM Detector Assemblies

sodium plenum was a ~2000 Ci source of  $^{24}\text{Na}$ . It was necessary, therefore, to reduce the  $\gamma$ -dose to the detectors themselves to ~10-20 R/hr so that they could operate reliably, and so that the degradation due to radiation damage would be acceptably slow. In addition, the 2.75-Mev  $^{24}\text{Na}$  gamma ray produced a background of neutrons by the  $(\gamma, n)$  reactor on hydrogen in the moderator. Monte Carlo calculations<sup>11</sup> predicted that the background from this source would be tolerable with the shield thickness chosen. Photoneutrons from the reactor water were minimized by shadow shielding and by careful placement of the DND enclosure against the loop surface to reduce the thickness of the water layer below 125 mm, because this region could not be shielded.

Transport time from the fuel bundle to the DND was ~2 s. Nearly all of the strong DN groups could be observed on the first pass; on second and subsequent passes of sodium around the loop only the longer-lived DN activity from the initial exposure or release would be observed; in particular,  $^{87}\text{Br}$  ( $t_{1/2} = 55.7$  s) could be followed for many minutes.

The DND was designed to detect both small failures and major events. The fission counters were most suitable to detect large-scale failures and the  $\text{BF}_3$  proportional counters for threshold and small-scale failures. To provide a wide range of sensitivity, the detectors were arranged, singly and in groups, in six recording channels. Particular fission detectors were of very low efficiency to accommodate, without limitation, the DN rates from major failures. Output from detectors operated in the pulse mode was recorded each second by the OLCS computer and periodically transferred to magnetic tape. One fission detector could be operated in the current mode and output recorded on high-speed digital memory to provide a finer time resolution of the DN signal. In addition, current-mode operation avoided dead-time problems which cause blocking of digital recording systems at extremely high count rates.

The DND was operated during SLSF experiment W2 and the response to the fission product source and the background effects were measured during the period prior to the transient. It was found that the background due to photoneutrons generated by  $^{24}\text{Na}$  corresponded to  $<0.2$  mm<sup>2</sup> of  $\text{U}^{235}$  recoil source equivalent, while the background due to the operation of the reactor itself (presumably leakage up the loop channels) was ~0.6 mm<sup>2</sup> recoil equivalent. The total background effects were less than 1 mm<sup>2</sup> recoil equivalent and therefore completely negligible in the case of the major effects seen in P4.

DND detector efficiencies were established by comparing the measured DN count rates in the loop sodium plenum resulting from steady-state irradiation of the fission product source with the corresponding DN activity calculated using a simulation model described in Ch. VI. Efficiencies for the most sensitive and least sensitive fission detectors were  $1.9(10^{-6})$  and  $1.5(10^{-7})$  counts per delayed neutron emitted in the sodium plenum volume, respectively, and  $9(10^{-5})$  for a  $\text{BF}_3$  detector.

### 3. On-Line Sodium Sampling System

The OLSS was a closed-loop sodium system, external to the in-reactor loop and the ETR, that continuously circulated a sodium sample stream from the loop sodium plenum through a 0.5-liter sample volume and back to the loop. The DNM in the OLSS monitored this sample volume for DN activity from failed fuel. The transport time to the DNM could be varied from 12 s to ~40 s. The OLSS/DNM observed only the longer-lived DN activity and was more prototypic of a reactor installation than the DND.

The main components and piping of the OLSS and the location of both the DNM and the DND are shown in Fig. 8. Most of the OLSS was contained under the ETR biological shielding within a skid enclosure housing the sodium pump, valves, sodium hold tank and expansion reservoir, vapor trap, oxide control equipment, instrumentation, and heated piping, as illustrated in Fig. 9. The DNM was located outside and adjacent to the skid within a shield block. A regenerative heat exchanger and air cooler connecting the loop to the OLSS skid reduced the temperature of sodium from the loop to 755 K (900°F) before it entered the skid.

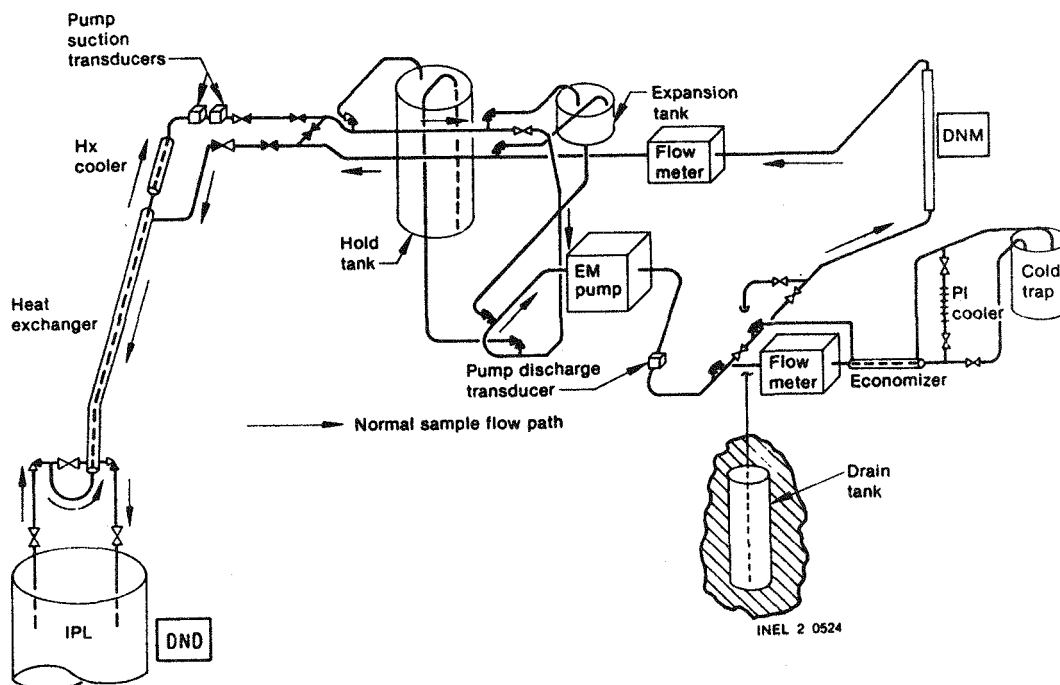
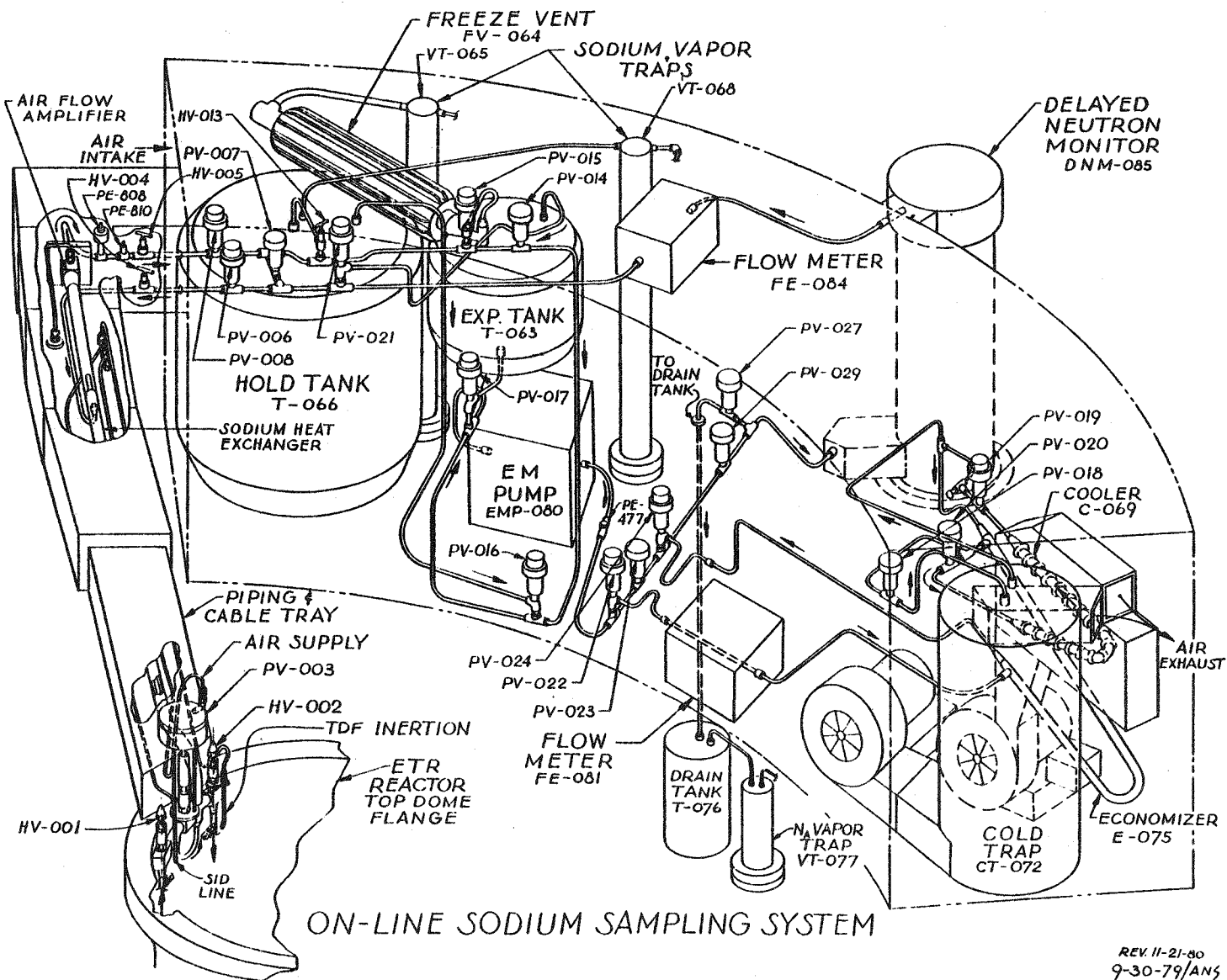


Fig. 8. Piping System and Major Components of the OLSS

The detector assemblies of both the DNM and the DND are shown in Fig. 7. The DNM detector assembly consisted of 27  $\text{BF}_3$  proportional counters and three fission detectors in a Benelux moderator. Both lead and tungsten alloy were used as shielding against the  $^{24}\text{Na}$  gamma ray background, for the same reason as in the DND. A wide range of sensitivity to fuel failure was provided in the DNM by utilizing the same channel arrangement and operating modes as employed in the DND.

The response of the DND and DNM was strongly dependent on the details of the loop flow and on the nature of the failure (see Ch. VI). Briefly, the input of DN precursors into the sodium would be continuous for a cladding failure that exposed solid fuel which remains in place. For a mass of molten fuel ejected into the sodium stream, the input is a single spike. The DND can see the onset of the signal after the transport delay of two s, the DNM only after a longer transport delay, 12 s or more. The DND signal is complicated by sodium circulation around the loop. For a DN spike due to molten fuel release, subsequent DN peaks would be seen at 10-s intervals (the sodium circulation time). This activity would decay, however, with the half-lives of the individual longer-lived DN groups. For a steady-state release, an equilibrium activity level would be approached in steps of decreasing amplitude at the sodium circulation interval.





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Fig. 9. On-Line Sodium Sampling System

### III. DESIGN OPERATING CONDITIONS

Careful consideration was given to the target molten fuel release and the fuel pin operating conditions in developing the bounds for the "worse-than-worst-case" nature of the P4 experiment. It was essential that the fuel canisters maintain integrity during low-power operation but generate and eject molten fuel into the fuel bundle at or near full-power conditions. The design goal for molten fuel release from a fuel canister was 10 to 30 grams, an order of magnitude larger than the minimum needed as an optimally located, short sleeve to cause cladding damage and fuel melting in a pin<sup>12</sup> or as fuel bridged between pins to cause sodium temperatures to exceed 1600°F.<sup>13</sup> This, coupled with the use of an ETR power ramp simulating a 5¢/s FTR reactivity insertion accident, used in the prior SLSF experiment W2, dictated the 56-g fuel inventory in a cylindrical canister. Even after extensive analysis and out-of-pile testing, the capability of fuel canisters to satisfy the experiment objectives could not be fully assured. To increase the potential for molten fuel release, both cylindrical and fluted canisters, predicted to fail by different mechanisms, were incorporated into the P4 fuel bundle.

The P4 design operating condition was a nominal bundle power of 1272 kW (1240 kW from 37 pins, plus 32 kW from the fuel canisters) and test sub-assembly coolant flow rate of 3.38 kg/s (7.45 lb/s) at an ETR power level of 156 MW. This operating condition simulated the nominal maximum FTR fuel pin linear power, but at a coolant flow rate 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 (20 to 21 ft/s) in the central coolant flow subchannels and resulted in a frictional pressure drop of 240 kPa (~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 (17.6 lb/s) to maintain the steady-state inlet temperature to the heat exchanger <839 K (1050°F). Bundle inlet temperature was 695 K (792°F), the inlet temperature for the FTR rated core. Corresponding temperatures around the loop at design operating conditions (no partial flow blockage in the fuel bundle) are shown in Fig. 10. Loop cover-gas pressure during most of the P4 irradiation was 160 kPa (22.8 psia) to avoid cavitation in the OLSS pump. Immediately prior to the power transient, the cover-gas pressure was reduced to 69 kPa (10 psia) to simulate LMFBR pressure levels in the P4 fuel bundle.

The power transient to initiate molten fuel release consisted of an ETR power increase from 40 to 175 MW, on an 18.8-s period, followed by a 5-s hold at 175 MW, a 5-s ramp down to 156 MW and continued steady operation. This was to be accompanied by an increase from <900 kg/s to 2600 kg/s in the helium flow to the loop heat exchanger. Because the power increase was rapid, a four-fold increase in less than 30 s, the bundle inlet temperature was predicted to increase by about 45 K before the heat exchange system reestablished steady state conditions ~200 s later. Both the ETR power ramp beyond 156 MW and the bundle inlet temperature overshoot contributed to the bounding nature of the P4 operating envelope, in addition to the undercooling of the fuel bundle and the high coolant inlet temperature.

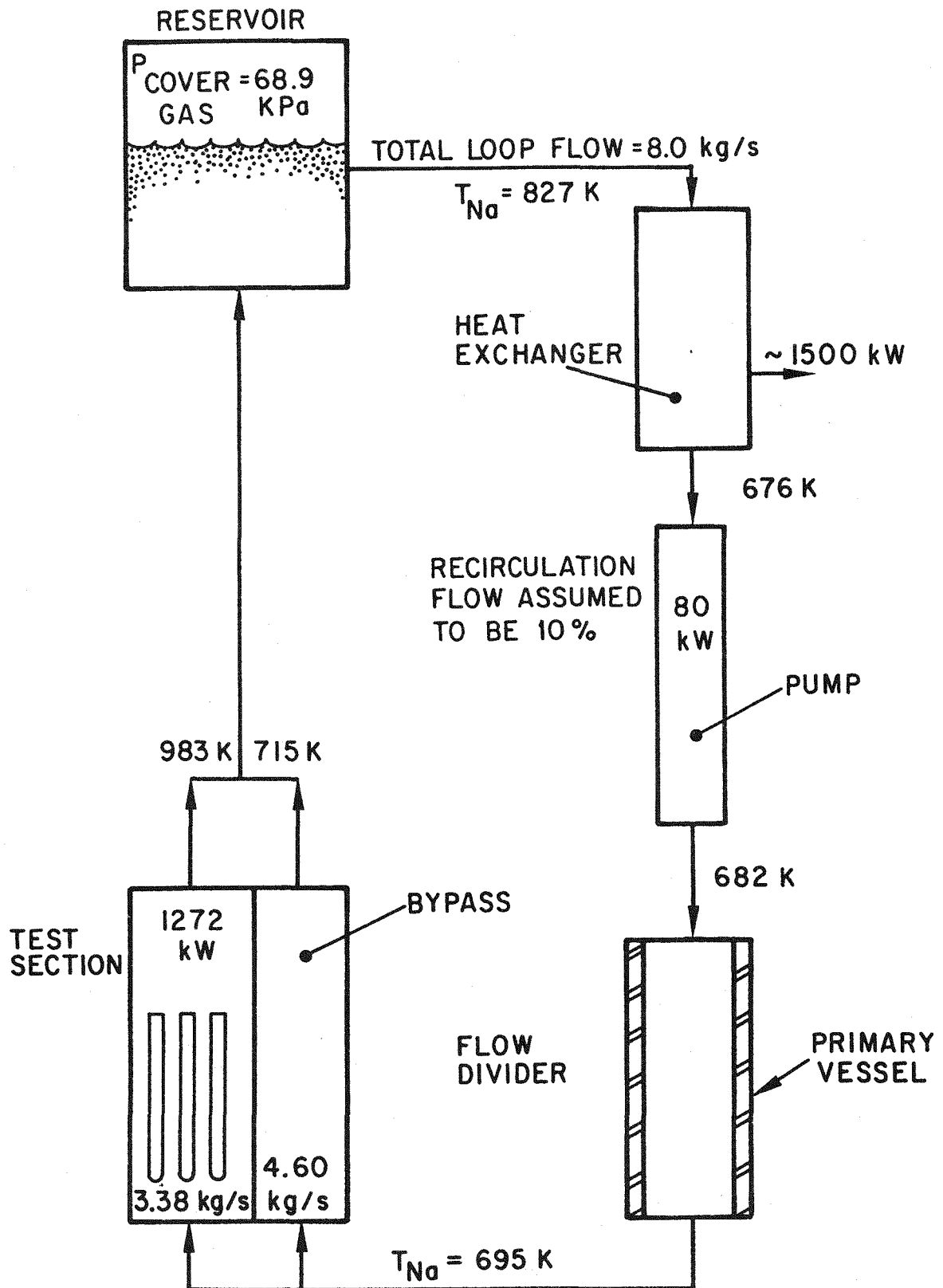


Fig. 10. Design Operating Conditions in the P4 Loop

P4 irradiation testing was planned to cover five 10-day irradiation cycles and reach a fuel burnup equivalent to 30 full-power days of ETR operation at 156 MW.<sup>14</sup> During the first 10-day cycle, ETR power was limited to 40 MW to avoid failing the fuel canisters while building up fission products in the initially fresh fuel for subsequent failure detection. After a 24-hour irradiation at 40 MW ETR power in Cycle two to saturate short half-life fission product activity, full reactor power would be reached with the planned power transient. Molten fuel would be released into the fuel bundle from failure of one or more of the fuel canisters during the power transient. Power operation would continue to the end of Cycle two following canister failure and fuel release into the bundle. Cycle three was to begin with a cautious approach to full power to determine DN characteristics from failed, sodium-logged fuel during startup. After steady power operation during most of Cycle three, fuel and DN response would be studied during an ETR scram and rapid return to power. Cycles four and five included short-term variations in both power and flow to yield operating conditions more severe than during previous steady-state operation and to complete irradiation to 30 full-power days.

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#### IV. EXPERIMENT PERFORMANCE

##### A. Prefailure Operations

Experiment P4 irradiation began on August 8, 1981, and continued through the first 10-day cycle at a maximum ETR power of 40 MW to prevent premature failure of the fluted fuel canister. In addition to building up a fission product inventory in the fuel pins and fuel canisters for failure detection purposes, several tests were performed. Data were collected during a pre-failure benchmark test at a fuel-bundle power of 326 kW (nominal ETR power of 40 MW) for bundle and sensor evaluation by comparison with postfailure results. A low-power practice transient was performed from 7 to 28 MW to verify the operation of the ETR power transient controller (PTC). The P4 Cycle-one irradiation, completed on August 15, 1981, was equivalent to 1.44 full power days.

Although the neutron level varied through the correct range during the low-power practice transient, the power ratio was only 3.1 instead of the desired 3.9. The low power ratio was attributed to an unequilibrium decay heat level during this test of the PTC.

P4 low-power operation resumed on August 19, 1981. The two days of operation prior to the P4 planned power transient was also limited to a maximum ETR power level of 40 MW and restored equilibrium levels of short half-life fission product activity for failure detection purposes. A second low-power practice transient was performed. The ratio of final power to initial power was improved over that observed during the first low-power practice transient but was still slightly lower than planned. To compensate, the power transient was initiated with the P4 fuel-bundle power on the high side of the reference 326 kW.

##### B. Planned Power Transient

###### 1. Summary of Events

The P4 power transient was initiated at 0644 MDT August 21, 1981, at a bundle power of 338 KW (40 MW ETR power) after the loop plenum-gas pressure was reduced to 69 kPa (10 psia). Initial conditions at the time of transient initiation are shown in Table 6. ETR power increased under the control of the preprogrammed PTC; helium circulator speed was controlled manually. The first disturbances in steady temperature, pressure and flow conditions, observed at 15.3 s, were attributed to initial failure of the fluted fuel canister. Single flow impulses of -0.65 kg/s and +0.3 kg/s, respectively, were measured at the bundle inlet and outlet. The fluted canister failure was also sensed by the inlet and outlet pressure sensors and by both acoustic detectors. The delayed neutron level "bumped" upward following initial failure of the fluted canister as the area of fuel exposed to the sodium increased. Signals of the bundle-outlet flowmeter, and later the combined-flow flowmeters, became noisy. There was no evidence of gross molten-fuel release, molten fuel-coolant interaction (MFCI), or flow blockage. Temperature perturbations or offsets of up to 20 K were observed.

Table 6. Initial Conditions Prior to the Planned Power Transient

|                           |      |
|---------------------------|------|
| ETR power, MW             | 40   |
| Fuel-bundle power, kW     | 338  |
| Coolant flow rate, kg/s   |      |
| - bundle inlet (FE 1-3)   | 3.45 |
| - bundle outlet (FE 2-1)* | 3.23 |
| - total loop              | 7.90 |
| Coolant temperature, K    |      |
| - bundle inlet            | 693  |
| - bundle outlet           | 780  |
| Loop plenum pressure, kPa | 69   |
| Pump power, kW            | 75   |
| Pump voltage, volts       | 274  |

\*Indicated fuel-bundle outlet flow rate was lower than the inlet flow rate in P4 and previous SLSF experiments, even though all flowmeters were calibrated prior to installation in the test train.

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 fuel bundle. Significant local temperature perturbations occurred downstream of the fuel canisters. Although failure had not occurred yet, strain in the cylindrical canisters distorted the bundle geometry and perturbed the local coolant flow.

Molten-fuel release occurred at ~34 s (170 MW ETR power). It was accompanied by perturbations in inlet and outlet flow, local coolant temperature levels, acoustic noise, inlet and outlet pressure, and by a sharp increase in DN level. The inlet flow rate decelerated to a minimum 2.4 kg/s and returned to 3.2 kg/s, or 93% of the initial flow rate. About half of the fuel bundle thermocouples indicated a response to the molten-fuel release, the others indicated little change. Thermocouple TE 3-14 on Pin 14 (adjacent to the cylindrical canister on Pin 5) indicated a temperature jump of 600 K. The response of thermocouple TE 3-14, whose junction was located 363 mm below the

fuel midplane, was typical of molten fuel contact and formation of a new junction. Thermocouples TE 6-1 and TE 6-2, whose junctions were located 145 mm above the fuel midplane on hex-can Flats 1 and 2, respectively, also were hit by molten fuel. Thermocouple TE 3-7, junction 94 mm 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 fuel bundle. This indicated that the molten-fuel release was from the cylindrical fuel canister on Pin 5 and was directed toward the outer-row pins and the hex-can. The minimum axial distance traveled by the molten fuel in reaching thermocouples TE 6-1 and 6-2 was ~100 mm. 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.

The rate of average power increase in the P4 loop exceeded 2.4 MW/-minute during the 27.75-s interval in which the ETR power increased from 40 MW to 175 MW. During this power transient, the helium flow to the SLSF heat exchanger was to be rapidly increased, within the operating limits of the circulators, to the full-power level. This operating procedure was predicted to increase the fuel-bundle inlet temperature to a maximum of 742 K at 70 s, followed by a leveling off and recovery to the design 695 K value.<sup>15</sup> However, an operational error in the manual control of the helium secondary coolant system caused a reduction in the total helium flow and prolonged the inlet temperature increase. 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 initial molten fuel release, and then slowly increased with the increasing inlet temperature. The fuel-bundle flow rate remained steady at 3.2 kg/s until 53 to 54 s, when the flow rate increased to 3.35 kg/s (97% of the initial flow rate). The flow increase was preceded by a 40 K drop in temperature at thermocouple TE 3-14 (Pin 14), beginning at 50 s, and was accompanied by a pause in the gradual temperature increase exhibited by thermocouples in the fuel bundle and at the bundle outlet. 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. Thermocouple TE 3-2 (Pin 3) failed at 80 s and temperature offsets were observed in a number of fuel bundle thermocouples, similar to the events a few seconds prior to the first cylindrical canister failure.

At 86 s, the fluted canister on Pin 12 ruptured, releasing molten fuel toward the center of the bundle. The cladding of the cylindrical canister on Pin 8 was locally melted in contact with this molten fuel, but the canister remained relatively intact (see Ch. V). Again, about half the fuel bundle 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 appeared to indicate steady sodium boiling thereafter. Thermocouple TE 6-2 also indicated sodium boiling. Molten-fuel release was accompanied by twin flow perturbations, 20 ms apart,



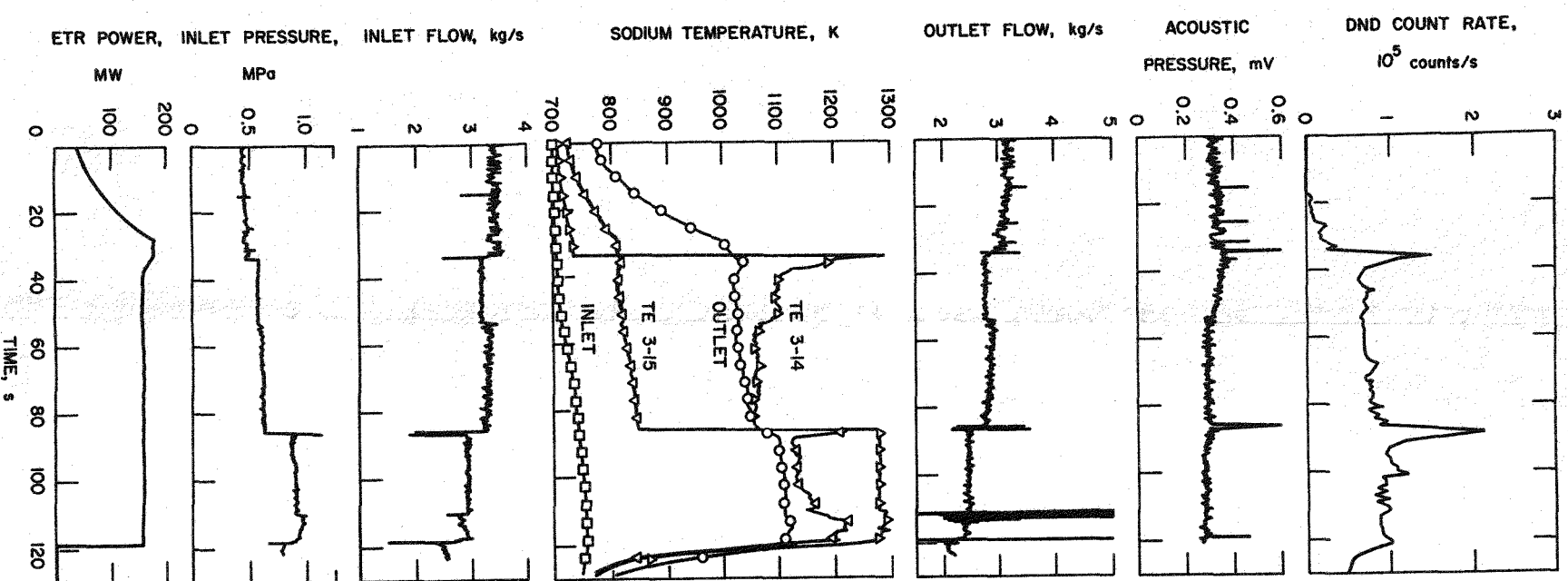


Fig. 11. Instrument Response During the P4 Power Transient

that reduced inlet flow to ~2.0 kg/s. The inlet flow rate recovered to 3.0 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. The flow rate then recovered to 2.95 kg/s (86% of the initial value). Two broadband noise peaks were indicated by the acoustic sensors. The fuel-bundle outlet temperature increased 40 K, from 1065 K to 1105 K. The inlet temperature at 86 seconds was 739 K. The DND signal peaked at 88 s.

Operation continued at a steady fuel-bundle flow rate of 2.95 kg/s until 110 s. Temperature increases of 30 to 60 K were indicated by thermocouples TE 3-2 (Pin 3), TE 3-7 (Pin 6), and TE 3-15 (Pin 18), beginning at 104 s. At 110 s, a large gas release occurred. [Because there had not been evidence of gas release earlier in the power transient, the experimenters believed this gas release was from fuel pins. It was not until neutron radiographs of the P4 test section were examined that a small failure of the hex-can was discovered.] The gas release at 110 s was through a failure in the hex-can where molten fuel released at 34 s had adhered. The gas released into the fuel bundle perturbed the coolant flow and caused large oscillations in the indicated flow rate through the bundle-outlet and combined-flow flowmeters. Fuel bundle temperatures increased while the released gas was being swept upward then began to return toward earlier values as the fuel bundle flow rate recovered to 2.95 kg/s.

At 117 s, the fuel-bundle flow rate began to drift lower. This coincided with a decline in fuel-bundle inlet pressure (PE 2-1). Thermocouples TE 3-1, 3-3, 3-10, 3-12, 4-3, 6-3, and 6-5 indicated increasing coolant temperatures beginning at 117 s, after reaching a maximum recovery from effects of the gas release. ETR scrambled at 118.4 s on a PPS PF-H signal that the time-averaged, low-flow setpoint of 80.3% had been reached. At scram, the fuel-bundle inlet temperature had risen to 750 K. One channel in the two-of-four logic for PPS PF-D (high sodium temperature in the meltdown cup) had tripped prior to scram and a scram from PPS PF-D was only seconds away. At scram, the pump was transferred to an emergency power supply. During this power-off transfer, loop and test-section flow dropped and the decrease in fuel-bundle pressure caused a second release of gas from the insulator region. Following ETR scram, the pump was transferred back to its regular power supply. The fuel-bundle flow rate at the pump benchmark voltage returned to its previous 2.95 kg/s (86% of initial flow rate) level. Because the fuel-bundle coolant flow rate was below the safety analysis and administrative operational limit of 90% initial flow for steady-state power operation, a restart for full-power operation was delayed until an expanded safety envelope was approved.

## 2. Sensor Response and Evaluation

A summary of the instrument response during the 118.4-s interval from initiation of the power transient to reactor scram is shown in Fig. 11. A detailed chronology of the instrument response is summarized in Table 7. The response of P4 test train instrumentation is illustrated and evaluated in the following pages. Instrument-response data are contained in fiche form in an envelope inside the back cover of this report.

Table 7. Chronology of Instrument Response During the P4 Power Transient

| Time<br>s    | Event and Instrument Response                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                            |
|--------------|----------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------|
| 0            | PTC initiated the power transient                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                        |
| 15.2 to 15.6 | <p>Fluted canister cladding failed in the cusp regions against adjacent pins. Fuel-bundle inlet flow (FE 1-3, FE 1-4) spiked down from 3.45 kg/s to 3.10 kg/s, outlet flow (FE 2-1) spiked up from 3.15 kg/s to 3.55 kg/s, and total flow (FE 3-2) spiked up to 8.5 kg/s from 7.9 kg/s.</p> <p>Fluctuations were observed in the bundle inlet (PE 2-1) and outlet (PE 4-1) pressure. An acoustic event was recorded by both the bypass (PE 9-1) and total flow (PE 9-2) acoustic sensors. Small temperature perturbations (10 to 20 K) were seen in the signals of thermocouples TE 3-10 and TE 4-3.</p> |
| 17-17.2      | The bundle outlet flowmeter (FE 2-1) signal became noisy.                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                |
| 17.2-17.5    | Total flowmeter (FE 3-1, FE 3-2) signals became noisy.                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                   |
| 17.5-18      | A sharp increase in DND signal resulted from the fluted canister failure.                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                |
| 25.5-25.8    | An increase in acoustic level at PE 9-1 was accompanied by small temperature perturbations on several thermocouples. Pressure spikes were measured at both the bundle inlet and outlet.                                                                                                                                                                                                                                                                                                                                                                                                                  |
| 26.6-26.7    | The DND signal had a small pulse on top of a general increase.                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                           |
| 27.8         | ETR power reached 175 MW and the start of a 5-s power hold.                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                              |
| 30-33        | Fuel-bundle inlet flow spiked down from 3.6 kg/s to 3.4 kg/s and exit flow spiked both up and down. An acoustic pulse was seen by PE 9-1; pressure spikes were measured at both the bundle inlet and outlet. Wire-wrap thermocouples on Pins 6, 7, and 8 showed temperature jumps and plateaus of up to 100 K, due to distortion of the bundle geometry by the strain in the cylindrical fuel canisters, followed by partial recovery.                                                                                                                                                                   |
| 32.8         | ETR-power downramp from 175 MW to 156 MW began.                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                          |
| 34           | The cylindrical fuel canister on Pin 5 failed and released molten fuel toward the hex-can. Two small downspikes in inlet flow, to 3.1 kg/s, then 3.0 kg/s, preceding the fuel release were followed by large inlet flow downspikes, to 2.5 kg/s, then 2.4 kg/s. Net inlet flow dropped from 3.45 kg/s to 3.2 kg/s. Pairs of up-spikes were observed in the outlet flow. Large acoustic pulses were detected by acoustic detectors PE 9-1 and PE 9-2. A bundle inlet pressure                                                                                                                             |

increase accompanied the drop in fuel bundle flow due to the partial flow channel blockage. Thermocouple TE 6-1 was hit by molten fuel and failed. TE 3-14 also was hit by molten fuel and formed a new junction, registering a 580 K increase. Molten fuel near the junction of TE 6-2 produced a 330 K jump in temperature.

- 34-37 Thermocouples TE 3-6, TE 3-7, TE 3-8, and TE 3-12 indicated temperature spikes of 50 to 130 K followed by a partial or complete recovery from these temperature offsets over 3 to 4 s. An acoustic noise spike was recorded by PE 9-1. The DND signal increased by nearly an order of magnitude in response to the molten fuel release.
- 38-53 Test section temperatures stabilized, reaching a minimum at 42 s followed by a gradual increase with increasing inlet temperature.
- 53-54 Inlet flow, steady at 3.2 kg/s since the molten fuel release at 34 s, increased to 3.35 kg/s and thermocouples TE 4-3 and TE 4-5 measured a slight drop in temperature.
- 54-79 Inlet flow gradually drifted lower, losing 0.05 kg/s in 25 s. Coolant temperature rose slowly in the fuel bundle, due to the gradual flow reduction and the increasing inlet temperature.
- 80 Thermocouple TE 3-2 on Pin 3, adjacent to the fluted canister, failed.
- 80-86 The downward drift in inlet flow increased. Thermocouples TE 3-1, TE 3-3, TE 3-7, and TE 3-10 measured temperature increases of 20 to 45 K.
- 86 The fluted fuel canister on Pin 12 ruptured and released molten fuel toward the center of the fuel bundle. Two acoustic noise peaks were accompanied by flow and temperature perturbations, reductions in fuel bundle and loop flow, and an increase in inlet pressure. Inlet flow spiked downward to 2.0 kg/s, then to 1.9 kg/s 20 ms later, and recovered to 3.0 kg/s. Outlet flow exhibited two up-spikes in flow of 0.75 to 1 kg/s 20 ms apart. About 0.45 s later, another flow perturbation occurred. Inlet flow spiked down to 1.3 kg/s and recovered to 2.95 kg/s. Total loop flow dropped from 7.6 kg/s to 7.4 kg/s.
- 86-89 Major changes in temperature occurred in the center of the bundle due to the increase in extent of the partial flow channel blockage. Thermocouple TE 3-15 indicated a temperature jump of 380 K followed by a persisting temperature increase of 275 K. Temperature "plateaus" of 220 K to 250 K were indicated by TE 3-7 and TE 3-14. TE 3-14 exhibited continuous boiling at 1280 K. Thermocouple TE 6-2 spiked, followed by an indication of boiling at 1322 K. The

DND signal began to increase at 86 s and peaked at 88 s. Thermocouples TE 3-1, TE 3-3, TE 3-6, TE 3-10, TE 3-12, and TE 6-3 showed temperature jumps of 80 to 120 K, and temperature "plateaus" of up to 80 K. At the bundle periphery, thermocouples TE 4-1, TE 4-3, TE 4-5, and TE 4-9 showed upward temperature offsets of 15 to 25 K. The bundle outlet temperature increased by 40 K.

- 89-110 Inlet flow remained steady at 2.95 kg/s and the DND signals decreased to an "exposed fuel" level. The DNM (OLSS) signals peaked at 108 s from the 34-s fuel release event. TE 3-14 continued to indicate boiling at 1280 K. Thermocouple TE 3-2 indicated a temperature increase of 100 K, from 1150 K to 1250 K, between 90 and 110 s; the temperature at TE 3-7 increased by 40 K, starting at 104 s.
- 110 The hex-can failed and argon gas entered the fuel bundle from the insulator region.
- 110-117 Inlet flow spiked down to 2.55 kg/s, stabilized at 2.8 kg/s during the gas release, and returned to the pre-gas-release level. Outlet flow increased by 0.6 kg/s followed by large oscillations in the flow signal indicating gas passage beginning at 110.6 s. Gas reached the combined-flow flowmeters at 110.9 s. The gas release caused a temporary increase in acoustic noise, bundle inlet pressure, and fuel-bundle local coolant temperatures. Thermocouples TE 3-3 and TE 3-12 indicated maximum temperature excursions of 100 K due to the gas release. TE 3-1 and TE 3-15 were perturbed by ~70 K; TE 2-3 and TE 3-10 by 30 to 40 K. Most bundle temperatures recovered to near their 110-s levels. The loop plenum pressure increased sharply from 78 kPa to 88 kPa.
- 117-118.4 Bundle inlet pressure and fuel bundle flow began to drift lower and fuel bundle temperatures began to increase. ETR scrambled at 118.4 s on a low-flow (PPS PF-H) trip.
- 118.4-122 At reactor scram, the pump was transferred to emergency power. The fuel-bundle coolant flow rate dropped from 2.75 kg/s to 0.7 kg/s during the 40-ms interval that pump power was off during the transfer. System pressure was reduced during the transfer and additional gas entered the fuel bundle. Gas passage persisted for about 3 s, as it had at 110 s.

a. ETR neutron level

The measured ETR neutron level during the P4 power transient, shown in Fig. 12, closely followed the planned power increase. The only deviation from the idealized neutron level profile was a 2% to 3% overshoot at 27 s, which had been expected. A very small ringing (<1%) in the steady-state neutron level gradually damped with time. This ringing had been also observed in the low-power transient and was well within the tolerances for the experiment.

b. Helium flow rate through the SLSF heat exchanger

During power operation, the sodium temperature was controlled by adjusting the fraction of helium flow routed to the loop heat exchanger via a three-way valve in the secondary coolant system. The speed of the helium circulators was under manual control to keep the high-speed gas-bearing circulators within their safe operating envelope; the three-way valve was operated in the automatic mode.

At the start of the power transient, the total flow rate in the helium system was 2420 kg/h (5330 lb/h), with 865 kg/h (1900 lb/h) passing through the loop heat exchanger. For the P4 power transient, an increase in helium system flow rate to ~2860 kg/h (6300 lb/h) was planned. The planned increase in helium flow rate through the loop heat exchanger and the predicted response of the fuel-bundle inlet and outlet sodium temperatures are shown in Fig. 13.

An operational error in the manual control of the helium circulator speed caused a sharp decrease in the total helium flow rate in the secondary coolant system during the ETR power increase, as illustrated in Fig. 13. The three-way valve opened to direct nearly all the helium flow through the loop heat exchanger. This was followed by a gradual increase in total secondary-system flow rate, but the target loop helium flow rate was not reached.

c. Inlet and outlet temperature

The fuel-bundle inlet and outlet temperatures were in good agreement with predictions up to ~80 s. As a result of the deficiency in loop cooling during the power transient, however, the predicted increase in fuel-bundle inlet temperature to 742 K and recovery to the initial 695 K was not realized. The fuel-bundle inlet temperature continued to increase during the power transient and reached 750 K (890°F) at scram.

The general increase in measured fuel-bundle inlet and outlet temperature is shown in Fig. 13. A perturbation in outlet temperature resulting from the 34-s molten fuel release and partial flow blockage, the slight increase in bundle flow at 53 s, a 40 K plateau in temperature at 86 s resulting from the second molten fuel release and further areal blockage, and a 15 K increase and recovery from gas release at 110 s can be seen superimposed on a general, increasing bundle-outlet temperature due to the increase in inlet temperature.

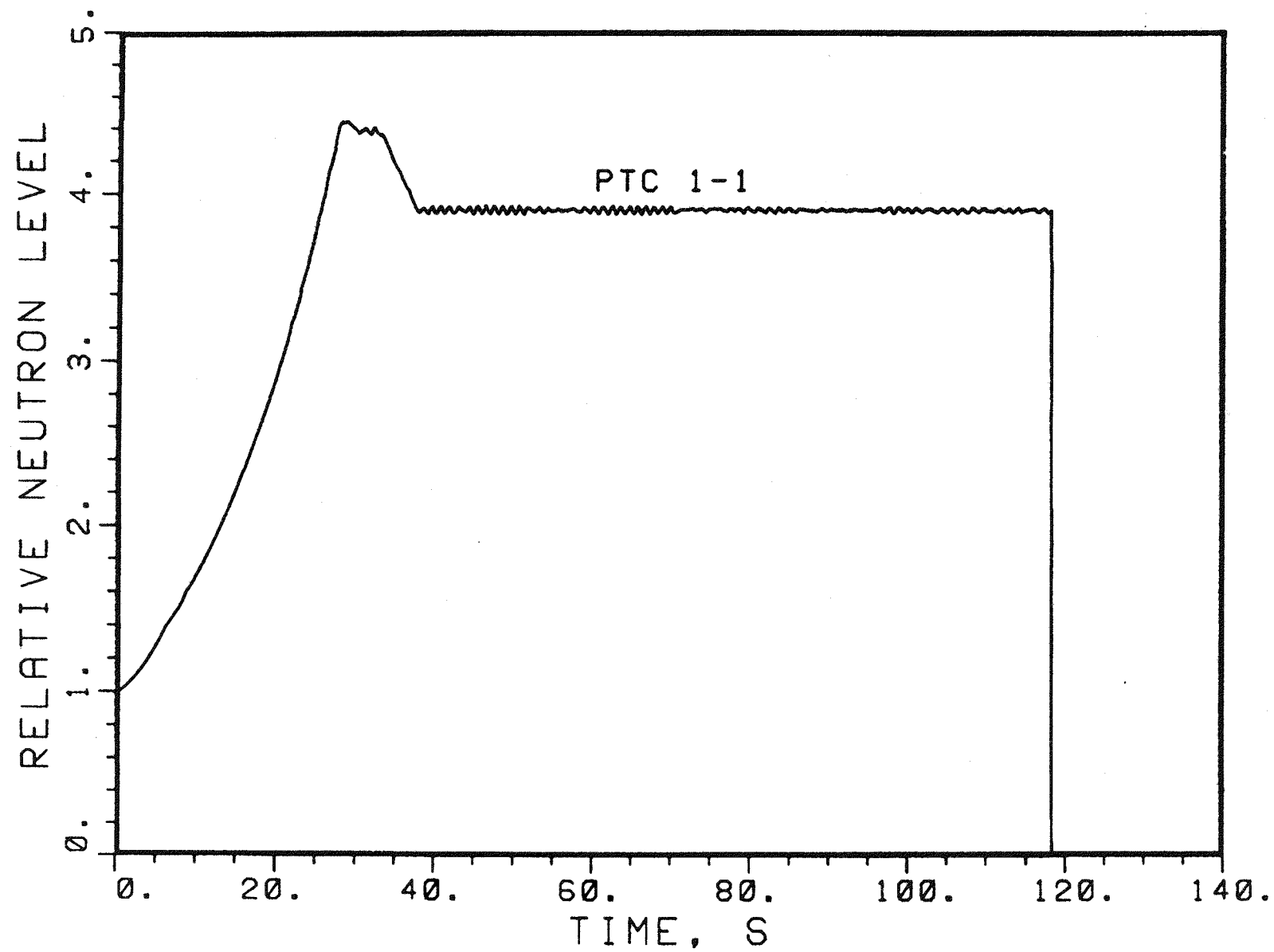


Fig. 12. ETR Neutron Level During the Power Transient

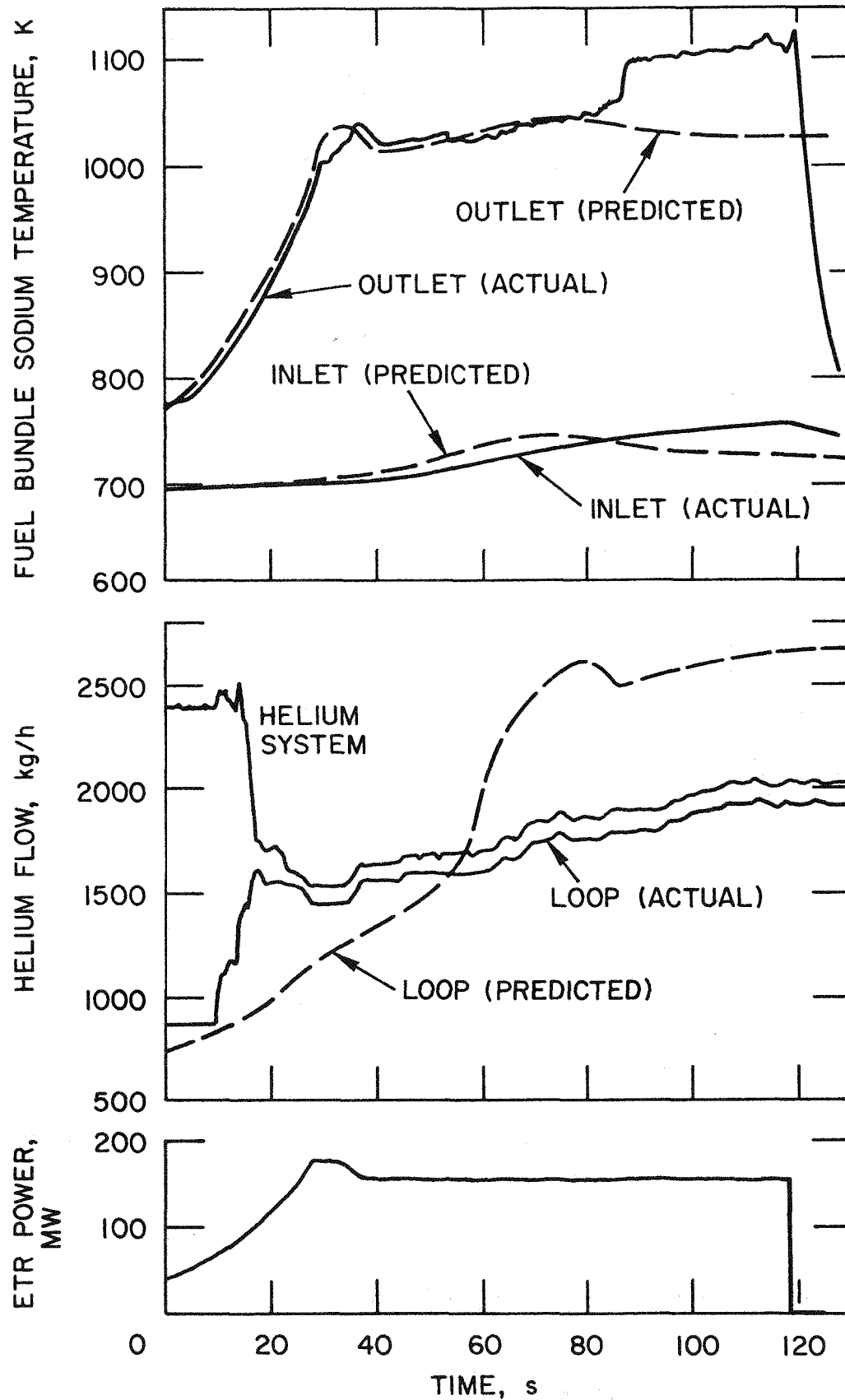


Fig. 13. Helium Flow; Fuel-Bundle Inlet and Outlet Temperatures

Sodium temperatures were elevated around the loop. The increased sodium temperature reduced performance of the pump, which was operating at a fixed voltage through variable transformers from line power, and also reduced the non-temperature-compensated signals from the inlet flowmeters to the low-flow PPS PF-H channel. Both the real reduction in loop flow and the apparent further reduction in inlet flow signal resulting from the increasing sodium temperature contributed to the ETR scram at 118.4 s.

d. Temperature within the fuel bundle and hex-can

As shipped, the P4 test train contained 30 active thermocouples to monitor the coolant temperatures at various elevations and radial positions in and around the fuel bundle. Five thermocouples, TE 3-5, TE 3-13, TE 4-4, TE 4-7, and TE 6-6, were known to have failed prior to initiation of the power transient. The number of thermocouples that provided information on local effects within the bundle can be further reduced. Five thermocouples at elevations above the top of the fuel columns, TE 3-11, TE 4-1, TE 4-2, TE 4-5, and TE 4-9, indicated 20 to 25 K 'plateau' temperature increases at 86 s and 10 to 20 K temperatures increases at 110 s, then recovery. This response parallels that of the bundle outlet thermocouples. Thermocouples TE 4-6 and TE 4-8 had junctions located below the fuel columns and responded like bundle inlet thermocouples. Thermocouple TE 6-5, located 12.5 cm (5.7 in) above the midplane of the test-pin fuel columns, also was relatively unaffected by the partial flow blockages or the gas release.

The response of 16 fuel bundle and hex-can wall thermocouples that observed local events during the power transient is summarized in Table 8. Thermocouple responses are also shown in Fig. 14. Thermocouple TE 6-3, located 145 mm above the fuel midplane, measured temperatures equal to or higher than the bundle outlet temperature throughout the power transient, as shown in Fig. 14. Addition of banding straps around the pins adjacent to the fluted canister caused an interference fit of the fuel bundle within the hex-can at the canister region. The local interference fit may have pushed fuel pins close to hex-can Flat 3. Thermocouple TE 6-3 was the only thermocouple to consistently indicate a high local coolant temperature throughout both the steady state and power transient operation.

Table 8 indicates little thermocouple response to the local melting of the fluted canister cladding beginning at 15.3 s. Between 24 and 29 s into the transient, temperature spikes ranging from 5 K to 30 K were measured by thermocouples TE 3-6, TE 3-7, TE 3-8, TE 3-9, and TE 3-10. Thermocouple TE 6-5 measured a 30 K temperature spike at 26 s and the junction of TE 3-4 opened at 29 s. Between 31 and 34 s, there was evidence of a growing bundle distortion, accommodated by fuel pins being moved out of their equilibrium positions by strain in the fuel canisters, which resulted in changes in local-flow-channel geometry. Significant temperature perturbations accompanied the bundle distortion three to four seconds prior to the first molten fuel release. This distortion appeared to be centered in the area between the two cylindrical canisters, but was also observed by thermocouple TE 6-3 on Flat 3. Temperature perturbations resulting from this distortion disappeared, as shown in Fig. 14 for thermocouples TE 3-7 and TE 3-10, when the cylindrical fuel canister on Pin 5 failed. Thermocouples TE 3-14 (Pin 14) and TE 6-1 (Flat 1) when hit by molten fuel released toward the outer rows of



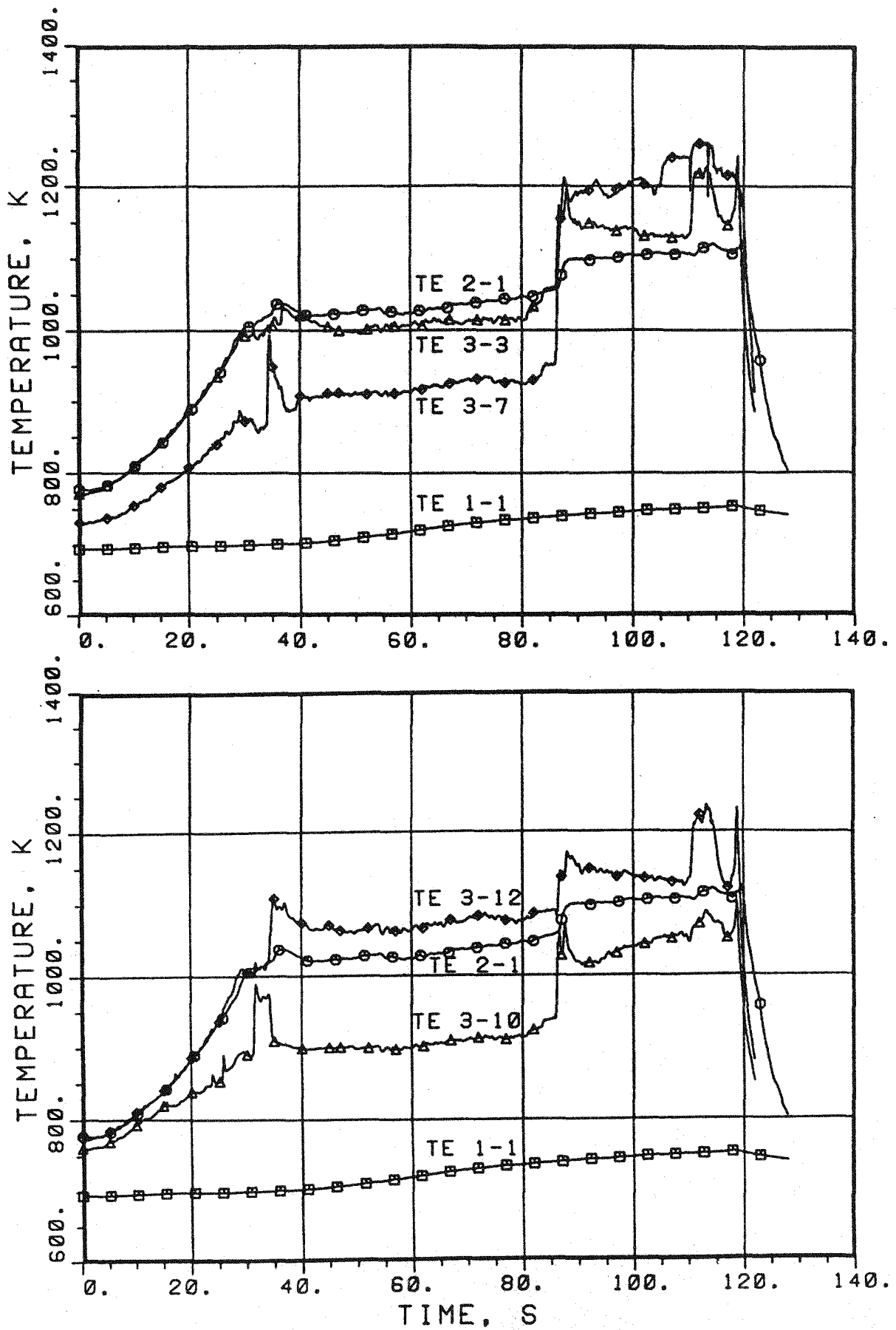


Fig. 14. Fuel-Bundle Thermocouple Response During the Power Transient

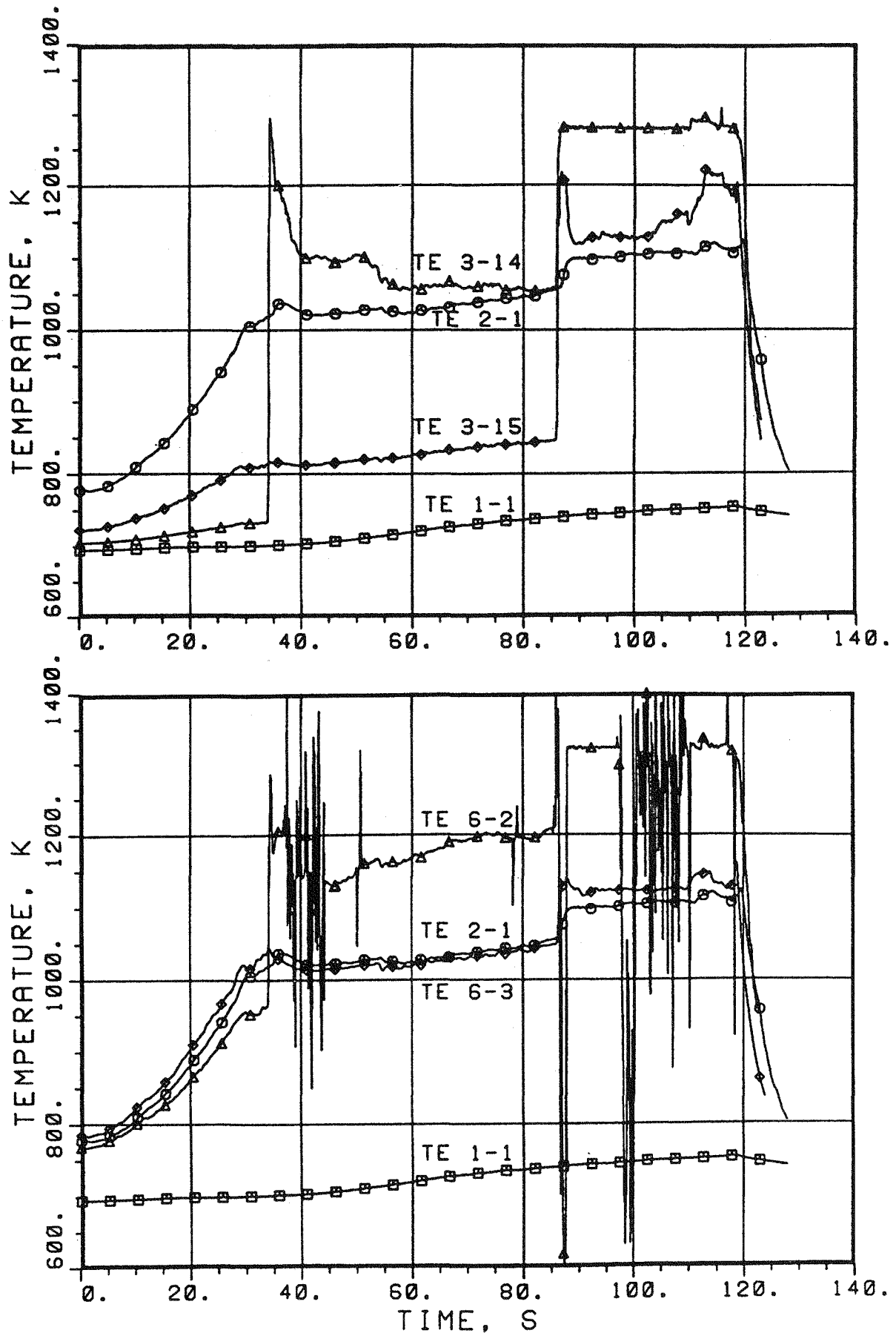


Fig. 14. Fuel-Bundle Thermocouple Response During the Power Transient (Cont'd)

Table 8. Response of Fuel-Bundle and Hex-Can-Wall Thermocouples During the P4 Power Transient

| Thermocouple | Fuel Pin | Junction Elevation* |            | Time, s           |                        |                                        |           |                              |                                                |                                 |
|--------------|----------|---------------------|------------|-------------------|------------------------|----------------------------------------|-----------|------------------------------|------------------------------------------------|---------------------------------|
|              |          | mm                  | 15-17      | 24-29             | 31-34                  | 34                                     | 53-54     | 80                           | 86                                             | 110-117                         |
| TE 3-1       | 1        | 450                 |            |                   |                        |                                        |           | gradual 20 K increase in 6 s | 100 K increase, +80 K plateau                  | 70 K increase, recovery         |
| TE 3-2       | 3        | -262                |            |                   |                        |                                        |           | failed-new junction formed   |                                                | 30 K increase, partial recovery |
| TE 3-3       | 4        | 94                  |            |                   |                        |                                        |           | gradual 45 K increase in 6 s | 115 K increase, +90 K plateau                  | 90 K increase, recovery         |
| TE 3-6       | 5        | 246                 |            | 5 K, 20 K spikes  |                        | +50 K plateau for 3 s                  | 10 K drop |                              | 80 K jump, +50 K plateau                       | +20 K plateau                   |
| TE 3-7       | 6        | 94                  |            | 7 K spike         | 30 K drop over 3 s     | 130 K jump, recovery in 3 s            |           | gradual 30 K increase in 6 s | +240 K plateau                                 | 15 K increase, 45 K drop        |
| TE 3-8       | 7        | 500                 |            | 25 K spike        | 60 K increase over 3 s | 45 K drop over 6 s                     |           |                              | 120 K jump, +45 K plateau                      | 10 K increase, partial recovery |
| TE 3-9       | 8        | 94                  |            | 10 K, 25 K spikes | +45 K plateau for 3 s  |                                        |           |                              | 30 K spike                                     | 15 K increase, recovery         |
| TE 3-10      | 8        | 145                 | 8 K spike  | 15 K, 30 K spikes | +100 K plateau for 3 s |                                        |           | gradual 20 K increase in 6 s | 120 K jump, decrease of 45 K, than ramp upward | 40 K increase, recovery         |
| TE 3-12      | 12       | 94                  |            |                   |                        | 100 K jump, gradual decrease to + 60 K |           |                              | 80 K jump, gradual decrease                    | 100 K increase, recovery        |
| TE 3-14      | 14       | -363                |            |                   |                        | 580 K jump, gradual decrease to +320 K |           |                              | 220 K plateau, boiling at 1280 K               | boiling continued at 1280 K     |
| TE 3-15      | 18       | -160                |            |                   |                        |                                        |           |                              | 380 K jump, +275 K plateau                     | 80 K increase, gradual decrease |
| TE 4-3       | 25       | 500                 | 20 K spike |                   |                        |                                        | 10 K drop |                              | +15 K plateau                                  | 40 K increase, recovery         |
| TE 6-1       | Flat 1   | 145                 |            |                   |                        | failed when hit by molten fuel         |           |                              |                                                |                                 |
| TE 6-2       | Flat 2   | 145                 |            |                   |                        | 330 K jump, fuel at junction           |           |                              | temp. spiking, boiling at 1322 K               | boiling continued at 1322 K     |
| TE 6-3       | Flat 3   | 145                 |            |                   | 35 K increase over 3 s | 40 K decrease over 8 s                 | 10 K drop |                              | 95 K jump, +80 K plateau                       | 20 K increase, recovery         |
| TE 6-4       | Flat 4   | 145                 |            |                   | 35 K spike, 55 K drop  |                                        |           |                              | 40 K drop, gradual recovery                    |                                 |

\*Measured from midplane of fuel canisters and fuel-pin pellet columns.

pins and hex-can Flat 2. Fuel appears to have located adjacent to the junction of thermocouple TE 6-2 (Flat 2) without causing thermocouple failure.

A 10 K drop in local temperature at 53 s coincided with an increase in fuel bundle flow. At 80 s, thermocouple TE 3-2 (Pin 3) failed. It was the only thermocouple on a fuel pin adjacent to the fluted fuel canister (Pin 12) that continued through the fuel canister region and had a junction located below the fuel midplane. This thermocouple failure, evidence of impending change of the fluted fuel canister, was also accompanied by gradual 20- to 45-K increases in sodium temperature in the center of the bundle. Rupture of the fluted fuel canister at 86 s released fuel toward the center of the fuel bundle. Thermocouple TE 3-15 (Pin 18) was hit by molten fuel and formed a new junction, registering a 380-K jump in temperature and a remaining 275 K plateau. The 86-s molten fuel release produced a 240-K plateau in local temperature at thermocouple TE 3-7 (Pin 6). Temperature perturbations in the wake of the partial flow blockage produced by the 34-s molten fuel release were also observed. There were a number of thermocouples registering temperature spikes of 80 to 120 K. Both thermocouples TE 3-14 (Pin 14) and TE 6-2 (Flat 2) indicated sodium boiling, at 1280 K and 1322 K, respectively.

Temperature perturbations of 10 K to 100 K were measured during the gas release beginning at 110 s. The highest temperature increases, 100 K, were measured by thermocouples TE 3-3 (Pin 4) and TE 3-12 (Pin 12), which were nearest to the hex-can failure location. Thermocouples TE 3-1 (Pin 1) and TE 3-15 (Pin 18) also measured major temperature increases, 70 K and 80 K, respectively. By 117 s, most thermocouples signals indicated complete temperature recovery from the gas release. Coolant temperatures then began to rise in the bundle, coinciding with a gradual decrease in fuel bundle flow, until power operation was terminated by the ETR scram at 118.4 s.

#### e. Sodium flow

Sodium leaked into the lower magnet section of the "quad" inlet flowmeter, rendering FE 1-1 and FE 1-2 inoperative before P4 power operation began. That reduced the number of operable fuel-bundle inlet flowmeters to two: FE 1-3 and FE 1-4. The sodium flow rates measured by fuel-bundle inlet flowmeter FE 1-3, fuel-bundle outlet flowmeter FE 2-1 and combined-flow flowmeter FE 3-2 during the power transient are shown in Fig. 15.

Small perturbations in signal were noted in the output of all flowmeters coincident with initial failure of the fluted fuel canister at 15.3 s. Noise in the signal of the bundle outlet flowmeter, and later the combined-flow flowmeters, resulted from slow gas release from the fluted fuel canister and the adjacent, pressurized fuel pins. Small perturbations in bundle flow were observed during the 5-s hold at 175 MW ETR power. Flow perturbations accompanied the ejection of molten fuel from the cylindrical fuel canister on Pin 5 at ~34 s. The inlet flow rate decelerated to a minimum 2.4 kg/s (70%) and returned to 3.2 kg/s, 93% of the initial steady state flow rate. The flow reduction accompanying the molten fuel release was best described as two sets of twin flow perturbations. Two small downspikes in inlet flow, to 3.1 kg/s, then 3.0 kg/s (4 ms apart) were observed at 33.97 s. They were accompanied by a pair of upspikes in bundle outlet flow. Subsequent

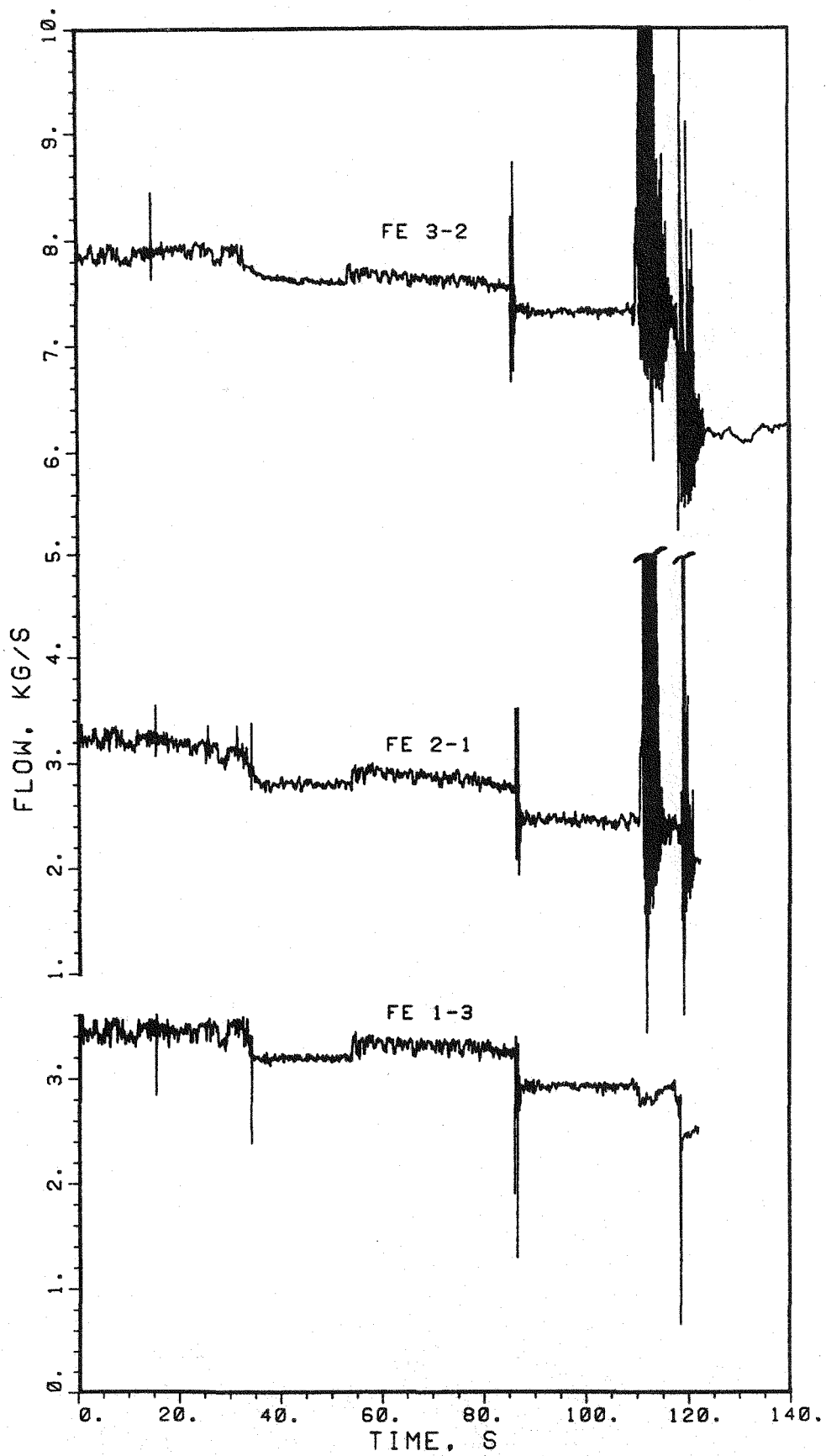


Fig. 15. Coolant Flow Rate During the Power Transient

inlet flow spikes down to 2.5 kg/s, then 2.4 kg/s (8 ms apart), also accompanied by upspikes in outlet flow, were observed between 33.98 and 33.99 s.

The envelope of the inlet flow response was in excellent agreement with pretest predictions. Pretest calculations predicted a 70% minimum inlet flow reached in 2.5 ms following rupture of one cylindrical fuel canister, based on a slug-response coolant behavior driven by the 14.4-MPa peak pressure calculated by the PLUTO code,<sup>16</sup> as shown in Fig. 16. Examination of Fig. 16 shows the code predicted a monotonic reduction and recovery in flow, whereas the measured flow showed an oscillatory behavior with a frequency of ~270 Hz and two major impulses. We theorize that the fine structure was caused by the expansion and contraction of a sodium vapor bubble, with the two major pulses caused by contact between hot fuel and liquid sodium.

Between 53 and 54 s, the fuel-bundle inlet flow increased from 3.2 kg/s to 3.35 kg/s. Then inlet flow began to gradually drift lower; the rate of inlet flow reduction increased at 80 s. At 86 s, inlet flow spiked downward to 2 kg/s, then to 1.9 kg/s 20 ms later, and recovered to 3 kg/s. Bundle outlet flow exhibited upspikes in flow of 0.75 to 1 kg/s 20 ms apart, as did the total loop flow. About 0.45 s later, a single flow perturbation occurred. Inlet flow spiked down to 1.3 kg/s and recovered to 2.95 kg/s. Bundle outlet flow was perturbed upward by 1.2 kg/s, as was the combined flow in the test train.

The bundle inlet flow remained steady at 2.95 kg/s until 110 s. Gas escaping from the pressurized insulator region into the fuel bundle through a local failure of the hex-can reduced the inlet flow for several seconds. Gas passing through the bundle-outlet and combined-flow flowmeters caused large oscillations in the indicated flow rate. The bundle coolant flow rate recovered to the pre-gas-release level by 115 s; passage of gas through the total-loop flowmeters ceased at ~116 s. At 117 s, the fuel-bundle flow rate began to drift lower. Additional gas entered the fuel bundle and swept through the bundle-outlet and combined-flow flowmeters following the ETR scram at 118.4 s.

#### f. Pressure

Fuel-bundle inlet and outlet pressure levels measured during the power transient by slow-response, eddy-current pressure sensors PE 2-1 at the inlet and PE 4-1 at the outlet are shown in Fig. 17. Both pressure sensors were sensitive to the fluted-canister failure beginning at ~15 s, bundle distortion events at 25 and 31 s that also produced temperature and acoustic noise perturbations, and pressurization that accompanied molten fuel release at 34 s and at 86 s. The bundle inlet transducer also sensed the slight change in bundle flow at 53 to 54 s, gas passage through the bundle beginning at 110 s, the "tailing off" of bundle inlet pressure beginning at 117 s, and the pump transfer at 118.4 s to a backup power supply. Step increases in inlet pressure accompanied the partial flow channel blockages at 34 and 86 s; recovery from the pressure increase due to gas release at 110 s was complete.

The reduction in inlet pressure coincident with a reduction in fuel-bundle flow at 117 s was atypical of either a flow-channel blockage

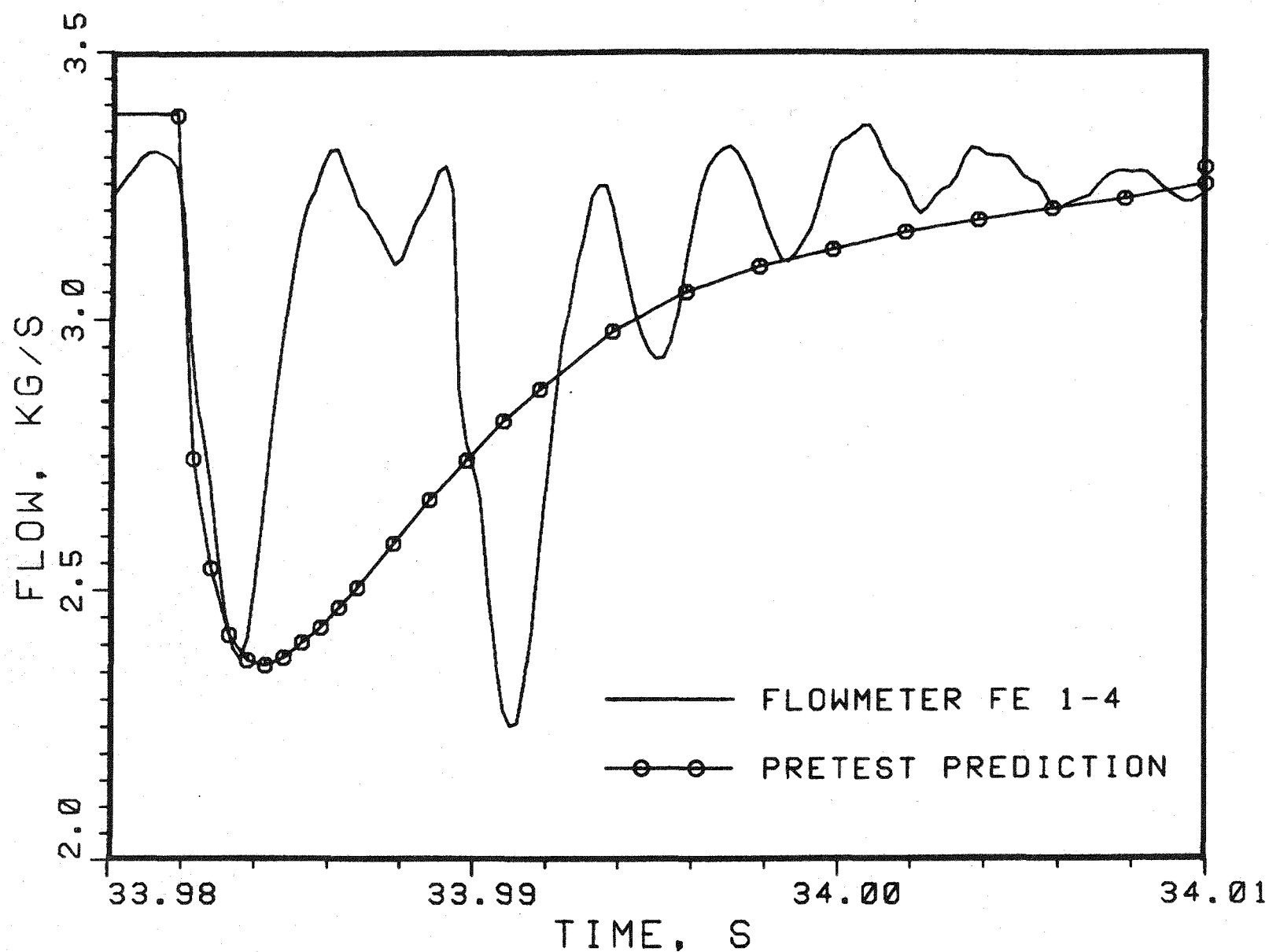


Fig. 16. Comparison of Inlet Flow Response to Molten Fuel Release with Pretest Prediction

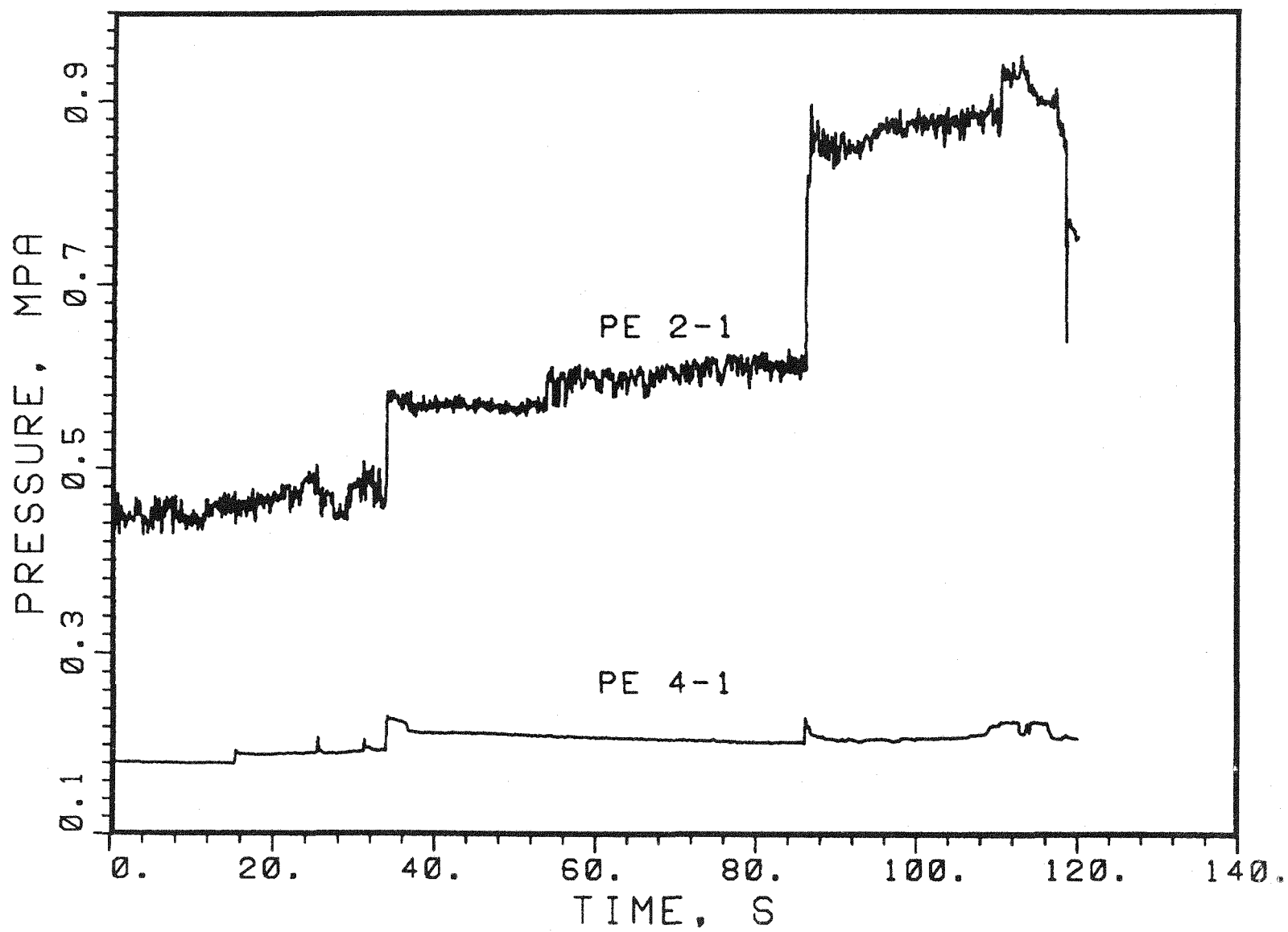


Fig. 17. Fuel-Bundle Inlet and Outlet Pressure during the Power Transient



extension (flow decrease, inlet pressure increase) or debris sweepout (flow increase, inlet pressure decrease). It was symptomatic of reduced pump performance at fixed power input due to both the rising inlet and system sodium temperature and gas entrainment in the pumped sodium, as shown in Ch. VI.

The loop gas-plenum volume and pressure were not static during experiment operation. Thermal expansion of the sodium in the loop due to the increasing sodium temperature would have increased the loop plenum pressure slightly as the sodium level increased. Because the operating sodium level was outside the active length of the sodium-level "J" probe, the plenum pressure change due to sodium expansion could not be evaluated. Automatic pressure control in the OLCS was provided via feedback control; the OLCS pressure/flow response had an ~50-s time constant.

The measured pressure in the OLCS upstream of the loop gas plenum (PE 17-1) is shown in Fig. 18. Also shown is the loop gas-plenum pressure corrected to remove the pressure adjustment of the OLCS (analysis described in Ch. VI). Figure 18 indicates a steady release of uncondensable gas to the loop plenum from failed fuel pins in the bundle after melting of fluted canister and adjacent pin cladding at ~15 s (and 2 to 3 s transport time to the plenum). This was followed by additional gas release from pin failure tied to the molten-fuel release at 34 s. Little additional gas was released coincident with the molten fuel release at 86 s. The major sharp increase in loop plenum pressure followed local failure of the hex-can at 110 s and blowdown of the pressurized insulator region volume through the fuel pin bundle. For a 30-liter loop gas-plenum volume, the 10-kPa pressure increase (corrected) corresponds to a release of 1280 cm<sup>3</sup> (STP) from the hex-can insulator region.

#### g. Acoustic Noise

The acoustic sensor located in the bypass region (PE 9-1) near the fuel-bundle outlet was closer to the fuel bundle than the acoustic sensor located in the combined flow region (PE 9-2) at a higher elevation, hence had a higher signal-to-background ratio. As shown in Fig. 19, acoustic sensor PE 9-1 was responsive to events occurring at 15, 25, 31, 34, 36, 86, and 118 s that were also sensed by the other types of instruments. The characteristics of the acoustic sensor response were varied. At 15 s, PE 9-1 response was low frequency (200-500 Hz). An acoustic "hissing" noise between 3 kHz and 8 kHz was followed, at 31 s, by a wideband acoustic pulse with frequency from 200 Hz to 20 kHz. The acoustic pulse reached its peak at 34 s, followed by a "crack" between 2 kHz to 4 kHz, at 36 s. Two wideband acoustic peaks and a single pulse occurred at ~86 s. "Hissing" acoustic noise of 5 kHz to 20 kHz accompanied gas release into the fuel bundle between 110 and 113 s, followed by a low-level "hissing" noise again with the gas release following reactor scram at 118.4 s.

### 3. Response of Failed-Fuel Detection Systems

The three fuel-failure detection systems indicated the presence of fission products in the loop following failure of the cladding of the test pins or the fuel canisters. They acted "globally", as opposed to most of the other sensors, and in making correlations between their signals and that of the other sensors, it was necessary to allow for the transport delay time. A

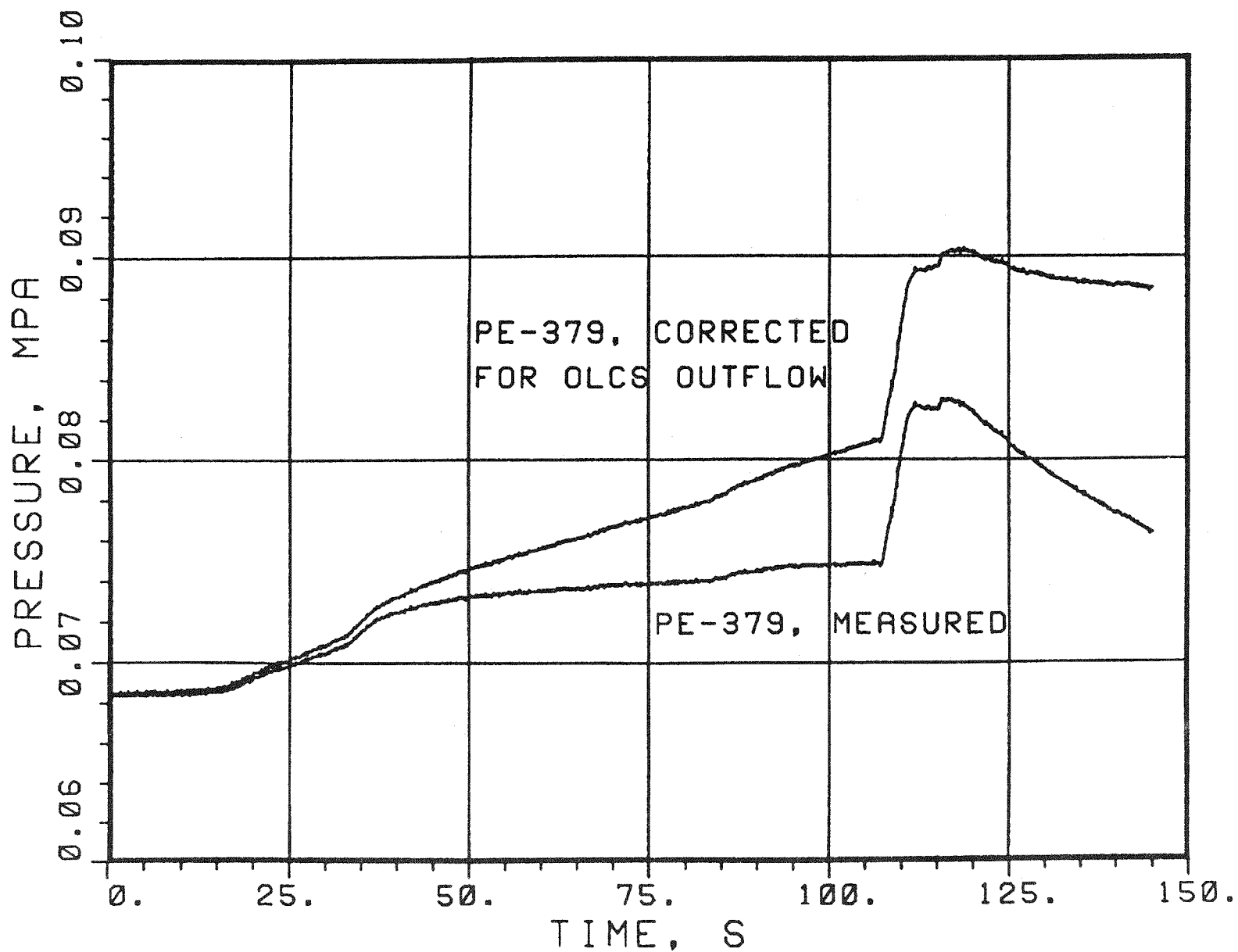


Fig. 18. Loop Plenum Gas Pressure - As Measured and Corrected for OLCS Outflow

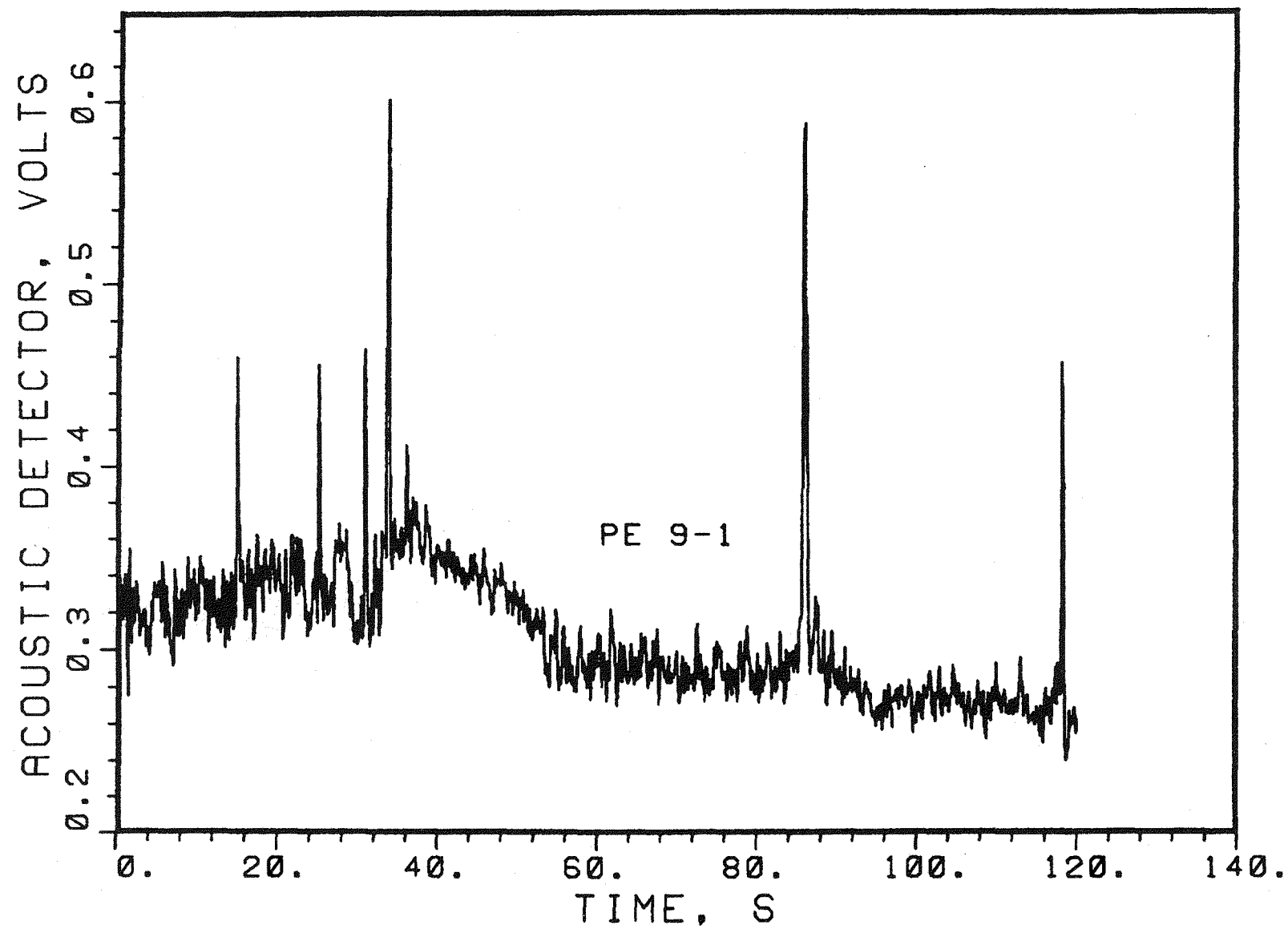


Fig. 19. Acoustic Sensor Response During the Power Transient

detailed correlation was developed only for the DN systems, in which the delay was measured in seconds, and not for the fission gases, where the delay was measured in minutes. In some cases, there was very clear correspondence with other signals, and in others little or none.

a. DND response

The DND gave the promptest indication of cladding failure. All of the sensitive channels containing  $\text{BF}_3$  detectors saturated due to the high count rates; detailed data were obtained from the fission-detector channels. Figure 20, similar to the top trace in Fig. 11, shows the digital data recorded from one fission detector. Figure 21 shows data from another fission detector which was operated in the current mode to avoid problems with dead time; these data were recorded on high-speed magnetic tape to preserve finer details in time.

Figure 20 shows the first indication of DN emitters in the sodium at 17 s. The strength of the signal increased strongly in a non-uniform fashion for the next 20 s. At 36 s, there was a DN spike indicating the expulsion of molten fuel. Roughly 6 s later, the DN rate fell from the peak to a value roughly three times that before the fuel release. This post-fuel-release DN level increased by about 20% over the next 46 s, but not evenly, having a number of minor peaks and bumps superimposed on it. At 88 s, there was a second major DN spike similar in magnitude to the first, except that the increase in the general level after this fuel release was not as great. The new level was maintained, subject to more fluctuation than before, until 120 s, when there was a distinct bump at the time of the scram, followed by decay with the complex time constant of the DN population.

Many of these events, both major and minor, can be correlated with other sensors, keeping in mind the two-s transit delay to the DND. There were six peaks closely following the initial fluted-canister failure, three of them were associated with other evidence of a mechanical disturbance in the system. The very first failure indication was accompanied by spikes in the acoustic, inlet and outlet flow, and pressure sensors. A close comparison of Figs. 19 and 21 shows a correlation between the initial DN rise and the first acoustic spike. The DN signal had a shoulder, before rising again to the first peak. It appears that there were two events, separated by one s (a similar behavior was seen during the W2 experiment<sup>8</sup>), and that the acoustic spike accompanied the first, as did spikes in the flow and pressure records. DN peaks at 27 s and 33 s had corresponding changes in acoustic and outlet flow signals but not in inlet flow. Three peaks were not echoed by any other sensor. Small peaks at 22, 24, and 30 s were not due to the recirculation of DN activity in the loop because they did not appear at the proper interval after any earlier peak. These six DN peaks appeared in the time interval during which failure of cladding in the cusp regions of the fluted canister was expected, due to sodium boiling, cladding dryout, and melting. The DN signal, therefore, provided confirmation that initial fluted canister failure followed expectations.

The major DN peak at 36 s was accompanied by strong acoustic signals, by permanent changes in the sodium flow and pressure, and also by an abrupt change in temperature measured by thermocouple TE 3-14. The peak DND signal alone indicated the expulsion of a large amount of fuel; the large

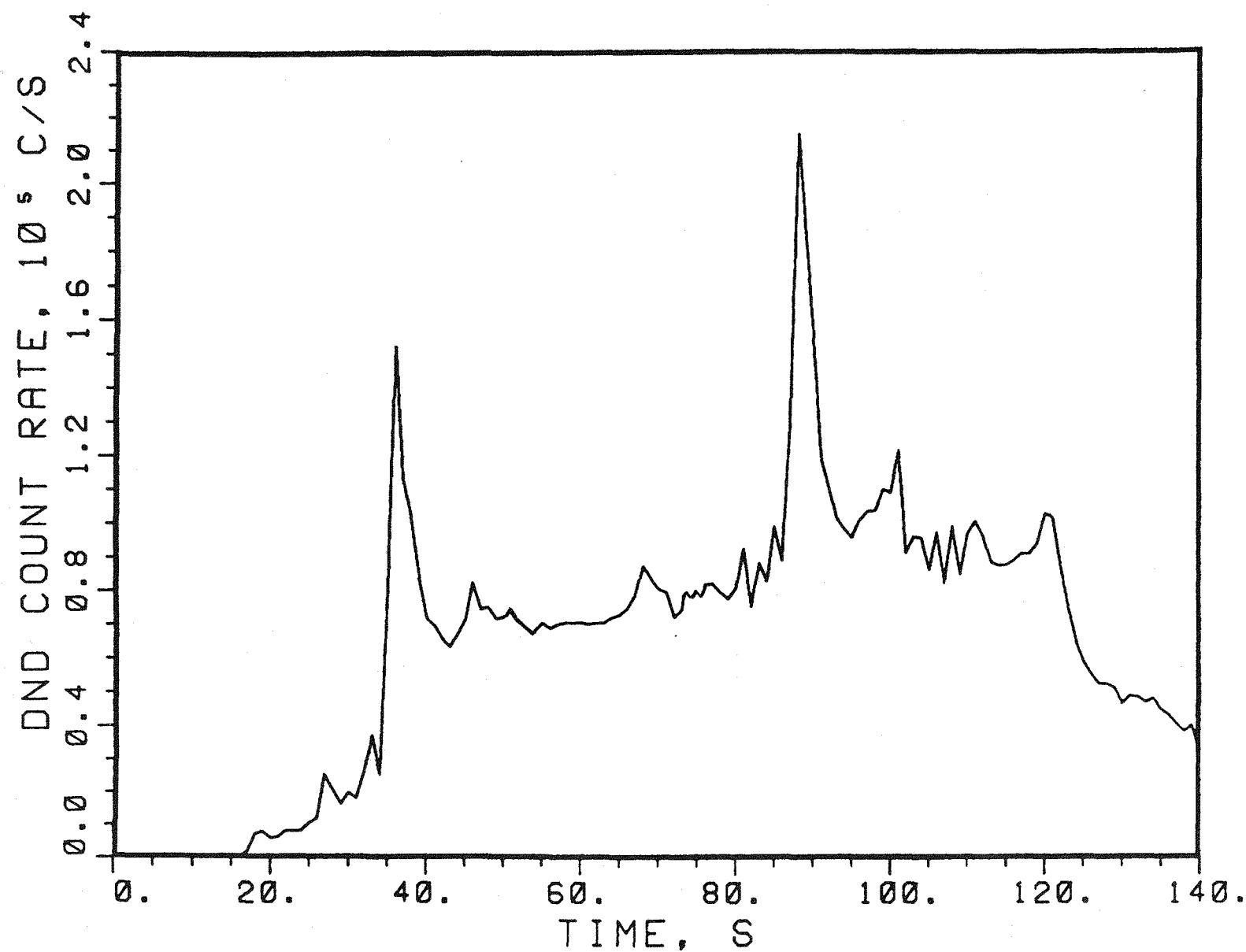


Fig. 20. Response of Fission Detector (DND) Operated in the Pulse Mode During the Power Transient

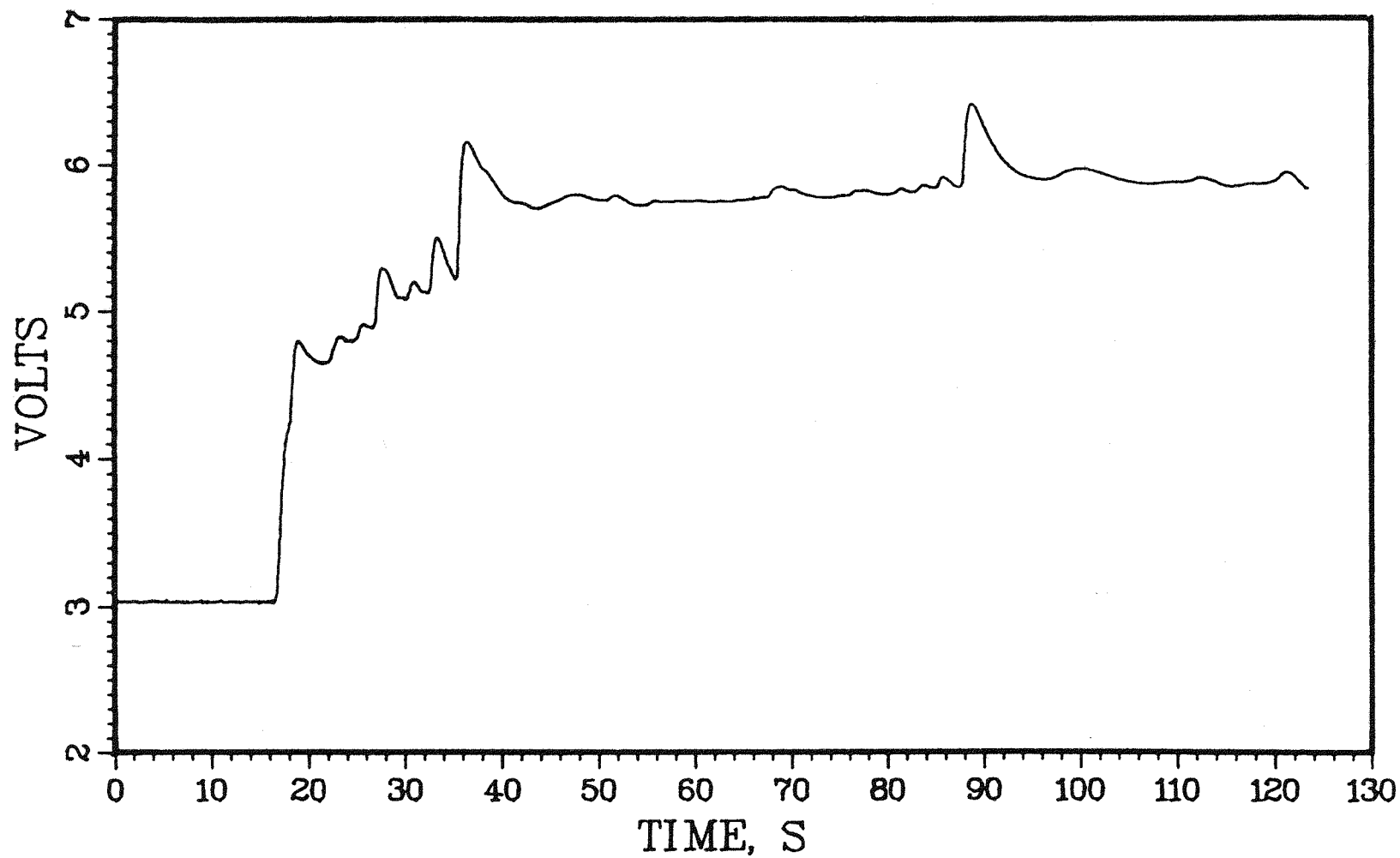


Fig. 21. Response of Fission Detector (DND) Operated in the Current Mode During the Power Transient

increase in the steady signal after the initial release indicated that this expelled fuel had not been swept away, but remained in the fuel bundle. Figure 19 shows that there was a second, minor acoustic signal 2 s later. In Figs. 20 and 21, a shoulder can be seen on the peak at ~38 s. This failure therefore, may have taken place in two steps. The bump at 46 s was due to the recirculation of the 36-s peak. However, small peaks at 50, 66, 76, 80, 82, and 84 s are genuine; they are not associated with significant signals from any other sensors. They could correspond to relatively minor adjustments of the fuel debris in the test section.

The major DN peak at 88 s was also accompanied by a strong acoustic signal, by changes in flow and pressure, and by an abrupt temperature change in thermocouple TE 3-15. The DND evidence that there was a major expulsion of fuel was, therefore, supported by other instruments. The peak at 96 s was a satellite due to recirculation in the loop sodium. The permanent signal due to the exposed fuel remaining in the bundle following this expulsion was not as great as in the case of the 36-s expulsion. Small DN peaks near 110 s seem to be genuine; that at 120 s is coincident with reactor scram.

b. DNM response

The signals from the DNM were closely related to those from the DND. The major difference was an additional transport delay of 12 s. This had two effects: there was a simple delay in seeing the onset of any event and, because the shorter-lived DN groups decayed during the transport delay, the DNM signal was weaker and less sharp. The DNM record during the transient is shown in Fig. 22; the relationship to the DND record is quite clear.

c. OLCS response

The fission gases were not expected to reach the detectors until 5 minutes after they had reached the gas plenum of the loop, because of the delay introduced to allow for the decay of  $^{23}\text{Ne}$ . Figure 23 shows the trace from an ionization chamber detector that observed the flow lines in the OLCS; it does not show the delay. Apparently the ionization detectors responded to a general background due to the reactor transient itself and failure products in the plenum of the loop. However, the count rate of the GeLi detectors did show the expected delay, demonstrating the advantage of spectroscopic measurements in a remote shielded area as opposed to simple measurements of radiation levels.

Figure 24 shows a spectrum recorded a few minutes after the power transient. The fission-gas sample was diluted to the maximum extent and the sample volume and collimators were set so as to produce minimum sensitivity. The OLCS, therefore, reinforced the evidence of the DN systems that major fuel failure had taken place.

C. Interim Shutdown and Low-Power Restart

Because the fuel-bundle coolant flow rate was below the safety analysis and administrative operational limit of 90% for steady-state power operation, a restart for full-power operations was delayed until an expanded safety envelope was prepared and approved. From August 21, 1981, until the return to full-power operation began on October 1, 1981, a variety of tests were

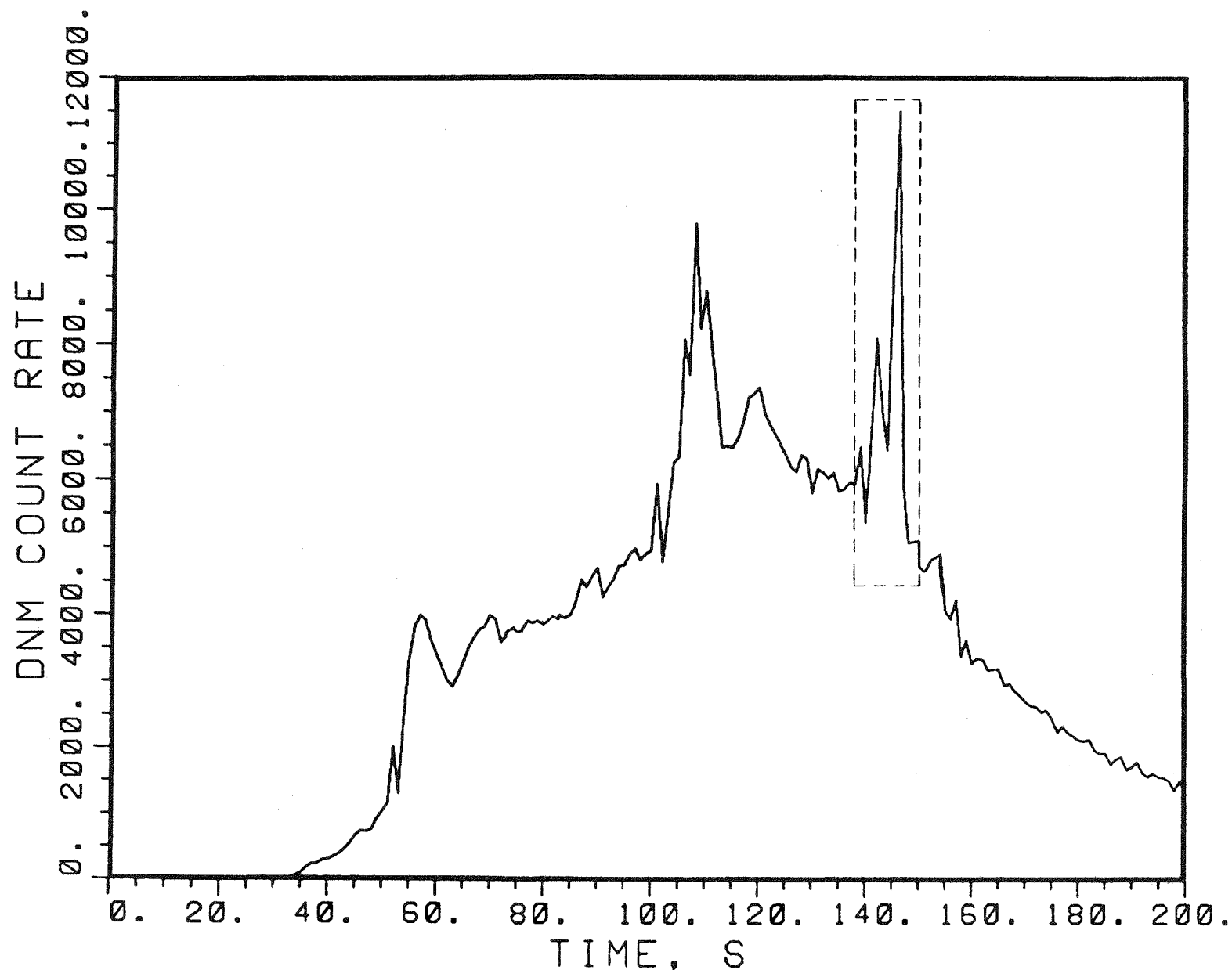


Fig. 22. Response of Fission Detector (DNM) Operated in the Pulse Mode During the Power Transient (The signal spiking between 138 and 150 s was a data-logging anomaly in the OLCS computer and occurred simultaneously on many channels).



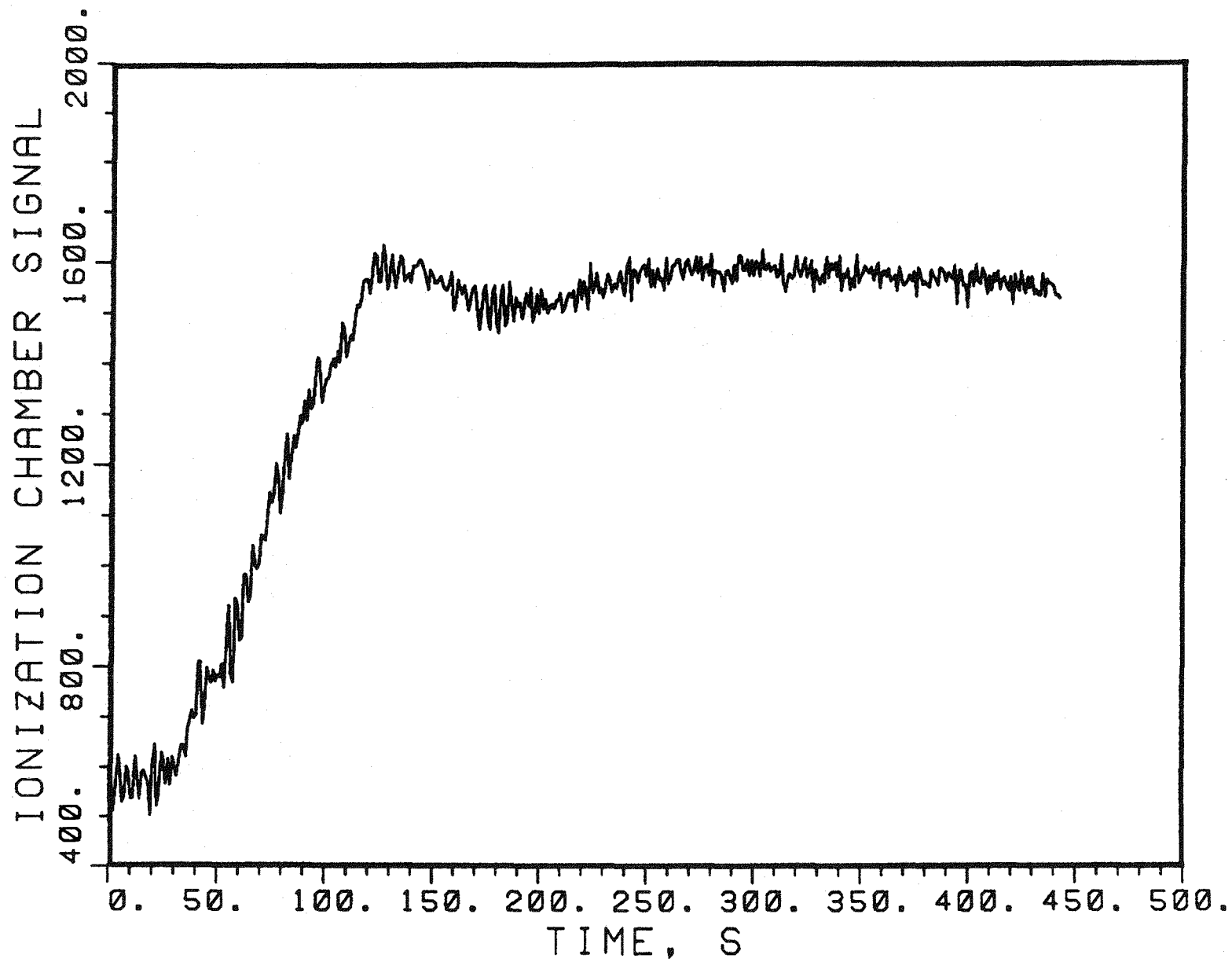


Fig. 23. Signal from OLCS Ionization Chamber

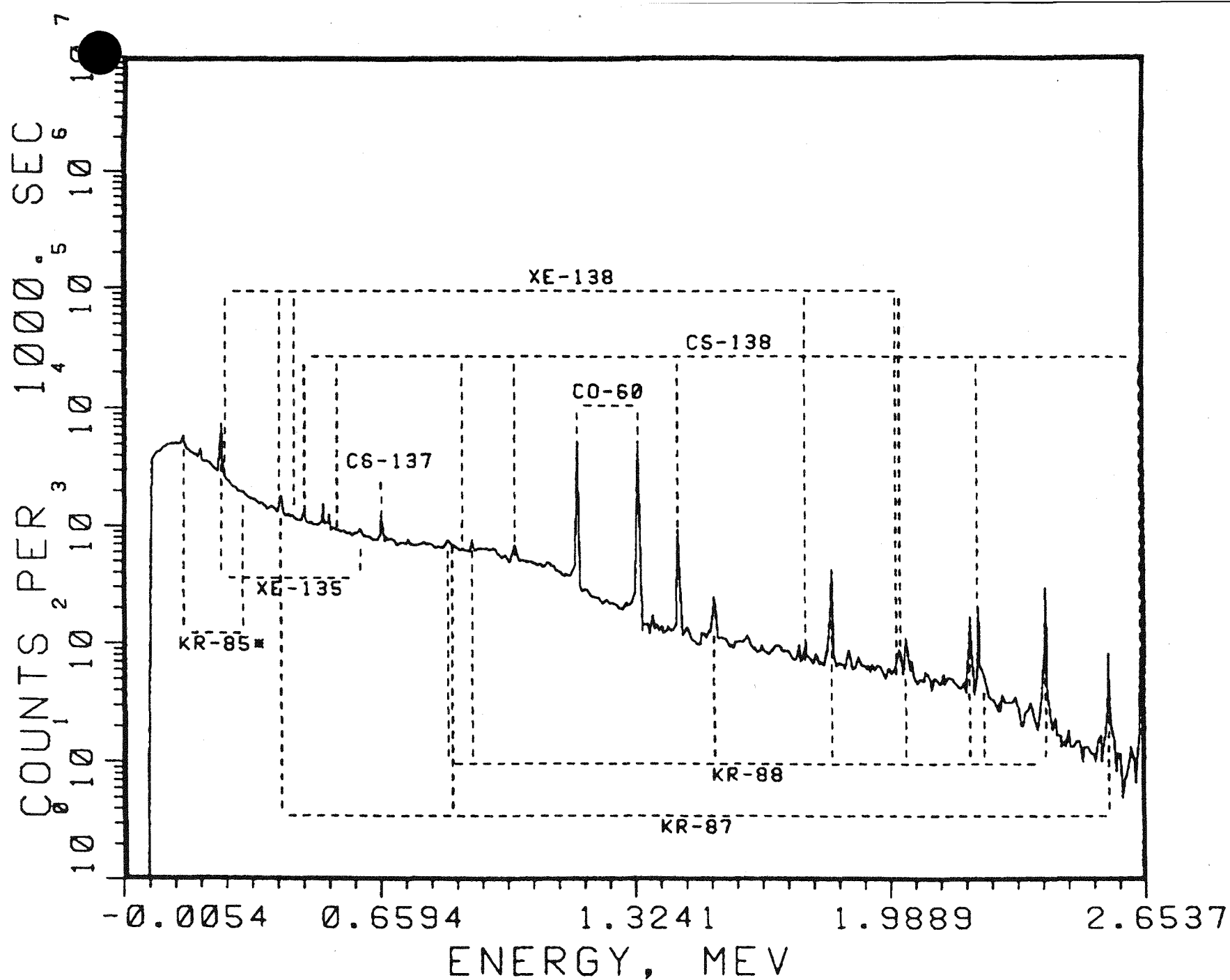


Fig. 24. Spectrum Taken from an OLCS GeLi Detector

performed in parallel with the revision of the P4 safety analysis document. These tests included a pump transfer-to-emergency-power test to confirm transfer time and determine coolant flow response, a loop plenum pressure test to identify failed pins via fuel-pin-plenum pressure-sensor response, delayed neutron mechanisms tests to study the DN signal as a function of the sodium temperature and reactor power (2.2 MW and 40 MW), and a repeat of the 40-MW benchmark test.

## 1. Fuel-Pin Pressure Sensor Test

The fuel-pin pressure sensors provided information on the extent of fuel pin failure. Over a four-h period on August 25, 1981, the loop plenum pressure was decreased from 157 kPa (22.8 psia) to 90 kPa (13.1 psia), then increased to 338 kPa (49 psia) followed by a return to 157 kPa. The fuel-pin pressure-sensor signals were monitored for change. If the pressure-sensor signal responded rapidly to the changing loop plenum pressure, the pin was assumed to be breached and open to the sodium. If the response was slow, the pin was assumed to have a pin hole leak through the cladding. Pins 4, 6, 7, and 14 were open to the sodium. Pins 3, 9, 11, and 13 had pin-hole leaks. Pin 19 was intact or any failure site sealed from the changing pressure in the sodium during the test. Five other fuel pins had pressure sensors; the sensors on Pin 1 and 2 failed prior to the power transient test and those on Pins 10, 15, and 16 were installed spares.

## 2. Delayed Neutron Mechanisms Tests

The existence of a large mass of failed fuel in the fuel bundle after the power transient presented a rare opportunity to observe the DN count rate from the debris. Such a large quantity of failed fuel is not normally available for study under steady-state conditions. Therefore, tests at low reactor power were performed to measure the dependence of the DN signal on the temperature of the sodium coolant.

ETR startup for Cycle-three irradiation began on September 3, 1981. During the initial DN mechanisms test, the ETR was operated at a constant 2.2 MW, corresponding to a bundle power of 18 kW. The loop sodium temperature was decreased from 700 K to 589 K in 1.5 h by changing the helium flow rate to the loop heat exchanger while loop sodium flow was held relatively constant. After reaching the 589 K sodium temperature, both loop flow and sodium temperature were held constant for two hours as additional DN data were taken. Then the loop sodium temperature was returned to 695 K. The data taken at a bundle power of 18 kW indicated a distinct temperature dependence, as shown in Fig. 25.

Operating the pump at a fixed power input did not yield a constant sodium flow rate during the DN mechanisms tests. As sodium temperatures decreased, sodium flow increased, and vice versa. The change in flow was about 3% per 55 K (100°F). It was necessary to adjust the pump voltage during the DN mechanisms tests to maintain a constant sodium flow rate.

After the initial measurements at 2.2 MW, the reactor power was increased to 4 MW, and then from 10 MW to 40 MW in steps of 5 MW. At each of the eight reactor power levels, the reactor conditions were held constant for four h to reach temperature equilibrium and to allow time for the DNM measurements described below. At 40 MW, the temperature was lowered for the second

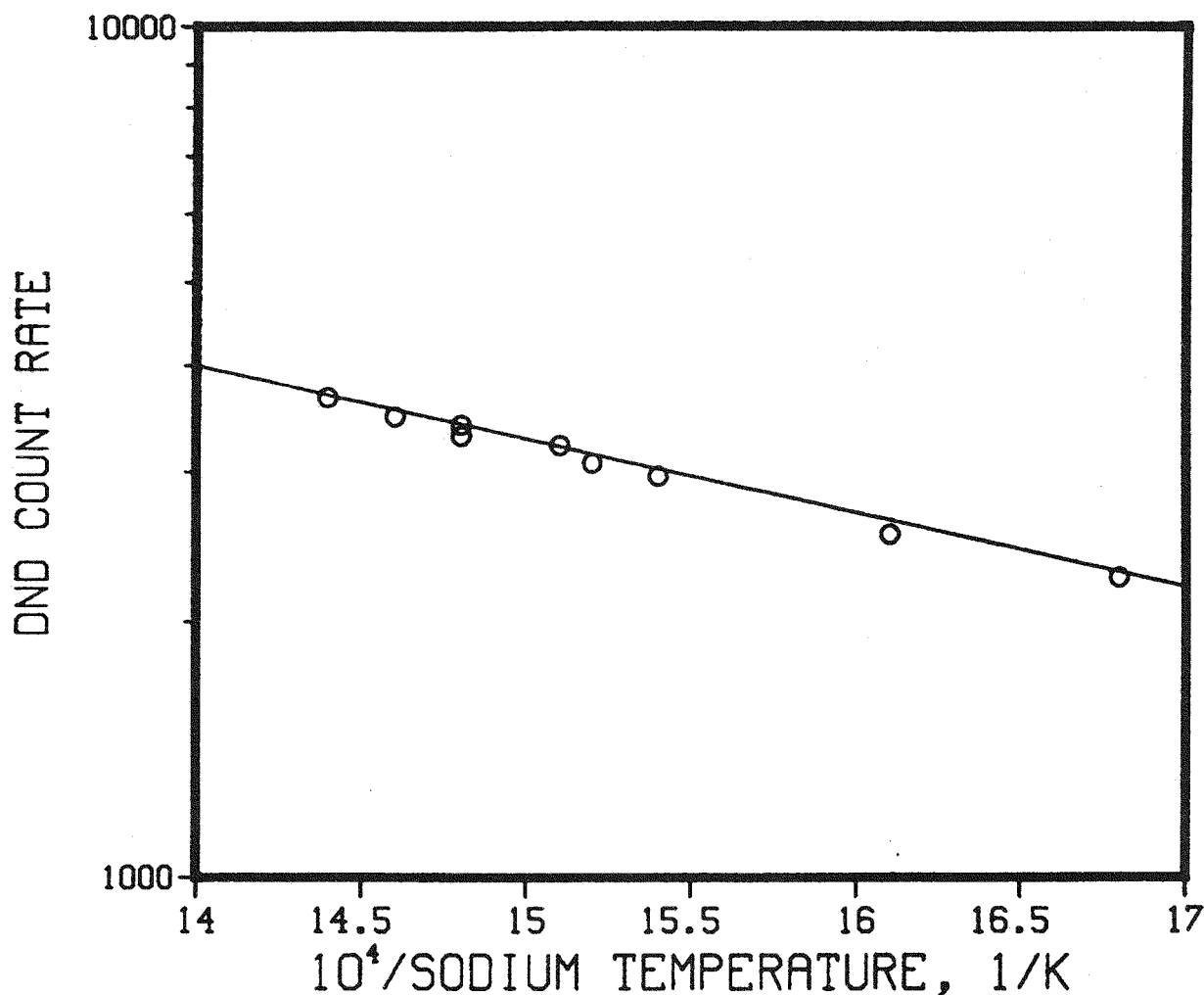


Fig. 25. DND Signal vs. Inverse Temperature (Narrow Range)

time from 700 K to 589 K for the final DN measurements. The rise to 40 MW was performed in a routine manner; no flow or temperature anomalies were observed. The DN signal from the DND at each power level is plotted in Fig. 26. The DN signal increased linearly with power, showing nothing abnormal, and indicated that, at least up to a reactor power of 40 MW (fuel-bundle power of 326 kW, ~25% design power), the exposed mass in the fuel bundle was quite stable. This stability gave confidence for the increase of power during Cycle four of experiment operation.

Measurements were also made of the dependence of the count rate in the DNM on the flow rate of the OLSS sample stream. The flow was varied from 1.2 gpm down to 0.3 gpm, corresponding to a delay range from 12 s to 40 s. This covered the range of time delay which is typical for an actual reactor

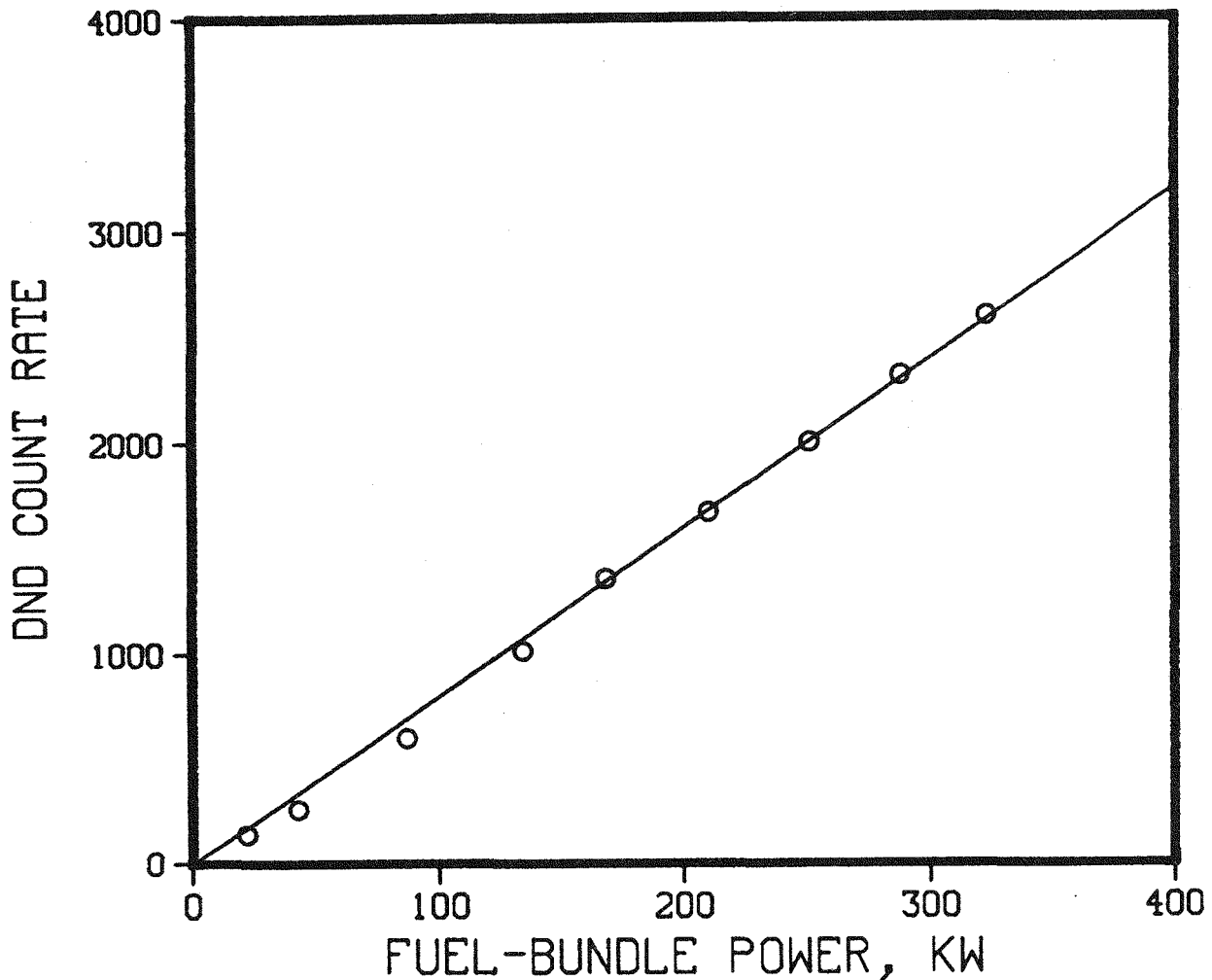


Fig. 26. DND Signal as a Function of Fuel-Bundle Power

installation. The effect of the time delay was to reduce the DN strength that reached the DNM because of decay during transit. The decay is complex because of the different half-lives of the several DN groups. Measurements should, therefore, indicate any changes in the population due to the effects, if any, of bundle power or sodium temperature. Measurements were made at each of the eight reactor power levels between 2.2 and 40 MW, and at temperatures of 700 K and 589 K. These data are compared with results of DNM measurements made during the approach-to-full-power operation in Sect. D of this chapter.

Additional DN mechanisms testing was performed in Cycle five of P4 power operation, after the approach to full power and blockage reconfiguration in Cycle four resulted in an even larger quantity of failed fuel and fuel debris to study. Chapter VI contains the interpretation of results from both series of DN mechanisms tests.

### 3. 40-MW Benchmark Tests

While at 40-MW ETR power during the DN mechanisms testing on September 5, 1981, data were collected during two 40-MW postfailure benchmark tests for bundle and sensor evaluation by comparison with prefailure results. One-h time-averaged data for all three 40-MW benchmark tests are summarized in Table 9. These results indicate that the temperature perturbations in the wake of the partial flow channel blockage formed by molten fuel release during the power transient persisted to the upper end of the heated region in the middle of the bundle (TE 3-3, TE 3-4, TE 3-1). Thermocouples above the blockage and near the periphery of the bundle such as TE 3-9 and TE 6-5, indicated lower temperatures during the full-flow, post-failure benchmark tests than prior to fuel canister failure and molten fuel release. This indicated that the partial flow-channel blockage formed during the power transient was located in the middle of the fuel bundle and toward the hex-can Flats 1 and 2, which was confirmed by the posttest examination (see Ch. V).

#### D. Blockage Reconfiguration

During the extended hot standby, from August 21, 1981, until the return-to-full-power operation began on October 1, 1981, the fuel-bundle inlet temperature was maintained at ~680 K, coupled with the intermediate operation to 40-MW ETR power, to enhance sodium ingress and egress at fuel-pin-failure sites and the formation of sodium urano-plutonate within the partial flow blockage. Both the extended hot standby and the subsequent return to power after significant fuel failure were atypical of LMFBR operation. The return to power was consistent with the bounding nature of local fault safety experiment P4 and provided the opportunity, in a heavily instrumented fuel bundle, to obtain information on the effect of sodium urano-plutonate dissociation.

#### 1. Summary of Events

Preparation for the return-to-full-power operation were completed with a safety analysis and administrative operational limit of 68% initial flow for steady-state power operation. At the reference pump power, the fuel-bundle flow rate of 2.95 kg/s (86% of initial) was unchanged from that during the later stages of the power transient.

ETR power operation resumed late on October 1. Reactor power was increased to 40 MW, followed by a four-h hold. Good agreement between the measured coolant flow rates and temperatures at 40 MW and the 40-MW benchmark test results were found in a comparison of the data. From 40 MW, reactor power was stepped up to 60 MW, followed by a two-h hold. Power plateaus at 70 MW and 80 MW were also maintained for two-h durations.

About seven minutes after reaching ~92 MW and coincident with the trimming of reactor power at 100 MW, the DND signals began to increase. This was the first nonequilibrium in the DN response during Cycle-four irradiation and signalled that events were building toward a change in the blockage configuration, which occurred 210 s later and was accompanied by a reactor scram. About 30 s later (180 s prior to the blockage reconfiguration), the DNM signals began to increase. The increase in DN signal was very gradual at first, building in an exponential manner as time passed, as shown in Fig. 27.

Table 9. 40-MW Benchmark Test Results

| Parameter             | August 21, 1981          |           | September 5, 1981       |          | Sensor              |
|-----------------------|--------------------------|-----------|-------------------------|----------|---------------------|
|                       | (before power transient) | full flow | (after power transient) | low flow | full flow           |
| Sodium Flowrate, kg/s |                          |           |                         |          |                     |
| Fuel Bundle Inlet     |                          | 3.40      | 2.91                    | 3.36     | FE 1-3              |
|                       |                          | 3.39      | 2.88                    | 3.34     | FE 1-4              |
| Fuel Bundle Outlet    |                          | 3.21      | 2.82                    | 3.24     | FE 2-1              |
| Loop                  |                          | 7.25      | 6.93                    | 8.02     | FE 3-1              |
|                       |                          | 7.91      | 7.36                    | 8.50     | FE 3-2              |
| Pump Voltage, volts   |                          | 275       | 276                     | 327      |                     |
| Pump Power, kW        |                          | 76        | 76                      | 105      | EJT-11,-12,-13      |
| Sodium Temperature, K |                          |           |                         |          |                     |
| Fuel Bundle Inlet     |                          | 694       | 695                     | 695      | Ave TE 1-1,-2,-3,-4 |
| Fuel Bundle           | @ - 46.5 cm*             | 691       | 693                     | 693      | TE 4-6              |
|                       | - 36.3 cm                | 703       | 722**                   | 719**    | TE 3-14             |
|                       | - 26.2 cm                | 717       | 728**                   | 725**    | TE 3-2              |
|                       | - 16.0 cm                | 721       | 776**                   | 767**    | TE 3-15             |
|                       | 9.4 cm                   | 768       | 795                     | 781      | TE 3-3              |
|                       | 9.4 cm                   | 746       | 790                     | 768      | TE 3-7              |
|                       | 9.4 cm                   | 767       | 765                     | 756      | TE 3-9              |
|                       | 9.4 cm                   | 772       | 750**                   | 747**    | TE 3-12             |
|                       | 14.5 cm                  | 758       | 782                     | 770      | TE 3-10             |
|                       | @ 14.5 cm                | 773       | 812**                   | 800**    | TE 6-1              |
|                       | 14.5 cm                  | 765       | ---                     | ---      | TE 6-2              |
| Hex-Can, PF-J         | 14.5 cm                  | 781       | 802                     | 788      | TE 6-3              |
|                       | 14.5 cm                  | 758       | 773                     | 761      | TE 6-4              |
|                       | 14.5 cm                  | 762       | 760                     | 752      | TE 6-5              |
| Fuel Bundle           | @ 24.6 cm                | 767       | 818                     | 799      | TE 3-4              |
|                       | 24.6 cm                  | 764       | 786                     | 774      | TE 3-6              |
|                       | 45.0 cm                  | 770       | 798                     | 784      | TE 3-1              |
|                       | 50.0 cm                  | 762       | 766                     | 758      | TE 3-8              |
|                       | 50.0 cm                  | 774       | 782                     | 771      | TE 4-3              |
|                       | 80.5 cm                  | 766       | 779                     | 767      | TE 3-11             |
| Fuel Bundle Outlet    |                          | 775       | 788                     | 776      | Ave TE 2-1,-2,-3    |
| Mixed Mean            |                          | 729       | 733                     | 729      | Ave TE 8-2,-3,-4    |
| Pump Exit             |                          | 693       | 694                     | 694      | Ave TE 21-1,-2      |

\* Location of thermocouple junction, measured from midplane of fuel column

\*\* New thermocouple junction formed during power transient

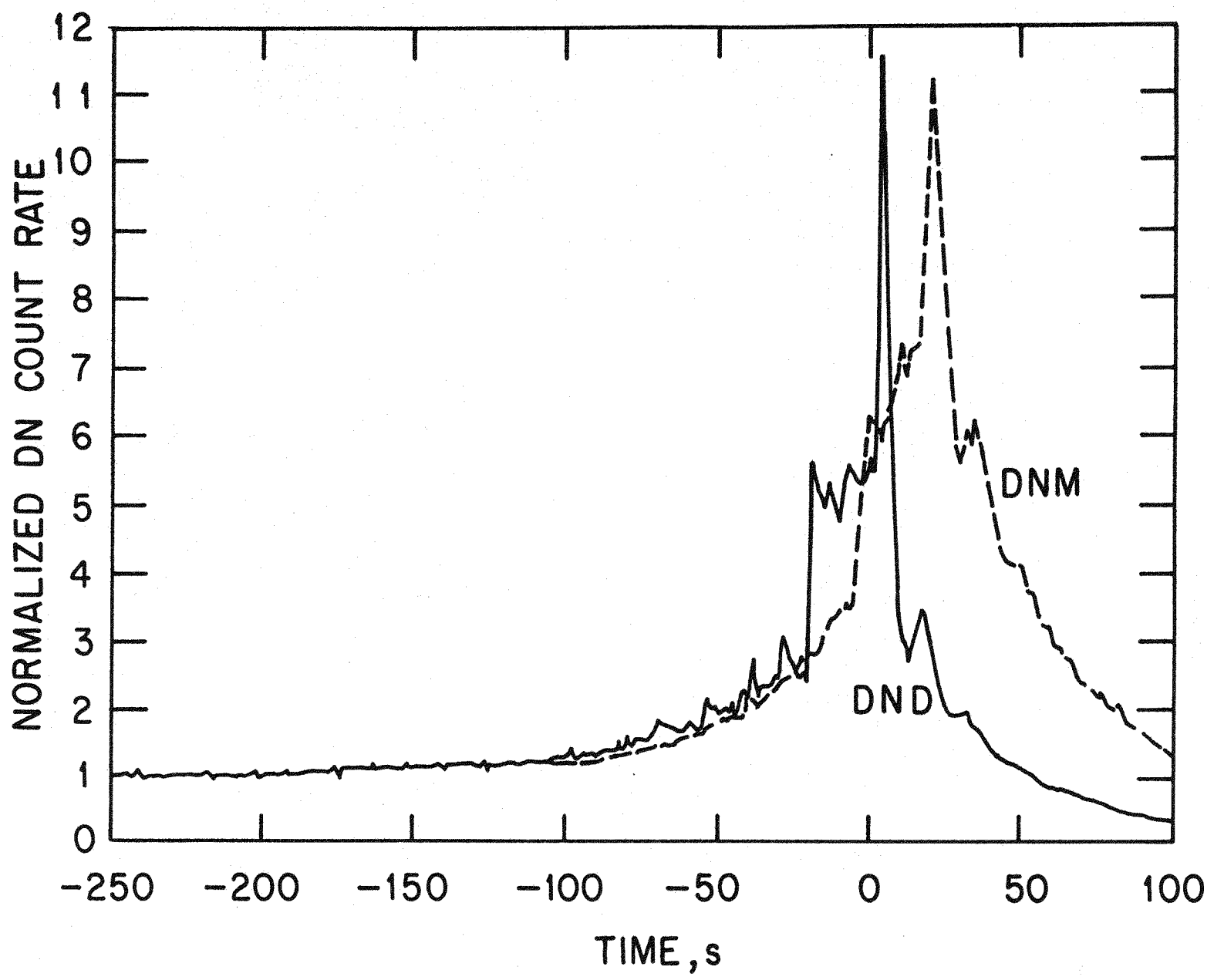


Fig. 27. DN Signal - Blockage Reconfiguration



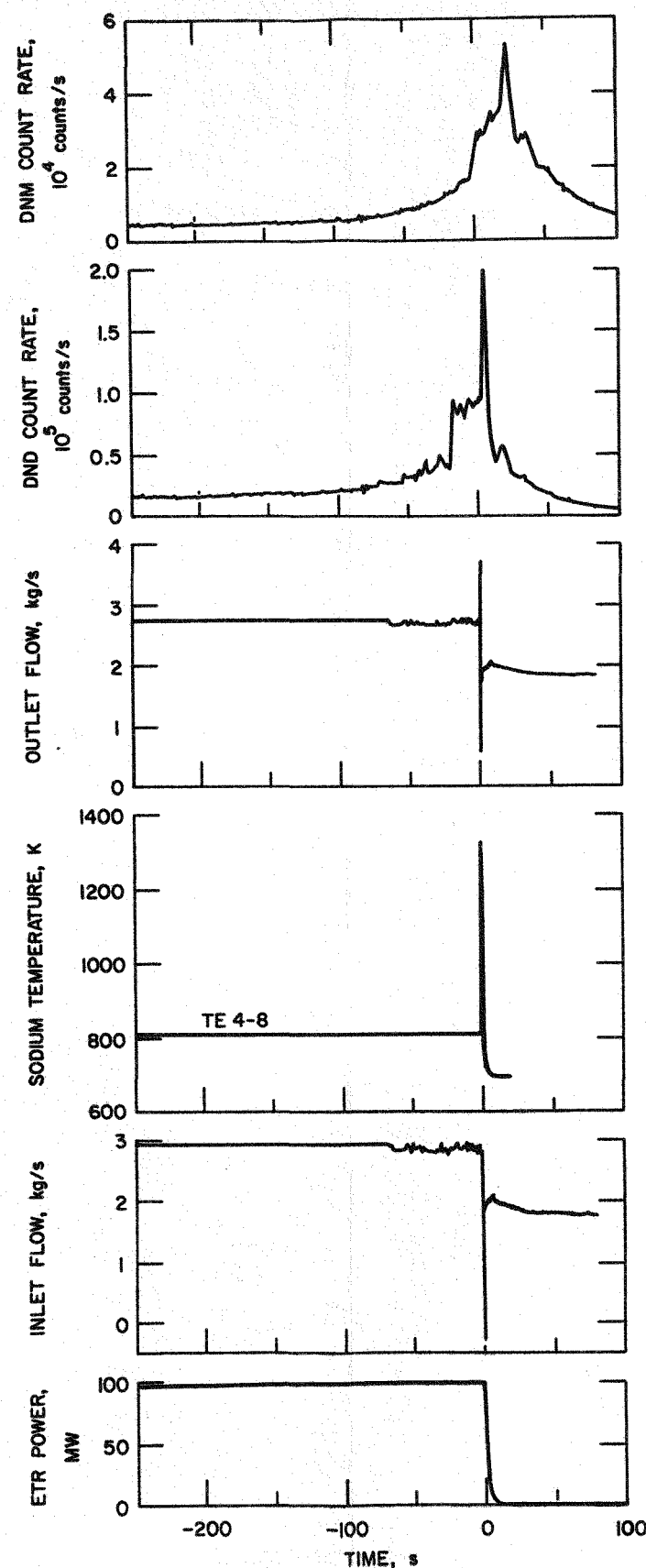


Fig. 28. Instrument Response During the Blockage Reconfiguration

Temperature perturbations of up to 15 K were indicated by thermocouples TE 3-1, TE 3-3, TE 3-9, and TE 4-1 coincident with the beginning of a sharp increase in the DND signal 21 s before the blockage reconfiguration. Fifteen s later the DNM signal also began to increase rapidly. About 5 s 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 0.5 s before the reconfiguration. This was followed by temperature increases at thermocouples TE 3-9 and TE 3-10 on Pin 8.

Basic characteristics of the blockage reconfiguration appeared similar to those for molten fuel release from the cylindrical fuel canister on Pin 5 and the fluted fuel canister on Pin 12 during the power transient. There were perturbations in temperature, pressure, flow, acoustic noise, and DN signal. A bundle inlet-flow deceleration was followed by a persisting flow reduction. Inlet flow drifted down to 2.55 kg/s, then dropped from 2.55 kg/s to 1.75 kg/s followed by a partial recovery to 2.25 kg/s. Flow held at 2.25 kg/s for ~0.35 s prior to a spike down to ~0.35 kg/s and recovery again to 2.25 kg/s. The fuel-bundle coolant flow rate steadied at 60%; this was accompanied by an ETR scram on low fuel-bundle inlet flow (PF-H). The ~0.4 s between the increase in the bundle flow resistance and a subsequent inlet-flow perturbation (that did not produce a further flow offset) was also observed during the P4 power transient on August 21.

Thermocouple TE 4-6 (Pin 32), which had a junction below the lower end of the fuel column, indicated a temperature jump of 320 K. This response indicated that molten fuel or hot fuel debris had contacted the sheath and formed a new junction. Although thermocouple TE 4-8 (Pin 35) had failed earlier and sodium entering the sheath failure had moved the thermocouple junction upward, it also indicated a substantial temperature spike, 510 K. These thermocouple responses indicated that fuel material moved outward toward hex-can Flats 3 and 4.

Following the blockage reconfiguration during this Cycle four of P4 irradiation, the coolant flow rate through the fuel bundle was below the low-flow setpoint stipulated in the extended safety envelope. No further major power operation was scheduled in order to preserve the condition of the fuel bundle, after three major molten-fuel-release/fuel-material-movement events, for the posttest examination.

## 2. Sensor Response and Evaluation

A summary of the instrument response to the blockage reconfiguration is shown in Fig. 28. A detailed chronology of the instrument response is summarized in Table 10. The response of P4 test train instrumentation is illustrated in the following pages. Instrument-response data are contained in fiche form in an envelope inside the back cover of this report.

The DAS systems were in operation in slow or standby mode at the time of the blockage reconfiguration. The primary DAS remained in slow mode for the entire event. The secondary DAS, using its disk-save feature, effectively recorded events until 14 seconds before scram in slow mode; the reconfiguration and post-scram events were recorded in fast mode. The OLCS DAS was in a standby recording mode.

Table 10. Chronology of Instrument Response  
During the Blockage Reconfiguration

| Time<br>s | Event and/or Instrument Response                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                            |
|-----------|-----------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------|
| -210      | Normalized DN signal = 1.00<br>DND signal began a gradual increase.                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                         |
| -180      | Normalized DND signal = 1.12, normalized DNM signal = 1.00.<br>DNM signal began a gradual increase.                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                         |
| -150      | Normalized DND signal = 1.21, normalized DNM signal = 1.08.                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                 |
| -100      | Normalized DND signal = 1.38, normalized DNM signal = 1.18.                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                 |
| -50       | Normalized DND signal = 2.13, normalized DNM signal = 1.83.                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                 |
| -21       | Normalized DND signal = 2.65, normalized DNM signal = 2.65.<br>Temperature perturbations of up to +15 K indicated by thermocouples TE 3-1, TE 3-3, TE 3-9, and TE 4-1. DND signal began to increase rapidly.                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                |
| -19       | DND signal reached a plateau.<br>Normalized DND signal = 5.5, normalized DNM signal = 2.75.                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                 |
| -6        | DNM signal began to increase rapidly.<br>Normalized DND signal = 5.5, normalized DNM signal = 3.5.                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                          |
| -0.5      | Increase in rate of change of inlet flow.<br>Normalized DND signal = 5.5, normalized DNM signal = 6.2.                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                      |
| -0.16     | Temperatures indicated by TE 3-9 and TE 3-10 began to increase.                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                             |
| 0-0.33    | <u>Blockage reconfiguration occurred.</u> Thermocouples TE 4-8 and TE 4-6 indicated "instantaneous" temperature spikes of 510 K and 320 K, respectively. TE 3-8 indicated an upward temperature spike from 885 K to 1000 K, then dropped back to 935 K. Thermocouples TE 4-7, TE 3-9 and TE 4-5 indicated upward temperature offsets of 60 K, 55 K, and 20 K, respectively. Fuel-bundle inlet flow, which had drifted down to 2.55 kg/s, dropped abruptly from 2.55 kg/s to 1.75 kg/s, then oscillated about 2.25 kg/s. Outlet flow increased momentarily, then also dropped to the 2.25 kg/s level. Pressure sensors on Pins 3, 4, and 11 (all touching the fluted fuel canister on Pin 12) indicated changes in pressure. |
| 0.33-0.6  | Sharp flow perturbations occurred. Fuel-bundle inlet flow dropped from 2.25 kg/s to -0.35 kg/s, then return to 2.25 kg/s. Fuel-bundle outlet flow increased to 3.65 kg/s, then dropped to 1.5 kg/s before recovering to 2.25 kg/s. Loop flow jumped from 7.1 kg/s to 9.4 kg/s, then dropped to 6.3 kg/s before returning to 7.1 kg/s. TE 3-8 indicated an upward temperature spike to 1025 K. Other thermocouples indicated that local sodium temperatures were still rising.                                                                                                                                                                                                                                               |

- 0.6-1.0 ETR scrambled on a low fuel-bundle flow (PPS-PF-H) trip and the pump transferred to emergency power. Pump power was off about 40 ms; fuel-bundle flow dropped to zero during the pump power-off (transfer) interval, then reached 2.1 kg/s after pump operation resumed at a lower power input. Most thermocouples indicated that coolant temperature peaked at ~1 s. TE 3-10 indicated a peak temperature of 1225 K (temperature increase of 350 K).
- 4 DND signal peaked at a normalized value of 11.5, then dropped sharply.
- 22 DNM signal peaked at a normalized value of 11.2, then dropped sharply.

a. Reactor power

Figure 29 shows the ETR power history prior to the blockage reconfiguration. The increase and partial power reduction at 0645 on October 2 was due to a secondary coolant (helium) system malfunction during the power increase. The increase from 80 MW terminated at 1024 at a level of approximately 92 MW. At 1034 (3 minutes prior to reconfiguration), a power change was initiated to bring ETR power to the specified 100 MW. At the time of scram, the ETR power was being trimmed at 100 MW.

b. Temperature within the fuel bundle and hex-can

Temperatures measured at the bundle inlet and outlet responded predictably to the blockage reconfiguration. The bundle inlet temperature remained at 698 K through reactor scram, then dropped slowly due to continued effects of helium-system operation and the absence of gamma heating of the loop structure. The bundle outlet temperature was steady at 940 K, rose to 980 K as a result of the flow reduction and perturbation accompanying the reconfiguration, then dropped smoothly to 700 K during a 20-s interval following scram.

Evaluating the local temperature response within the fuel bundle was complicated by the prior failure of a number of thermocouples. On failure, some thermocouple sheaths were open to the sodium. Sodium had "wicked" its way upward in the ceramic insulator in those thermocouples, which elevated the effective location of the thermocouple junction. Table 11 contains an appraisal of the condition of thermocouples known to be failed prior to the blockage reconfiguration. Also shown are the elevations of the original junctions of the thermocouples and the subsequent elevations of the junctions of thermocouples with sodium in the sheath, as inferred from resistance measurements. Other thermocouples may also have been impaired. Some thermocouples thought to have failed earlier appeared responsive.

Temperature perturbations of up to 15 K were indicated by thermocouples TE 3-1 (Pin 1), TE 3-3 (Pin 4), TE 3-9 (Pin 8), and TE 4-1 (Pin 21) coincident with the beginning of a sharp increase in the DND signal 21 s before the blockage reconfiguration. Shortly before the reconfiguration, temperatures indicated by thermocouples TE 3-9 (Pin 8) and TE 3-10 (Pin 8) began to increase. The first perturbation in bundle and loop flow was accompanied by temperature spikes of 510 K, 320 K, and 115 K, respectively, in the signals from thermocouples TE 4-8 (Pin 35), TE 4-6 (Pin 32), and TE 3-8 (Pin 7). Thermocouples TE 4-7 (Pin 34), TE 3-9 (Pin 8), and TE 4-5 (Pin 30) indicated upward temperature offsets of 60 K, 55 K, and 20 K, respectively. About 0.35 s later, a sharp flow perturbation occurred. Thermocouple TE 3-8 (Pin 7) indicated an upward temperature spike to 1025 K which accompanied a general increase in local sodium temperature within the bundle. Thermocouples TE 3-2 (Pin 3), TE 3-12 (Pin 12), TE 3-13 (Pin 12), TE 3-14 (Pin 14), and TE 4-9 (Pin 37) appeared insensitive to the flow perturbations. Most of these were known to have sodium in the sheath; all but TE 4-9 (Pin 37) indicated a steady 790 K at 100 MW and dropped smoothly to 700 K following scram.

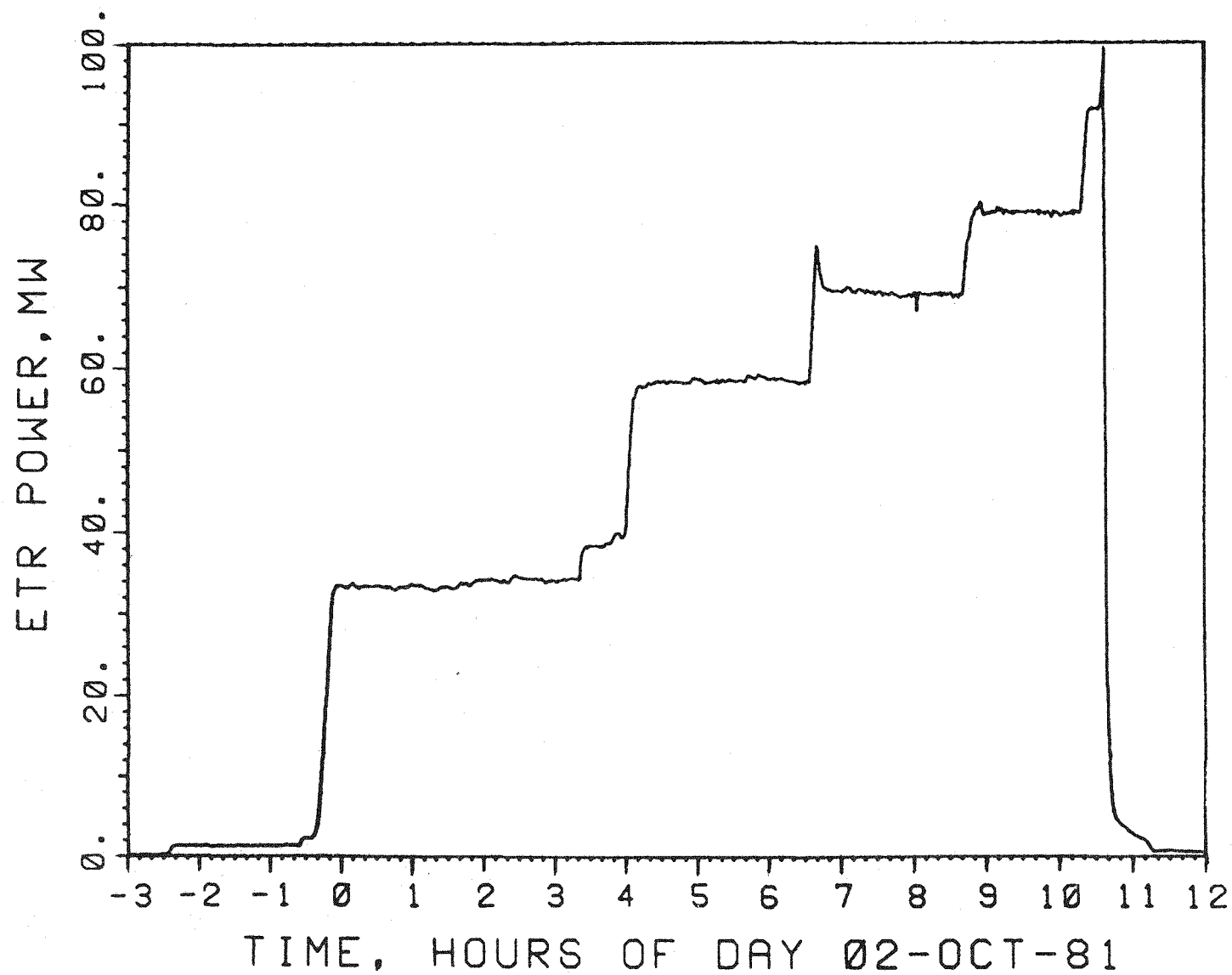


Fig. 29. ETR Power-Time History During Return-to-Power Operations

Table 11. Failed Thermocouples in the Fuel Bundle Prior to the Blockage Reconfiguration

| Thermocouple | Nature of Failure<br>or Symptom | Elevation of Junction*<br>m |         |        |
|--------------|---------------------------------|-----------------------------|---------|--------|
|              |                                 | original                    | 8/22/81 | 9/9/81 |
| TE 3-2       | sodium in sheath                | -0.26                       | 0.40    | 3.45   |
| TE 3-4       | intermittent                    | 0.25                        |         |        |
| TE 3-5       | erratic behavior                | 0.09                        |         |        |
| TE 3-7       | erratic behavior                | 0.09                        |         |        |
| TE 3-12      | sodium in sheath                | 0.09                        | 1.72    | 3.77   |
| TE 3-13      | sodium in sheath                | 0.25                        | 3.28    | 4.56   |
| TE 3-14      | sodium in sheath                | -0.36                       | 0.41    | 3.02   |
| TE 3-15      | sodium in sheath                | -0.16                       | -0.16   | 0.40   |
| TE 4-4       | open circuit                    | 0.45                        |         |        |
| TE 4-7       | sodium in sheath                | -0.21                       | 0.03    | 1.14   |
| TE 4-8       | sodium in sheath                | -0.61                       | -0.61   | 0.22   |
| TE 4-9       | alumel wire grounded            | 1.62                        |         |        |
| TE 6-1       | erratic behavior                | 0.14                        |         |        |
| TE 6-2       | open circuit                    | 0.14                        |         |        |
| TE 6-6       | shorted                         | 0.14                        |         |        |

\* measured from fuel midplane

Sodium temperatures reached a peak in the bundle ~1 s after scram. The largest temperature increase was measured by thermocouple TE 3-10 (Pin 8), which reached a peak temperature of 1225 K (temperature increase of 350 K). Other temperature peaks and temperature changes were as follows:

| Thermocouple         | Max. Temperature<br>K | Temperature Rise<br>K |
|----------------------|-----------------------|-----------------------|
| 3-1 (Pin 1)          | 1020                  | 60                    |
| 3-3 (Pin 4)          | 1070                  | 120                   |
| 3-8 (Pin 7)          | 1020                  | 140                   |
| 3-9 (Pin 8)          | 940                   | 70                    |
| 3-11 (Pin 10)        | 970                   | 45                    |
| 3-15 (Pin 18)        | 970                   | 45                    |
| 6-3 (hex-can Flat 3) | 1015                  | 45                    |
| 6-4 (hex-can Flat 4) | 1000                  | 110                   |
| 6-5 (hex-can Flat 5) | 905                   | 45                    |

Figure 30 shows thermocouple responses to the blockage reconfiguration. The apparently erratic signals before -14 s are due to the slow-mode sampling program which was active only 0.1 s each second.

### c. Sodium flow

Sharp flow perturbations were evident in all flowmeter signals, as shown in Fig. 31. The fuel-bundle inlet flow drifted down to 2.55 kg/s, then dropped from 2.55 kg/s to 1.75 kg/s and recovered to 2.25 kg/s. Outlet flow increased momentarily, then also dropped to the 2.25 kg/s level. A reduction in flow through the loop was also evident. The flow rate through the fuel bundle oscillated around 2.25 kg/s for ~0.35 s, followed by a major perturbation. Inlet flow spiked down to -0.35 kg/s, outlet flow spiked up to 3.65 kg/s, and loop flow spiked up to 9.4 kg/s before a temporary recovery. This was followed by a shallow, but broader downcycle in coolant flow rate. Fuel-bundle coolant flow rate steadied at 60%; this was accompanied by an ETR scram. During the 40-ms duration of the pump power transfer, the fuel-bundle inlet flow dropped to zero, then reached 2.1 kg/s after pump operation resumed. There was no evidence of any gas release in the bundle-exit or loop flowmeter signals.

## 3. Response of Failed-Fuel Detection Systems

When reactor operation was resumed at 40 MW, the count rate in the DND and the DNM agreed well with those at the end of the previous period of

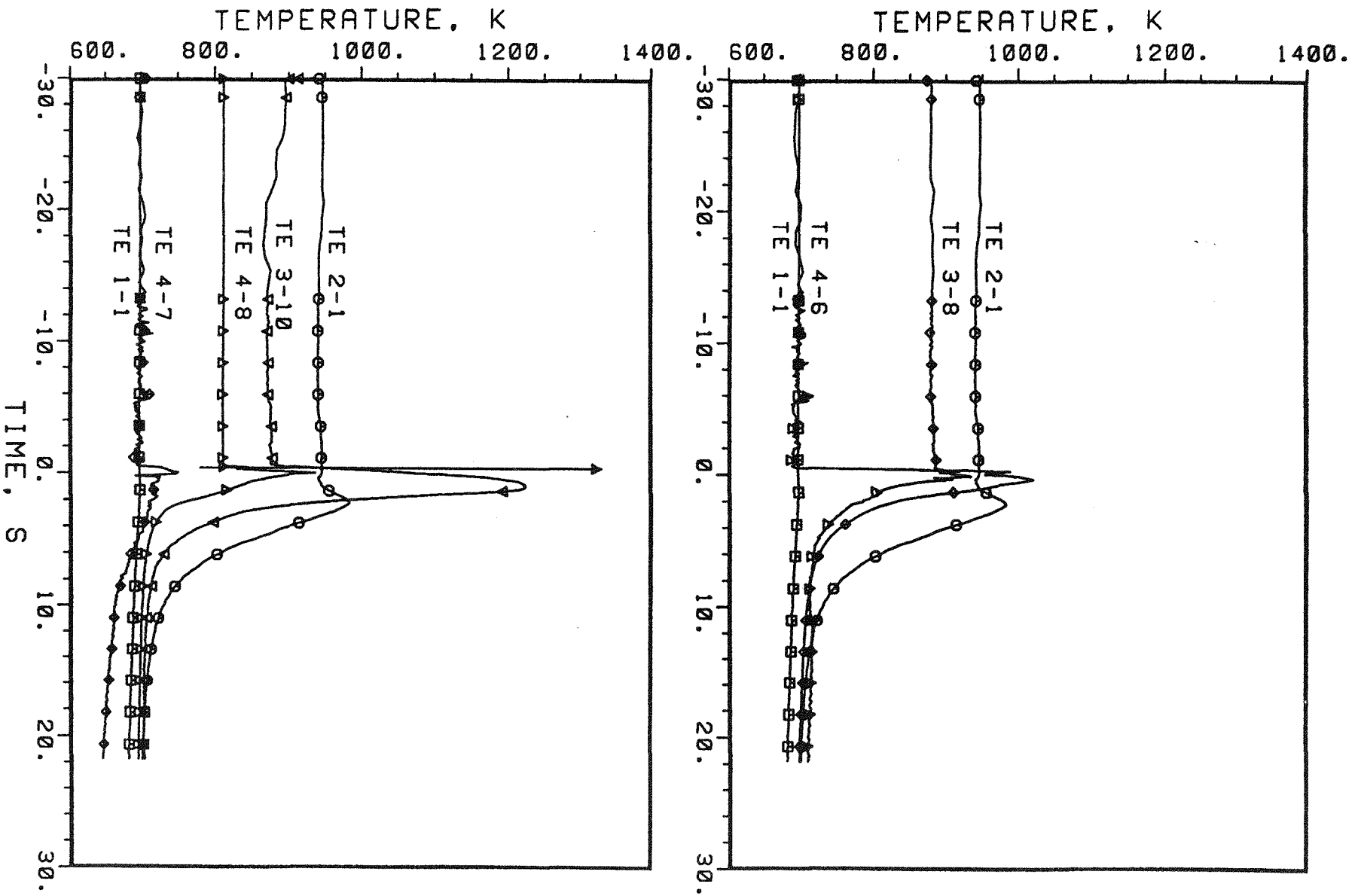


Fig. 30. Fuel-Bundle Thermocouple Response to the Blockage Reconfiguration



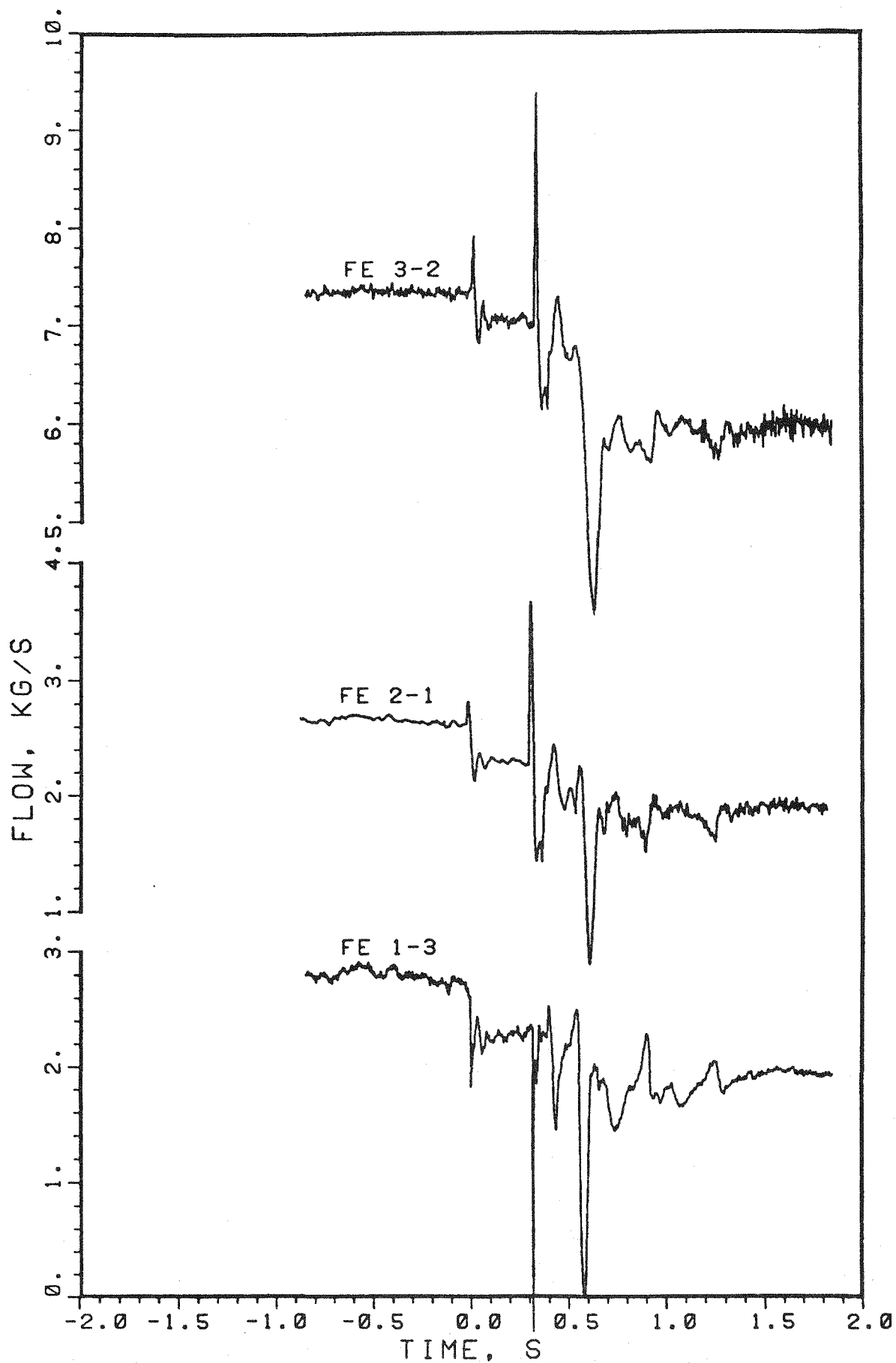


Fig. 31. Sodium Flow During the Blockage Reconfiguration

operation. This indicated that the area of exposed fuel in the fuel bundle had not undergone any major change during the 25 days at zero power.

During the four-h hold at 40 MW (326-kW fuel-bundle power), the DNM count rate was recorded as a function of OLSS flow rate, in the same fashion as during the DN mechanisms tests, and the same was done during the two-h holds at ETR power levels of 60, 70 and 80 MW. These data are included in Fig. 32, as are the previous DNM data obtained between 2.2 and 40 MW.

In all of the data taken up to a power of 80 MW, there was no indication of any variation from a normal increase of DN signal proportional to reactor power. A possible small effect which can be attributed to the increasing sodium temperature will be discussed later. The DND response deviated from proportionality at a reactor power of 92 MW, as shown in Fig. 33. Because this was not a scheduled hold point, no flow-variation data were obtained from the DNM.

Shortly after, with reactor power near 100 MW, the DN signals from both the DND and DNM began to increase slowly, as shown in Figs. 27 and 28. The change was very slow; the DND rate increased by a factor of 2.5 in about three minutes. This change was not accompanied by perturbations in signals from any other sensor (Fig. 28), which indicating that there was not, up to this point, any detectable mechanical change taking place in the fuel. At the end of this gradual change, there was a further sudden increase by a further factor of two, to a level five times the original steady-state value. This level was maintained, with some fluctuation in the case of the DND, for 21 s. Figure 28 still indicates essentially no disturbance from any other sensor. Finally, a sudden increase to eleven times the original DN level was accompanied by major disturbances in sodium temperature and sodium flow rate, which caused a reactor scram. The final sharp peak in the DN signal was that to be expected from a sudden expulsion of fuel. Because a reactor scram occurred, the signature is less characteristic than the peaks in the earlier power transient, when continued operation showed also the continuous signal from the exposed mass of fuel. The additional peaks after the main peak are simply due to the recirculation of the DN emitters in the loop. This DN record is an unambiguous indication of a change in the partial flow blockage that began slowly and ended in a rearrangement of the fuel debris.

#### E. Concluding Operations at Low Power

##### 1. Fuel-Pin Pressure Sensor Test

A loop plenum pressure test to identify any additional failed pins via response of the fuel-pin plenum pressure sensors was performed on October 15. Loop plenum pressure was varied between 90 kPa (13 psia) and 255 kPa (37 psia). Pin 19, thought to be intact following the 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-can Flats 3 and 4.

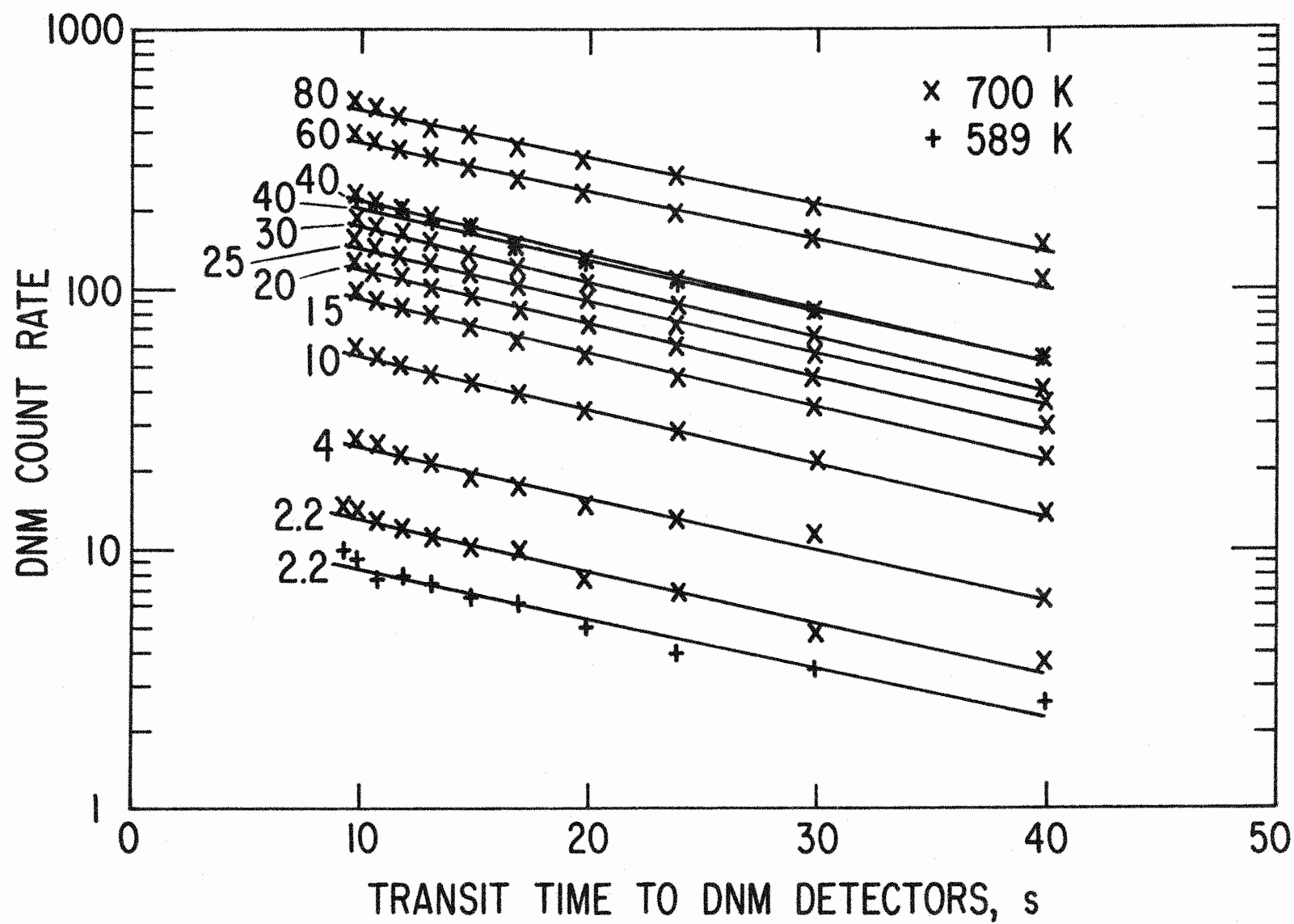


Fig. 32. DNM Signal vs. Delay Time as a Function of Reactor Power

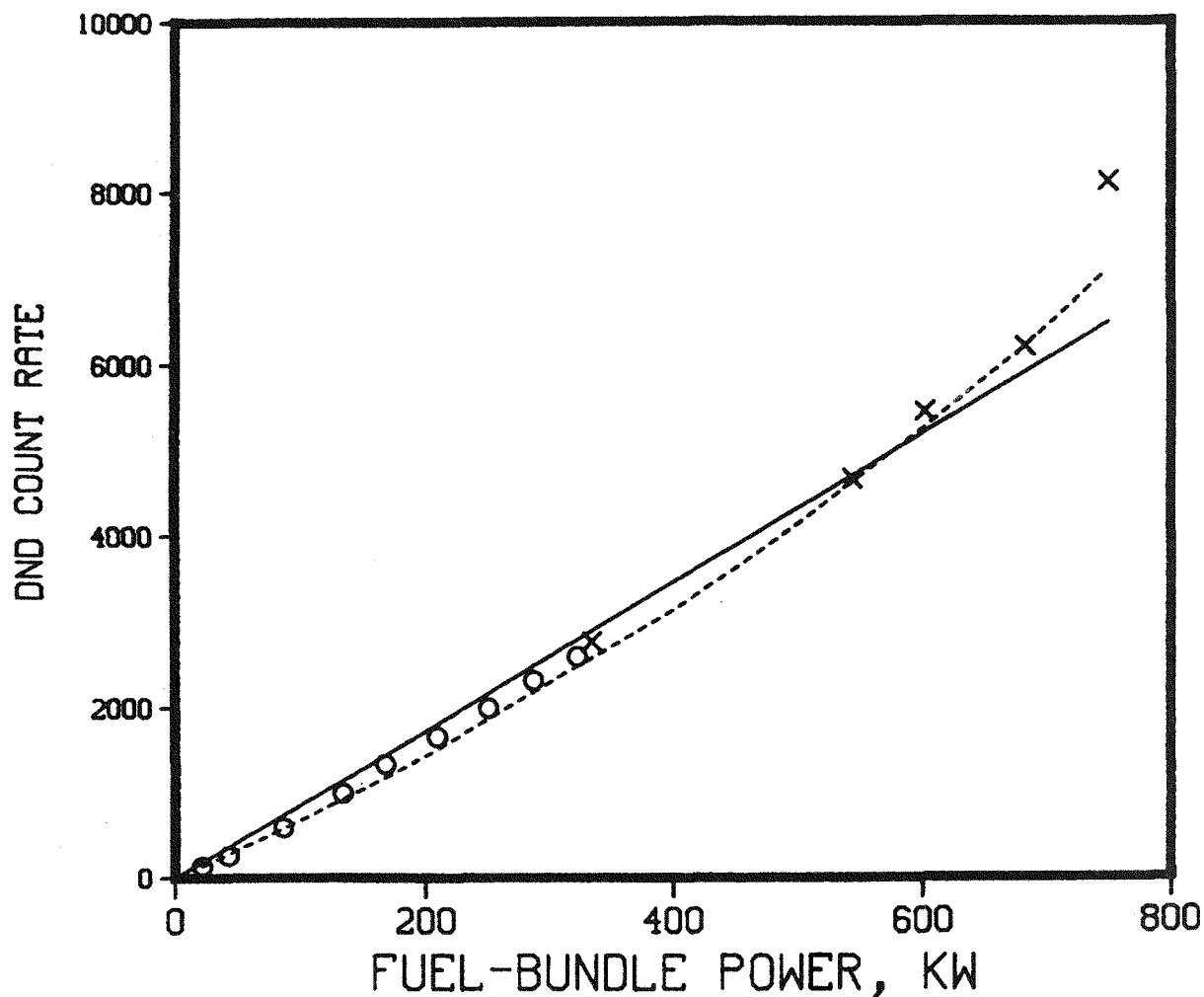


Fig. 33. DND Signal as a Function of Fuel-Bundle Power

## 2. Delayed Neutron Mechanisms Test

Because the coolant flow rate through the fuel bundle was below the low-flow setpoint stipulated in the extended safety envelope, low-power operation to study the coolant-temperature sensitivity of the DN signal from the larger partial flow channel blockage after the blockage reconfiguration concluded the experiment. This DN mechanisms test, performed on October 27 and 28, was the only reactor operation during P4 Cycle-five irradiation. The same measurements were repeated in the final DN mechanisms test as had been made in the earlier tests, but over a wider temperature range, from 700 K (800°F) to 458 K (365°F). The reactor and bundle power were lower than in the first tests and some uncertainty was introduced into the results by the difficulty in obtaining a precise value of the bundle power. The reactor power was held constant at ~1.3 MW. The corresponding fuel-bundle power was ~11 kW, but this

value decreased slowly by a total of ~40% during the period of the measurements (~24 hours); presumably this was due to local flux changes caused by control rod adjustments.

The amount of fuel exposed in the reconfigured blockage appeared to be larger than that in the first DN mechanisms test. The DN count rates were about the same as measured earlier, although the reactor and fuel-bundle power were lower. No attempt was made to go to higher power, but a very determined effort was made to go to the lowest possible sodium temperature. It was hoped to reach a temperature low enough that the DN signal would no longer be temperature dependent. This was not achieved; however, the range of the DND data was considerably extended.

The final DN mechanisms test started at a fuel-bundle sodium temperature of 700 K (800°F). Sodium temperature was decreased in 28 K (50°F) increments, followed by two-h holds to ensure equilibrium conditions had been

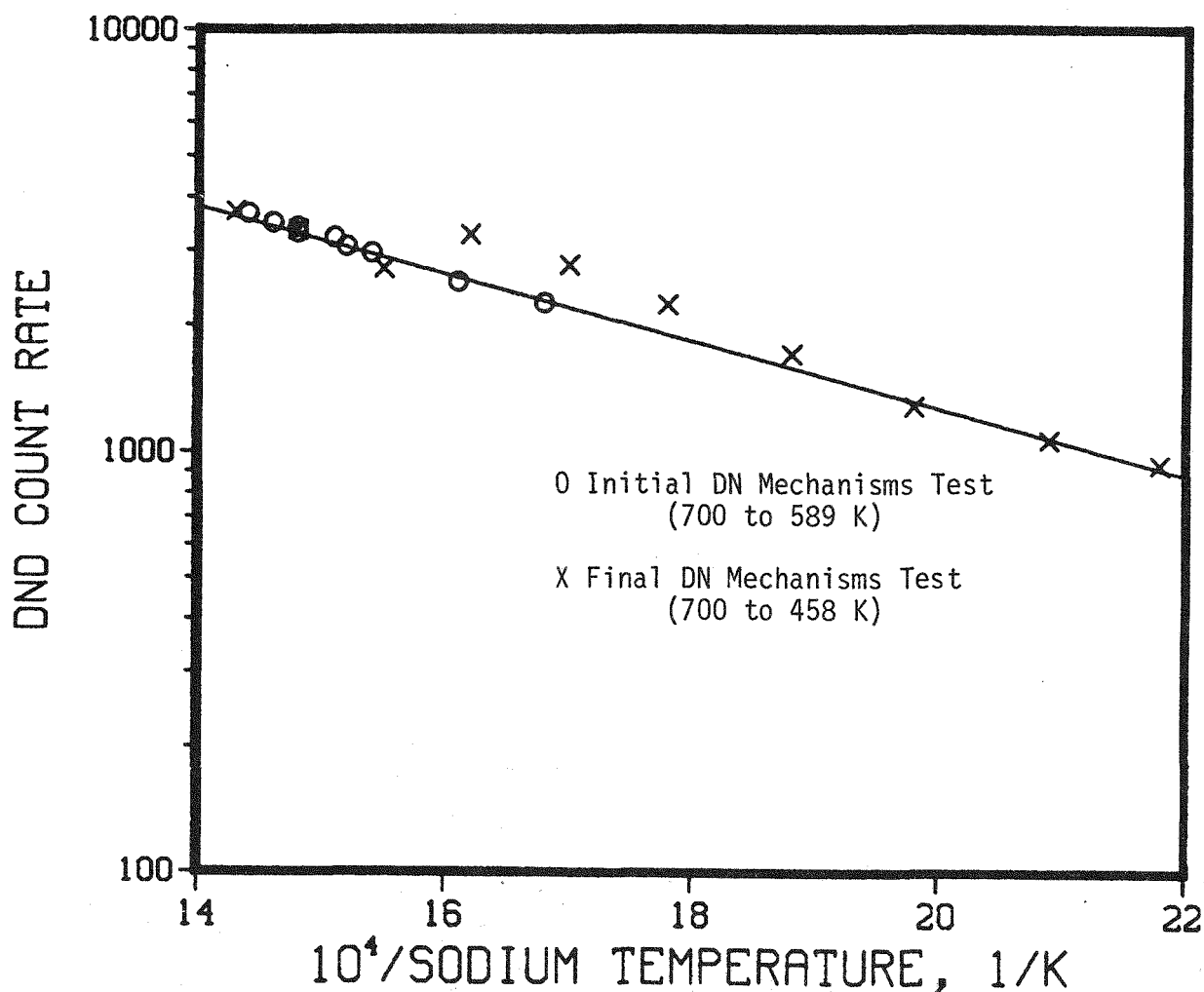


Fig. 34. DND Signal vs. Inverse Temperature (Wide Range)

established. It was intended to work down to a 450 K (350°F) sodium temperature. However, as the last temperature plateau was being approached, sodium flow in the OLSS became erratic. The lowest sodium temperature at which DND data were obtained was 458 K (365°F). These DND and DNM data are shown in Figs. 34 and 35.

As had been observed in the earlier DN mechanisms tests, sodium flow in the loop changed with changes in sodium temperature. The sensitivity in sodium flow to a changing sodium temperature in the pump was about 3% per 55 K (100°F). To maintain a constant sodium flow rate, it was necessary to decrease the pump voltage (and power) as sodium temperatures were reduced and to increase pump voltage during increases in sodium temperature.

The interpretation of test results from this DN mechanisms test, as well as that from the previous tests, is contained in Ch. VI.

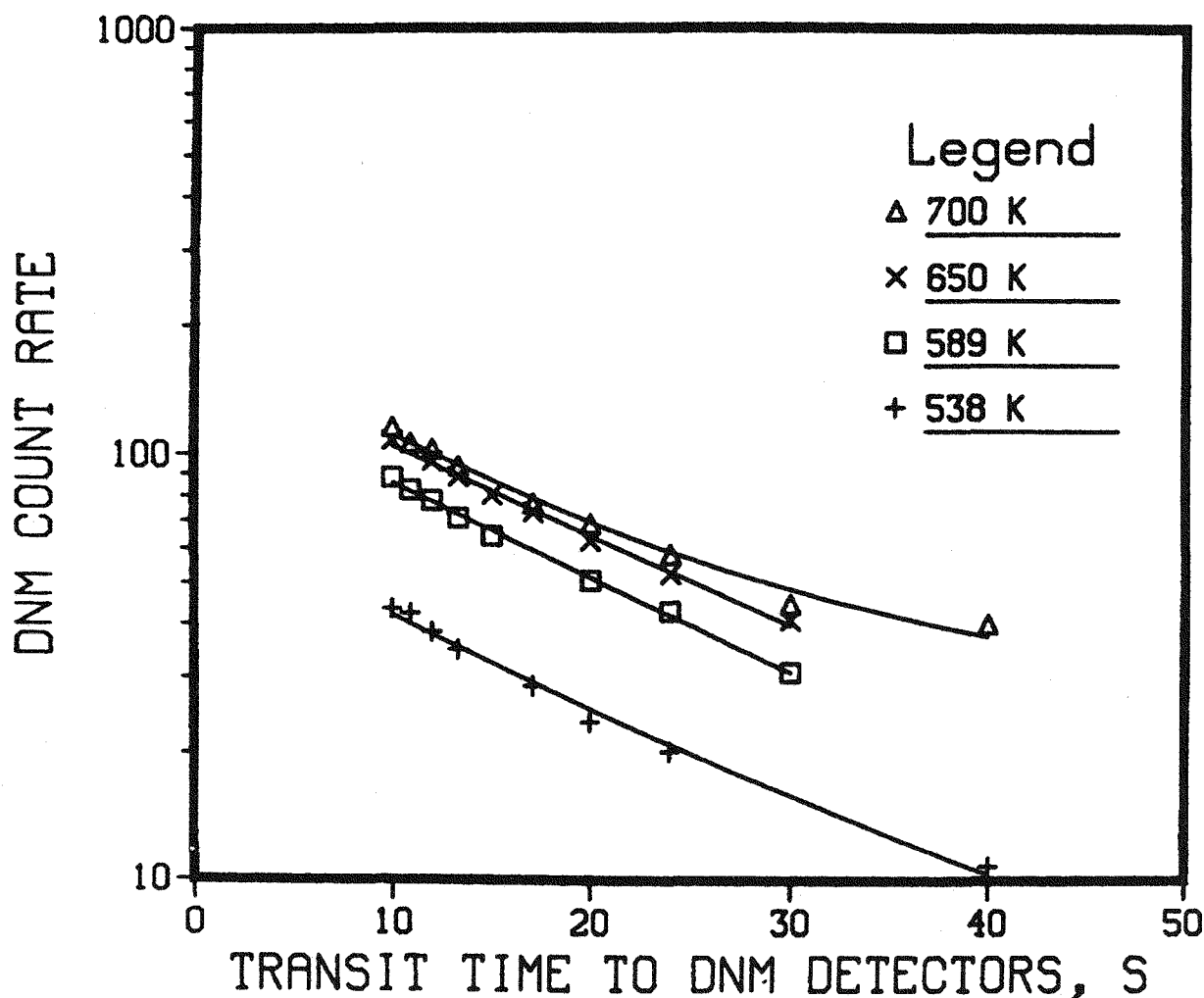


Fig. 35. DNM Signal vs. Delay Time as a Function of Sodium Temperature

78-100



## V. POSTTEST EXAMINATION

The distribution of materials and the condition of the components in the fueled portion of the P4 test train at the end of the in-reactor experimentation were determined by a comprehensive posttest examination (PTE) at the Alpha-Gamma Hot Cell Facility (AGHCF) at ANL-East following nondestructive examination and preliminary disassembly at the Hot Fuel Examination Facility (HFEF) at ANL-West.

The general methods used throughout the disassembly and examination of the P4 test train were essentially those used for the P3 test train described in Ref. 17. A quantitative description was made of the condition and distribution of all materials and components based on direct observations of the many transversely cut surfaces of thin sections from the fuel-bundle canister region. The extent and nature of sodium urano-plutonate formation in failed fuel was evaluated.

### A. Disassembly and Examination at HFEF (ANL-West)

#### 1. Loop Disassembly and Nondestructive Examination

The P4 loop and test assembly were transported from the ETR to the HFEF for initial disassembly and nondestructive examination. SLSF personnel developed comprehensive requirements for the P4 handling, disassembly, and examination while HFEF personnel were responsible for developing detailed operating procedures and equipment to satisfy those requirements. Goals of the HFEF-phase of the P4 PTE were to retrieve the fuel bundle by dismantling the loop and withdrawing the test assembly, to assess the macrocondition of the fuel bundle by nondestructive examination techniques, and to prepare the fuel bundle for shipment to the AGHCF at ANL-East.

The dismantling operation was initiated when the fission-product decay heat level had diminished to the point where removal of the test train from sodium into the HFEF argon atmosphere would not result in overheating the fuel or structural materials. A controlled sodium-freezing sequence (top to bottom) had been used to maximize the void fraction in the test section prior to removal of the loop from the ETR, thereby minimizing possible degradation of experimental evidence due to expansion of loop sodium during the remelt operation. A controlled sodium remelt sequence was used (also top to bottom) at the HFEF to decrease the stresses imposed on loop and test-assembly structures during sodium melting and expansion.

The test train was cut approximately in thirds as it was withdrawn from the loop following the sodium remelt. Sodium drained from the bottom test-train segment as it was removed from the loop. Several nondestructive examinations, including neutron radiography, isotopic gamma scanning, and visual examination, were performed to permit assessing the extent of fuel-bundle disruption so that procedures for the remainder of the disassembly could be defined. In addition, this information provided the first feedback from the PTE evaluating the actual response of the test-train components to the LMFBF local-fault simulation.



Full-length neutron radiographs were produced of the P4 fuel bundle at the Neutron Radiography Facility (NRAD) within the HFEF. The fuel bundle was radiographed from three azimuthal views (60° apart) aligned with the hex-can flats thus allowing sight down the rows of fuel pins. Figure 36 shows the 120° clockwise view of the NRAD neutron radiograph and an elevation scale that is used throughout this report to axially locate features of interest as positive distances above the midplane of the pretest fuel column and negative distances below. Also included in Fig. 36 are the HFEF and subsequent AGHCF sectioning diagrams, the original locations of the fuel-pin components, and the Nb-95 (fuel) and Co-58 (metal) isotopic gamma-ray scans.

The neutron radiographs provided evidence that a significant portion of the bundle flow area became blocked by fuel expelled from the fuel canisters and relocated from adjacent pins during the power transient. The 28-cm-long blockage region extended from 13 cm below to 15 cm above the fuel midplane; disrupted fuel contacted the inner surface of the hex-can at the fuel midplane.

## 2. Fuel-Bundle Disassembly

A 37-cm-long fuel-bundle segment, containing the entire blockage region, was cut from the P4 test section with a 2.4-mm thick, large-diameter aluminum wheel in a high-speed, abrasive cutoff machine. The work was not rotated, and no coolant was used. In order to reduce the spread of contamination within the HFEF, the in-cell vacuum system was attached to a makeshift plastic glove box that shrouded the work area. Sufficiently high temperatures were reached at the edge of the cutoff wheel to melt sodium and possibly to affect the distribution and composition of fission gases adjacent to the cuts so that materials from near the HFEF cuts were assumed to be non-representative of the real posttransient conditions. The AGHCF cutting method, to be described later, alleviated the problem of overheating.

Each cut surface was examined and photographed at 2X magnification. Protective caps were attached to the ends of the fuel-bundle segment; the segment was placed in an aluminum shipping container and transferred in a shielded cask to ANL-East for detailed examination at the AGHCF.

## B. Disassembly and Macroexamination at AGHCF (ANL-East)

### 1. Objectives and Methods

Two major uncertainties persisted after evaluation of the on-line instrument data and nondestructive examination results: namely, blockage morphology and composition, and the extent of failure in fuel pins adjacent to the UO<sub>2</sub>-filled fuel canisters.

Information was collected during the macroexamination phase of the PTE to permit assessment of the following:

- cladding disruption and relocation,
- fuel disruption and relocation in fuel canisters and fuel pins, and
- nature and extent of blockage formation.

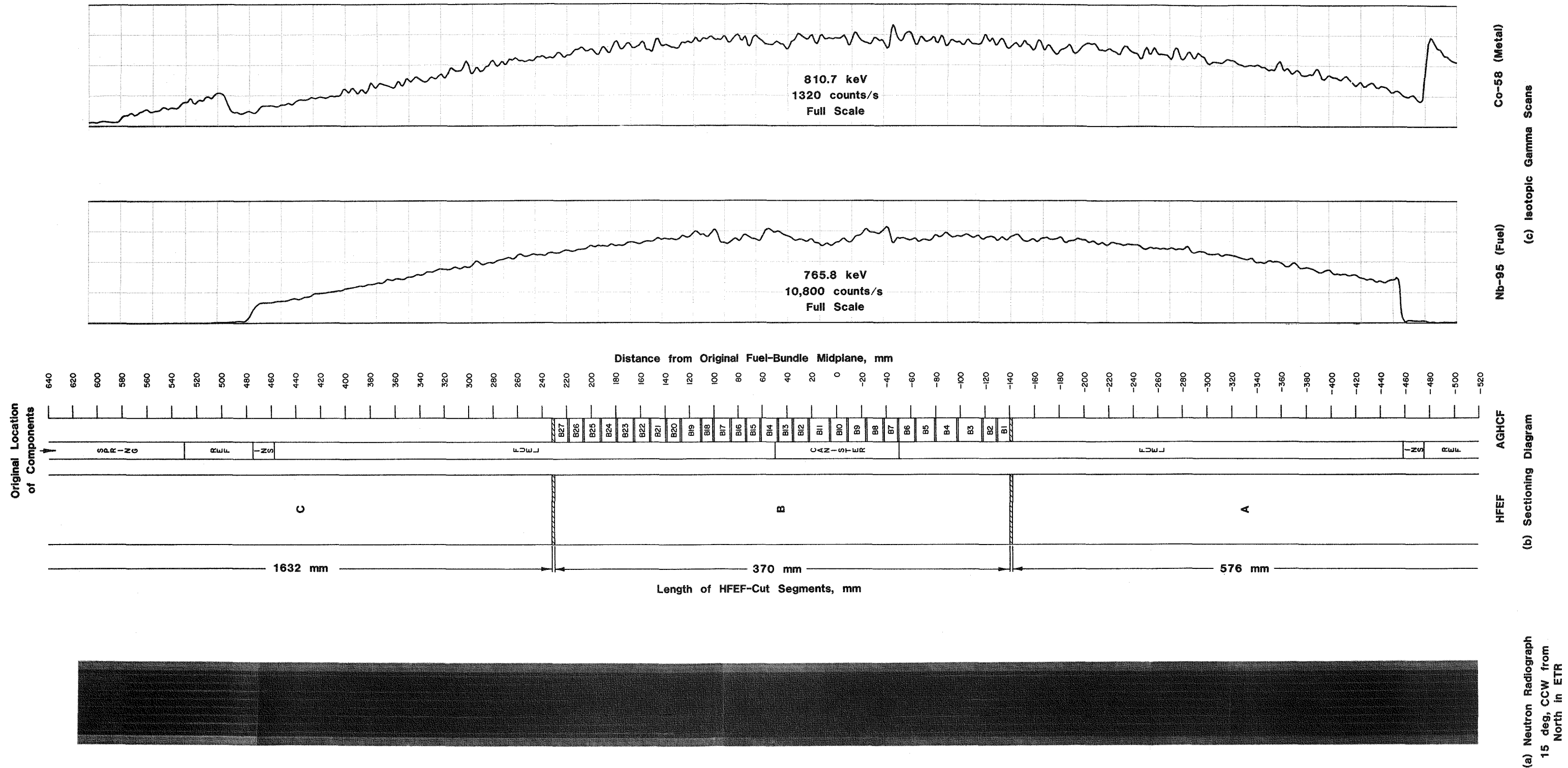


Fig. 36. Nondestructive Examination Results and Disassembly Diagram of the P4 Fuel Bundle

The detailed disassembly of the 37-cm-long fuel-bundle segment was done by the same transverse cutting method used so successfully in retaining the as-received condition and position of the fuel and structural materials in the disassembly of fuel bundles from previous SLSF experiments. Epoxy was used as a stabilizing agent to preserve the posttest condition and configuration of fuel-bundle components and disrupted materials during macroexamination of the blockage region.

A high-speed SiC cutoff machine was used to produce a series of closely spaced transverse cuts that divided the fuel-bundle segment into 27 thin cross sections. The cutoff machine had features that minimized heating the work, permitted accurate location of cuts, and resulted in smooth transversely cut surfaces. The epoxy-filled macrosections were metallographically ground to produce smooth, flat surfaces that were photographed at 2X magnification to permit critical, visual examination and quantitative evaluation of fuel-bundle disruption.

The axial position of the fuel-bundle sections, cut and prepared at the AGHCF, are illustrated in the cutting diagram of Fig. 36 as positive or negative distances above or below the fuel midplane. Careful examination of the 2X macrophotographs provided information concerning the blockage morphology and extent of failure in fuel pins adjacent to the fuel canisters. Alpha autoradiographs were produced from the metallographically ground macrosections to obtain information about the distribution of  $UO_2$  and mixed-oxide fuel in the coolant channels and fuel canisters.

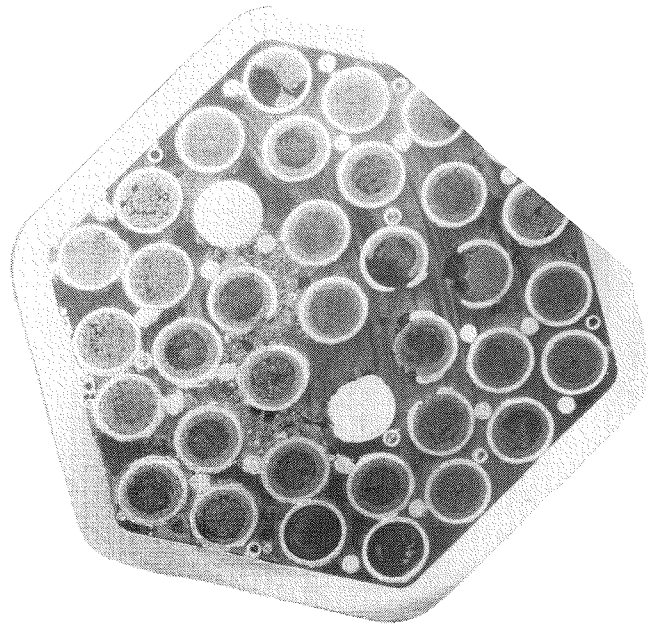
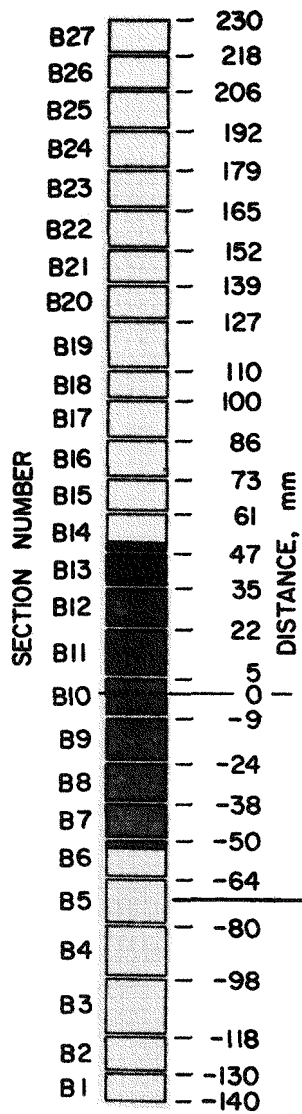
## 2. Observations

The 2X photographs and  $\alpha$ -autoradiographs of the transversely cut and metallographically ground macrosections served to document the posttest condition of the materials and components throughout the blockage region and were used as the basis for macroevaluations and the location of samples for microexaminations. The clarity of the macrophotographs and  $\alpha$ -autoradiographs shown in Fig. 37 permitted the evaluation of several major posttest features including the axial and radial extent of cladding and fuel disruption; the probable path of the significant fuel-motion events; the shape, extent and location of the partial coolant-channel blockage; the posttest condition of certain fuel-bundle components; and the presence of sodium urano-plutonate formation and subsequent dissociation.

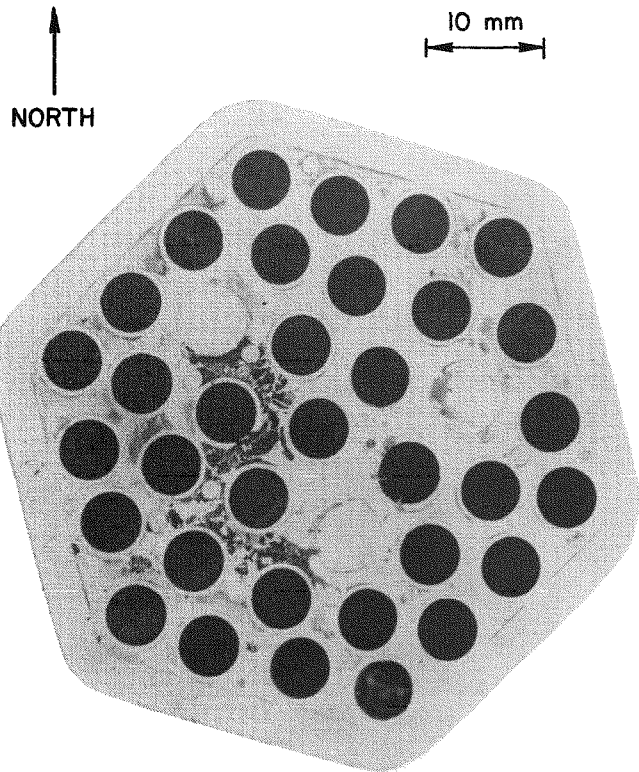
### a. Fuel-motion events

The three canisters were breached over their entire lengths with the 10% CW, row-of-12 cylindrical canister sustaining the least disruption at all elevations. Melting at the cusps of the fluted canister in macrosection B7, see Figs. 37 and 38(a), typifies the nature of its initial failure at 15 s. Most of the  $UO_2$  fuel filling the fluted canister remained at this elevation; however, nearly all of the fuel in the remainder of the fluted canister became disrupted and relocated so that only two tips of the canister remain at higher elevations, shown in macrosections B9, 10, and 11 of Fig. 37.

The 20% CW, row-of-six cylindrical canister failed at 34 s and ejected molten fuel radially outward toward two hex-can flats and axially upward between outer-row pins and along the hex-can wall, as illustrated in



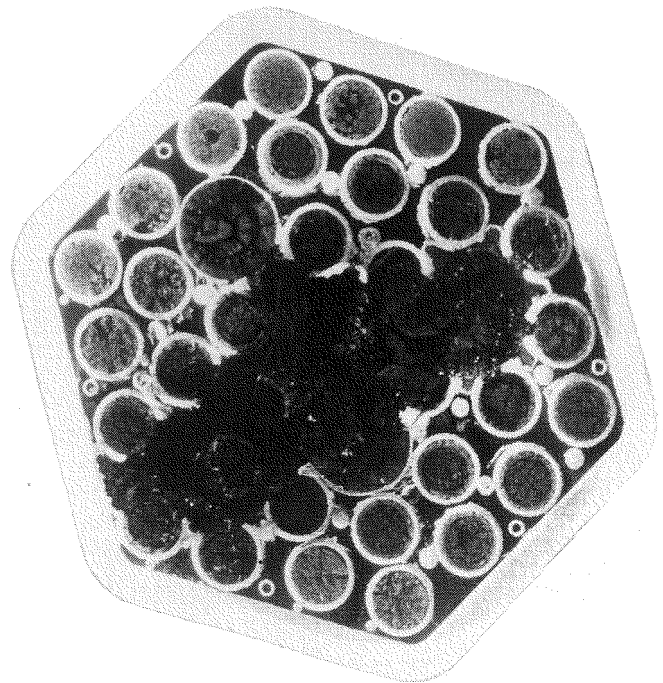
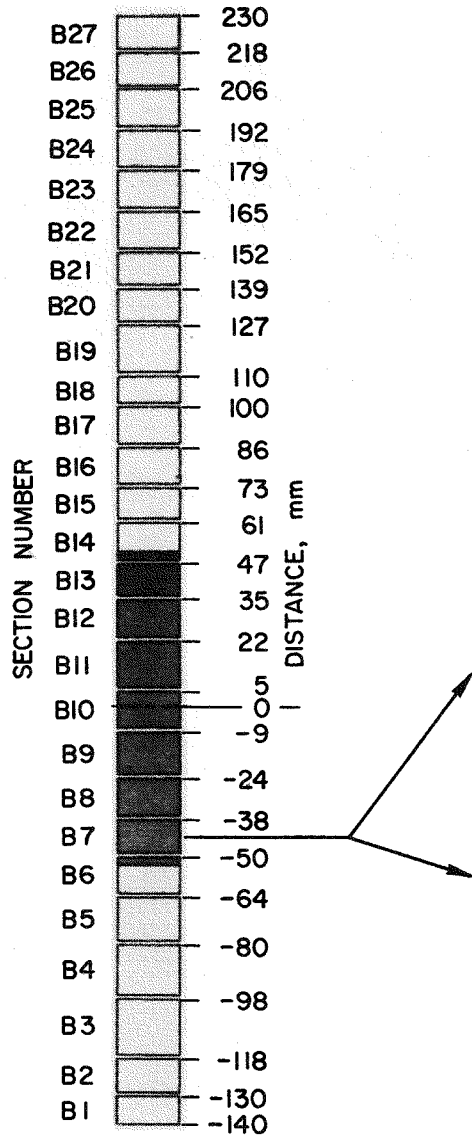
B5 MACROPHOTOGRAPH



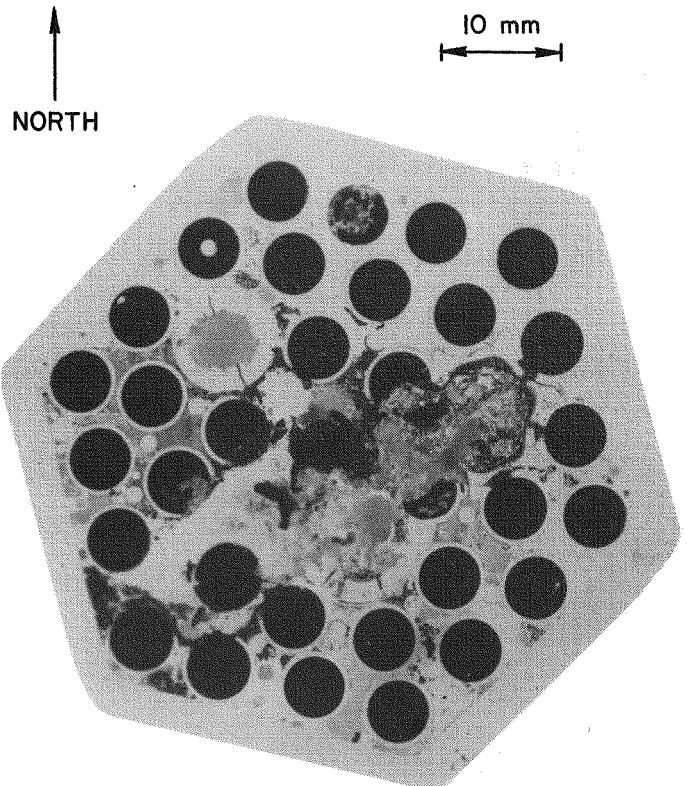
B5  $\alpha$  AUTORADIOGRAPH

(a)

Fig. 37. Selected Macrophotographs and  $\alpha$ -Autoradiographs of Transverse Surfaces Through the P4 Fuel Bundle



B7 MACROPHOTOGRAPH



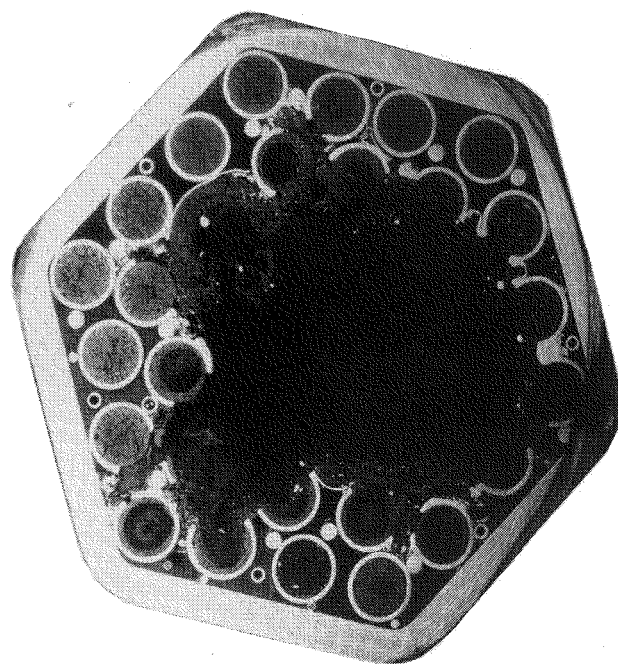
B7  $\alpha$  AUTORADIOGRAPH

Fig. 37b.

|     |      |
|-----|------|
| B27 | 230  |
| B26 | 218  |
| B25 | 206  |
| B24 | 192  |
| B23 | 179  |
| B22 | 165  |
| B21 | 152  |
| B20 | 139  |
| B19 | 127  |
| B18 | 110  |
| B17 | 100  |
| B16 | 86   |
| B15 | 73   |
| B14 | 61   |
| B13 | 47   |
| B12 | 35   |
| B11 | 22   |
| B10 | 5    |
| B9  | 0    |
| B8  | -9   |
| B7  | -24  |
| B6  | -38  |
| B5  | -50  |
| B4  | -64  |
| B3  | -80  |
| B2  | -98  |
| B1  | -118 |
|     | -130 |
|     | -140 |

SECTION NUMBER

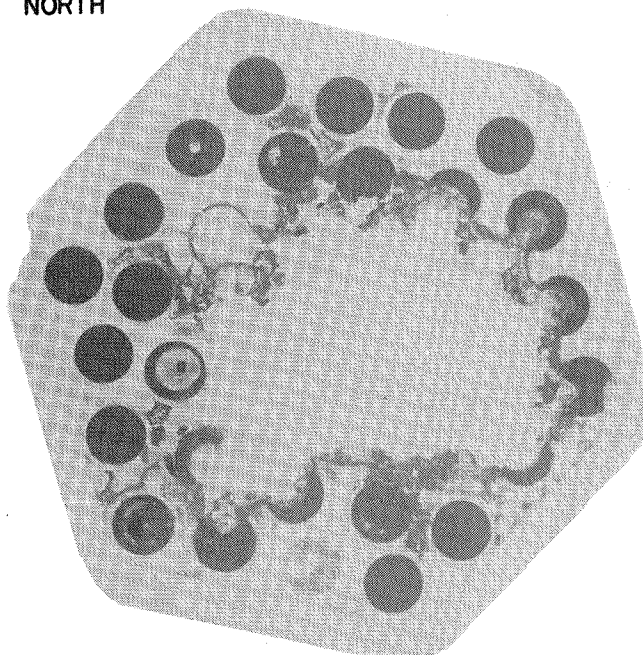
DISTANCE, mm



B9 MACROPHOTOGRAPH

NORTH ↑

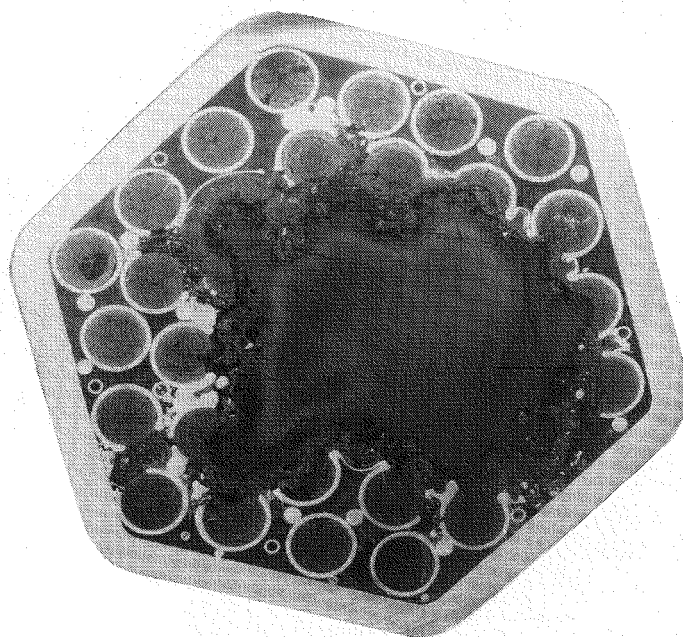
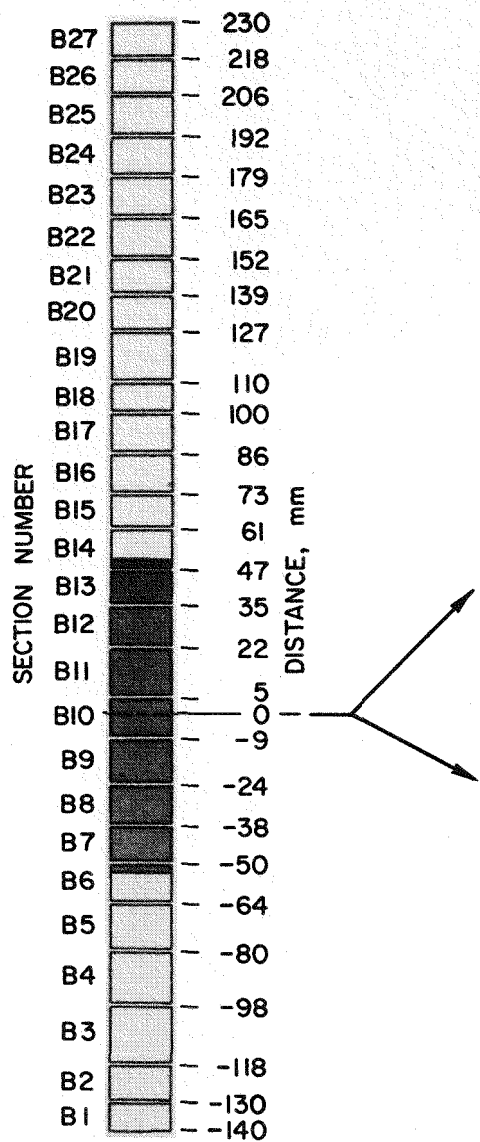
10 mm



B9  $\alpha$  AUTORADIOGRAPH

Fig. 37c.

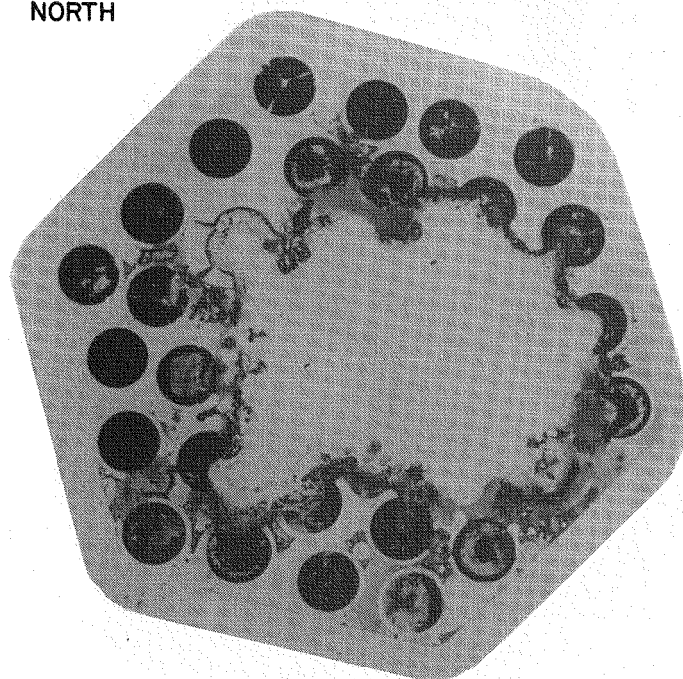




BIO MACROPHOTOGRAPH

NORTH  
↑

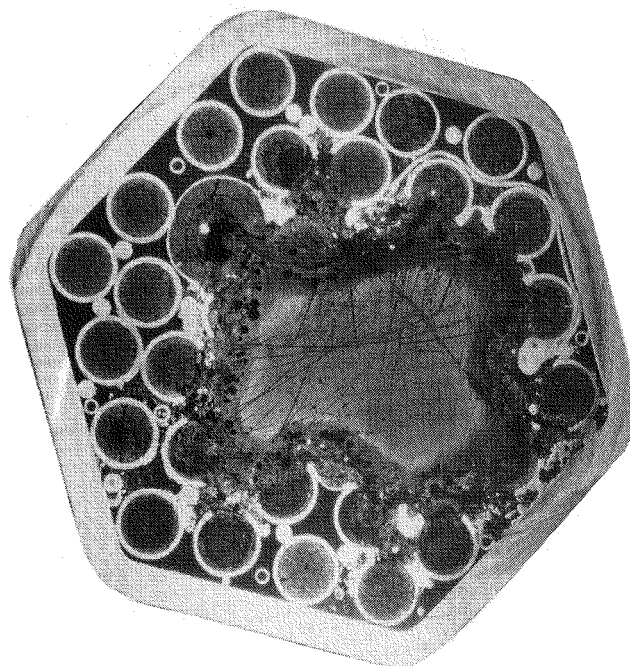
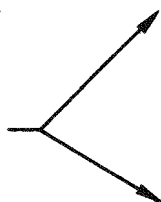
10 mm  
↔



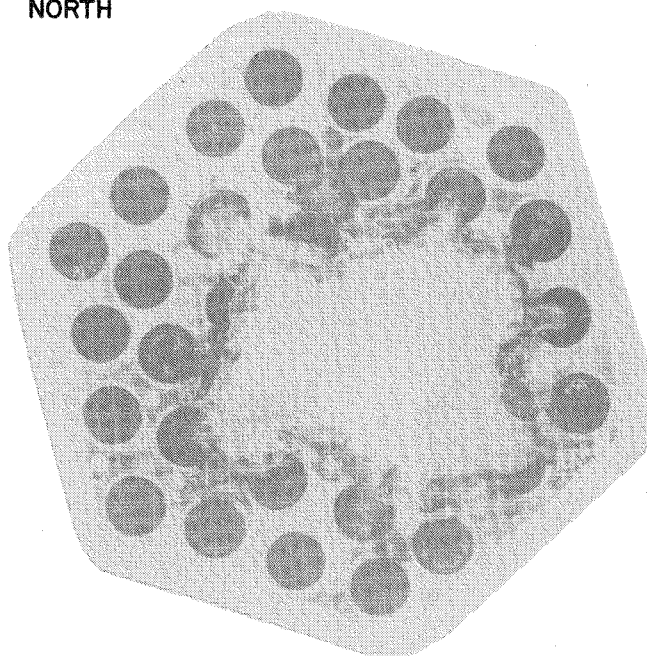
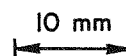
BIO  $\alpha$  AUTORADIOGRAPH

Fig. 37d.

| SECTION NUMBER | DISTANCE, mm |
|----------------|--------------|
| B27            | 230          |
| B26            | 218          |
| B25            | 206          |
| B24            | 192          |
| B23            | 179          |
| B22            | 165          |
| B21            | 152          |
| B20            | 139          |
| B19            | 127          |
| B18            | 110          |
| B17            | 100          |
| B16            | 86           |
| B15            | 73           |
| B14            | 61           |
| B13            | 47           |
| B12            | 35           |
| B11            | 22           |
| B10            | 5            |
| B9             | 0            |
| B8             | -9           |
| B7             | -24          |
| B6             | -38          |
| B5             | -50          |
| B4             | -64          |
| B3             | -80          |
| B2             | -98          |
| B1             | -118         |
|                | -130         |
|                | -140         |



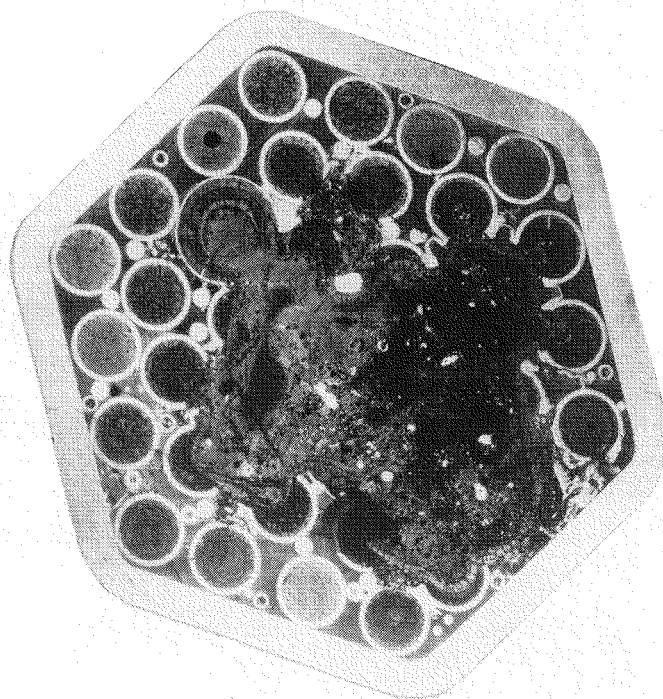
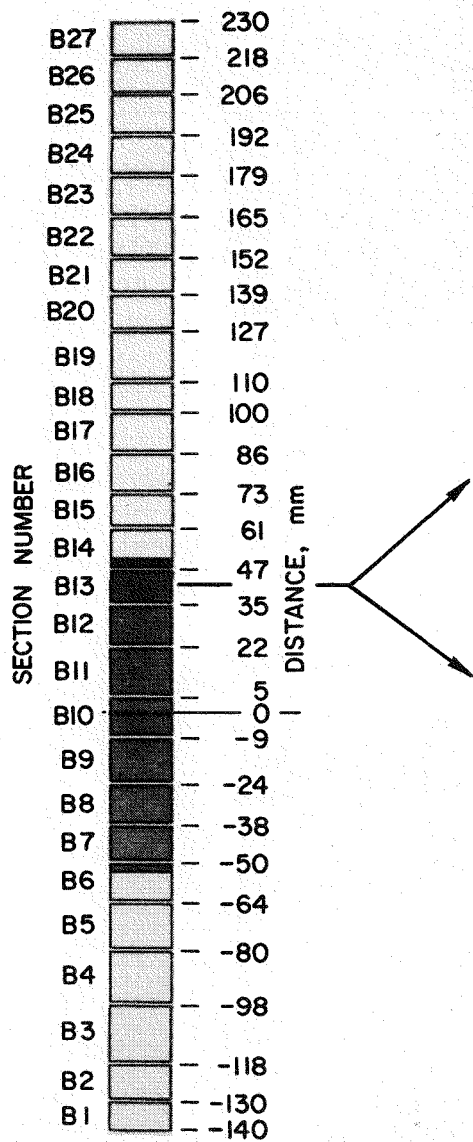
BII MACROPHOTOGRAPH



BII  $\alpha$  AUTORADIOGRAPH

Fig. 37e.

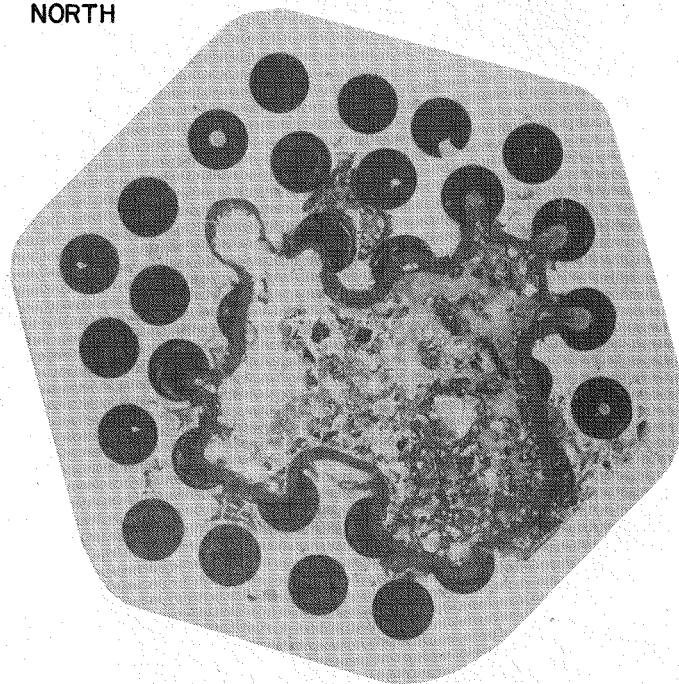




B13 MACROPHOTOGRAPH

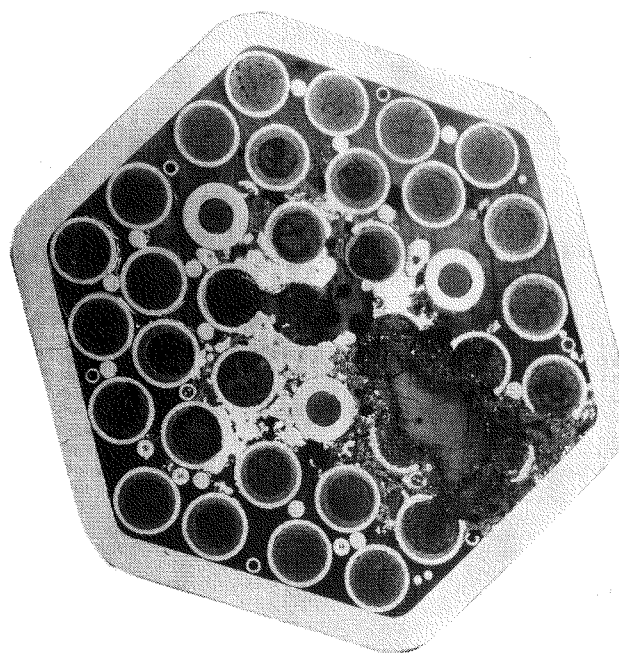
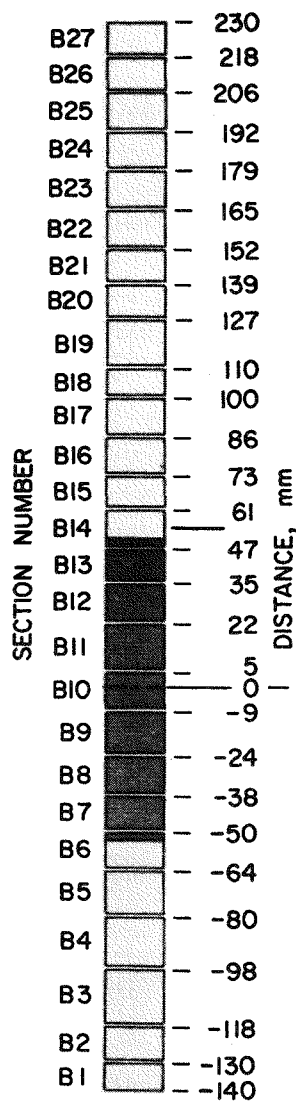
NORTH ↑

10 mm



B13  $\alpha$  AUTORADIOGRAPH

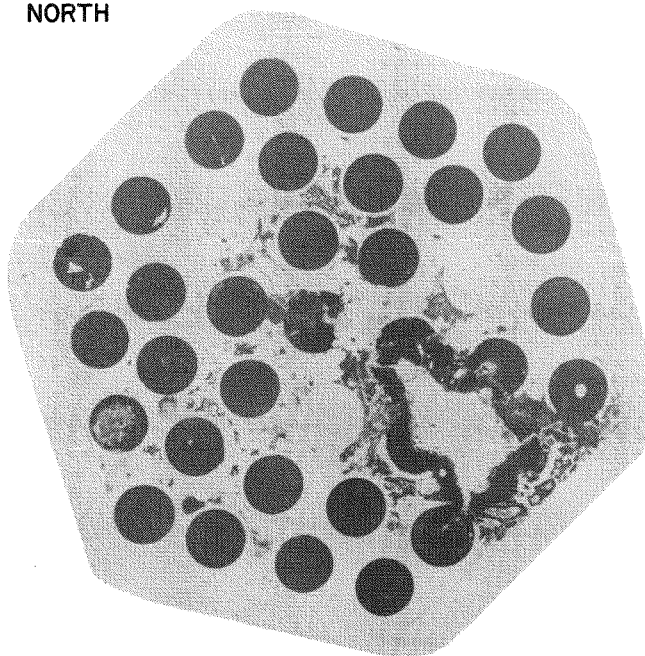
Fig. 37f.



B14 MACROPHOTOGRAPH

NORTH ↑

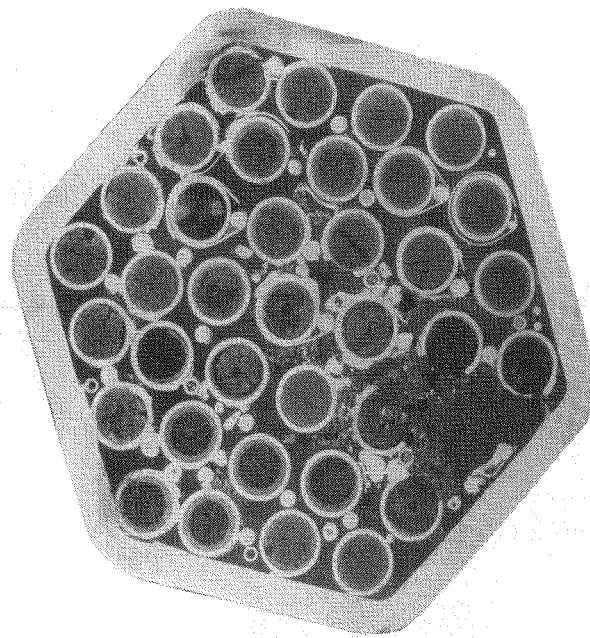
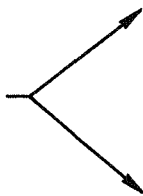
10 mm



B14  $\alpha$  AUTORADIOGRAPH

Fig. 37g.

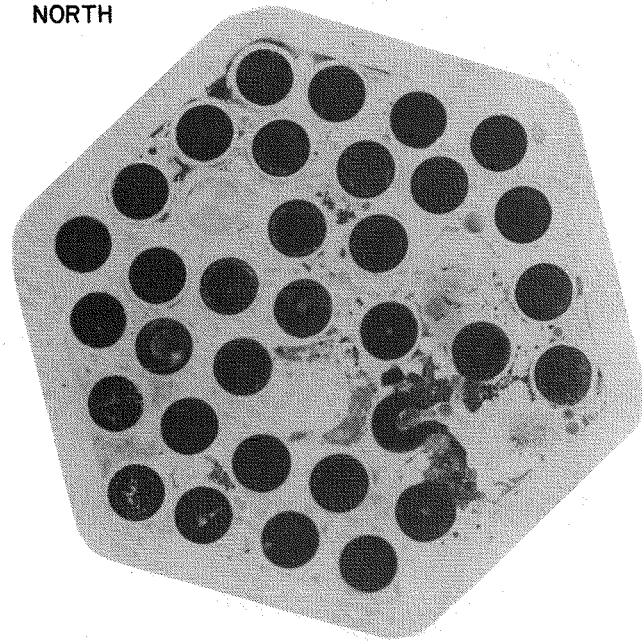
|     |      |
|-----|------|
| B27 | 230  |
| B26 | 218  |
| B25 | 206  |
| B24 | 192  |
| B23 | 179  |
| B22 | 165  |
| B21 | 152  |
| B20 | 139  |
|     | 127  |
| B19 | 110  |
| B18 | 100  |
| B17 | 86   |
| B16 | 73   |
| B15 | 61   |
| B14 | 47   |
| B13 | 35   |
| B12 | 22   |
| B11 | 5    |
| B10 | 0    |
| B9  | -9   |
| B8  | -24  |
| B7  | -38  |
| B6  | -50  |
| B5  | -64  |
| B4  | -80  |
| B3  | -98  |
| B2  | -118 |
| B1  | -130 |
|     | -140 |



B15 MACROPHOTOGRAPH

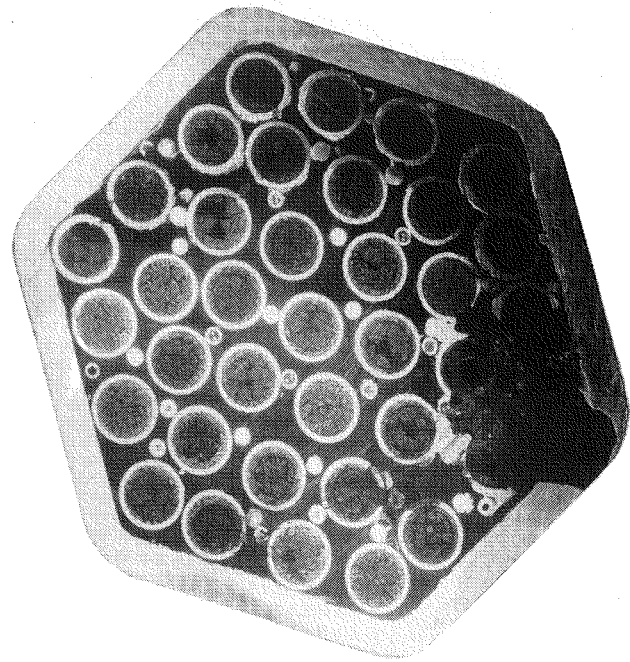
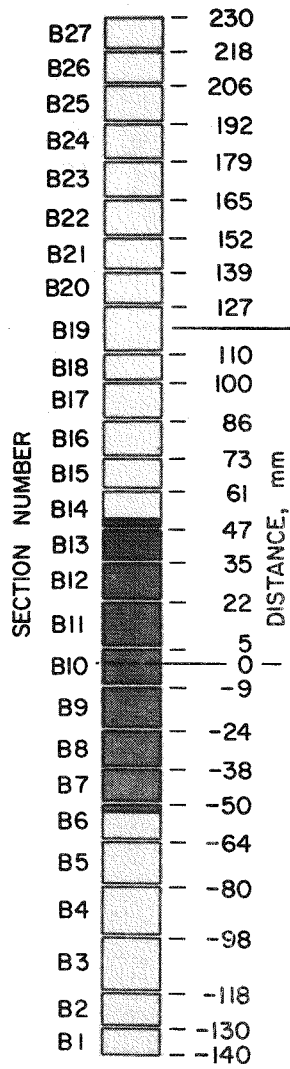
NORTH ↑

10 mm

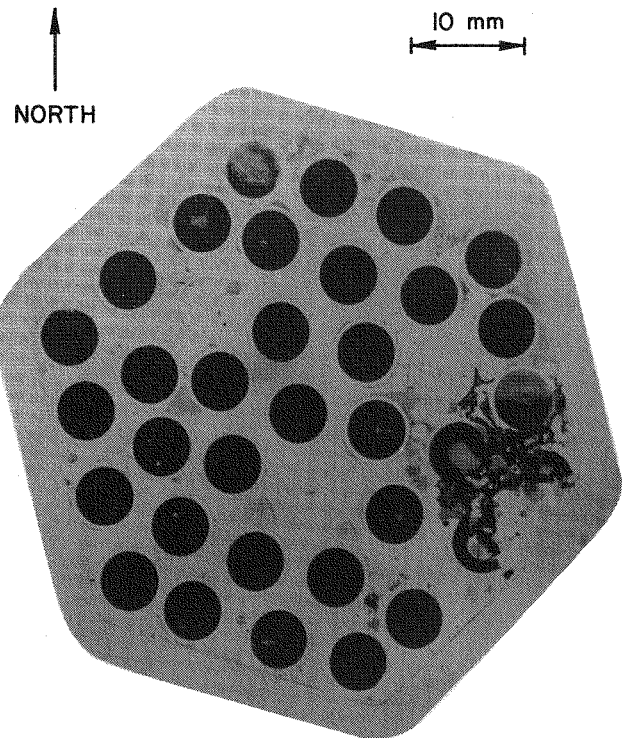


B15  $\alpha$  AUTORADIOGRAPH

Fig. 37h.

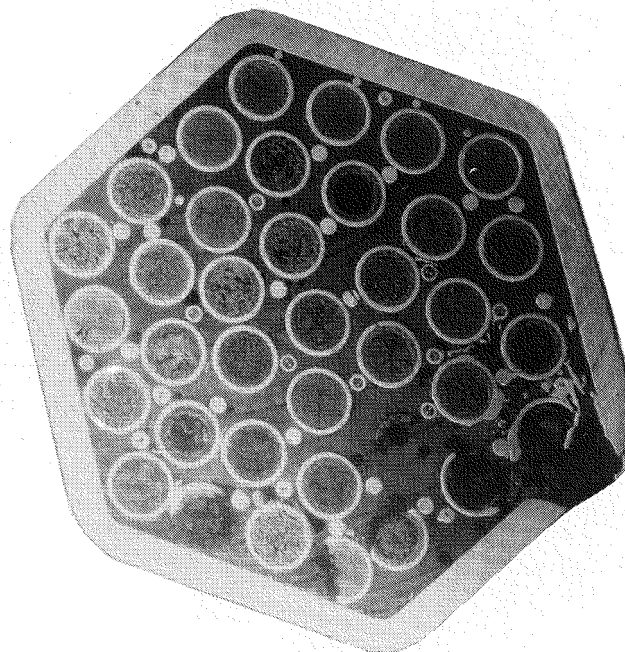
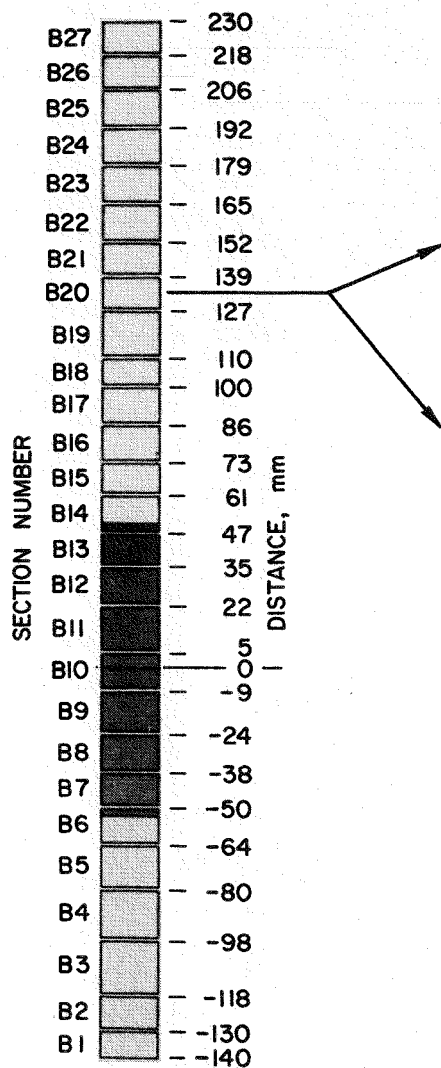


B19 MACROPHOTOGRAPH

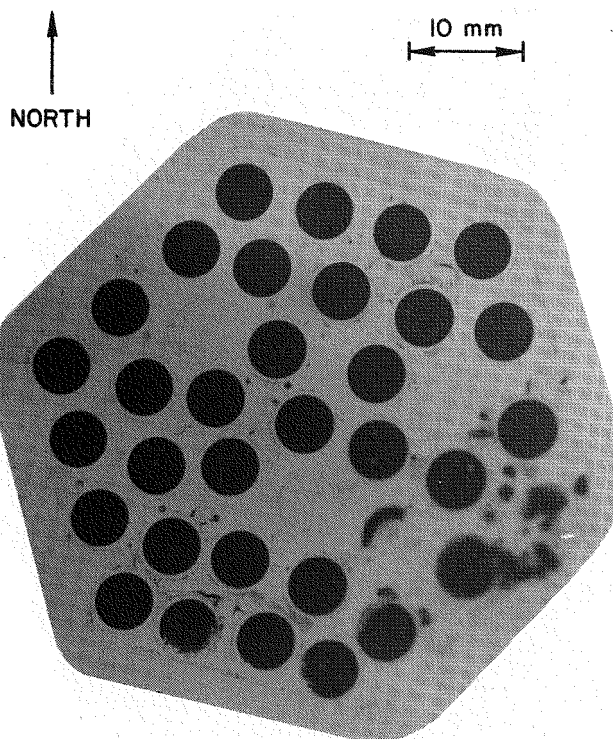


B19  $\alpha$  AUTORADIOGRAPH

Fig. 37i.



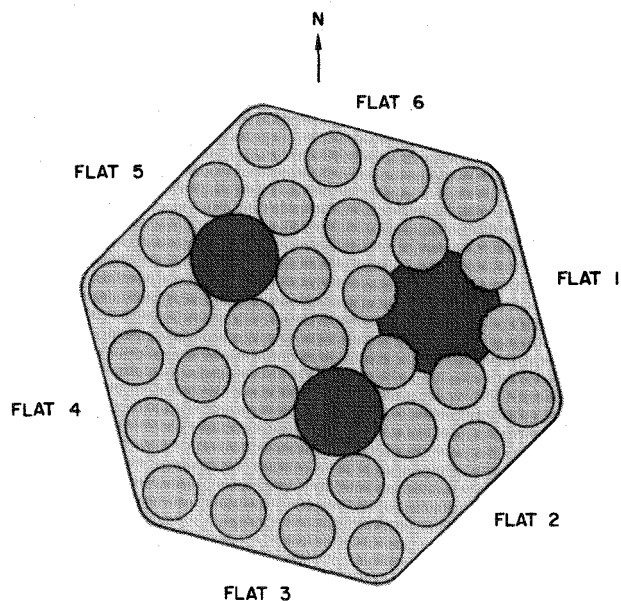
B20 MACROPHOTOGRAPH



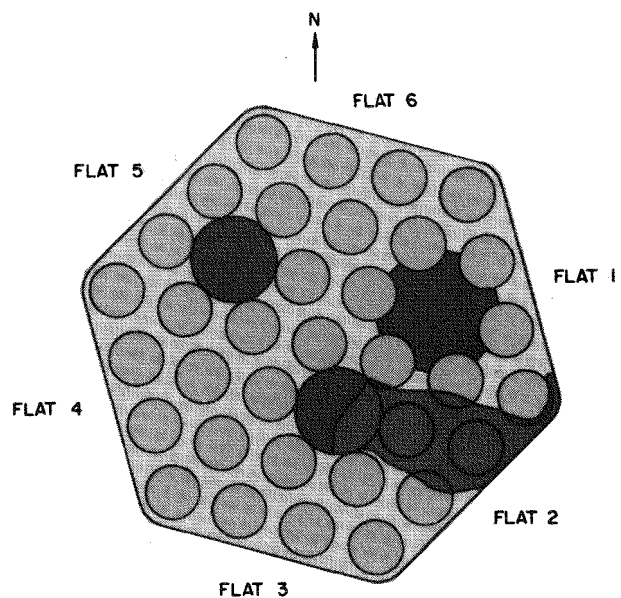
B20  $\alpha$  AUTORADIOGRAPH

Fig. 37j.





FUEL CANISTER REGION AT 15.2 s  
FAILURE OF FLUTED CANISTER  
(NO FUEL RELEASE)



FUEL CANISTER REGION AT 34 s  
FAILURE OF THE 20 % C.W. CYLINDRICAL CANISTER  
(MOLTEN FUEL RELEASE)

Fig. 38. Pictorial Illustrating the 15.2-s Failure of the Fluted Canister and the Approximate Fuel Motion Resulting from the 34-s Failure of the 20% CW Canister

Fig. 38(b). Molten fuel and fuel debris were directed upward by the upward, counterclockwise spiral of the spacer wires on the fuel pins. Disrupted fuel and metal relocated to and contacted the inner surface of the hex-can adjacent to Pins 27, 28, 29, and 30 within a continuous axial region that extended from just below the top of macrosection B8 (-24 mm) to slightly above the top of macrosection B15 (73 mm). Upward movement of the molten fuel was impeded by the reduced-diameter spacer wires on the outer-row fuel pins and a safety-function thermocouple (TE 6-1) adjacent to the hex-can wall. Molten fuel caused cladding failure and fuel disruption of adjacent pins before lodging against the hex-can, the safety-function thermocouple, and adjacent fuel pins within a short axial region centered 130 mm above the fuel midplane, as shown in macrosection B19 of Fig. 37.

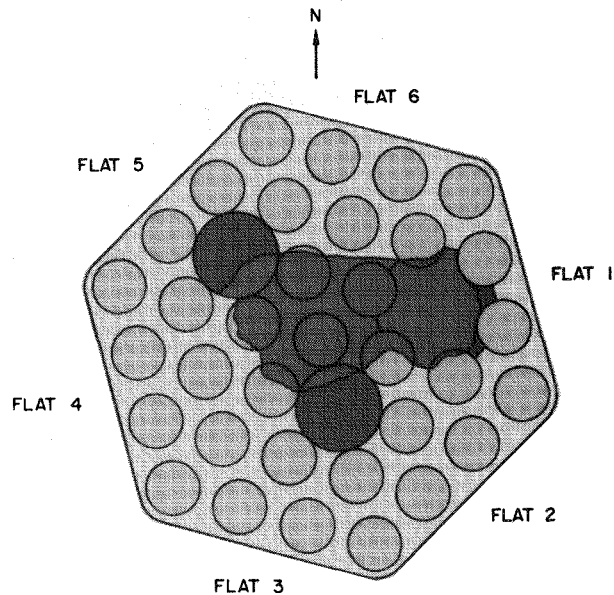
Rupture of the fluted canister at 86 s (see Fig. 39(a)) expelled molten fuel into the center of the bundle, causing fuel-pin failures and locally melting cladding on the intact row-of-12 cylindrical canister. Maximum damage to the canister occurred near the fuel midplane, as shown in macrosections B9 and 10 of Fig. 37, where slightly more than half of the  $\text{UO}_2$  became disrupted.

The severely damaged center of the bundle, bounded by the failed fuel canisters, contained little, if any, pin structure following rupture of the fluted canister. A significant amount of cast metal and a lesser quantity of failed fuel, shown in macrosection B14 of Fig. 37, formed a cap over the voided region below. The damaged region at the lower right of the cross section was caused by the relocation of molten  $\text{UO}_2$  from the row-of-six cylindrical canister (the 34-s event). Movement of this molten  $\text{UO}_2$  caused little cladding damage for 40 mm and then lodged against the hex-can wall in sections B19 and 20 (see B19 in Fig. 37), leading to local meltthrough of the hex-can wall at 110 s.

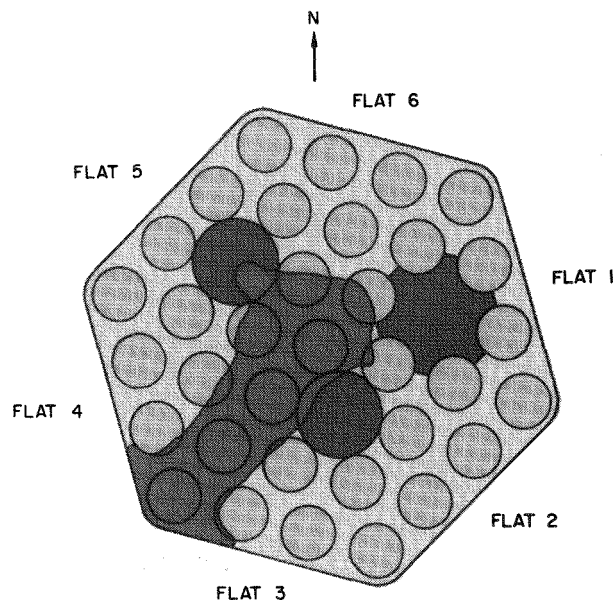
PTE evidence indicated extensive formation of sodium urano-plutonate in failed pins and disrupted fuel in flow channels during the lengthy ETR shutdown following the power transient. Dissociation of the sodium urano-plutonate led to a reconfiguration of the disrupted materials during the subsequent return-to-full-power operation. Fuel and metal debris relocated downward from the center of the bundle just below the fuel midplane toward the hex-can Flats 3 and 4 opposite the fluted canister. Evidence of the reconfiguration event is shown in Fig. 37 (macrosections B5, 7, and 9) and illustrated in Fig. 39(b), where additional fuel-pin damage and disrupted material are visible at the lower left of the cross sections.

#### b. $\text{UO}_2$ -filled canisters

Cladding of the fluted canister melted at 15 s without releasing molten fuel to the coolant channel; this was followed by a rupture at 86 s that ejected molten  $\text{UO}_2$  into the center of the bundle and caused extensive fuel-pin failures and melting of cladding on the row-of-12 cylindrical canister. Evidence of the initial failure can be seen in macrosection B7 (-38 mm) of Fig. 37 even though subsequent fuel disruption obscured much of the evidence at higher elevations. The meltthrough at the center of three cusps in B7 is indicative of the initial failure mechanism predicted for the fluted canister. The adjacent fuel pins are similarly melted, and the fuel-pin cladding has fused with the canister cladding. The



FUEL CANISTER REGION AT 86 s  
DISRUPTION OF FLUTED CANISTER  
AND SUBSEQUENT FAILURE OF THE  
10 % C.W. CYLINDRICAL CANISTER



FUEL CANISTER AT TIME OF  
THE RECONFIGURATION  
(FUEL RELOCATED TO THE HEX-CAN WALL)

Fig. 39. Pictorial Illustrating the Approximate Fuel Motion Resulting from the 86-s Disruption of the Fluted Canister and the Reconfiguration Event



welding of the fluted canister to the six adjacent fuel pins must have maintained the integrity of the canister until its 86-s rupture. Macrosections B9, 10, and 11 in Fig. 37 also provide evidence of welding of the canister cladding to the cladding of adjacent fuel pins. In these macrosections, melting of the cusps progressed from the initial meltthrough toward the tips of the flutes.

Nearly all of the  $\text{UO}_2$  fuel disrupted and relocated from the fluted canister at 86 s. A large voided region, identified earlier and shown in macrosections B9, 10, and 11, covered the center of the fuel bundle and extended to the fluted canister where the tips of only two flutes remained above B7 (-38 mm). Even in macrosection B7, the center of the canister cross section is filled with a mixture of disrupted  $\text{UO}_2$  and mixed-oxide fuel dotted with small globules of cast metal.

The 20% CW, row-of-six cylindrical canister failed at 34 s and ejected molten fuel into the coolant channel in the manner previously described. The condition of the canister following its planned failure was obliterated by the subsequent failure of the fluted canister at 86 s. The centrally located voided region, shown in macrosections B9, 10, and 11 of Fig. 37, encompasses all but a small portion of the row-of-six canister. All that remains of the canister above B7 (-38 mm) is a short arc of cladding that extended between two adjacent pins and a thin shell of  $\text{UO}_2$  fuel. The amount of molten fuel ejected from the canister during the 34-s event could not be determined by the posttest examination.

The 10% CW, row-of-12 cylindrical canister sustained the least disruption at all elevations. Maximum damage occurred near the fuel midplane, as shown in macrosections B9 and 10 of Fig. 37, where slightly more than half of the  $\text{UO}_2$  became disrupted. Prior to 86 s, fuel at the center of the cylindrical canister was molten causing high thermal and stress loading of the cladding. Molten fuel, ejected from the fluted canister during the 86-s fuel-motion event, impacted the cylindrical-canister cladding and caused its immediate failure.

Evidence from macrosections B9, 10, and 11 of Fig. 37 indicates that once the cladding failed, hot fuel from the molten core of the canister broke through the thick shell of unmelted fuel. The flow of melted and unmelted fuel out of the canister forced the cladding to open or spread, causing local bundle distortion. A central, voided cavity was formed in the row-of-12 canister, and a mixture of melted  $\text{UO}_2$  and mixed-oxide fuel drained to the bottom of the cavity in B7 (-38 mm) and froze there.

Many of the radial cracks present in the unmelted fuel shell of the canister in macrosections B9, 10, and 11 are filled or decorated with cast metal from the failed cladding. The horseshoe-shaped shell of unmelted fuel that surrounds the voided cavity is an extension of the much larger voided region of the central part of the bundle.

#### c. Mixed-oxide fuel pins

Expulsion of molten fuel from the three  $\text{UO}_2$ -filled canisters severely tested the integrity of the fuel pins. Pins directly impacted by the molten  $\text{UO}_2$  experienced a range of damage varying from minor cladding melting

to complete cladding and fuel disruption. The lowest elevation displaying disruption was -64 mm, where cladding failed. Damage to the fuel-pins was most extensive throughout the canister region (-50 mm to +50 mm), where the central region of the bundle cross section became completely disrupted, leaving an irregularly shaped void that extended upward from -30 mm to the tops of the canisters. An axial region containing little or no fuel-pin damage extended from +60 mm to +100 mm, above which the effects of the 34-s fuel motion and the hex-can failure at 110 s are evident.

The macrophotographs in Fig. 37 illustrate the extent and nature of the fuel-pin damage. Table 12 depicts the fuel-pin condition on a pin-by-pin basis for each macrosection. Pins experiencing only cladding melting are labeled "M", those pins with less than 100% fuel disruption are labeled "D", and pins that were completely disrupted (e.g., those once located in the voided region) are labeled "V". Fuel pins not experiencing damage are left unlabeled.

Seven pins (~20% of the fuel bundle) survived the in-reactor testing without failure at any elevation; three others experienced minor cladding melting near the bundle midplane. Fuel was disrupted in 23 pins. Seven pins adjacent to the hex-can breach failed within an axial region extending from +110 mm upward to +165 mm.

Many of the fuel pins shown in the macrophotographs in Fig. 37 displayed a distinct ring at the periphery of the fuel pellet surrounding a somewhat damaged interior region. Only pins known to have been breached at some elevation demonstrated this condition so it was believed to be related to sodium urano-plutonate formation. Later metallographic examination confirmed that the ring of fuel at the perimeter of the pellet contained sodium urano-plutonate while the interior did not contain any. More will be said about this in the micro-examination discussion of the PTE. Pins which did not fail displayed as-fabricated structures, except for those fuel pins that must have operated at slightly elevated temperatures with respect to the others. The pin just above the row-of-12 canister in macrosection B7 (-38 mm), Fig. 37, illustrates the typical restructuring that occurred.

Throughout the canister region, the large void identified earlier encompassed from six to nine fuel-pin sites, depending on the elevation, and nearly all of the fluted and row-of-6 cylindrical canisters. The void was irregularly shaped and extended almost to the hex-can wall in macrosections B9, B10, and B11 of Fig. 37. The void penetrated into the row-of-12 cylindrical canister in these same three macrosections. The molten fuel released from the fluted canister during the 86-s event led to the failure of the row-of-12 canister, and the canister's inventory of molten fuel broke through its shell of unmelted fuel leaving an internal void that extended outward to the fuel-bundle void. The remainder of the large void was surrounded by disrupted fuel and severely damaged fuel pins. In many instances, intact segments of cladding on adjacent pins were welded together at their ends, as illustrated in Fig. 37.

#### d. Hex-can

The 34-s failure of the row-of-six cylindrical canister ejected molten  $\text{UO}_2$  away from the center of the bundle toward adjacent mixed-oxide pins

Table 12. Posttest Condition of Each Pin at the Many Elevations Studied During the Macroexamination

| Macrosection<br>Number | Distance from<br>Fuel Midplane<br>mm | Fuel-Pin Location Within the P4 Bundle |   |   |   |                |   |               |                |   |    |    |                 |    |    |    |    |    |    |                 |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |   |   |   |   |   |   |   |   |
|------------------------|--------------------------------------|----------------------------------------|---|---|---|----------------|---|---------------|----------------|---|----|----|-----------------|----|----|----|----|----|----|-----------------|----|----|----|----|----|----|----|----|----|----|----|----|----|----|----|----|----|----|---|---|---|---|---|---|---|---|
|                        |                                      | Row of Six                             |   |   |   |                |   | Row of Twelve |                |   |    |    |                 |    |    |    |    |    |    | Row of Eighteen |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |   |   |   |   |   |   |   |   |
|                        |                                      | 1                                      | 2 | 3 | 4 | 5 <sup>1</sup> | 6 | 7             | 8 <sup>2</sup> | 9 | 10 | 11 | 12 <sup>3</sup> | 13 | 14 | 15 | 16 | 17 | 18 | 19              | 20 | 21 | 22 | 23 | 24 | 25 | 26 | 27 | 28 | 29 | 30 | 31 | 32 | 33 | 34 | 35 | 36 | 37 |   |   |   |   |   |   |   |   |
| B22                    | 165                                  |                                        |   |   |   |                |   |               |                |   |    |    |                 |    |    |    |    |    |    |                 |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    | M  |    |   |   |   |   |   |   |   |   |
| B21                    | 152                                  |                                        |   |   |   |                |   |               |                |   |    |    |                 |    |    |    |    |    |    |                 |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    | M  | M  |   |   |   |   |   |   |   |   |
| B20                    | 139                                  |                                        |   |   |   |                |   |               |                |   |    |    |                 |    | D  |    |    |    |    |                 |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    | D  | M  |   |   |   |   |   |   |   |   |
| B19                    | 127                                  |                                        |   |   |   |                |   |               |                |   |    | M  | D               |    |    |    |    |    |    |                 |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    | M  | V  | D |   |   |   |   |   |   |   |
| B18                    | 110                                  |                                        |   |   |   |                |   |               |                |   |    |    | D               | D  |    |    |    |    |    |                 |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    | D  | D  | M |   |   |   |   |   |   |   |
| B17                    | 100                                  |                                        |   |   |   |                |   |               |                |   |    |    |                 |    |    |    |    |    |    |                 |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |   |   |   |   |   |   |   |   |
| B16                    | 86                                   |                                        |   |   |   |                |   |               |                |   |    |    |                 | M  | M  |    |    |    |    |                 |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |   |   |   |   |   |   |   |   |
| B15                    | 73                                   |                                        |   |   |   | M              |   |               |                |   |    |    |                 | M  | D  |    |    |    |    |                 |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    | M  | V | D |   |   |   |   |   |   |
| B14                    | 61                                   |                                        |   |   |   |                |   |               |                |   |    |    |                 |    |    |    |    |    |    |                 |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    | D  | V | D |   |   |   |   |   |   |
| B13                    | 47                                   |                                        |   |   |   |                |   |               |                |   |    |    |                 |    |    |    |    |    |    |                 |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    | D  | D | M | V | D | M |   |   |   |
| B12                    | 35                                   |                                        |   |   |   |                |   |               |                |   |    |    |                 |    |    |    |    |    |    |                 |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    | D  | D | D | D | D | M |   | M |   |
| B11                    | 22                                   |                                        |   |   |   |                |   |               |                |   |    |    |                 |    |    |    |    |    |    |                 |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    | D  | D | D | D | M | M |   | M |   |
| B10                    | 5                                    |                                        |   |   |   |                |   |               |                |   |    |    |                 |    |    |    |    |    |    |                 |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    | D  | D | D | V | M |   | D | M | M |
| B9                     | -9                                   |                                        |   |   |   |                |   |               |                |   |    |    |                 |    |    |    |    |    |    |                 |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    | D  | D | D | D | M |   | D | M | M |
| B8                     | -24                                  |                                        |   |   |   |                |   |               |                |   |    |    |                 |    |    |    |    |    |    |                 |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    | D  | D | M | M |   | M | M | M |   |
| B7                     | -38                                  |                                        |   |   |   |                |   |               |                |   |    |    |                 |    |    |    |    |    |    |                 |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    | M  | M |   |   | M | M | M |   |   |
| B6                     | -50                                  |                                        |   |   |   |                |   |               |                |   |    |    |                 |    |    |    |    |    |    |                 |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    | M  | M |   |   | M | M |   |   |   |
| B5                     | -64                                  |                                        |   |   |   |                |   |               |                |   |    |    |                 |    |    |    |    |    |    |                 |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |   |   |   | M | M |   |   |   |

<sup>1</sup>Pin contains 20% CW canister

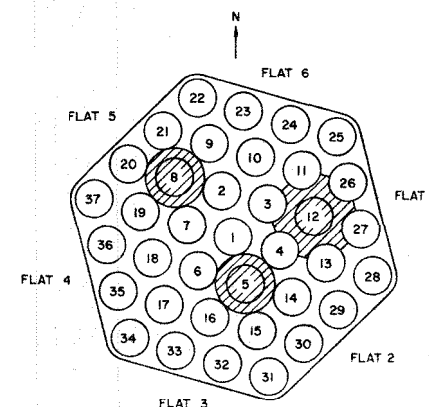
<sup>2</sup>Pin contains 10% CW canister

<sup>3</sup>Pin contains fluted canister

<sup>4</sup>"V" identifies a pin with complete fuel disruption

<sup>5</sup>"D" identifies a pin with less than complete fuel disruption

<sup>6</sup>"M" identifies a failed pin with only melted cladding



and the hex-can wall. The path of the fuel motion was outward and upward, as described earlier, and can be observed at the lower right of the macrosections in Fig. 37. Fuel and metal debris contacted the hex-can wall from just below the top of macrosection B8 (-24 mm) to slightly above the top of macrosection B15 (+73 mm). Molten  $\text{UO}_2$  moved even higher and lodged against the inner surface of the hex-can wall in macrosections B19 and B20, trapped by the small spacing between the outer-row pins and the hex-can wall and the presence of thermocouple TE 6-1. The failed pins at the lower right of macrosection B14 (+61 mm), Fig. 37, indicate the upward motion of molten  $\text{UO}_2$  and fuel debris within the coolant channels.

Nearly 76 s elapsed between the time molten fuel contacted the hex-can wall in sections B19 and 20 and the wall failed by local meltthrough at 110 s. The breach caused by the gradual melting of the hex-can was approximately 10 mm wide by 25 mm long. The curved surface at the right of the breach in B19 (Fig. 37) was indicative of a slow melt front that progressed through the hex-can wall. Subsequent metallographic examination of the site, described later, showed extreme grain growth of metal structures adjacent to the breach that was promoted by the long time the region experienced elevated temperatures. The rapid depressurization of the insulator region at the time of the  $\sim 2.5\text{-cm}^2$  meltthrough forced the fuel away from the breach and explains why no disrupted material remained at the hex-can failure site.

The disrupted material on hex-can Flats 1 and 2 between macrosections B8 and B15 covered a continuous area of  $\sim 13.4\text{ cm}^2$ . This material remained in contact with the hex-can wall throughout the rest of the experiment. Damage to the hex-can within this region was minimal and went unnoticed until the microexamination phase of the PTE. While investigating the structure of fuel contacting the hex-can, it was observed that the inner surface of the hex-can wall was locally melted in macrosections B8, B11 and B15. Melting was slight, limited to a maximum depth of only 0.25 mm. This is in sharp contrast to the  $\sim 2.5\text{-cm}^2$  meltthrough of the hex-can wall in macrosections B19 and B20 and indicates the importance of sodium in the fuel debris to maintain coolability.

#### e. Distribution of materials

The quantitative posttest axial distribution of metal and fuel within the hex-can has been evaluated from the 2X macrophotographs. The data were obtained by cutting the nominally 2X photos of the transverse, macrosection surfaces and weighing the pieces. The data were corrected for actual magnification of the photos and normalized against the design-drawing dimensions of the components. Figure 40 shows plots that describe the survivability of fuel-bundle components, including mixed-oxide and  $\text{UO}_2$  canister fuel and structural metal from fuel-pin and canister cladding, spacer wires, hex-can safety thermocouples, and fluted-canister banding straps. The plots show the fraction of the original materials that remained intact throughout the experiment. Materials that experienced some melting were considered intact as long as relocation had not occurred (e.g., local melting of cladding that had not relocated was considered as intact). The area of the disrupted fuel and metal blockage at the elevation of each macrosection is shown graphically in Fig. 41. In like manner, the size of the large void shown in the macrosections of Fig. 37 is also plotted as an area in Fig. 41. The data shown in Fig. 41 were used to calculate an approximate void volume of  $29.4\text{ cm}^3$ .

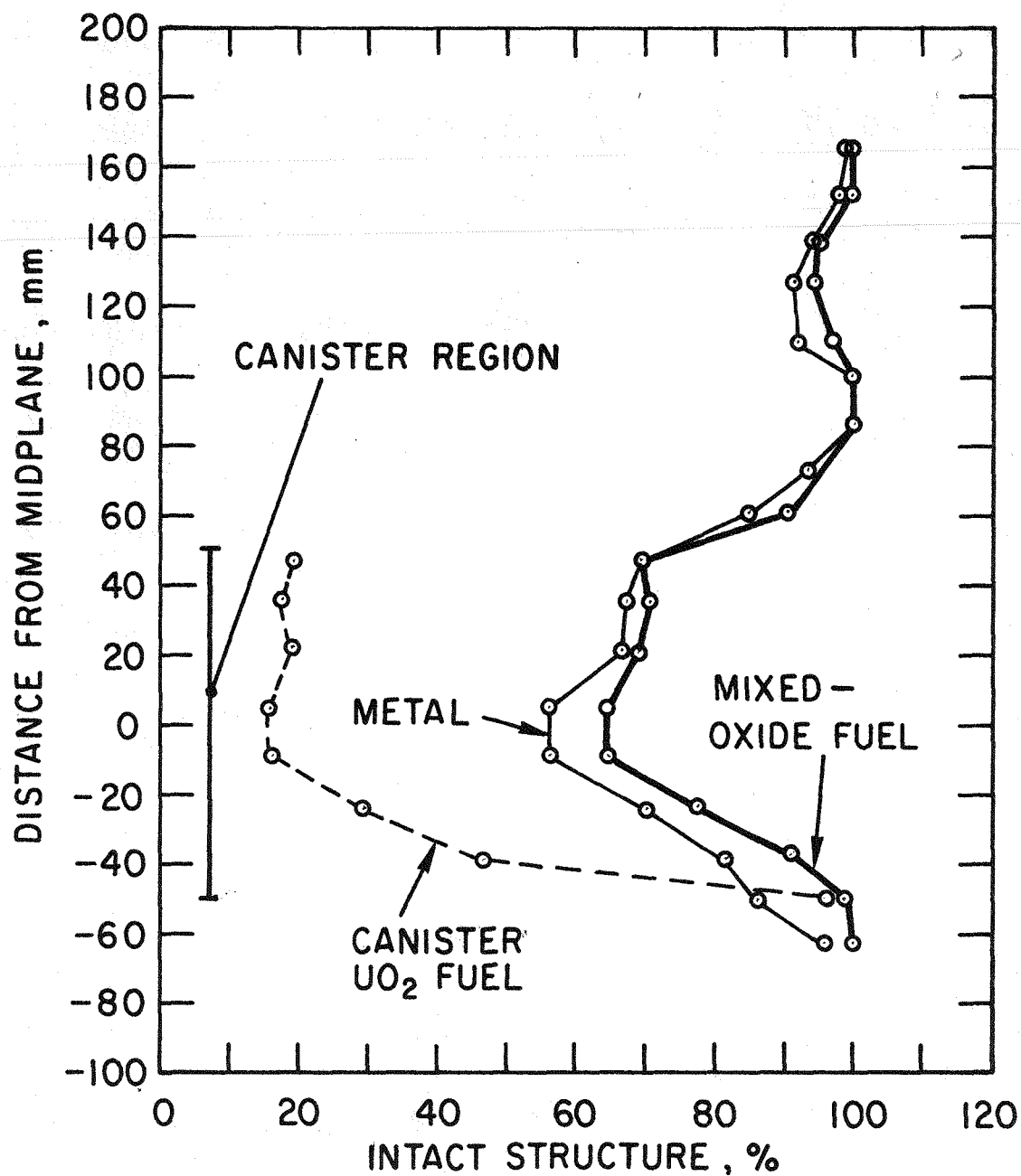


Fig. 40. Fraction of Original Fuel and Metal Within the Hex-Can That Remained Intact Throughout the Experiment

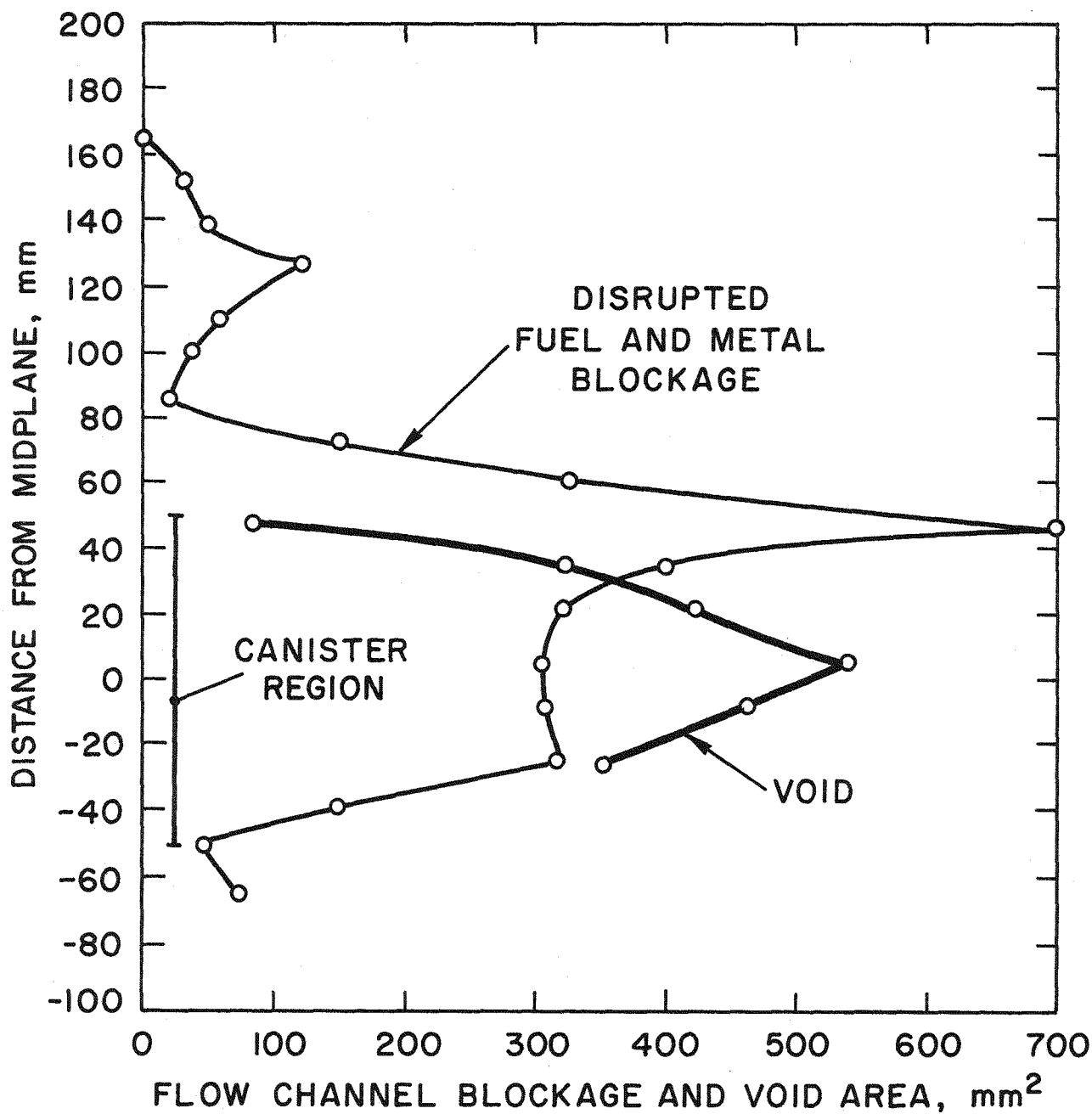


Fig. 41. Area of Disruption (Blockage and Void) Within the Original Fuel-Bundle Cross Section

The lowest elevation displaying disruption was -64 mm, where cladding failed and ~10% of the flow channel was blocked with disrupted fuel and metal from the reconfiguration event. Damage to the fuel bundle was most extensive throughout the canister region (-50 mm to +50 mm). Nearly one-third of the mixed-oxide and slightly more than 80% of the  $\text{UO}_2$  fuel were disrupted in the upper two-thirds of the canister region. Likewise, nearly one-third of the initial metal structure became disrupted in this same axial region. The extent of areal flow-channel blockage was relatively constant throughout the canister region. The most extensive blockage occurred at the top of the canisters in sections B13 and 14; the cap of the voided region in B14 is shown in Fig. 37. An axial region containing little or no disruption extended from +60 mm to +100 mm, above which the effects of the 34-s fuel motion and subsequent hex-can failure at 110 s were observed. The amount of disrupted fuel (canisters plus fuel pins) was estimated at  $270 \pm 27$  g in the 10-cm-long canister region and  $310 \pm 31$  g total.

### C. Microexamination and Evaluation

The detailed examination of the 2X macrophotographs and  $\alpha$ -autoradiographs provided sufficient information about the extent of fuel disruption in mixed-oxide fuel pins caused by the expulsion of molten  $\text{UO}_2$  from the fuel canisters. The flows of molten fuel and metal were easily traced and identified with the major disruption events, which were detected by various test train sensors. However, three major areas of uncertainty persisted after completion of the macroexamination; namely,

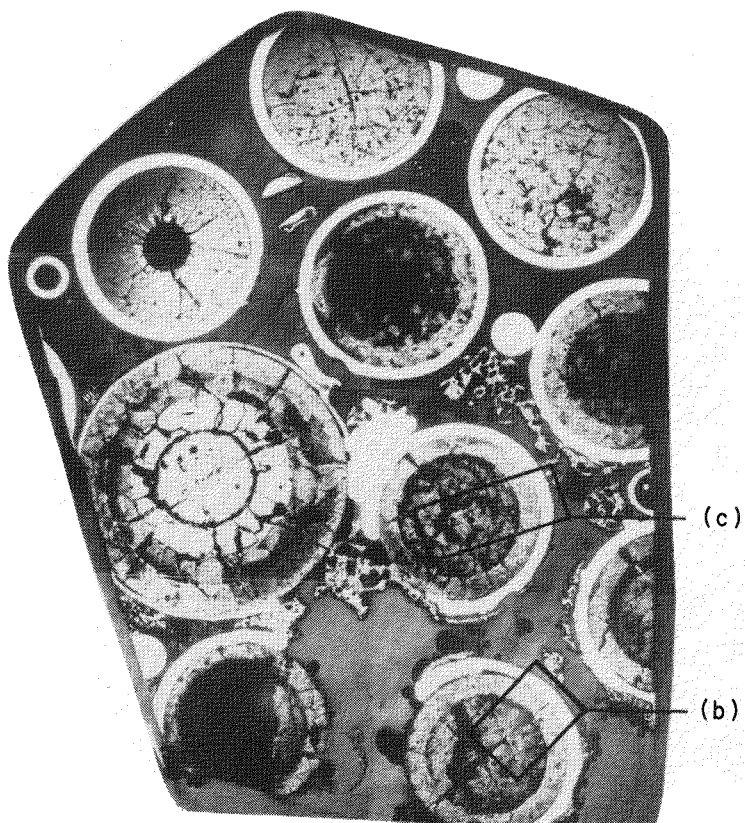
- formation of sodium urano-plutonate in failed mixed-oxide pins and disrupted fuel,
- nature of disrupted fuel and metal in the coolant channels and fuel canisters, and
- microstructural response of metal structures adjacent to the hex-can breach.

Numerous metallography specimens were cut and prepared from nearly every macrosection, and optical microexamination provided information related to the major areas of uncertainty. The electron microprobe analyzer was used to identify the compositions of various microstructural formations, and the Quantimet was used to determine the relative proportion of fuel, steel, and porosity in various flow-channel sites containing disrupted materials.

#### 1. Sodium Urano-Plutonate

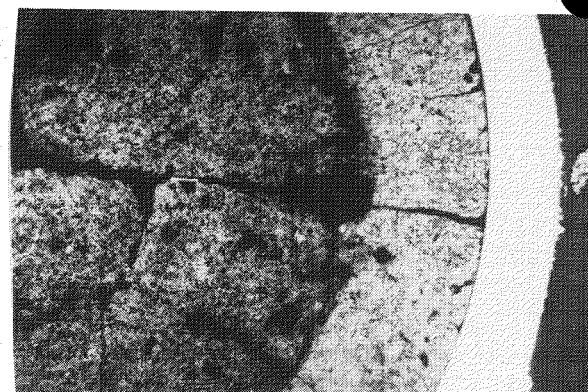
The loop and test train were maintained in a hot standby condition following the ETR scram on August 21, 1981, until ETR power operation was resumed on October 1. During the extended hot standby, the fuel-bundle inlet temperature was maintained at ~680 K, coupled with intermediate operation to 40 MW ETR power, to enhance sodium ingress and egress at fuel-pin failure sites and the formation of sodium urano-plutonate within failed fuel pins and the partial flow blockage.

Fuel pins known to have cladding breaches were carefully studied during the macroexamination for evidence of sodium urano-plutonate formation. Many of the pins displayed a wide band of intact fuel at the periphery of fuel



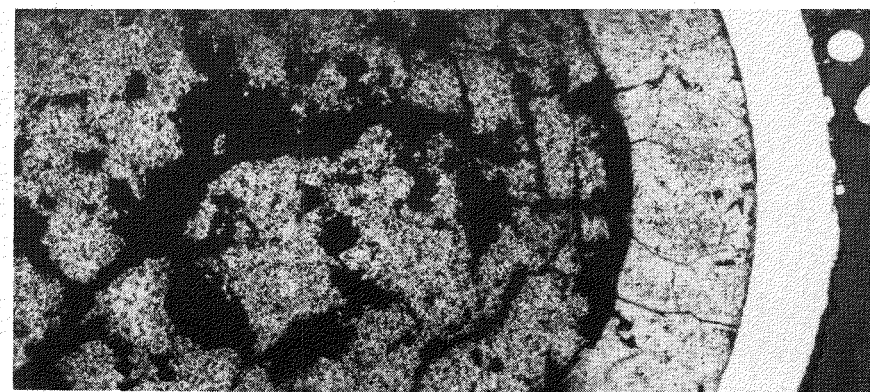
→ | ← 1 mm

(a)



→ | ← 0.5 mm

(b)



→ | ← 0.5 mm

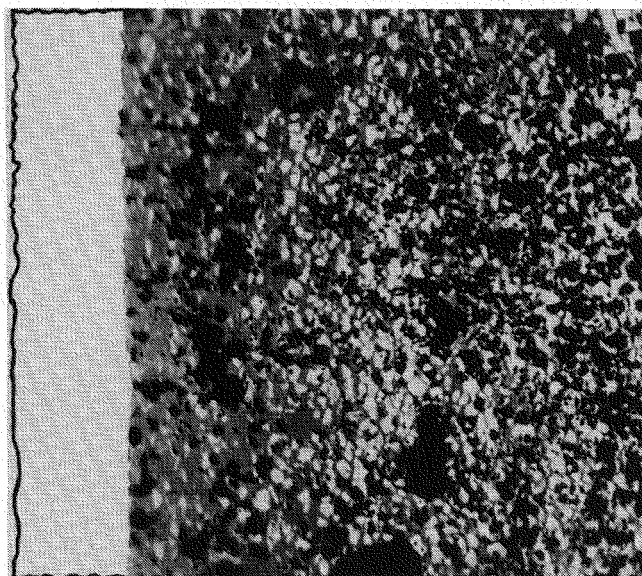
(c)

Fig. 42. Microstructure of Fuel from Section B6 (-50 mm) Containing Sodium Urano-Plutonate



pellets surround a damaged interior region, as shown in Fig. 42(a). Since this condition existed only in failed pins, it was believed that the wide peripheral ring of fuel contained sodium urano-plutonate.

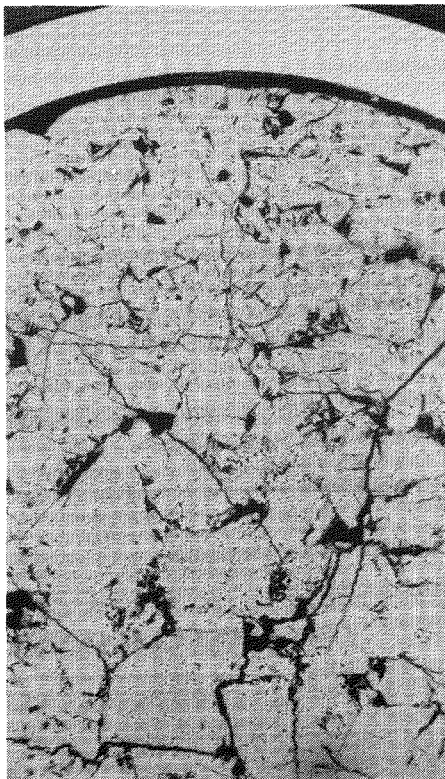
Higher-magnification micrographs, shown in Figs. 42(b) and 42(c), illustrate more clearly the nature of the ring and the damaged interior. A gray phase, thought to be sodium urano-plutonate, filled many of the cracks, grain boundaries, and voids within the rings shown in these figures. Electron microprobe analysis of these and other similar regions confirmed that the gray phase was sodium urano-plutonate. Figure 43 illustrates the presence of this material at the periphery of a mixed-oxide fuel pellet where sodium urano-plutonate extended from the inner cladding surface at the left of the figure for some distance into the fuel-pellet grain boundaries at the right of the micrograph. Grain boundaries and cracks provided paths for the ingress of sodium into the fuel during the hot standby. Small individual mixed-oxide grains were embedded in a matrix of sodium urano-plutonate at the left of Fig. 43, where sodium continued to be supplied for the reaction that consumed the mixed-oxide grains.



→ | ← 20  $\mu$ m

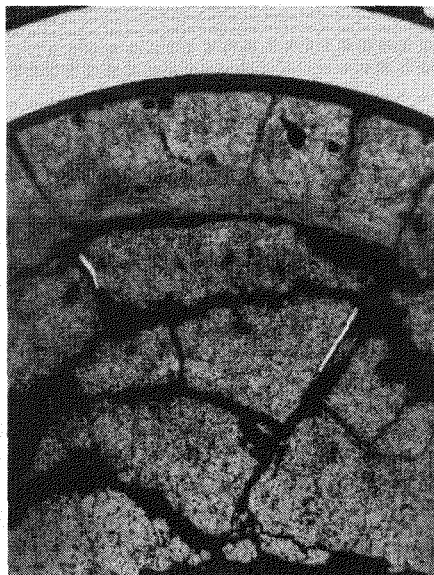
Fig. 43. Micrograph of a Mixed-Oxide Fuel Pin 100 mm Above the Fuel Midplane Showing Sodium Urano-Plutonate to the Right of the Cladding

The two micrographs on the right of Fig. 44 more fully illustrate the presence of sodium urano-plutonate within cracks and grain boundaries of the fuel. A broad band of sodium urano-plutonate (darker gray phase) spans a wide crack at the periphery of a fuel pellet in Fig. 44(c). Fig. 44(a) was included to illustrate the fuel microstructure in an unfailed fuel pin.



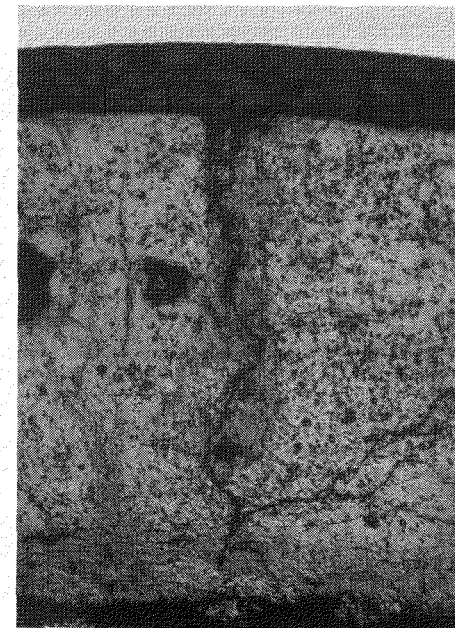
→ | ← 0.5 mm

(a)



→ | ← 0.5 mm

(b)



→ | ← 0.1 mm

(c)

Fig. 44. Micrograph of Fuel from an Unfailed Pin Shown with Two Micrographs of Failed Fuel Containing Sodium Urano-Plutonate at Grain Boundaries and Cracks

For P4 system pressures, sodium urano-plutonate is known to dissociate at  $\sim 1500$  K, which is above the boiling point for unreacted sodium. As ETR power operation was resumed, the isotherm for dissociation moved radially outward within the individual fuel pellets for each power increase. Mixed-oxide fuel and sodium vapor resulted in the dissociated region (inner portion of the pellet), and mixed-oxide fuel and sodium urano-plutonate remained in the outer, fuel-pellet shell. Reactor operation was concluded at 100 MW ETR power, at which time the isotherm for dissociation had moved radially outward to the ring defined by the inner surface of the sodium urano-plutonate band of Figs. 42 and 44. Measurements made on many micrographs determined that the isotherm for dissociation was at 75% of the original fuel-pellet radius.

The dissociation of sodium urano-plutonate and immediate vaporization of free sodium caused severe damage to the fuel microstructure. As a result, existing cracks were widened and grain boundaries were significantly weakened so that much fuel was lost during cutting and metallographic preparation. The severely damaged centers of the fuel were evidence that sodium urano-plutonate formed across the entire mixed-oxide fuel pellets during the extended, hot standby.

Conditions during the hot standby were suitable to promote the formation of sodium urano-plutonate in both failed pins and disrupted fuel within the bundle flow channels; however, the sodium urano-plutonate in the flow-channel material dissociated during the reconfiguration event because of higher temperatures there. The resultant vaporization of sodium caused the relocation of disrupted material. Only one site, of the many that were examined within the disrupted material, contained any sodium urano-plutonate.

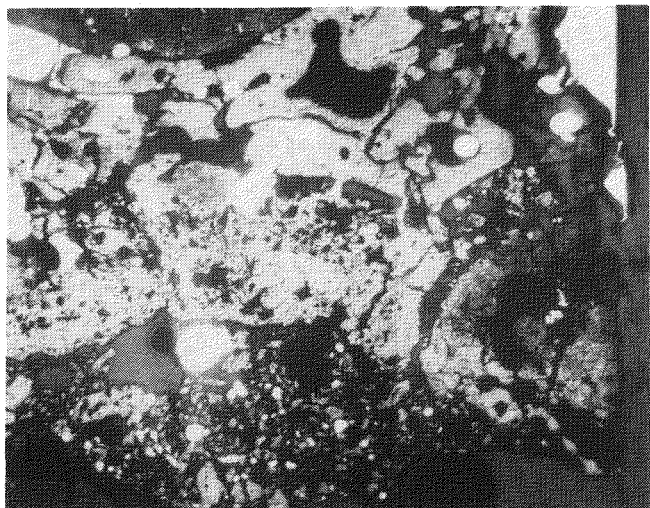
Careful diametral measurements were made on many fuel-pin micrographs to determine if the formation of sodium urano-plutonate caused swelling. In all cases, no permanent fuel-pin swelling was detected.

## 2. Disrupted Material in Coolant Channel

The micrographs shown in Figs. 45, 46, and 47 illustrate structures that typify the nature of disrupted materials in the flow channels of the P4 bundle. The large void that occupied the central part of the bundle and most of the fuel canisters was bordered by a porous fuel structure intermixed with stainless steel particles of varying sizes and geometries. A dense band of once-molten fuel, shown in Figs. 46 and 47, separated the disrupted fuel and metal at the periphery of the voided region from the damaged but otherwise intact mixed-oxide fuel. The dense band contained a mixture of  $\text{UO}_2$ , mixed-oxide fuel, and micron-size stainless-steel particles. Cool-down cracks traversed the dense fuel band in some regions.

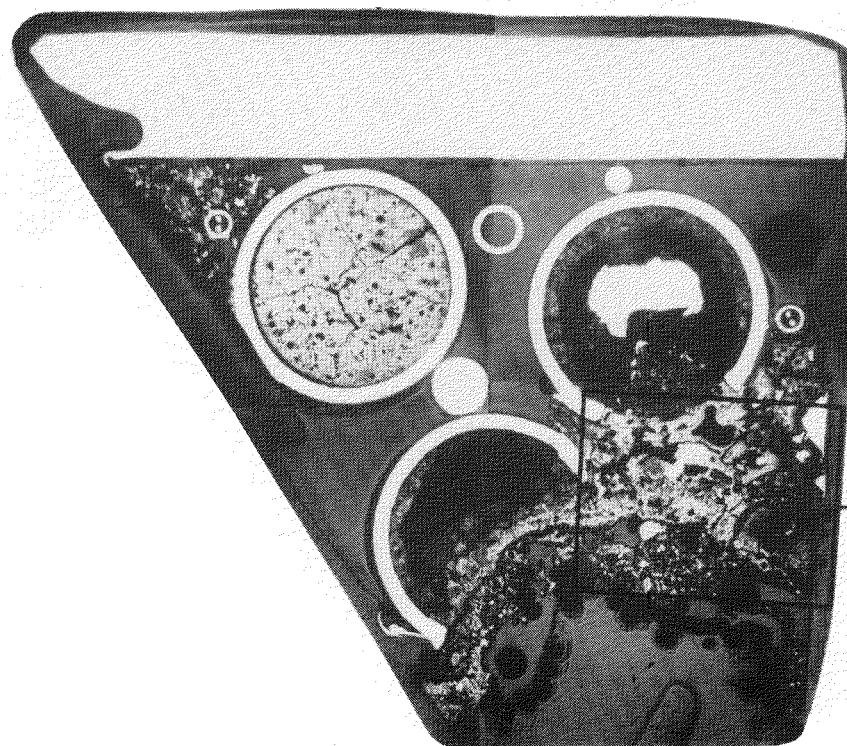
In many instances, the dense band of fuel penetrated deeply into failed fuel pins. One tip of the fluted canister, shown in Fig. 46(a), contained unmelted  $\text{UO}_2$  that was forced away from its cladding by the intrusion of molten stainless steel. The cladding of the fluted canister and two adjacent pins have fused and provided support to the disrupted fuel and metal bordering the large voided region.

As stated earlier, essentially all of the sodium urano-plutonate, which formed in the disrupted fuel during the extended standby, dissociated



→ | ← 1 mm

(a)



→ | ← 1 mm

(b)

Fig. 45. Micrographs of Failed Mixed-Oxide Fuel Pins and Disrupted Fuel and Metal in Section B11 (+22 mm)

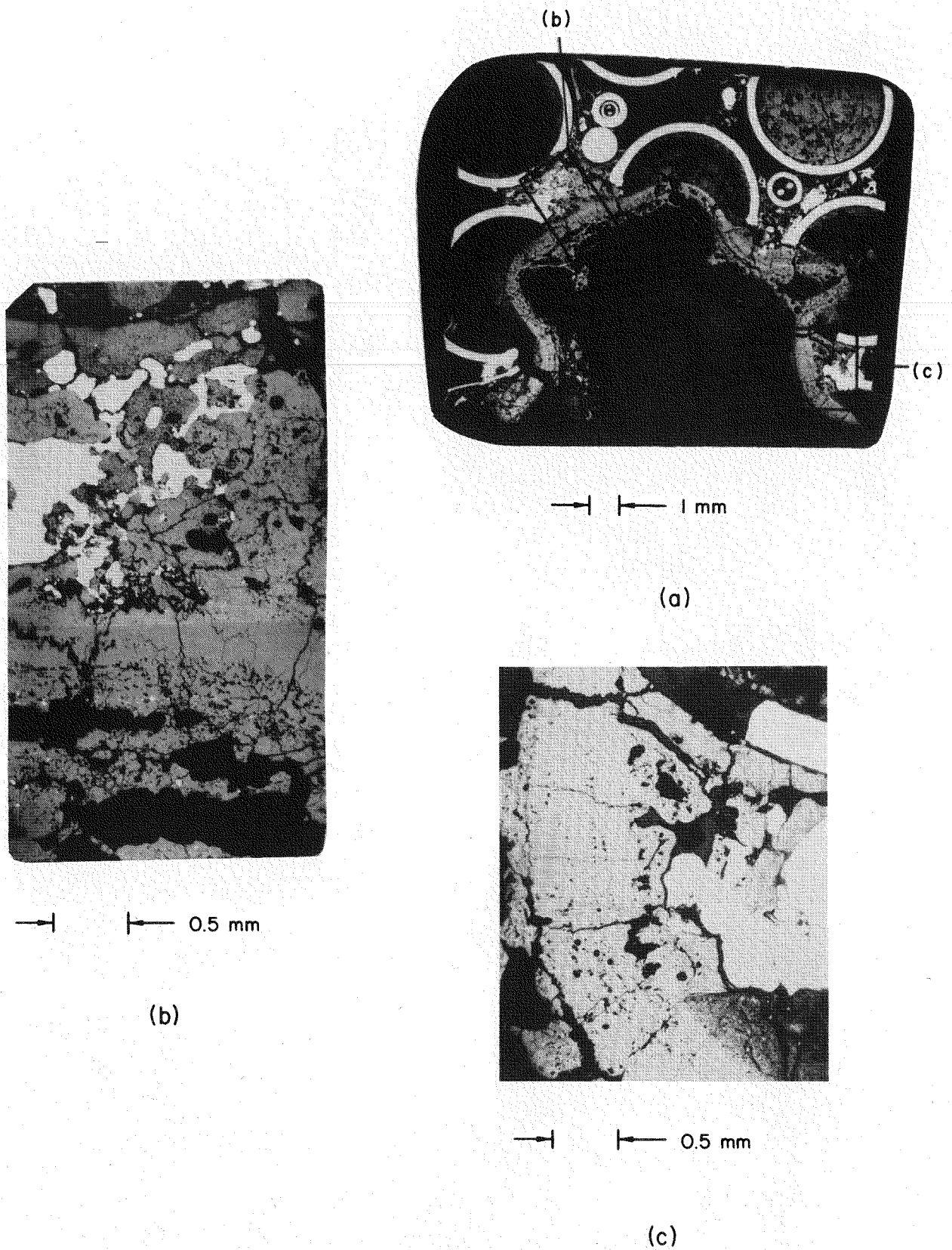


Fig. 46. Microstructure Showing a Dense Band of Disrupted Fuel Surrounding the Voided Region in Section B13 (+47 mm)



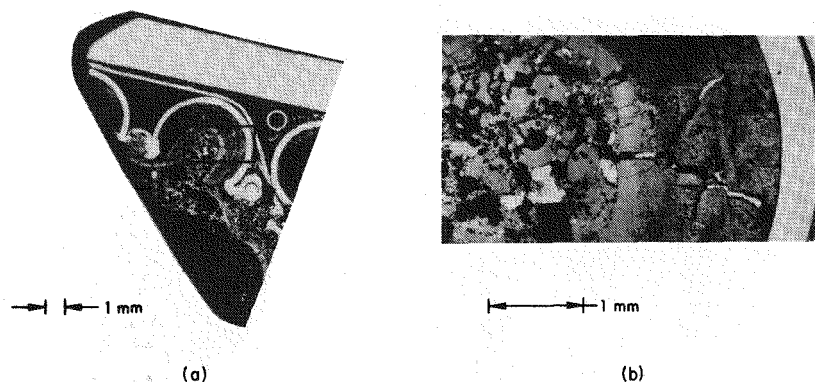


Fig. 47. Micrographs of a Dense Band of Disrupted Fuel Within a Partially Intact Fuel Pellet

during the return-to-full-power operation. However, additional metallurgical processes must have occurred in the disrupted coolant-channel fuel after the dissociation because its posttest microstructure was radically different than was observed for intact fuel pellets (compare the microstructure of the central part of Fig. 42(c) with those of Figs. 45(b) or 46(c)). Continued heating of the disrupted coolant-channel fuel, following its dissociation, and its subsequent motion during the reconfiguration event may explain its different microstructural appearance.

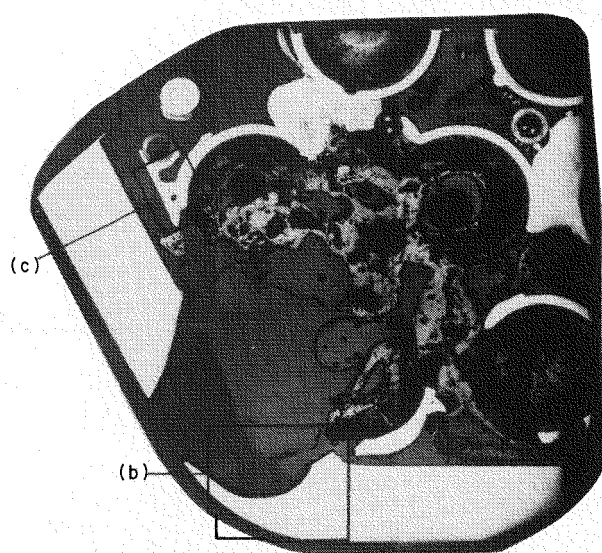
The relative proportion of fuel, steel, and porosity was measured at some 17 various locations that contained disrupted material from the different fuel-motion events. It was thought that the proportions of these components might be characteristic of the particular fuel-motion event. However, the results were so varied that a correlation could not be made.

The central disk of the 10% CW, row-of-12 canister shown in Fig. 42(a) is a mixture of once-molten  $UO_2$  and mixed-oxide fuel. The mixture drained and froze there at the end of the 86-s fuel-motion event. The outside diameter of the wide band of once-molten  $UO_2$  that surrounds the disk marks the extent of the molten  $UO_2$  inventory for the row-of-12 canister. Measurements made on this and similar micrographs led to a calculated melt fraction of 49%.

### 3. Hex-Can Failure

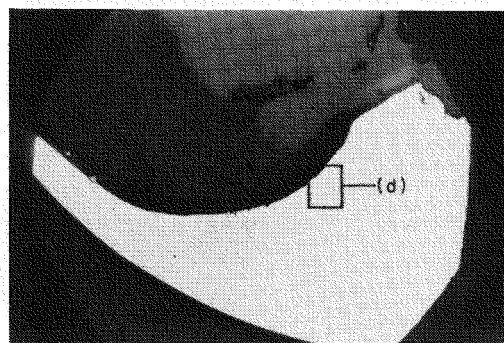
Examination of the metal structures adjacent to and in the vicinity of the hex-can failure was important in understanding the nature of the hex-can failure and behavior of molten stainless steel from failed cladding.

Figure 48(a) shows a low-power micrograph of the hex-can failure and higher magnification micrographs of selected regions from macrosection B19



1 mm

(a)



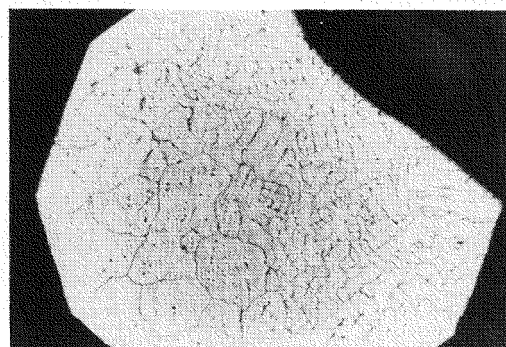
1 mm

(b)



0.5 mm

(c)



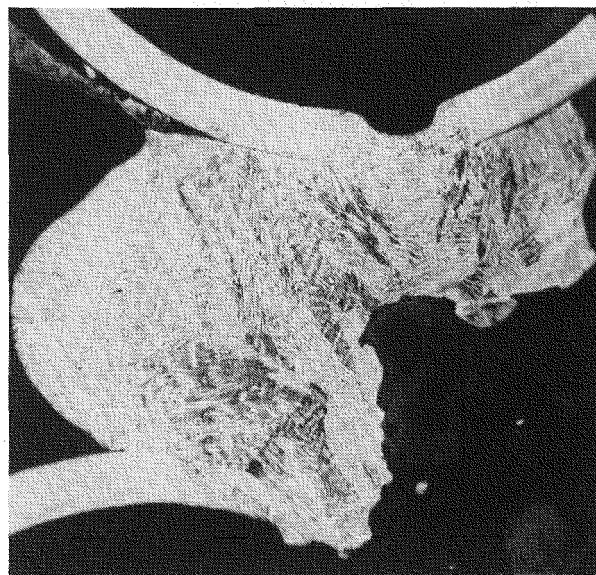
0.1 mm

(d)

Fig. 48. Microstructure of Metal Structure Adjacent to the Hex-Can Failure  
127 mm Above the Fuel Midplane

(+127 mm). Failure of the hex-can resulted when molten fuel lodged against the hex-can during the 34-s fuel motion event and slowly heated the hex-can wall to its melting temperature, causing its failure at 110 s. Rapid depressurization of the insulator region forced this fuel towards the upper right of Fig. 48(a). The shape of the melted hex-can surface shown in Fig. 48(b) and the large grain growth shown in Fig. 48(c) was indicative of a slow meltthrough.

Figure 48(d) shows how molten metal from the hex-can and a fuel pin at the left of Fig. 48(a) flowed away from the disrupted fuel along the intact surfaces of the hex-can and cladding. Various separate material flows are visible in the micrograph. Figure 49 illustrates how molten metal from failed cladding moved away from the cladding breach, bridged the space between the pins, and fused to adjacent cladding. Similar behavior was observed in the voided canister region where fusing of cladding from adjacent pins provided radial support to disrupted materials in the center of the bundle. The reconfiguration occurred after sufficient pressure was generated to break through the structure there.



1 mm

Fig. 49. Micrograph Showing How Molten Metal from Failed Cladding Fused to the Cladding of an Adjacent Pin

Figure 50 illustrates the cladding microstructure of a failed pin adjacent to the hex-can failure. The composite micrograph in Fig. 50(c) shows the progression of grain growth from the as-fabricated microstructure at the left to extremely large grains at the right. The grains were so large that the 0.381-mm cladding wall thickness was spanned by only three grains. The large metal grains at the right of the micrograph were surrounded by grain boundaries that melted and contributed to the cladding failure.



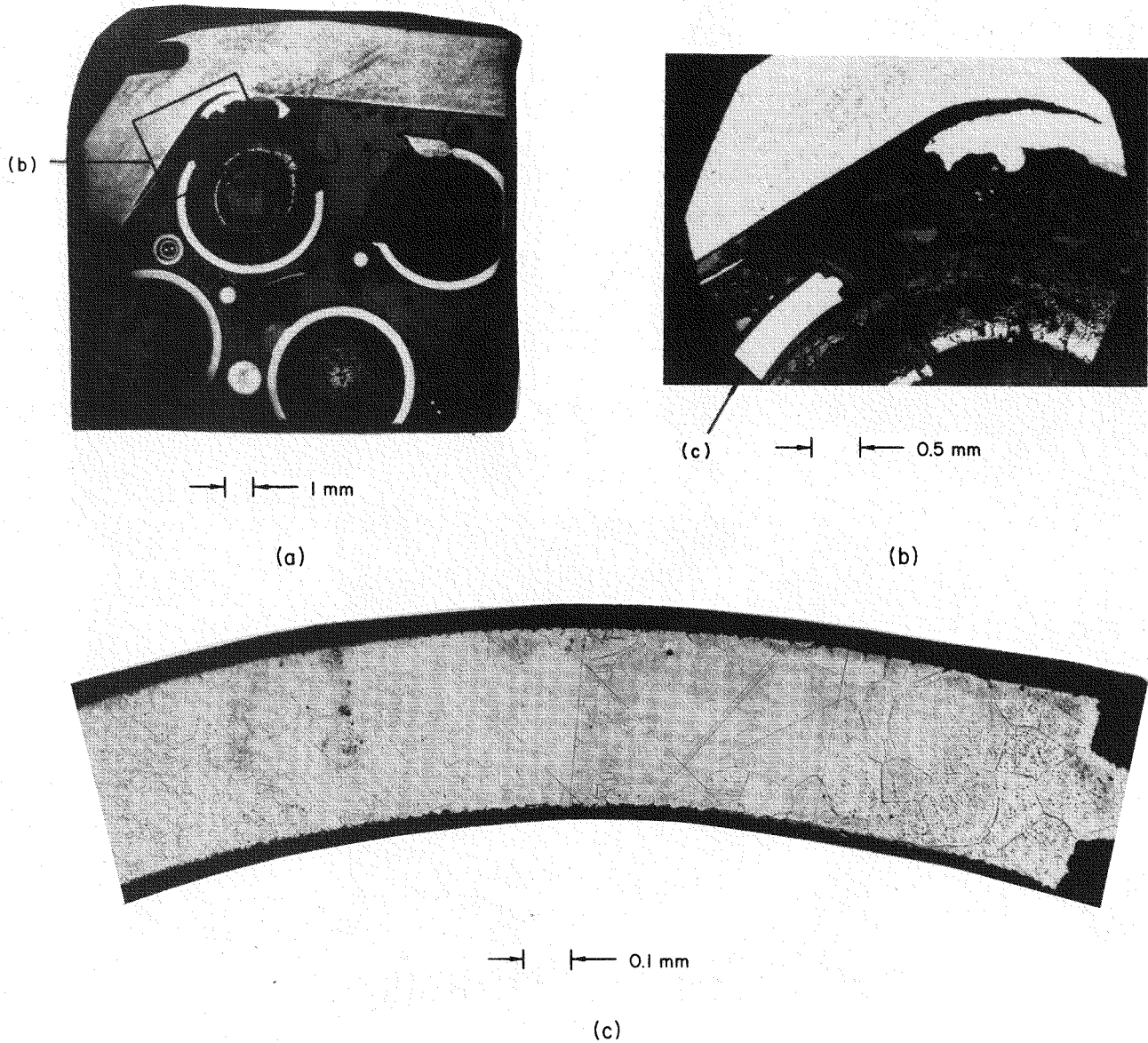


Fig. 50 Cladding Microstructures Showing Grain Growth in a Failed Pin Near the Hex-Can Failure, 139 mm Above the Fuel Midplane

## VI. POSTTEST ANALYSIS

### A. Delayed Neutron Behavior

#### 1. DN Methods and Analysis

##### a. Model

To interpret the DN signals recorded during any stage of the experiment, it was necessary to have a model of the behavior of the DN-precursor emission from the fuel and transport in the loop to the location of the detection systems. The experimental observations were compared with the predictions of this model to make clear the major features of the fuel failure events.

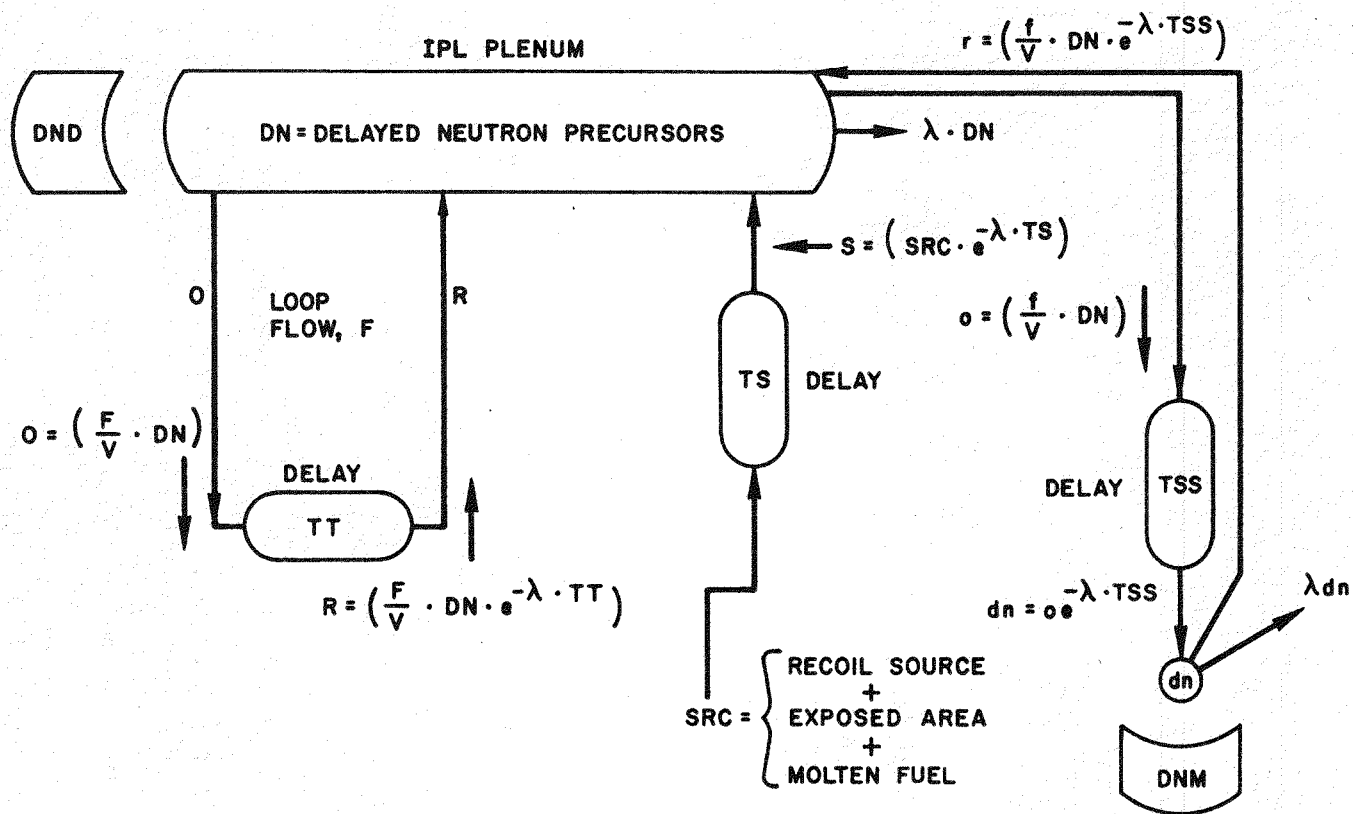
It should be made clear at the outset that there is much that is not clearly established at the most basic scientific level about the release of fission products from fuel under extreme conditions, and therefore a detailed description is not to be expected. Nevertheless, a model was developed which gave a good qualitative description of events; the same basic model had previously been used to interpret the DN signals from experiment W2. The time of initial failure and subsequent developments were in agreement with other sensors. Quantitative agreement with the amount of disrupted fuel identified during the W2 posttest examination was within a factor of two. A recalculation of W2 with the current model, using more accurate estimates of fission yields, has led to near perfect agreement between the calculated and measured fuel release.

There are two basic aspects of the model: the transport of DN emitters in the loop flow, and the source strength and release from the fuel.

##### 1) Loop flow

Figure 51 shows a schematic of the DN loop model. Consider a DN source (SRC) located in the test section. For the moment, let us regard it as a single radioactive species, with decay constant  $\lambda$ . If the DN-emitting fission products from the source are mixed or dissolved in the sodium coolant, the upward flow of the coolant carries them to the sodium plenum at the top of the loop; the delay in transit is labelled TS. In the plenum, the incoming sodium is assumed to mix completely and uniformly throughout the volume, V. The outflow, F, from the plenum therefore removes a constant fraction,  $F/V$ , of the plenum contents per unit time and recirculates it. That fraction therefore passes completely around the loop and through the neutron flux. It divides, with part passing again through the test section and part through the bypass, then returns to the plenum, where it again mixes throughout the plenum volume. The delay time for a complete circulation is TT. The transit delays TS and TT are a function of the fixed distances travelled through the loop piping and the flow rate F, which is variable. The mean residence time in the plenum is a function of the flow rate F and the plenum volume V. These times are, therefore, calculable from the mechanical details of the loop.

It is assumed that the DN precursors and emitters remain with the sodium and do not plate out on any of the surfaces; however, as they travel they undergo radioactive decay. At design flow conditions, TS was 2 s,



$$d(DN) = S_{(T>TS)} \cdot e^{-\lambda \cdot DN} - O - o + R_{(T>TT)} + r_{(T>TSS)}$$

$$DND \text{ COUNTS} = EFF_{(DND)} \cdot \lambda \cdot DN ; DNM \text{ COUNTS} = EFF_{(DNM)} \cdot \lambda \cdot dn$$

Fig. 51. Schematic of the P4 DN Loop Model

TT was 10 s, and the mean residence time in the plenum was 1.5 s. The half lives of DN emitters ranged from a maximum of 54 s down to ~1 s and the effect of series decay during transport to the detectors was of major importance.

The DN population in the plenum was obtained from Eq. 1

$$\frac{d}{dt} (DN) = S - O + R - o + r - \lambda DN \quad (1)$$

where

- DN = DN population in plenum at any instant,
- S = DN source corrected for series decay during the transport from the failure site to the plenum,
- O = (F/V) DN, (F = loop flow, V = plenum volume), outflow from the plenum due to loop flow,
- R = returning flow corrected for series decay during transport around the loop,
- o = (f/V) DN, (f = OLSS flow, V = plenum volume), outflow from the plenum due to OLSS flow,
- r = returning OLSS flow corrected for series decay during transport around the OLSS,
- $\lambda$  DN = DN decay rate in plenum.

This is illustrated on Fig. 51.

In the simplest picture, there is one such equation for each DN-emitting species; the total DN decay rate in the plenum is the sum of all contributions

$$DN_{tot} = \lambda_1 DN_1 + \lambda_2 DN_2 + \dots + \lambda_n DN_n \quad (2)$$

(The analysis is slightly more complex when both parents and daughters are considered.) Since the plenum is the source volume for the DND, the count rate in the DND is obtained by multiplying this source rate by the DND efficiency.

The OLSS stream samples the DN population in the plenum and transfers it to the DNM sample volume. There is a further transit delay in the OLSS of 12 to 40 s, during which the DN species undergo further series decay.

## 2) Neutron source strength and release from fuel

The DN isotopes have traditionally been lumped for reactor applications into six groups.<sup>18,19</sup> Table 13 lists the half-lives and yields of the six groups for <sup>235</sup>U and <sup>239</sup>Pu that were used in one version of the model. Table 14 lists actual emitters and their parents that have been proposed as a standard set for interpretation of all fuel failure work.<sup>20,21,22</sup> These isotopes have been used in a second version of the model; they essentially correspond to groups 1-4 of Table 13. The halogen isotopes, Br and I, dominate the longer-lived groups which, because of the loop transit times, are the most important contributors to the count rates. They form chemical compounds with the sodium; therefore, the assumption that they travel with the sodium is a good one. The same assumption is made for all other isotopes, but with less certainty that it is justified.

Table 13. Characteristics of the Six Delayed-Neutron Groups

| Group | <sup>235</sup> U      |            | <sup>239</sup> Pu     |            |
|-------|-----------------------|------------|-----------------------|------------|
|       | t <sub>1/2</sub><br>s | Yield<br>% | t <sub>1/2</sub><br>s | Yield<br>% |
| 1     | 54.5                  | 0.063      | 53.8                  | 0.025      |
| 2     | 21.8                  | 0.355      | 22.3                  | 0.181      |
| 3     | 6.0                   | 0.314      | 5.2                   | 0.139      |
| 4     | 2.2                   | 0.679      | 2.1                   | 0.212      |
| 5     | 0.5                   | 0.214      | 0.6                   | 0.066      |
| 6     | 0.2                   | 0.043      | 0.2                   | 0.023      |

The tables indicate the populations of DN emitters in irradiated fuel, before they are transferred to the sodium. It is at this point that most of the complexity and uncertainty enter into the calculations. We can consider three basic situations which probably span all possibilities.

#### Recoil from a surface

Fission fragments from within 10 microns of the surface that are aimed within a certain range of angles to the surface can leave the fuel. This mechanism leads to a continuous source strictly proportional to the neutron flux and to the exposed physical area, with no dependence on the temperature of the fuel or the coolant. This seems to give an adequate description of the DN source from the U-Ni-alloy fission product source. For oxide fuel, most recent data from reactors is inconsistent with this mechanism,<sup>23,24,25</sup> the signal strength being larger than expected by a factor which varies greatly. Although this mechanism is no longer considered to be physically realistic, for purposes of comparison it is common and convenient to describe any signal in terms of an equivalent "recoil area". Because fission fragments are released as they are created, the inventory of DN emitters is as shown in the tables. This mechanism, however, does apply to the specially constructed sources, such as the metal-alloy fission product source which was part of the P4 test train.

#### Finely fragmented or molten fuel

For sudden exposure of finely fragmented or molten fuel, it is considered extremely probable that the entire inventory of DN isotopes is released into the sodium and carried away. The quantity of each isotope depends on the previous irradiation history. However, because all of the half lives are less than 1 minute long, an irradiation of ~5 minutes is sufficient to bring all to equilibrium. The effect of a sudden expulsion of molten or finely fragmented fuel is, therefore, to release a large instantaneous spike

Table 14. Production and Decay Constants for Delayed-Neutron Emitters

Precursor Isotopes

| Isotope           | $t_{1/2}$<br>s | $\lambda$<br>s <sup>-1</sup> | Fission Yield<br>% |                   |
|-------------------|----------------|------------------------------|--------------------|-------------------|
|                   |                |                              | <sup>235</sup> U   | <sup>239</sup> Pu |
| <sup>87</sup> Se  | 16.0           | 0.043                        | 0.605              | 0.057             |
| <sup>88</sup> Se  | 1.5            | 0.462                        | 0.254              | 0.011             |
| <sup>137</sup> Sb | 0.82           | 0.845                        | 0.014              | 0.002             |
| <sup>137</sup> Te | 3.5            | 0.198                        | 0.297              | 0.085             |
| <sup>89</sup> Se  | 0.41           | 1.691                        | 0.059              | 0.002             |
| <sup>93</sup> Kr  | 1.27           | 0.546                        | 0.282              | 0.029             |
| <sup>138</sup> Te | 1.6            | 0.433                        | 0.054              | 0.012             |
| <sup>94</sup> Kr  | 0.21           | 3.301                        | 0.092              | 0.004             |
| <sup>139</sup> Te | ~0             | 10 <sup>9</sup>              | 0.006              | 0.001             |

Daughter Isotopes

| Isotope           | $t_{1/2}$<br>s | $\lambda$<br>s <sup>-1</sup> | Fission Yield<br>% |                   | Probability<br>of Neutron<br>Emission<br>% |
|-------------------|----------------|------------------------------|--------------------|-------------------|--------------------------------------------|
|                   |                |                              | <sup>235</sup> U   | <sup>239</sup> Pu |                                            |
| <sup>87</sup> Br  | 55.7           | 0.012                        | 1.305              | 0.440             | 2.58                                       |
| <sup>88</sup> Br  | 15.9           | 0.044                        | 1.609              | 0.267             | 6.35                                       |
| <sup>136</sup> Te | 21.0           | 0.033                        | 1.189              | 0.491             | 1.10                                       |
| <sup>137</sup> I  | 24.6           | 0.028                        | 2.421              | 0.156             | 7.10                                       |
| <sup>89</sup> Br  | 4.5            | 0.154                        | 1.084              | 0.120             | 14.20                                      |
| <sup>93</sup> Rb  | 5.8            | 0.120                        | 2.564              | 0.704             | 1.36                                       |
| <sup>130</sup> I  | 6.5            | 0.107                        | 1.264              | 0.002             | 5.14                                       |
| <sup>94</sup> Rb  | 2.7            | 0.257                        | 1.555              | 0.294             | 10.20                                      |
| <sup>139</sup> I  | 2.4            | 0.289                        | 0.424              | 0.181             | 9.90                                       |

of DN products, in which all of the isotopes are enhanced above the steady-state production rate, and the longer-lived isotopes are the most enhanced. This mechanism gives a very large recoil equivalent (about 400 cm<sup>2</sup>/g), but only instantaneously, and with the individual isotopes in different ratios than in the recoil case.

### Diffusion

An alternate mechanism for release from solid fuel is diffusion from the fuel matrix, in a fashion similar to the known diffusion of the fission gases. Diffusion leads to a source from a volume of fuel connected to a surface and, therefore, to a signal much larger than that due entirely to recoil from the surface itself. It accounts for the enhancement of the DN signal which is so commonly observed<sup>23,24,25</sup> and is relevant to measurements made on the blockage. The diffusion of gases has been studied intensively, but not nearly so much is known about the corresponding processes for non-gaseous fission products. Diffusion constants for both noble gases and halogens have been measured over a range of temperatures.<sup>26</sup> The diffusion constants increase rapidly with temperature. The value for Br is greater than that for I. This means that different species will diffuse at different rates and that the process will be faster at higher temperatures. The diffusion out of fuel grains has been discussed by Booth.<sup>27</sup> For each radioactive species, he has derived a formula for the released fraction, as follows:

$$R/B = 3 (D/a^2\lambda)^{1/2} [\coth(a^2\lambda/D)^{1/2} - (D/a^2\lambda)^{1/2}] \quad (3)$$

where

- R = release rate,
- B = birth rate,
- a = an adjustment parameter, nominally the equivalent grain diameter,
- D = diffusion constant,
- $\lambda$  = decay constant.

The diffusion constants are not well known but probably have a temperature dependence of the form of the Arrhenius equation

$$D = D_0 e^{-Q/T} \quad (4)$$

where

- Q = activation constant,
- T = temperature of fuel grain.

From Eq. 3, when  $(D/a^2\lambda) \ll 1$ , the term in square brackets is close to unity; we then see that the DN release from the fuel grains is proportional to  $(D/\lambda)^{1/2}$ , so that the longer-lived isotopes are enhanced over the shorter-lived isotopes, as compared with recoil. Also, due to the temperature dependence of D in Eq. 3, a simple calculation for Br<sup>87</sup> shows that R/B increases markedly with temperature and approaches 100% near the melting point.<sup>28</sup> Thus, in the high-temperature limit, the diffusion release approaches the complete release that we have assumed above accompanies fuel melting. This picture is obviously incomplete because it stops at the boundary of the grain and neglects other necessary processes which move the

isotopes along cracks and fissures to the surface of the fuel and sodium contact. However, it serves as a point of comparison for the experimental data.

#### b. DN simulation

Two models of the operation of the loop, the OLSS, and the DND and DNM DN detectors were constructed, using the digital simulation system CSMP. They were essentially identical in their treatment of the sodium flow, and differed in the method by which the DN inventory was calculated. In the older model, the data of Table 13 were used in a separate calculation, using a special library in RIBD<sup>29</sup>, and were then input to the CSMP calculation. In the second model, the DN inventory was calculated in the simulation program directly, using the data of Table 14. This latter development brings the calculations into line with other recent DN analyses.<sup>20,21,22</sup> Similar results are obtained from both variants of the model; minor differences presumably indicate sensitivity to details of the radioactive decay chains during transport. An essential assumption made in both cases is that all of the chemical species are, in fact, transported completely and equally with the sodium, and all effects of deposition and plate-out are neglected. While this is probably justified for the halogens, I and Br, for other elements it is more questionable.

Both simulations evaluate Eq. 1 for each DN group or DN emitter, so as to determine the DN population in the loop sodium plenum. Input data and parameters include masses of molten fuel exposed at chosen times, areas exposed at chosen times, the variable history of the loop and OLSS flow, and the reactor power history. The different isotopes make quite different contributions to the DN count rate; the shortest-lived isotopes decay during the first journey to the plenum and the intermediate-lived isotopes during the first circulation in the loop. Figure 52 shows, for illustration, the result of a calculation for a sudden exposure of a mass of molten fuel. The large peak is due to the first passage of the DN population through the plenum, and the smaller peak to the remaining DN population in the sodium following circulation through the loop and past the detectors the second time. Hence, the signature for molten fuel is a major peak is followed by satellites of decreasing amplitude separated by the loop circulation time, ~10 s. Figure 53 illustrates the DN signal characteristic of an exposed fuel area (equivalent recoil source). Continuous DN release from an exposed fuel area results in a steady signal which approaches an asymptotic value in increments equal to the circulation time of sodium in the loop. Multiple exposures are clearly distinguished from one another because of the prominence of the DN peaks.

## 2. Evaluation of DN Response During Power Operation

#### a. Low-power transient

During the low-power transient tests (to 40 MW), the three fuel failure detection systems responded to the signal from the U-Ni fission product source. The ratio of the DN count rate between the transient peak and the original level agreed with the ratio of reactor power. This meant that the DN systems were operating properly and that there had been no premature failure of the fuel canisters.



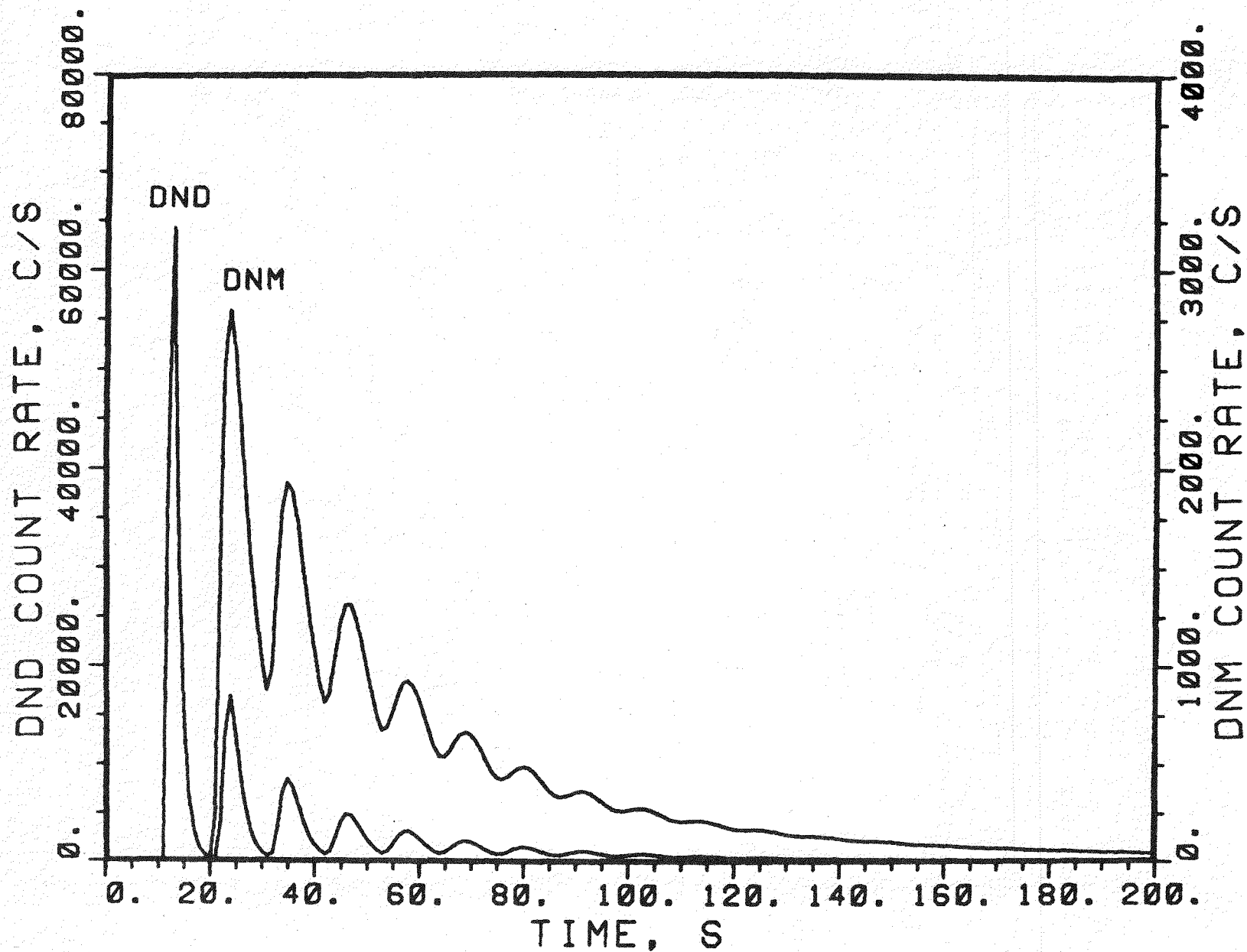


Fig. 52. Calculated DN Response Characteristic for Sudden Release of Molten Fuel

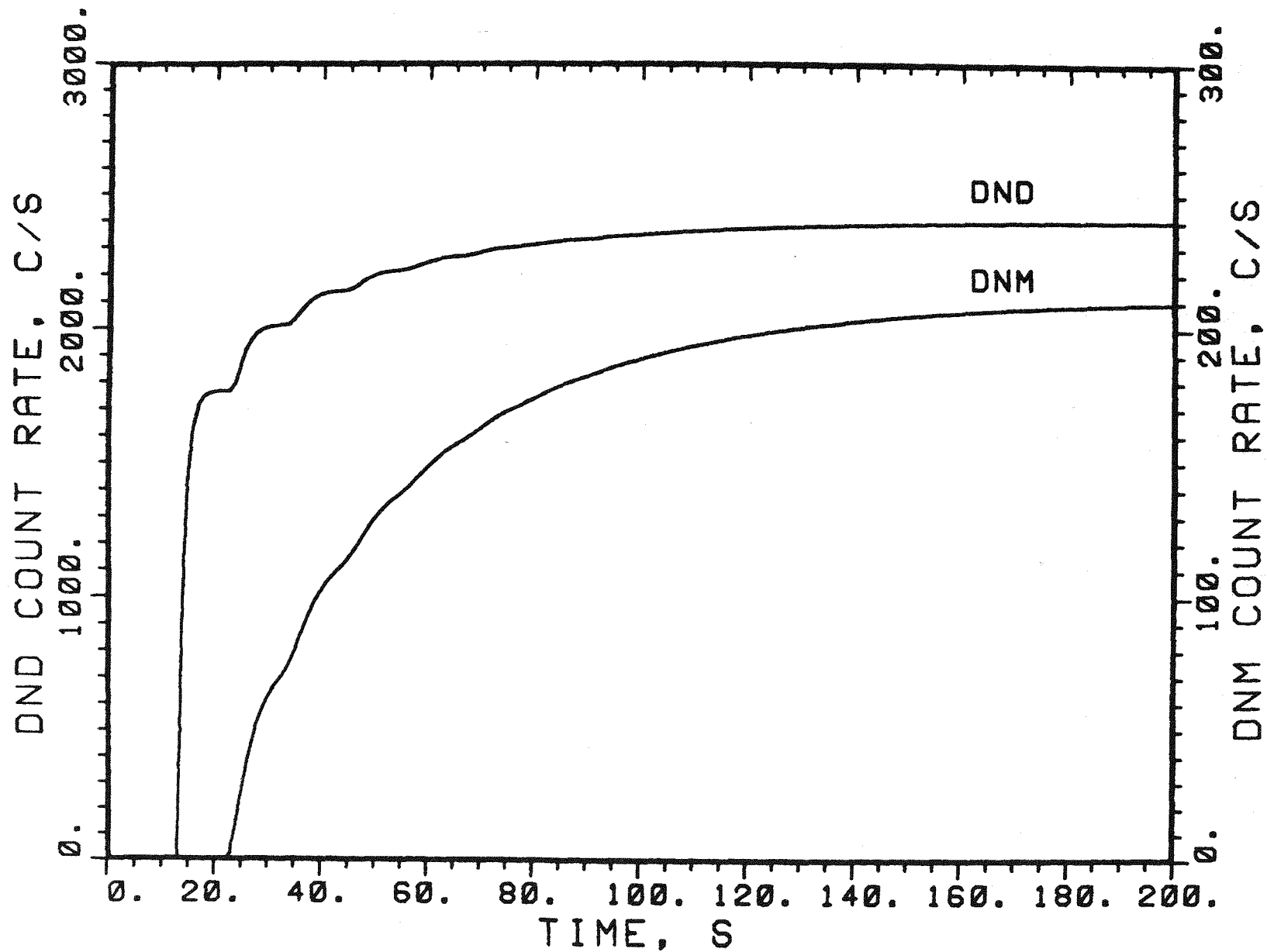


Fig. 53. Calculated DN Response Characteristic for Solid Exposed Fuel

b. Power transient

The DN signals from the DND and the DNM during the power transient were evaluated using both DN simulation models. Figure 54 shows the results from a calculation using the first model, assuming that there was an initial cladding failure at 16 s which exposed solid fuel, that the two large peaks at 36 s and 88 s corresponded to expulsion of molten fuel, and that, after the expulsion, the fuel assumed a solid form which continued to release delayed neutrons. The magnitudes of the molten fuel release and the exposed fuel areas were all adjusted independently to fit the data.

Results from a calculation using the second model are shown in Fig. 55. In this case, the calculation was run so as to deduce the masses of fuel released at any time from the actual count rates, so that there are no independent parameters in the calculation. The "exposed mass" in this case was based on the sudden release of the accumulated DN inventory on fuel melting and the continued steady release of the instantaneous production from the same mass after it was exposed to the sodium coolant. Therefore, the fuel mass corresponding to the peak and the area exposed afterwards are not independent. It is noticeable that after the first large peak corresponding to molten fuel release, the area exposed remains roughly constant, while after the second major expulsion the mass actually falls. The two calculations are in general agreement; the first results in an estimate of about 40 g of fuel for each of the two major peaks, the second, about 60 and 50 g, respectively. In the calculation with the first model, the exposed area after the first major expulsion was  $\sim 63,000 \text{ cm}^2$ . After the second major expulsion, the exposed area was  $\sim 75,000 \text{ cm}^2$ , an increase of only  $\sim 12,000 \text{ cm}^2$ . In the calculation with the second model, the area actually falls after the second expulsion. This suggests that the first molten fuel released remained at high temperature and in a form in which all of the new products could be released, while after the second the fuel mass assumed a form with a smaller effective area, or was cooled sufficiently to retard diffusion of the fission products.

The initial failures of the fluted canister were also studied in some detail. Figure 56 shows the six peaks between 16 and 36 s, each corresponding to a sudden fuel exposure followed by a continuous exposed area. A fuel exposure equivalence of 17 to 20 g was estimated from calculations using the two models.

The minor bumps and peaks between the two major molten fuel expulsions and following the second seem to be due to movements of the blockage mass resulting in changes in the exposed area, or changes in temperature (see temperature record) that result in changes in diffusion rates and therefore changes in effective area. The behavior at and after reactor scram presents difficulties in analysis, and will be discussed later.

c. Blockage reconfiguration

The history of the reconfiguration was discussed earlier. During the increase in reactor power, the mass of fuel exposed earlier produced a substantial DN signal. DN measurements at  $\sim 92 \text{ MW}$  had already shown some indication of a signal increase that was not proportional to power. The slow increase in the DN signal was fitted in the simulation (Fig. 57) by a slow change in effective exposed area or in equivalent exposed mass beginning

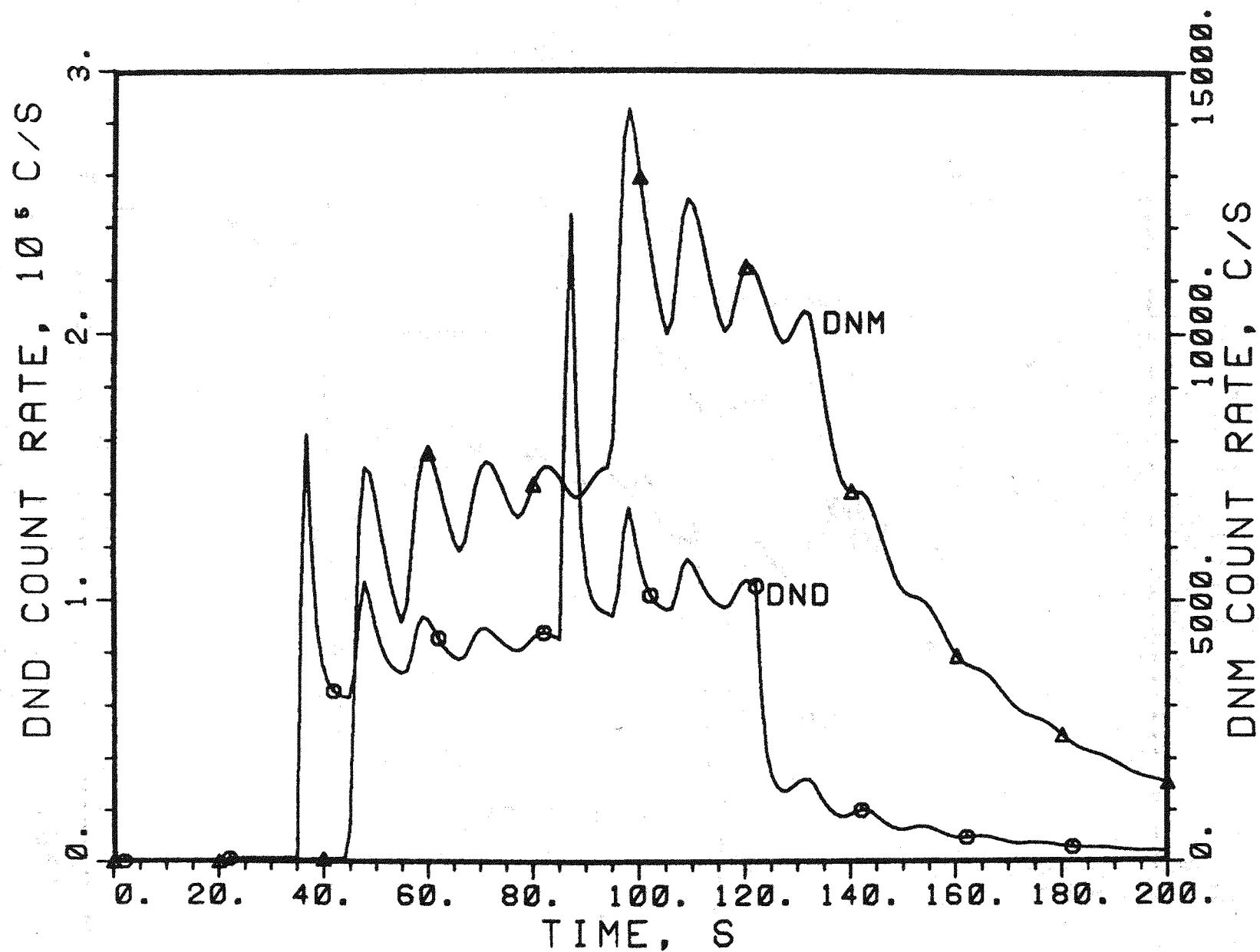


Fig. 54. Simulation of Power Transient Events Using First DN Model

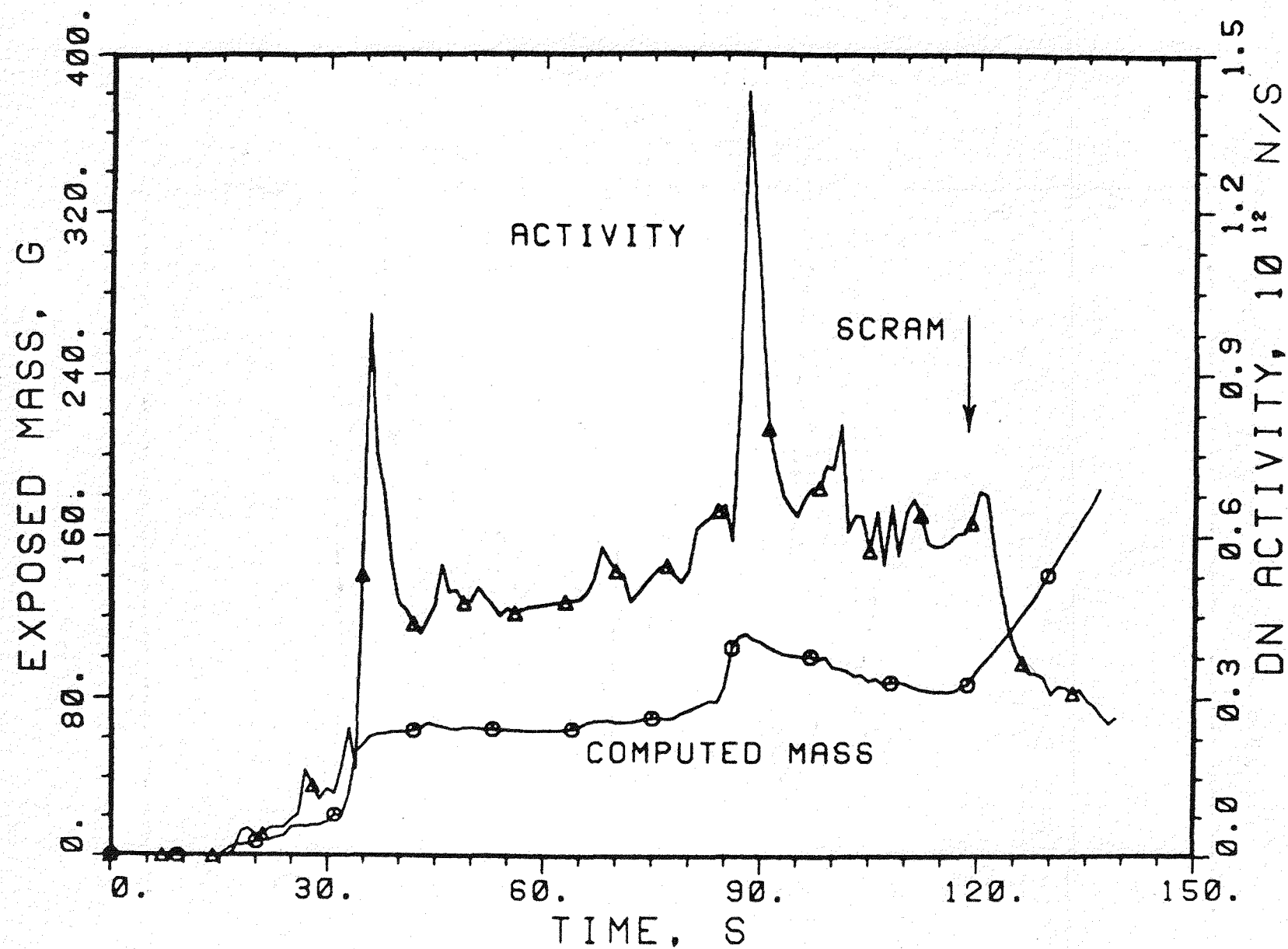


Fig. 55. Simulation of Power Transient Events Using Second DN Model  
 (The DN activity is the DND fission-detector count rate divided by  
 a combined detector sensitivity and geometric efficiency of  $1.5(10^{-7})$   
 counts per delayed neutron emitted in the sodium plenum volume)

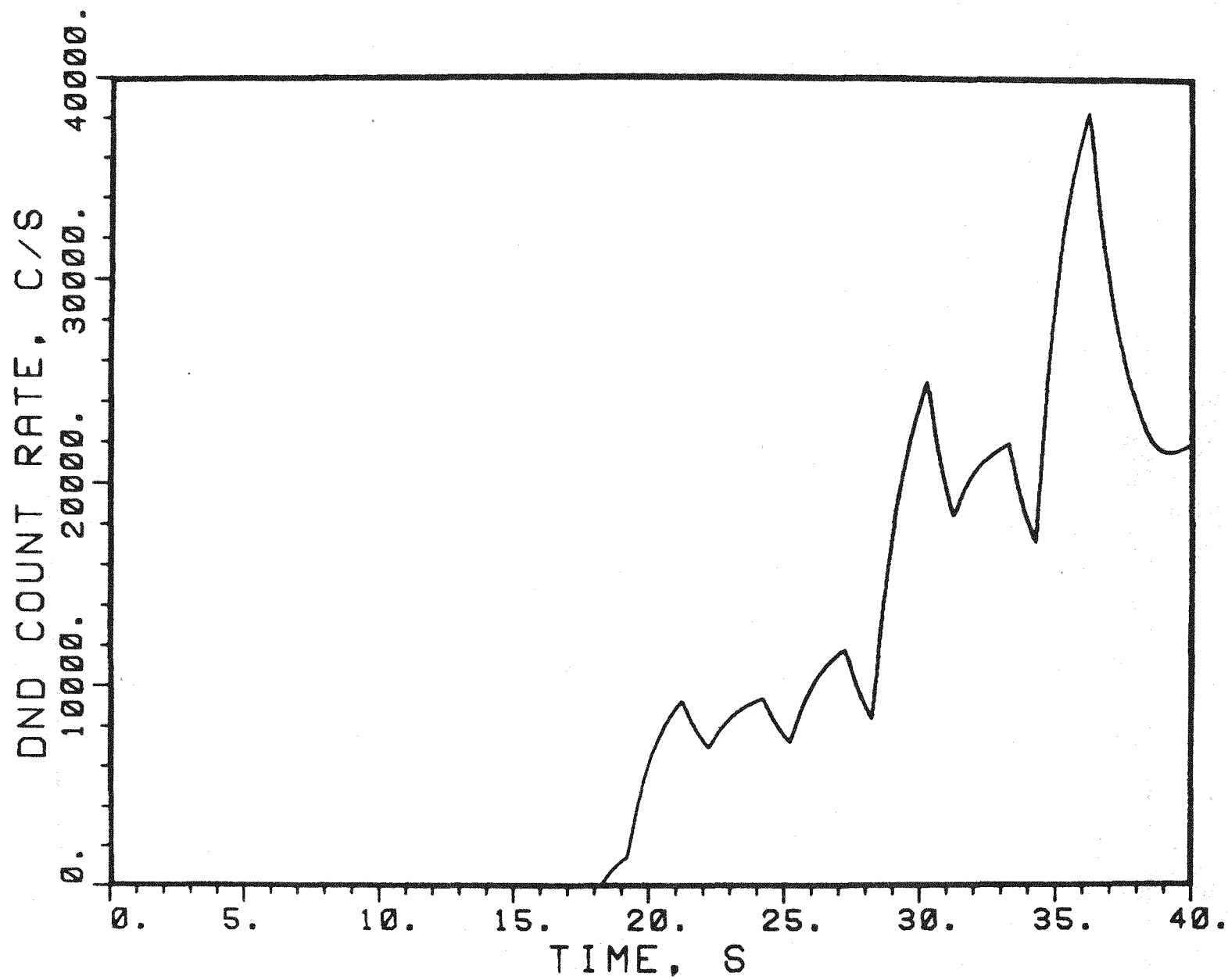


Fig. 56. Simulation of Fluted-Canister Cladding Melting and Fuel Exposure

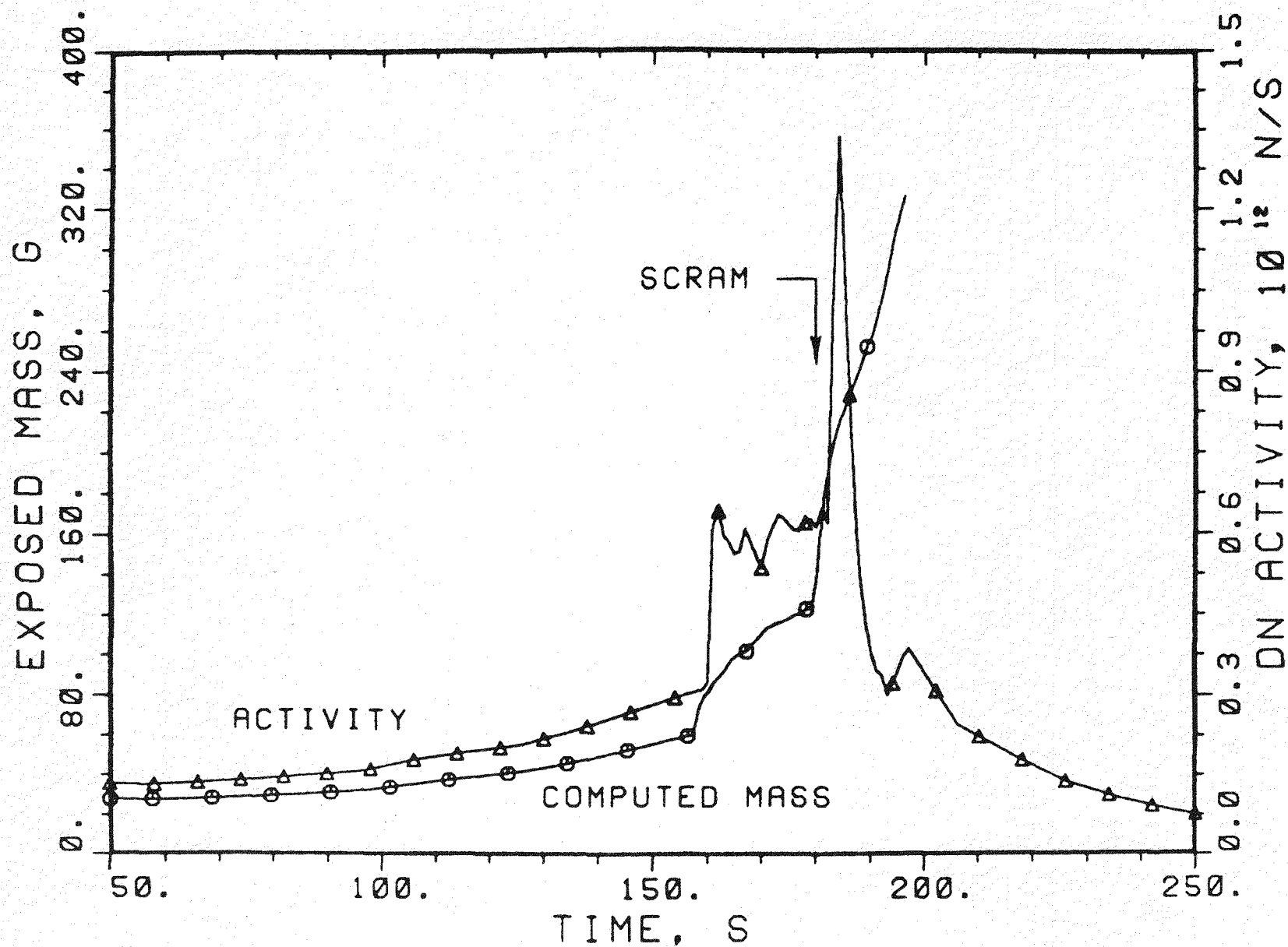


Fig. 57. Simulation of Blockage Reconfiguration Using Second DN Model  
 (The DN activity is the DND fission-detector count rate divided by  
 a combined detector sensitivity and geometric efficiency of  $1.5(10^{-7})$   
 counts per delayed neutron emitted in the sodium plenum volume)

at 20 g and rising to 65 g. Isotope release rates are believed to be diffusion limited at this point as evidenced by a lower calculated exposed mass than at the end of the power transient. Because power, temperature, and flow conditions were uniform, the slow increase in exposed mass could represent subtle structural changes in the exposed fuel, perhaps related to the dissociation of sodium urano-plutonate discussed in Ch. V.

The step increase that occurred at the end of this period is unlike the major failures in the power transient, which were characterized by spikes in DN signal that were compellingly interpreted by molten fuel exposed in a single instant. At first appearance, the DN record indicated a sudden step increase in signal, followed by a 20 s plateau during which no major change occurred. However, any attempt to simulate the DN signal necessitated a continuing process throughout the period. It is not possible to account for both the sharp onset and the plateau on the basis of an event occurring at one single instant. The sudden leading edge can only be explained by a sudden event which exposed 35 g of fuel. The plateau which followed required a continuing increase in the exposed fuel mass amounting to an additional 35 g. It therefore appears that the blockage was changing in some manner throughout this period, which presaged the major event leading to the reconfiguration and reactor scram. The DN peak due to that event corresponded to the exposure of a further 100 g of fuel, either molten or highly fragmented. The further development of the DN signal is confused because of the reactor scram (see d. Postscram events); it cannot be known whether this would have been a spike, as in the two major failures in the power transient, or the leading edge of a plateau as in the event 20 s earlier. The total exposed fuel mass of ~240 g indicated at the end of this event was greater than twice that exposed at the end of the power transient, depending on the reliance which can be placed on the calculation just after reactor scram.

#### d. Postscram events

Analysis of the DN signal becomes increasingly difficult after reactor scram. After scram, the computed exposed mass is extremely sensitive to the exact neutron power shutdown curve as well as background in a rapidly falling DN signal. Current modeling makes use of a "generic" ETR shutdown history<sup>30</sup> applied to isotope generation in the DN loop, which explicably followed parents and daughters in the decay chain. Basically, at times >15 s after scram, measured activity is higher than is to be expected on the basis of the modeling assumptions, and all calculations eventually lead to a physically unreasonable "runaway" fuel exposure. Whether the difficulty lies in the modeling (i.e. the production, release, or transport of DN emitting isotopes) or the present knowledge of fission-counter background or neutronic shutdown of the reactor is not known at present.

Had the P4 power transient and return-to-full-power operations terminated normally with simply a shutdown to a low power, as planned,<sup>14</sup> this difficulty in analysis would not exist. Therefore, some loss of useful information must be attributed to both the power-transient and blockage-reconfiguration reactor scrams.

Despite the difficulty in making quantitative estimates, there is some evidence in the Fig. 55 data of the P4 power transient for subsequent mass exposure of uncertain magnitude at reactor scram, in addition to those



described above. In this case, the DN data indicated a small spike. The calculations after scram show a monotonically increasing exposed fuel mass and, for the reasons given above, are of dubious significance. At later times, when the DN signal was falling smoothly, further mass exposure was implausible. In a subsequent review of the DN data, a signal spike between 138 and 150 s was found to be a data-logging anomaly in the OLCS computer, as indicated in Fig. 22. This anomaly was not included in the current analysis. In the P4 blockage reconfiguration case, the single postscram DN spike observed coincided in timing with a recirculation of isotopes from the large spike observed at scram; therefore postscram mass exposure is unlikely.

Analyzing the SLSF W2 experiment<sup>8</sup> in the same fashion as above yielded a computed plateau of exposed mass for about 10 s postscram. Here spacing of postscram DN peaks also corresponded closely to recirculation times, so any further mass exposed after scram was also unlikely.

The inference of postscram mass exposure after the P4 power transient may be correlated with the loop depressurization and gas release from failed pins at scram (see Table 7). Significant fission product release could have occurred without major changes in the form or quantity of exposed fuel. However, evaluation of the DN data is clouded by the general analysis difficulties described above and the issue must, therefore, remain somewhat unsettled.

### 3. DN Mechanisms Tests

On three separate occasions, tests of the response of the DN systems were made at steady reactor power to evaluate the mechanism of release of the DN fission products. The data were obtained from

- the fission product source before failure,
- the partial flow blockage after the power transient, and
- the partial flow blockage after the reconfiguration.

The measurements were described in their correct historical sequence in Ch. IV and the results are evaluated below.

The measurements that were made were interpreted on the basis of predictions of the CSMP model of the DND and the DNM for the different assumptions which were made about the nature of the source, as discussed previously.

#### Recoil

Fission products leave the fission product source by simple recoil; therefore the process is confined to a thin surface layer, of thickness equal to the fission product range. The effect must be independent of temperature. For fuel, the same effect must exist and should also be independent of temperature.

## Diffusion

Diffusion can account for a much larger signal strength because it is a volume effect. In this case, we expect a temperature dependence of some sort because diffusion constants are temperature dependent. Also, the time delay inherent in the diffusion process affects the distribution of the different isotopes in the population, yielding a dependence on decay constant.

The DN response was investigated by changing the coolant temperature, the transport delay, and the reactor power.

### a. Effect of sodium temperature

The DND count rate was observed while the sodium temperature was varied, on several occasions, as previously described. The first measurements were made with the blockage remaining after the power transient; the data obtained are plotted in Figs. 25 and 34 (circles). The reactor power was 2.2 MW, and the total bundle power was 18 kW. The same measurements were made over a wider temperature range after the reconfiguration; the points are shown in Fig. 34 (crosses). The reactor power was ~1.3 MW and the bundle power was only 11 kW at this time; because the power was lower, the data were normalized to fit on the same curve. The DN signal was approximately the same, indicating that the effective mass of the exposed fuel was ~70% larger after the reconfiguration than after the power transient. ETR power was not increased above 1.3 MW after the reconfiguration, so counting statistics were rather poor. It was difficult to keep the reactor power quite constant over a period of hours; the individual points were normalized to a constant bundle power, as monitored by the rise in temperature of the sodium. Even then, the recorded data show some scatter. It is possible that there was some variation of local neutron flux and the coupling of the fuel to the reactor flux may not have remained constant after the reconfiguration.

Note that the CSMP model is not required for interpretation in this case, as the plot is simply one of count rates at steady flow conditions. The DN population in the plenum reached a steady value within a few minutes of reaching a constant temperature and flow. There is an obvious effect of temperature, which is the same for the initial and reconfigured blockage. The temperature dependence agrees with that deduced from the Arrhenius form of the diffusion constants (Eq. 4) inserted in Eq. 3, with  $Q = 4000$  K and the temperature that of the sodium. Exactly what this means is not easy to say. The temperature in the equation is that of the fuel grains, while the measured temperature was that of the sodium. Sodium temperature is presumably a good indicator of the surface temperature but not of that in the interior; however it may be an adequate approximation at such low power.

The DN data shown in Figs. 25 and 34 were obtained at a fuel-bundle power of 11 to 18 kW. Sodium temperature was also decreased from 700 K to 589 K at a reactor power of 40 MW to further investigate the effect of a reduced internal fuel temperature on the DN signal. The temperature effect at 40 MW was much smaller than that measured at low power. This will be discussed further in the next section.

Even at the lowest temperatures attained there was no sign of a recoil mechanism becoming an important component of the total release; i.e. a dependence on temperature was always apparent. Therefore the data are completely consistent with diffusion from a volume of fuel as the dominant release mechanism.

The change of count rate with temperature is due to the change of  $D$  in  $(D/a^2\lambda)^{1/2}$  in the Booth equation, as discussed earlier. The measurements of  $D$  or  $D/a^2$  that have been made at higher temperatures by Friskney<sup>26</sup> and Prussin<sup>31</sup> indicate differences between different elements and a much higher temperature dependence than inferred from the present measurements. The change of the average value of  $D/a^2$  was roughly 16 over the temperature range 700 K to 458 K in the blockage. This can be compared to a change of 100 for  $^{131}\text{I}$  over the temperature range 1587 to 2083 K.<sup>31</sup>

#### b. Effect of transit delay

In order to investigate the contribution of the different isotopes to the total signal, the transit time to the DNM was changed from 12 to 40 s by varying the OLSS flow rate. Due to the different decay rate of the individual isotopes, the population in the DNM sample chamber changed with delay time. Ideally, it should be possible to deduce from these data the population of isotopes in the loop sodium plenum (as sampled by the OLSS) and ultimately the rate of release of each isotope from the fuel. In particular, we can hope to distinguish between recoil and diffusion release.

Neglecting differences in  $D/a^2$ , the diffusion rates from the fuel grains are enhanced by  $(1/\lambda)^{1/2}$  as compared with unity for simple recoil. Diffusion thus increased the contributions of the first four groups of Table 13 by factors of 8.8, 5.6, 2.9 and 1.8, respectively. Application of diffusion to the decay chains in Table 14 involved somewhat more complexity than using the six DN groups, but no difference in principle. The predictions of the CSMP model are shown in Fig. 58 for both recoil and diffusion. The two calculations were normalized at  $t = 10$  s. The two curves show a complex decay curve as the shorter-lived isotopes die out at longer delays. The effect of diffusion on the calculation is less striking than might be expected and is discussed below. Even at the longest delay times available, the longest-lived isotope contributes only about a third of the total, while two-thirds come from the second group of isotopes. Included in the modeling are the time delay from the fuel to the loop plenum, the recirculation in the loop, and the longer dwell time in the DNM sample volume; all of these tend to blur the distinction between the recoil and diffusion predictions. However, the diffusion curve indicates the expected enhancement of the longer-lived components.

Figure 58 also shows the data obtained from the fission product source before any fuel failure, and one set of data from the fuel debris following the power transient. The magnitudes of the count rates in these two cases were very different but they, too, have been normalized at 10 s to emphasize shape differences. The fission-product-source data were expected to be a good calibration of a pure recoil source.<sup>32</sup> However, the measured data do not agree well with either recoil or diffusion calculations. These data show a distinctly longer-lived component than either calculation and also more than the data from the fuel debris. The reason is not clear. During the W2 experiment, a similar set of measurements were made of the DND count rate as a

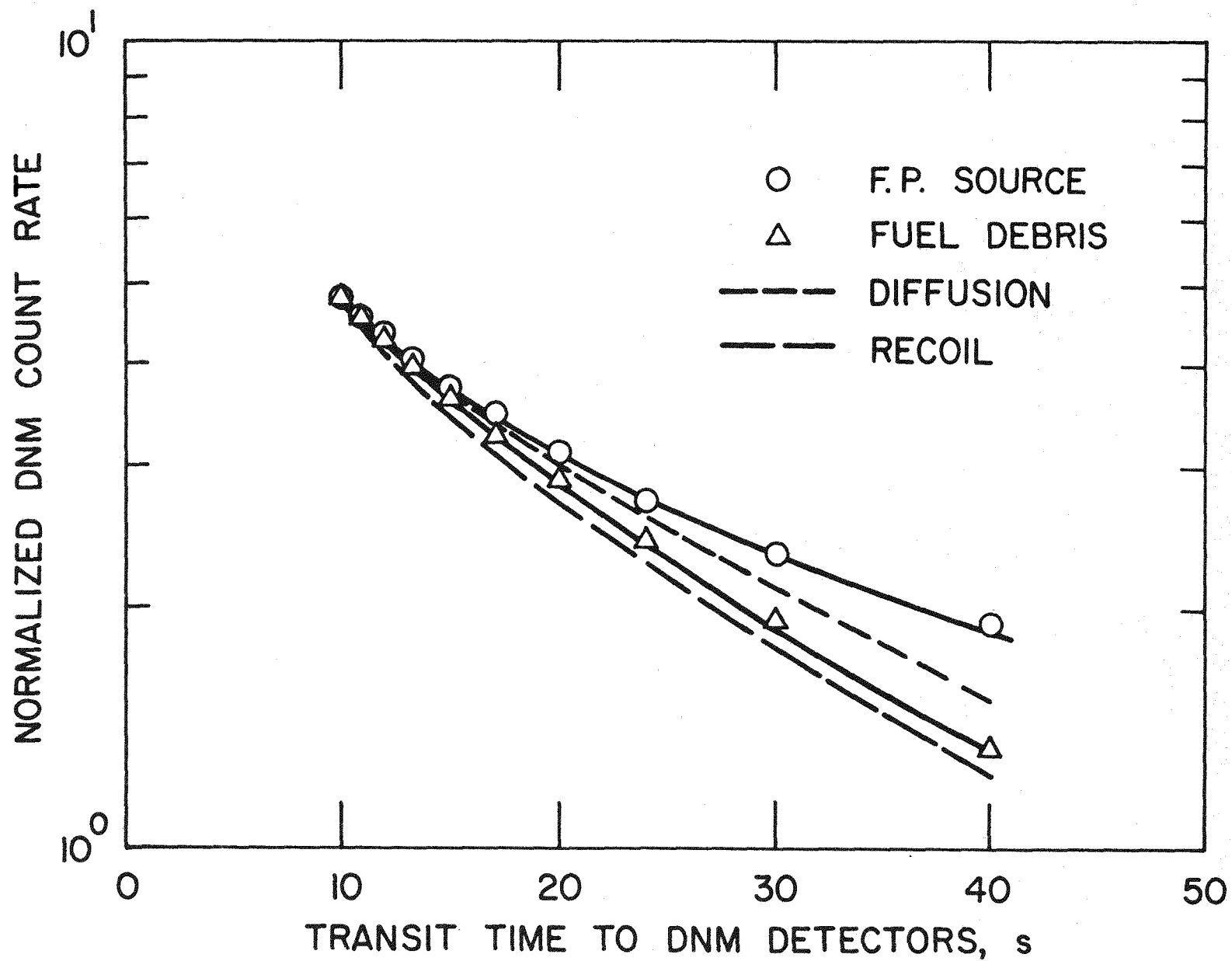


Fig. 58. Comparison of DNM Signal from the Fission Product Source with that from Postfailure Fuel Debris

function of flow in the loop. In that case, quite good agreement was obtained between the fission-produce-source data and recoil calculations. The only explanation that can be advanced is that W2 was operated without a cadmium neutron filter on the secondary vessel and had a higher signal from the fission product source than in P4. The fission-product-source data in P4 must include an unmeasured background due to the neutrons from the reactor core. Background due to photoneutrons was subtracted, but no measurement was feasible of the core background because the fission product source, once installed, could not be removed or turned off. The count rate in the OLSS from the fission product source was quite low at 40-s delay; that from the blockage was ~10 times as great. Therefore a background that was negligible in the case of the fuel could have been a serious factor in discerning the signal from the fission product source.

The blockage DN data fall closer to the calculations describing diffusion than that for recoil. The agreement is not perfect, but that is not unreasonable in view of key simplifying assumptions that have been made in the calculations. The first is that the diffusion term  $D/a^2$  is identical for all isotopes. There is no reason for this to be true; in fact, there is empirical evidence at higher temperatures that they differ.<sup>26,31</sup> The second is that all chemical species are transported equally well in the sodium, and that all survive transport, avoiding plate-out or deposition equally well. In addition, the model assumes uniform velocity in the OLSS piping; laminar flow would have the effect of enhancing the fraction of shorter-lived isotopes at any point in the analysis.

Figure 59 shows a least-squares fit to the blockage data for both recoil and diffusion, normalizing both calculations to the data. It is clear that the diffusion curve is a better fit. An attempt was made to resolve the data into the individual isotopic components. It failed to produce definite and unambiguous results. It seems surprising that the difference between the shape of recoil and diffusion flow dependence is not more marked. The explanation lies partly in the effects of transport delay and recirculation in the loop, which already builds up the longer-lived isotopes in the plenum regardless of release mechanism, partly in the minimum flow delay of 12 s to the DNM, and partly in the fact that the longest-lived isotope is not one of the strongest contributors. The major effect of diffusion is a much stronger and temperature-dependent release.

The DNM response as a function of flow rate (transit delay) was measured at a sodium temperature of 700 K for power levels from 2.2 MW to 80 MW, also at 2.2 MW at a sodium temperature of 589 K. These data are shown in Fig. 32; all show the same dependence on OLSS transit delay. The data obtained at low power down to a temperature of 538 K from the fuel after the reconfiguration (Fig. 35) also show the same transit-delay dependence.

There is evidence<sup>33</sup> that the diffusion effects are different when the fuel is heated internally by fission and when it is heated by external means. The longer-lived isotopes can be coming from deeper in the fission-heated fuel, where the temperature is less affected by the sodium. Such an effect does not seem to be significant for the data already discussed. However, one set of data, at 40 MW and a lower temperature of 589 K, shows a weaker temperature dependence and a slightly different slope than the data taken at 2.2 MW. This appears to indicate that at higher power the internal temperature of the fuel becomes more important to the diffusion, and the sodium temperature less so.

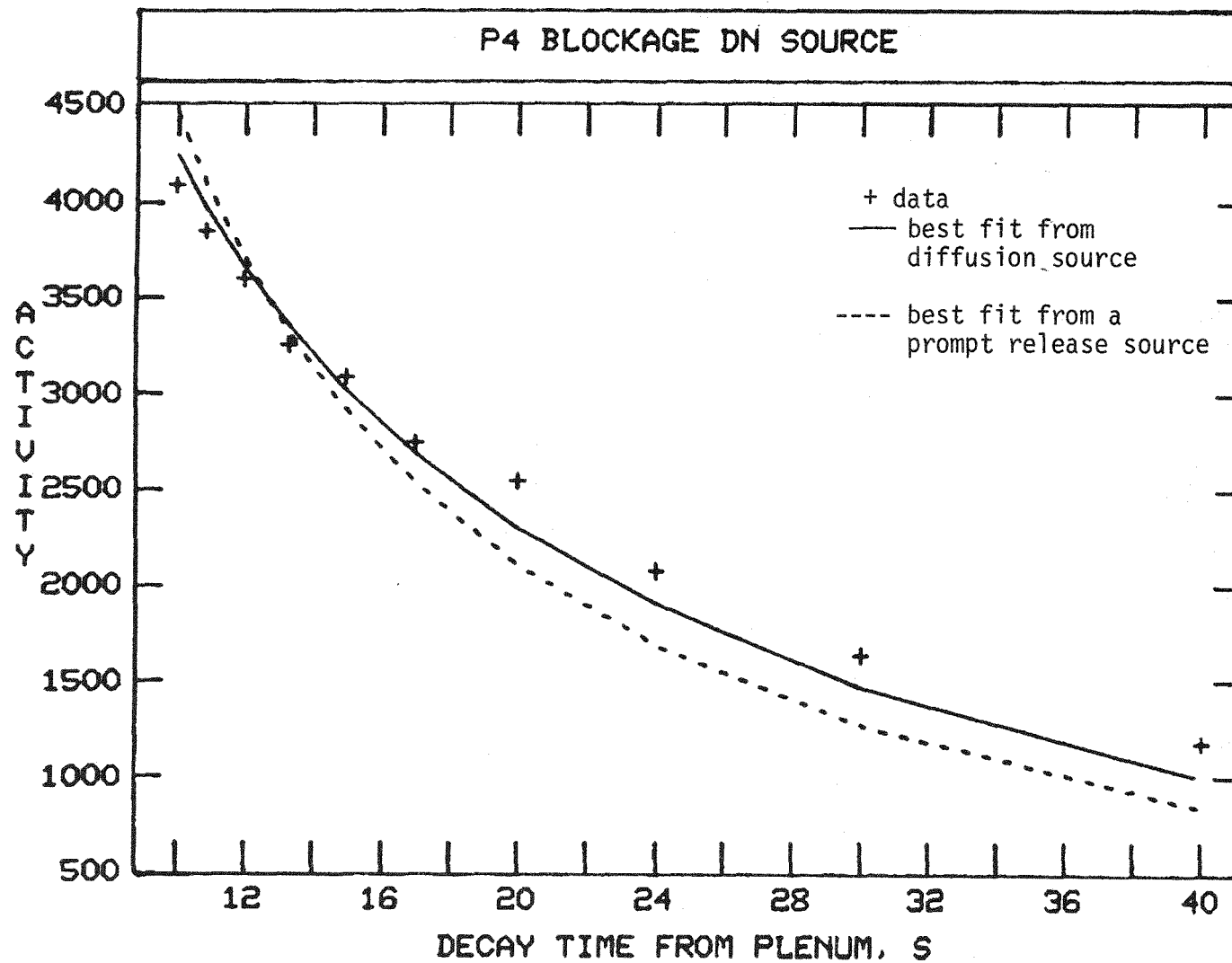


Fig. 59. Comparison of DND Signal During DN Mechanisms Test with Recoil and Diffusion DN Response Characteristics

### c. Effect of reactor power

DND data were obtained from failed fuel and fuel debris at bundle powers ranging from 11 kW to 750 kW (ETR power 1.3 MW to 92 MW) during postfailure operation. The DND count rate as a function of fuel-bundle power is shown in Fig. 33. The data shown as circles are from the first DN mechanisms test and the data illustrated by crosses are from the return-to-full-power operation that was terminated by the reconfiguration event. The data from the two different occasions agree well, suggesting that the blockage had remained undisturbed while the reactor was not at power (25 days).

The DND count rates up to 80 MW were in fairly good agreement with a simple proportionality to reactor power. However, in view of the previous measurements and discussion, it is not clear that this should be so. The sodium temperature increases with reactor power. Since, at constant low power, a definite temperature dependence was measured, it is reasonable to examine whether there is any such effect at higher power. The sodium temperature at midplane (assuming a temperature change linear with power, and the temperature dependence previously measured) was used to calculate R/B from the Booth equation, where B is proportional to the reactor power (Eq. 3) and  $Q = 4000$  K. The resulting R/B relationship is plotted in Fig. 33 as a dotted line; the straight line simply indicates a proportionality of DN signal to reactor power with no sodium temperature effect. The effect of the correction can be seen more clearly in Fig. 60, where the temperature correction has been applied to the DN count rate. Only one point, at the highest power, does not lie on the straight line through the temperature-corrected data. It was obtained at 92 MW, just below the power at which the reconfiguration occurred.

### 4. Summary of DN Analysis Results

- The DN signals exhibited a number of characteristics which strongly suggest that the parent fission products come from a volume of fuel by diffusion.
- There was a clear dependence of the DN signal on sodium temperature, which follows a simple Arrhenius form.
- At steady-state conditions, the DN signal was proportional to fission rate when the above correction for temperature was included.
- The isotopic mix of the DN precursors at the detectors does not depend strongly on either temperature or power.
- Lack of detailed fundamental information on the diffusion and transport of individual isotopes limited the detailed analysis which could be made; nevertheless, estimates of the disrupted fuel mass agree well with those from PTE evaluations.
- DN detectors are particularly sensitive to changes in the fuel morphology and can detect events unseen by other sensors. Multiple fuel exposure events are easily distinguished.
- Small changes in DN responses may precede major changes in fuel and could be used to provide a warning if compared with reactor power measurements in an on-going manner.

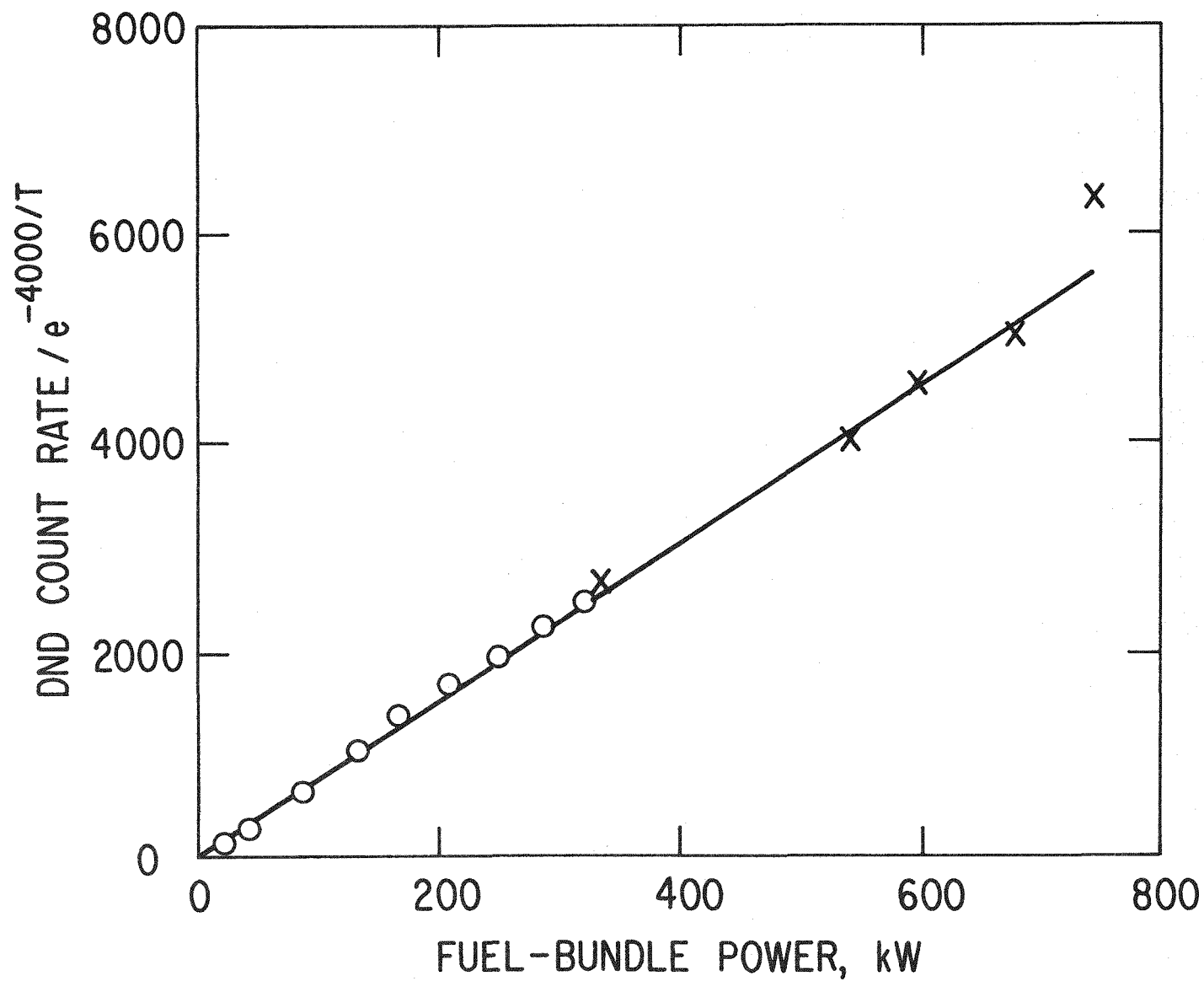


Fig. 60. DND Signal (Temperature Corrected) as a Function of Fuel-Bundle Power



## B. Fuel Canister Thermal-Mechanical Behavior

### 1. FPIN2 Analysis of Cylindrical Fuel Canisters

The FPIN model, developed to analyze the complex thermal-mechanical phenomena which govern fuel and cladding behavior during fast reactor accident transients, was used to analyze the cylindrical fuel canisters during the P4 power transient. FPIN1,<sup>34</sup> an early version of the FPIN2 computer code,<sup>35,36</sup> was used for the pretest analysis. The 20% and 10% CW Type 304 SS cladding on the cylindrical canisters were predicted to reach failure strains of ~8.8% at 24.9 s (150 MW) and 26 s (160 MW), respectively.<sup>16</sup> The failure strain was determined from results of tensile tests on canister cladding material. However, it should be noted that much of the tensile-test data showed anomalous behavior because of difficulties with recovery during the tests and because many of the thin specimens failed in the gripping holes.<sup>16</sup>

During the P4 power transient, the 20% CW cylindrical fuel canister failed somewhat later than predicted and the 10% CW canister did not fail on its own but suffered a cladding meltthrough from fuel ejected from the fluted canister. A posttest analysis of the cylindrical fuel canister was performed using the measured experiment conditions, stress-rupture correlations for Type 316 SS, and the much improved, current developmental version FPIN2 code. The posttest analysis took advantage of much more rigorous treatment of fuel and cladding mechanics and cavity pressurization contained in FPIN2. Results of the posttest analysis are contained in a separate report,<sup>37</sup> which includes a description of the FPIN2 code, selection of input parameters that characterize the cylindrical canisters, thermal-mechanical response of the canisters during the power transient, and appropriate failure criteria to interpret failure of the canisters. This section is a summary of that report.

#### a. FPIN description

The FPIN model simulates the fuel and blanket regions of a single fuel pin which is divided into a number of axial segments (as many as 20 in the FPIN2 version). In general, each segment can have regions containing central cavity gas and/or molten fuel, solid fuel, cracked fuel, and solid cladding. When the heat transfer option to calculate the coolant temperature is invoked, the fuel pin is surrounded by a circular coolant channel and outer wall. Each axial segment of fuel and cladding is further divided radially into a number of finite element rings (20 in the fuel and 10 in the cladding). Axial symmetry of temperatures and external loads is assumed and gradients of temperature and loads in the axial direction are assumed small. At the start of the analysis, the fuel pin is assumed to be stress-free with steady-state thermal conditions.

The basic FPIN program includes a group of subroutines to calculate the temperature of the fuel, cladding, coolant, and outer wall, a separate group of subroutines to calculate the mechanical behavior of the fuel and cladding, and facilities to couple to other related programs. The earlier FPIN1 code used a semi-analytical method to calculate stresses and displacements; material displacements were limited to small perturbations of the initial pin geometry. FPIN2 can handle large strains by virtue of a completely new fuel-cladding mechanics section. The major modifications include changes in the heat transfer subroutines to account for deformed fuel geometry

and implementation of a finite-element mechanical analysis that allows for great flexibility in handling various constitutive equations and cracking models. Simple linear elements are used that provide the framework needed to handle the complications due to complex material behavior and moving boundaries due to fuel cracking and melting. The finite-element method was chosen because it allows for convenient modular coding of the fuel pin mechanics such that different models for material behavior and improvements in specific algorithms can be implemented easily in the code.

b. Characterization of cylindrical fuel canisters

In both the pretest and posttest analyses, only the 10-cm length of the cylindrical fuel canisters was modeled, not the entire pin. Because little axial variation was expected over this short length, only one axial segment was used. The initial room-temperature, zero-power dimensions of the canisters are shown in Table 15. Because the smeared density of the fuel within the canisters was approximately 91%, the molten cavity pressure would increase rapidly due to the 12% expansion on melting. The volume between the fuel column and the end cap must be made available to the molten cavity. FPIN2 does not have a plenum model that would apply so it was assumed that the fuel column extended the full inside length of the canister and contained a central hole with volume equal to the end volume. In addition, some of the volume in the fuel cladding gap was incorporated into the central hole to overcome computational difficulties with the FPIN2 cavity pressure calculation when relatively fresh fuel was specified with little fission gas. The total adjustments in volumes available for fuel expansion were made in such a way as to maintain the specified smear density.

To establish the hot dimensions and fuel crack pattern just before the transient, the pin power and inlet coolant temperature were raised from room temperature and zero power to operating conditions over a period of 60 s. Although FPIN2 does not calculate phenomena such as fuel restructuring and steady-state fission gas swelling, the calculated "hot" conditions should represent a reasonable characterization of the fuel that had only six days of irradiation at a power level of about 60 W/g. The only additional steady-state parameter that needed to be specified was the fission gas content of the fuel. Assuming no gas had been released, the gas retained in the fuel was about  $5.2 \times 10^{-4}$  g/cm<sup>3</sup>. Any additional gas trapped in the 5% porosity of the fuel during sintering would have added only  $2 \times 10^{-6}$  g/cm<sup>3</sup> to the total. The cavity pressure prior to fuel melting was calculated using the option that allowed no escape of gas to a plenum. The mass of the gas in the central hole was determined from the initial pressurization of 2.5 bar.

Properties for Type 316 SS cladding were used in the posttest analysis. Differences in material properties between the Type 304 SS cladding of the canisters and Type 316 SS were considered of minor significance. The amount of initial cold work in the canisters was handled through the hardness parameter in the FPIN2 cladding deformation model. For the cases of 10 and 20% reduction in wall thickness, the appropriate strain hardening values are given by  $|\ln(1.0 - 0.1)| = 0.105$  and  $|\ln(1.0 - 0.2)| = 0.223$ , respectively.

The fill gas in the two cylindrical canisters was 100% Kr. Although an arbitrary mix of He, Kr, and Xe can be specified in the more complex gap conductance models in FPIN, the use of pure Kr is outside the data base for

Table 15. Cylindrical Canister Dimensions and Volumes

|                                  | <u>10% CW (Pin 8)</u> | <u>20% CW (Pin 5)</u> |
|----------------------------------|-----------------------|-----------------------|
| Canister inside length, cm       | 10.17                 | 10.17                 |
| Canister ID, cm                  | 0.84                  | 0.84                  |
| Canister OD, cm                  | 0.87                  | 0.87                  |
| Fuel column length, cm           | 9.964                 | 9.982                 |
| End volume, cm <sup>3</sup>      | 0.114                 | 0.104                 |
| -- Radius of equivalent hole, cm | 0.0597                | 0.0571                |
| Radial fuel-cladding gap, cm     | 0.0051                | 0.0025                |
| -- F-C Gap for FPIN2, cm         | 0.002                 | 0.001                 |
| -- Radius of FPIN2 hole, cm      | 0.0780                | 0.0703                |

which the models have been verified. Furthermore, the more complex models make the FPIN results more difficult to interpret because the thermal and mechanical analyses are strongly coupled through the dependence of the gap conductance on fuel-cladding contact pressure. For these reasons, the gap conductance was assumed to be constant. The value chosen,  $0.40 \text{ W/cm}^2\cdot\text{K}$ , is consistent with the range of results predicted by the complex gap-conductance models.

#### c. FPIN2 results

The FPIN2 thermal analysis was performed using the ETR power-time history during the power transient and the FPIN option in which cladding outer surface temperatures are input. This option was chosen because the fuel canisters blocked off much of the flow in several subchannels, producing a flow pattern that was clearly three-dimensional, with bypass flow and possible hot spots. The cladding surface temperatures were based on an interpolation of bulk sodium temperatures at the fuel canister lower and upper end caps. A 10 K temperature drop from the cladding to the coolant was assumed for heat transfer from the heat-generating fuel canisters. With no correction for bypass flow or hot spot conditions, the cladding temperatures used in FPIN2 were probably low.

Areal melt fractions (based on the solidus radius) for the two cylindrical canisters are shown in Fig. 61. Melting of the fuel in the 10% CW canister began at 17.8 s. Melting of the fuel in the 20% CW canister was delayed until 19.3 s because of the lower power coupling for its position one row closer to the center of the subassembly. Areal melt fractions continued to increase rapidly until the reactor power leveled off. Some additional melting occurred after peak reactor power was reached at 28 s because of heat capacity effects and the increasing coolant inlet temperature.

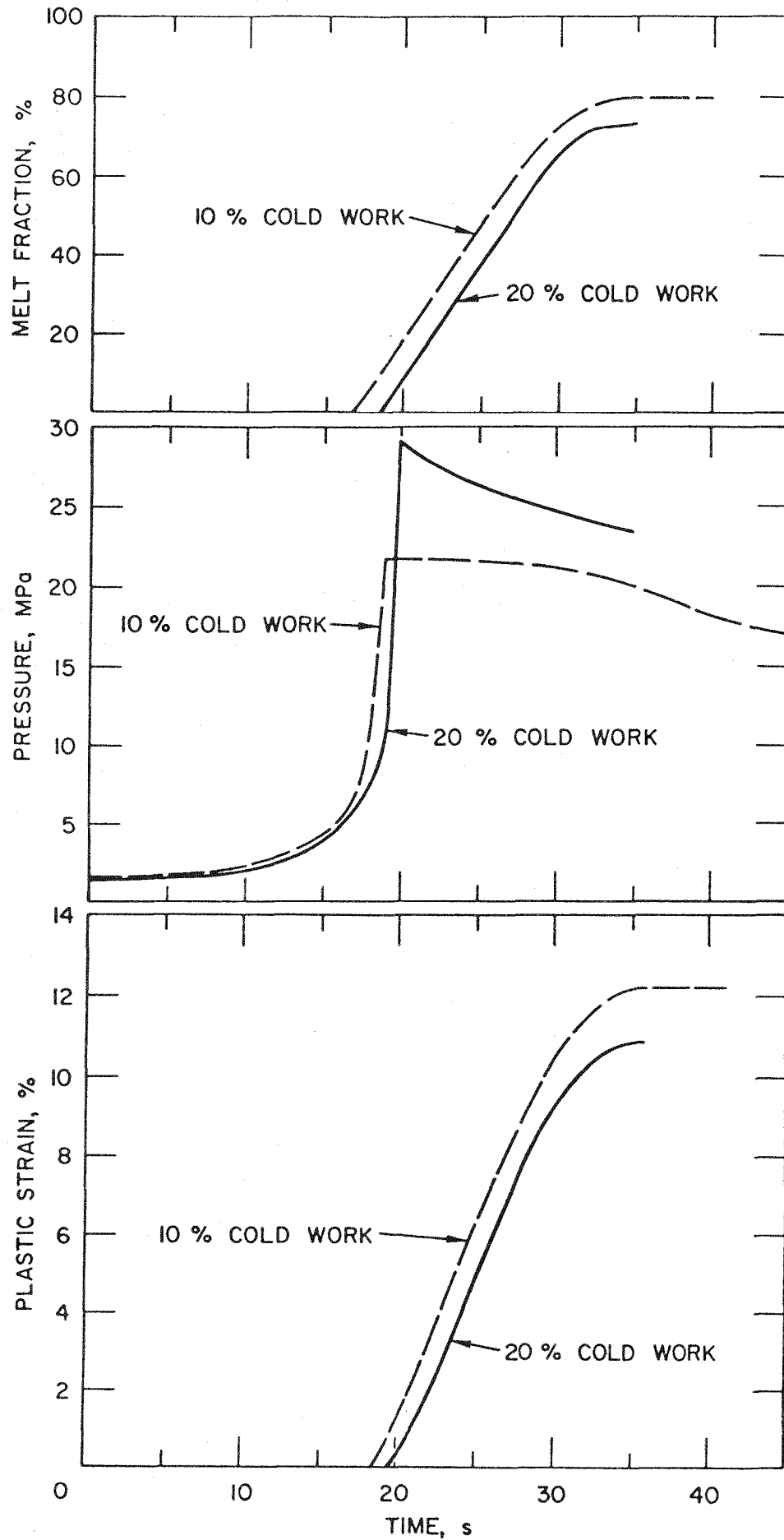


Fig. 61. Calculated Fuel Melt Fraction, Cavity Pressure, and Plastic Strain in the Cylindrical Fuel Canisters During the Power Transient

The molten fuel cavity pressure in the canisters is also shown in Fig. 61. Initially the pressure was about 6 times the fill gas pressure because of the increase in the gas temperature at steady-state conditions. When melting began, the pressure increased rapidly as the available volume inside the canisters was taken up by the expansion of the melting fuel. During this stage, the cavity pressure was estimated by assuming that the walls of the cavity were rigid and that only the fill gas contributed to the pressurization. It is easy to show that, in this case, the ratio of the cavity pressure  $P$  to the initial cavity pressure  $P_0$  is approximately

$$P/P_0 = \left\{ \left[ \frac{R_M}{R_C} \right]^2 [1 - (1 - \bar{P})(1 + \Delta)] + (1 - \bar{P})(1 + \Delta) \right\}^{-1} \quad (5)$$

where  $R_M$  is the melt radius,  $R_C$  is the cavity radius prior to melting,  $\Delta$  is the expansion of  $UO_2$  on melting (0.12) and  $\bar{P}$  is the fuel porosity (0.05). The pressure tends to infinity in Eq. 5 when  $R_M/R_C = 4.1$ . Using the cavity radius and fuel dimensions given in Table 15, the areal melt fractions corresponding to the pressure asymptote are 56% and 46% for the 10% CW and the 20% CW canisters, respectively. In fact, rapid pressurization begins somewhat earlier in the FPIN2 calculations because the cavity radius at fuel melting has decreased by about 50% due to inward creep of the cavity wall. This deformation is primarily due to the stresses from the differential expansion between the fuel and cladding. If the actual cavity dimensions at fuel melting are used in the above equation, the melt fractions at which the pressure tends to infinity are 20% and 12% for the 10% CW and 20% CW canisters, respectively. These values are consistent with the FPIN2 results shown in Fig. 61.

The cavity pressure cannot increase without limit as melting proceeds. Eventually the fuel must crack and/or soften to the point where most of the cavity pressure is transmitted to the cladding. As a result of this loading the cladding begins to deform plastically, as shown in Fig. 61. The cladding deforms at an ever-increasing rate until dislocation glide becomes the dominant deformation mechanism at which point the cavity pressure is dominated by the high strain-rate flow stress  $\sigma_{\theta y}$ . If the outer fuel is cracked such that it can no longer support a tangential stress, a simple force balance shows that the cavity pressure is related to the cladding stress by

$$P = \frac{\sigma_{\theta y} h}{R_m} \quad (6)$$

where  $h$  is the cladding thickness and  $R_m$  is the melt radius. As melting proceeds the pressure decreases. If, on the other hand, cracks in the outer fuel are closed, but the outer fuel is sufficiently hot so that its stress state is nearly hydrostatic, the same force balance gives a constant pressure

$$P = \frac{\sigma_{\theta y} h}{R_0} \quad (7)$$

where  $R_0$  is the cladding inner radius.

The FPIN2-calculated cavity pressure illustrated in Fig. 61 shows a behavior intermediate between results given by Eqs. 6 and 7. Of course, the 10% CW canister yields at a lower flow stress than the 20% CW canister. The cavity pressure for the 20% CW canister drops off more rapidly than the cavity pressure for the 10% CW canister because the fuel cracks are

more open in this case, giving a stronger dependence of the pressure on the melt radius. It is important to note from Fig. 61, however, that the total plastic strain in the two canisters was nearly equal by the time the power leveled off. The reason for this is that most of the plastic strain is determined by the excess volume that must be generated to accommodate the expansion of molten fuel. Because the melt fractions and the available volumes are nearly the same for both canisters, the plastic strains must be nearly the same.

The above result can easily be understood by calculating how much the canisters would have to deform if all the fuel inside of them melted. From room temperature to melting the volume expansion of  $UO_2$  is 27%.<sup>38</sup> The free volume inside the canisters can be determined from the end volume and fuel-cladding gap, as given in Table 15, plus the pore volume in the 95% dense fuel. Subtracting the expansion volume from the available volume shows that the cladding must deform to accommodate about  $0.82 \text{ cm}^3$ . Part of this deformation comes from thermal expansion of the cladding, but most of it must come from plastic deformation. Because the biaxial stress state in the cladding is nearly 2:1 after fuel melting begins, most of the plastic deformation is diametral with only small changes in the length of the canisters. The increase in volume  $\Delta V$  due to the plastic strain is then just

$$\Delta V = 2\pi R_0^2 L \epsilon_{\theta p} \quad (8)$$

where  $R_0$  is the inner radius of the canister,  $L$  is its length and  $\epsilon_{\theta p}$  is the plastic hoop strain. Equation 8 shows the  $\epsilon_{\theta p} = 7.3\%$  for  $\Delta V = 0.82 \text{ cm}^3$ . The plastic strains from the FPIN2 calculations shown in Fig. 61 are somewhat larger because of early differential expansion strains, rate-dependent plastic deformation and because fuel vapor pressure loads the cladding in addition to the loading caused by molten fuel expansion.

The FPIN2 calculations for the 20% CW canister were terminated at 34 s. FPIN2 calculations for the 10% CW canister were carried out to 86 s, but results are shown in Fig. 61 out to only 40 s. Because the reactor power had leveled off by 40 s, subsequent cladding deformation occurred very slowly and the cavity pressure decreased only slightly. For instance, between 40 and 86 s the cladding plastic strain for the 10% CW canister increased only from 12.65 to 12.71%.

FPIN2-calculated results of interest for the interpretation of the P4 experiment are shown in Table 16 for the experimentally determined failure times of the cylindrical canisters.

#### d. Cladding failure

Appropriate failure criteria were applied to the FPIN2-calculated cylindrical-canister cladding response during the P4 power transient to interpret canister failure. During the transient, the cylindrical cladding temperature is estimated to have increased from about 750 K to 880 K at an average heating rate of 5 K/s. The most applicable experimental data from which to predict cladding failure under these conditions was from the HEDL FCTT tests.<sup>39,40</sup>

Table 16. Calculated Cylindrical Canister Conditions at the Experimentally Determined Failure Times

|                                                                             | 20% CW | 10% CW |
|-----------------------------------------------------------------------------|--------|--------|
| Failure time, s                                                             | 34     | 86     |
| Areal melt fraction, %                                                      | 72     | 79     |
| Fraction of fuel that has reached the liquidus, %                           | 69     | 76     |
| Molten fuel volume, cm <sup>3</sup>                                         | 5.6    | 6.1    |
| Cavity gas volume, cm <sup>3</sup>                                          | 0.072  | 0.20   |
| Approximate molten fuel mass, g<br>(areal melt fraction times initial mass) | 40     | 43     |
| Cavity pressure, MPa                                                        | 24     | 16.3   |
| Fuel vapor pressure, MPa                                                    | 3.3    | 7.6    |
| Cladding plastic strain, %                                                  | 11.3   | 12.7   |
| Average cladding temperature, K                                             | 890    | 927    |

The use of a constant, rather small, value of the gap conductance may tend to overestimate the amount of fuel melting. Table 17 displays the reduction in fuel melting for larger values of gap conductance. These values were calculated by FPIN2 using an option that allows the extremely fast calculation of only the thermal history without any mechanical analysis.

Table 17. Effect of Fuel-Cladding Gap Conductance on Thermal Conditions at the Experimentally Determined Failure Times

|                                                   | 20% CW |       |       | 10% CW |       |       |
|---------------------------------------------------|--------|-------|-------|--------|-------|-------|
| Failure time, s                                   | 34     | 34    | 34    | 86     | 86    | 86    |
| Fuel-clad gap conductance, W/cm <sup>2</sup> ·K   | 0.4    | 0.8   | 1.2   | 0.4    | 0.8   | 1.2   |
| Areal melt fraction, %                            | 72     | 59    | 53    | 79     | 66    | 60    |
| Fraction of fuel that has reached the liquidus, % | 69     | 55    | 49    | 76     | 62    | 56    |
| Radius where fuel is at solidus temp., cm         | 0.357  | 0.321 | 0.305 | 0.373  | 0.338 | 0.322 |

The FPIN2 calculations above show that the 20% CW cylindrical canister failed under a state of 2:1 biaxial stress due to the nearly hydrostatic pressure of expanding molten fuel. If this pressure were constant, as in the FCTT tests, one would expect the 10% CW canister to fail first because it had the largest melt fraction and the lowest ultimate strength. However, the pressure was not constant in P4. Instead, the loading was due to a nearly incompressible mixture of molten fuel and small amounts of gas. Any increase in pressure could easily be relieved by plastic deformation of the cladding. As mentioned previously, that is why the cavity pressure, shown in Fig. 61, was dictated by the cladding flow stress once plastic deformation began. This also meant that unstable deformation would not occur at the same plastic strain at which it would occur under the constant pressure conditions or conditions where the loading medium was highly compressible. Under these latter conditions, the pressure would not change significantly with deformation. Once gross plastic instability occurred, the increase in hoop stress due to wall thinning was greater than the increase in flow stress due to the plastic deformation.

If the unstable deformation was localized, such as formation of a bulge or necking of the wall, the resulting small increase in volume would not greatly reduce the pressure. The P4 canisters were short; formation of a bulge that was much smaller than the overall dimensions of the canister was unlikely. Local necking of the cladding wall along the axial direction, however, could occur. The neck that formed would be under the same state of plane strain as occurs in a thin sheet under tension. This is because the stress through the thickness is zero and the axial deformation is constrained by the neighboring material which did not deform axially. Uniform elongation before this mode of deformation becomes unstable is much greater than the elongation that would produce unstable deformation under constant pressure. This is the reason that elongations and strengths are greater when tubes are loaded with an expanding mandrel than when they are loaded with gas pressure. In the P4 experiment, it meant that the strains to failure would be expected to be larger than data from pressurized tube experiments would suggest.

At the given temperatures of the cladding, the above analysis of failure by local necking predicted diametral failure strains of 24% and 12% for the 10% and 20% CW canisters, respectively. Comparing these strains with the calculated strains at the failure times given in Table 16 shows that the 10% CW canister would not have been expected to fail mechanically. The plastic strain (11.3%) in the 20% CW canister at the experimentally determined failure time was somewhat less than the above failure strain (12%). However, given the uncertainties in the sodium temperatures and gap conductances that were used in the FPIN2 calculations and the uncertainties in the mechanical properties of the P4 canister materials, the agreement between calculated and experimentally determined failure conditions is very good.

In summary, the canister cladding strain was almost entirely due to the volumetric expansion of the fuel on melting. The mode of canister cladding failure was plastic instability by necking of the canister wall. The failure time of the 20% CW canister and the non-mechanical failure of the 10% CW canister are consistent with the FPIN2 calculations using the plastic instability failure criterion.



## 2. THTB Analysis of Fluted Fuel Canister

A three-axial-node THTB model of a 30° segment of the fluted canister and adjacent fuel pin was developed during P4 experiment preparation to study coolant, cladding, and fuel temperatures. It had 51 fuel nodes (15 in the fluted canister, 36 in the fuel pin), 15 cladding nodes, and 15 sodium coolant nodes in each of the three axial planes. This model correctly predicted that integrity of the fluted canister would be maintained through the low-power ( $< 40$  MW) testing and that cladding failure would result from bulk sodium boiling in the annular coolant channels between the fluted canister and adjacent pins at  $\sim 14.5$  to 16 s into the power transient.<sup>16,41</sup>

The THTB model was modified to study the temperature field of the fluted canister and adjacent pins beyond the initial cladding melting. Following coolant boiling and cladding melting, the thin annular cooling channel between the fluted canister and the adjacent pins would disappear and the molten cladding would be displaced as the pin and canister fuel came into contact. Rather than revise the model to remove the annular cooling channel and both the fuel-pin and fluted-canister cladding in the cusp region, the model geometry and nodal structure were kept intact and only a material change made. The cusp-region cladding and annular cooling channel were replaced with an anisotropic material having high thermal conductivity in the radial direction, low thermal conductivity in the circumferential direction, and low specific heat. This combination of properties simulated the characteristics of heat transfer from the fluted-canister fuel to the adjacent fuel pins in the absence of the cladding and annular cooling channel.

First melting of fuel in the fluted canister occurred at about 10 s into the power transient. The fuel temperature and fuel melt fraction increased rapidly. At 28 s, when reactor power reached 175 MW, the fuel melt fraction was estimated to be  $\sim 0.6$  and the peak fuel temperature  $\sim 5000$  K. As the power transient continued, the fuel melt fraction increased until the only fluted canister fuel remaining unmelted was a thin strip at the tip of each flute and a shell adjacent to the top and bottom end caps. Peak fuel temperatures exceeding 5810 K ( $10,000^\circ\text{F}$ ) were calculated. Fuel was also melted in the adjacent fuel pins from the contact point in each cusp out to the centerline of each fuel column.

The calculated temperatures of the molten fuel were not very meaningful because the extrapolation was well beyond the range of validity for both the model and the fuel thermal properties. However, the results indicated that peak fuel temperatures in the fluted canister were exceedingly high. The damage potential, due both to the volume of molten fuel and the extreme fuel temperatures, was substantial and bounded that resulting from postulated local fault events.

### C. Partial Flow Blockage

#### 1. Effect of Partial Blockage on Coolant Flow Rate and Pressure Levels

Substantial changes in the fuel-bundle coolant flow rate arising from partial flow blockages were observed on four occasions during P4 power operation. They were as follows:

- Reduction in flow rate from 100% to 93% at 34 s in the power transient due to molten fuel ejection from the cylindrical fuel canister on Pin 5,
- Increase in flow rate from 93 to 97% at 53 to 54 s in the power transient,
- Reduction in flow rate from 97% to 86% at 86 s in the power transient due to molten fuel ejection from the fluted fuel canister on Pin 12,
- Reduction in flow rate from 86% to 60% accompanying the blockage reconfiguration.

The influence of a partial flow blockage on fuel-bundle and loop coolant flow rates had been evaluated during the P4 pretest analysis.<sup>16</sup> Results were based on a 10.2-cm-long, smooth, planar blockage blocking from 14% (same areal blockage as the three fuel canisters) to 100% of the fuel-bundle coolant flow area. These results were used to estimate the extent of each partial flow channel blockage observed in P4, as shown in Fig. 62. Were these results correct, the 86% bundle coolant flow rate existing at the end of power transient operation corresponded to a 69% areal flow blockage and the 60% bundle coolant flow rate after the reconfiguration corresponded to an 85% areal flow blockage.

To determine the actual flow blockage, each macrosection of the fuel bundle was scrutinized to identify individual coolant subchannels as being either completely blocked, partially (half) blocked, or fully open. Details on the partial flow blockage found in each macrosection of the fuel bundle in the fuel canister region are shown in Fig. 63. Then the total flow channel area blocked in each macrosection was estimated from the known areas of center, edge, and corner channel in an undisrupted pin-bundle geometry. The resulting fraction of local coolant flow area blocked by fuel or cladding debris in each macrosection through the P4 fuel bundle is shown in Table 18. Through the fuel canister region, shown bounded by brackets joining macrosections B6 and B13, the fraction of coolant channel flow area blocked varied from a minimum of 7% at the lower end cap to a maximum of 35% at the elevation of the lower banding strap around the fluted canister and six adjoining pins. For macrosections B8 through B13, the coolant flow channel areal blockage was surprisingly uniform at ~30%. The total areal blockage of the 37-pin bundle due to both the disrupted fuel and cladding and the canisters was a maximum of about 43%. The actual areal flow blockage due to disrupted fuel and cladding in the coolant flow channels through the canister region of the P4 fuel bundle, then, was only half that inferred from Fig. 62.

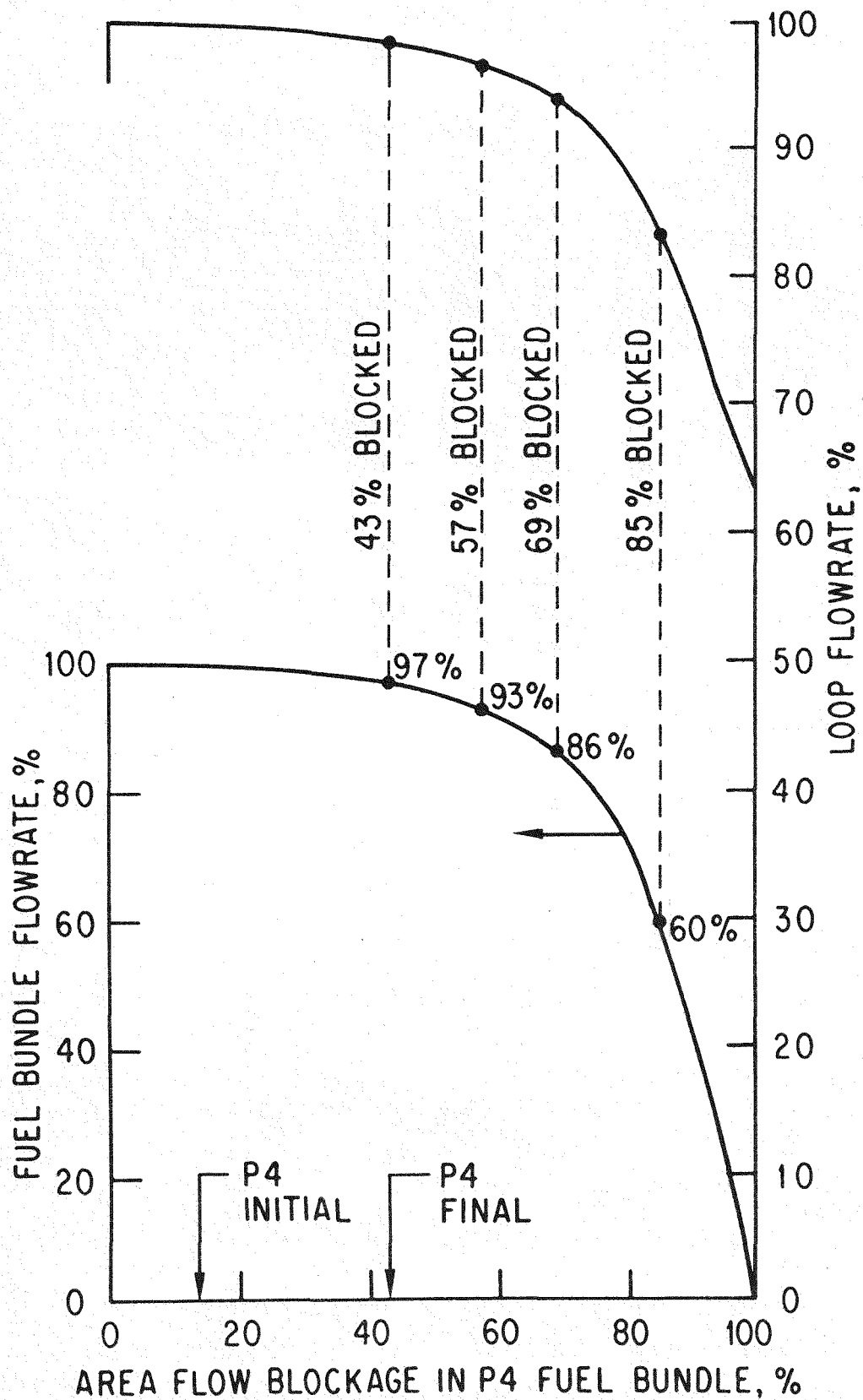


Fig. 62. Coolant Flow Rate as a Function of Area Flow Blockage

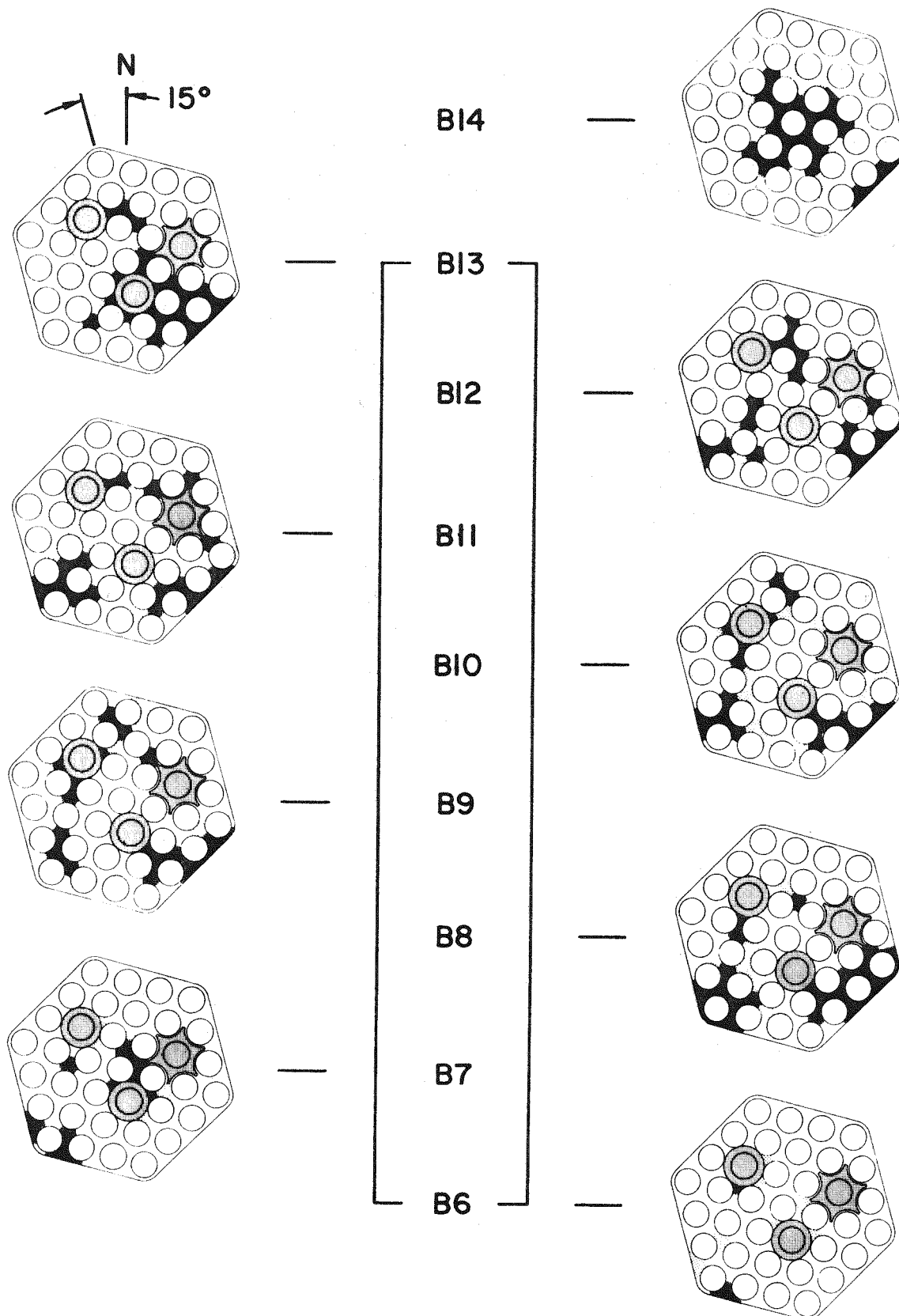


Fig. 63. Partial Blockage Configuration in the Fuel Canister Region

Table 18. Areal Blockage through the P4 Fuel Canister Region

|     | Initial Coolant<br>Flow Area, cm <sup>2</sup> | Coolant Channel<br>Areal Blockage, cm <sup>2</sup> | Area Blocked,<br>% |
|-----|-----------------------------------------------|----------------------------------------------------|--------------------|
| B22 | 7.03                                          | 0.00                                               | 0                  |
| B21 | 7.03                                          | 0.48                                               | 7                  |
| B20 | 7.03                                          | 0.26                                               | 4                  |
| B19 | 7.03                                          | 0.43                                               | 6                  |
| B18 | 7.03                                          | 0.42                                               | 6                  |
| B17 | 7.03                                          | 0.21                                               | 3                  |
| B16 | 7.03                                          | 0.00                                               | 0                  |
| B15 | 7.03                                          | 0.60                                               | 8                  |
| B14 | 7.03                                          | 1.46                                               | 21                 |
| B13 | 6.04                                          | 1.70                                               | 28                 |
| B12 | 6.04                                          | 1.74                                               | 29                 |
| B11 | 5.88                                          | 1.87                                               | 32                 |
| B10 | 6.04                                          | 1.81                                               | 30                 |
| B9  | 6.04                                          | 1.39                                               | 23                 |
| B8  | 5.88                                          | 2.05                                               | 35                 |
| B7  | 6.04                                          | 0.87                                               | 14                 |
| B6  | 6.04                                          | 0.40                                               | 7                  |
| B5  | 7.03                                          | 0.47                                               | 7                  |

The direction that expelled fuel material and debris moved was influenced by the spiral of the spacer wires. Looking down on the fuel-bundle cross section, as shown in Figs. 4 and 63, the spacer wires spiraled upward along the fuel pins in a counterclockwise direction. Molten fuel ejected toward hex-can Flats 1 and 2 was guided upward by the spiral of the fuel-pin spacer wires. Likewise, molten fuel ejected toward hex-can Flats 3 and 4 was guided downward by the spiral of the fuel-pin spacer wires.

It appears that the flow reduction accompanying the blockage reconfiguration did not result from an increase in areal blockage within the fuel bundle, but from a lengthening of the partial flow blockage region. One can conclude that the molten fuel released from the cylindrical and fluted fuel canisters during the power transient resulted in a 28 to 32% flow channel blockage between the upper banding strap and the upper end cap of the fuel canister region. Material disrupted during the blockage reconfiguration produced a 23 to 35% flow channel blockage from the fuel canister midplane to the lower banding strap.

The frictional pressure drops and pressure levels within the P4 test train were estimated for the range of flow conditions measured during experiment operation, as shown in Table 19. Results of the posttest hydraulic calculations using the FEPSTY code, based on a bundle coolant flow rate equal to the average of FE 1-3 and FE 1-4 measurements and loop coolant flow rate measured by flowmeter FE 3-2, are shown in Table 20. The estimated frictional pressure drop through the fuel canister and partial blockage region of the fuel bundle is shown in Table 20 to increase from 155 kPa after the power transient to 436 kPa following the reconfiguration. Comparable blockage pressure drop estimates drawn from the pretest calculations on which Fig. 62 was based were 177 kPa after the power transient and 450 kPa following the reconfiguration. The reconfiguration of the partial flow blockage during the return-to-full-power operation increased the estimated blockage frictional resistance by a factor of 2.5 to 2.8.

The increase in pressure drop through the reconfigured blockage is consistent with the extent and location of debris found in the bundle macrosections. The reconfiguration relocated fuel debris against hex-can Flats 3 and 4 below the fuel canister midplane. This partial blockage was similar in extent to the power-transient blockage extending from hex-can Flats 1 and 2 to the center of the bundle above the fuel canister midplane and would cause a doubling of the blockage-region pressure drop. The remainder of the pressure drop is accounted for by the increase in cross flow within the fuel canister and partial flow blockage region.

## 2. Temperature Field in the Blockage Wake

The SABRE-1 (Subchannel Analysis of Blockage in Reactor Elements) code<sup>42,43</sup> was used to estimate the temperature field in the wake of the partial flow blockage following molten fuel release from the fluted canister at 86 s into the power transient. This information was needed as a boundary condition for the THTB analysis of local hex-can failure. Although the P4 fuel bundle was heavily instrumented, only three thermocouples (TE 3-12, TE 6-1, TE 6-2) were near the location where once-molten fuel adhered to the hex-can. Both thermocouples TE 6-1 and TE 6-2 were hit by molten fuel. Their junctions were either buried within the debris or relocated to the upper blockage surface and could provide no information on sodium temperatures in the blockage wake.

Table 19. Measured Isothermal, Time-Averaged P4 Coolant Flow Rates

Sodium Temperature = 687 K (776°F)

Pump Voltage = 278 volts

Coolant Flow Rate, kg/s (lb/s)

| Date                                      | bundle<br>inlet |             | bundle<br>outlet | total<br>loop |              |
|-------------------------------------------|-----------------|-------------|------------------|---------------|--------------|
|                                           | FE1-3           | FE1-4       | FE2-1            | FE3-1         | FE3-2        |
| Aug. 14, 1981<br>(before power transient) | 3.38 (7.43)     | 3.37 (7.41) | 3.20 (7.05)      | 7.33 (16.14)  | 7.91 (17.40) |
| Aug. 28, 1981<br>(after power transient)  | 2.94 (6.47)     | 2.91 (6.42) | 2.77 (6.09)      | 6.96 (15.32)  | 7.50 (16.50) |
| Oct. 1, 1981<br>(before reconfiguration)  | 2.97 (6.53)     | 2.94 (6.46) | 2.91 (6.39)      | 7.18 (15.80)  | 7.50 (16.51) |
| Oct. 7, 1981<br>(after reconfiguration)   | 2.02 (4.45)     | 1.99 (4.37) | 2.07 (4.54)      | 6.38 (14.03)  | 6.70 (14.73) |

Table 20. Calculated Pressure Levels and Pressure Drops for  
Measured Isothermal Operating Conditions

|                              | Aug. 14      | Aug. 28      | Oct. 1       | Oct. 7       |
|------------------------------|--------------|--------------|--------------|--------------|
| Flow Rate, kg/s (lb/s)       |              |              |              |              |
| fuel bundle                  | 3.37 (7.42)  | 2.93 (6.45)  | 2.95 (6.50)  | 2.00 (4.41)  |
| loop                         | 7.91 (17.40) | 7.50 (16.50) | 7.50 (16.51) | 6.70 (14.73) |
| Pump                         |              |              |              |              |
| voltage, volts               | 277          | 274          | 273          | 273          |
| power, kW                    | 77           | 76           | 76           | 76           |
| net head, kPa (psia)         | 710 (103)    | 710 (103)    | 703 (102)    | 731 (106)    |
| Pressure Drop, kPa (psia)    |              |              |              |              |
| upper pin bundle             | 162 (23.5)   | 128 (18.5)   | 130 (18.8)   | 64 (9.3)     |
| blockage                     | 10 (1.5)     | 155 (22.5)   | 143 (20.8)   | 436 (63.2)   |
| lower pin bundle             | 62 (9.0)     | 48 (6.9)     | 48 (7.0)     | 24 (3.5)     |
| inlet flowmeter              | 146 (21.2)   | 110 (16.0)   | 112 (16.2)   | 52 (7.5)     |
| inlet orifice                | 203 (29.4)   | 153 (22.2)   | 156 (22.6)   | 72 (10.4)    |
| Pressure Levels, kPa (psia)* |              |              |              |              |
| combination of fuel          |              |              |              |              |
| bundle and bypass flow       | 146 (21.2)   | 142 (20.6)   | 142 (20.6)   | 133 (19.3)   |
| top of fuel pin              | 181 (26.3)   | 170 (24.7)   | 170 (24.7)   | 150 (21.8)   |
| top of blockage              | 361 (52.3)   | 312 (45.3)   | 314 (45.6)   | 229 (33.2)   |
| bottom of blockage           | 371 (53.8)   | 467 (67.8)   | 458 (66.4)   | 665 (96.4)   |
| bottom of fuel pin           | 426 (61.8)   | 521 (75.5)   | 511 (74.1)   | 694 (100.7)  |
| inlet flowmeter entrance     | 579 (84.0)   | 636 (92.3)   | 629 (91.2)   | 822 (119.2)  |

\*calculation based on loop cover gas pressure of 69 kPa (10.0 psia)



There are numerous computer codes for analyzing the coolant temperature and flow fields in the wake of partial flow blockages in reactor subassemblies. SABRE-1 was used for P4 posttest analysis because it was operational at ANL and had been used to define the temperature field boundary conditions as one step in the P4 pretest thermal analysis of the fluted fuel canister.<sup>44</sup>

SABRE-1 attempts to solve the three-dimensional Navier-Stokes equations and the energy equation for single-phase, incompressible, steady-state flow conditions in a reactor subassembly containing a partial flow blockage. The blockage must be planar, i.e., occupy the same radial subchannels between two horizontal planes. Wire-wrap spacers are not addressed. Both restrictions on the application of SABRE-1 and lack of detail about the three-dimensional shape and extent of the blockage suggest prudence in use of the results.

The flow subchannels blocked in macrosection B13, the uppermost section through the canister region, were modeled as the reference blockage configuration. Fuel in the four subchannels blocked between the outer-row pins and the hex-can wall was assumed to have twice the power density of fuel in the pins; fuel debris blocking eight subchannels between the canisters and the outer-row pins was at 1.4 times the nominal power density. The sodium flow rate in the bundle was 2.95 kg/s.

For an impermeable blockage, SABRE-1 predicted a negligible temperature difference across the blockage. Because this result was contrary to measured results, a means was sought to predict the temperature field for a porous blockage. One approach was to treat the blockage as having a discrete permeability. This is done by identifying certain subchannels as unblocked in the program input. SABRE-1 results for non-heat-generating blockages analyzed using this discrete "open subchannel" approach have compared favorably with experimental data.<sup>45</sup>

For simulation of a porous, heat-generating partial flow blockage, however, it was found necessary to vary the number of blocked channels through a sequence of problem restarts. The best convergence was obtained for sequential restart runs beginning with open flow (no initial blockage). First, the fuel canisters were constructed by inserting single-subchannel blockages, one by one, in successive runs. Then the reference partial flow blockage was modeled. The blocked subchannels were added in pairs, beginning with those believed to be the least porous and ending with the addition of those only partially blocked. This approach appeared successful, although heavily based on engineering judgment.

Temperature fields in the wake of the P4 partial flow blockage, as calculated by SABRE-1, are shown in Figs. 64 and 65 for two different "slices" through the bundle. The positions of these blockage-wake temperature fields and a subchannel-by-subchannel description of sodium temperatures downstream of the partial flow blockage are shown in Fig. 66. The temperature of the sodium bathing the once-molten fuel on the hex-can wall was calculated to be ~245 K higher than the temperature of the sodium in free-stream flow.

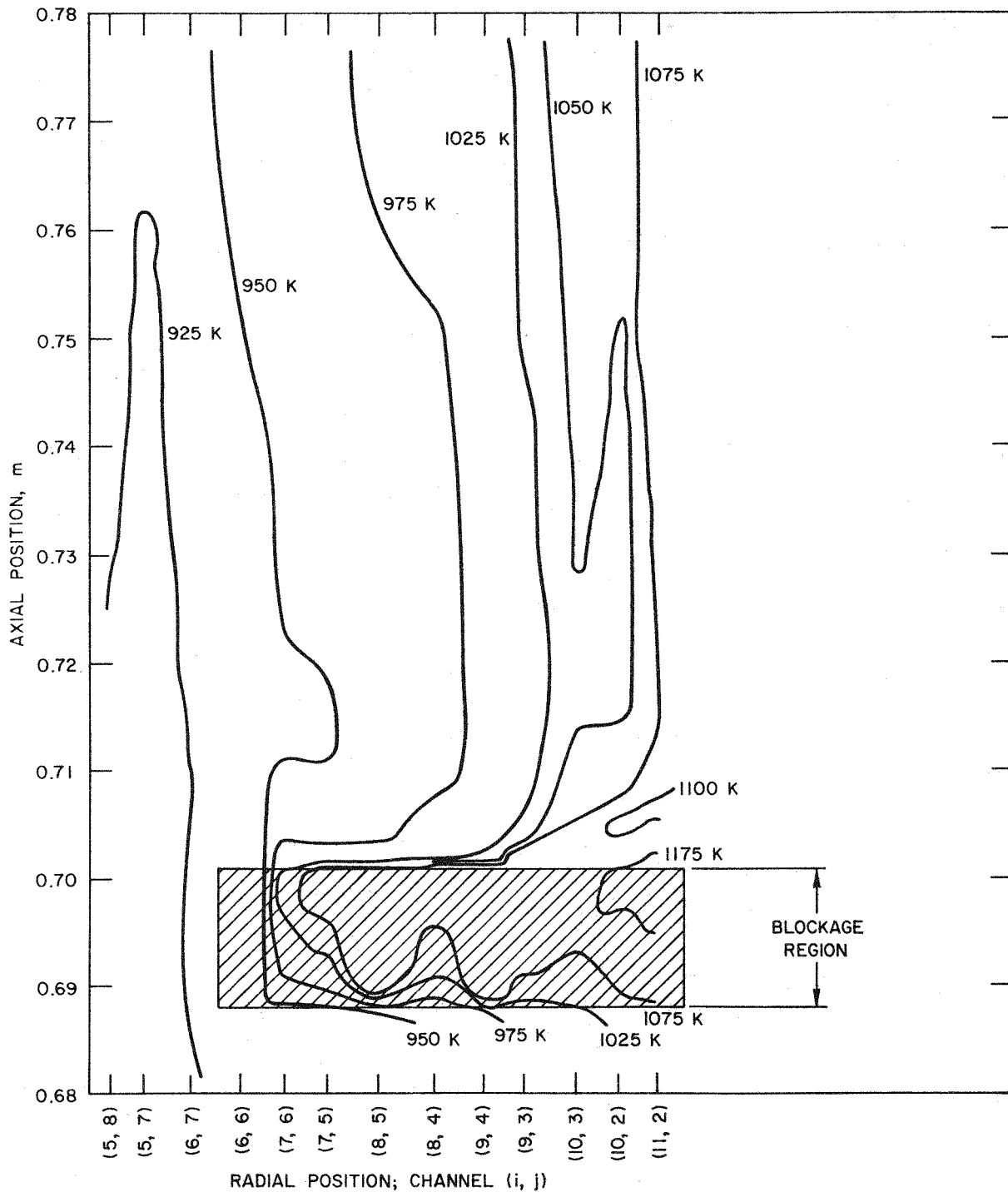


Fig. 64. Temperature Field in the Wake of the P4 Partial Flow Blockage (Sect. A-A) Prior to Local Failure of the Hex-Can Wall

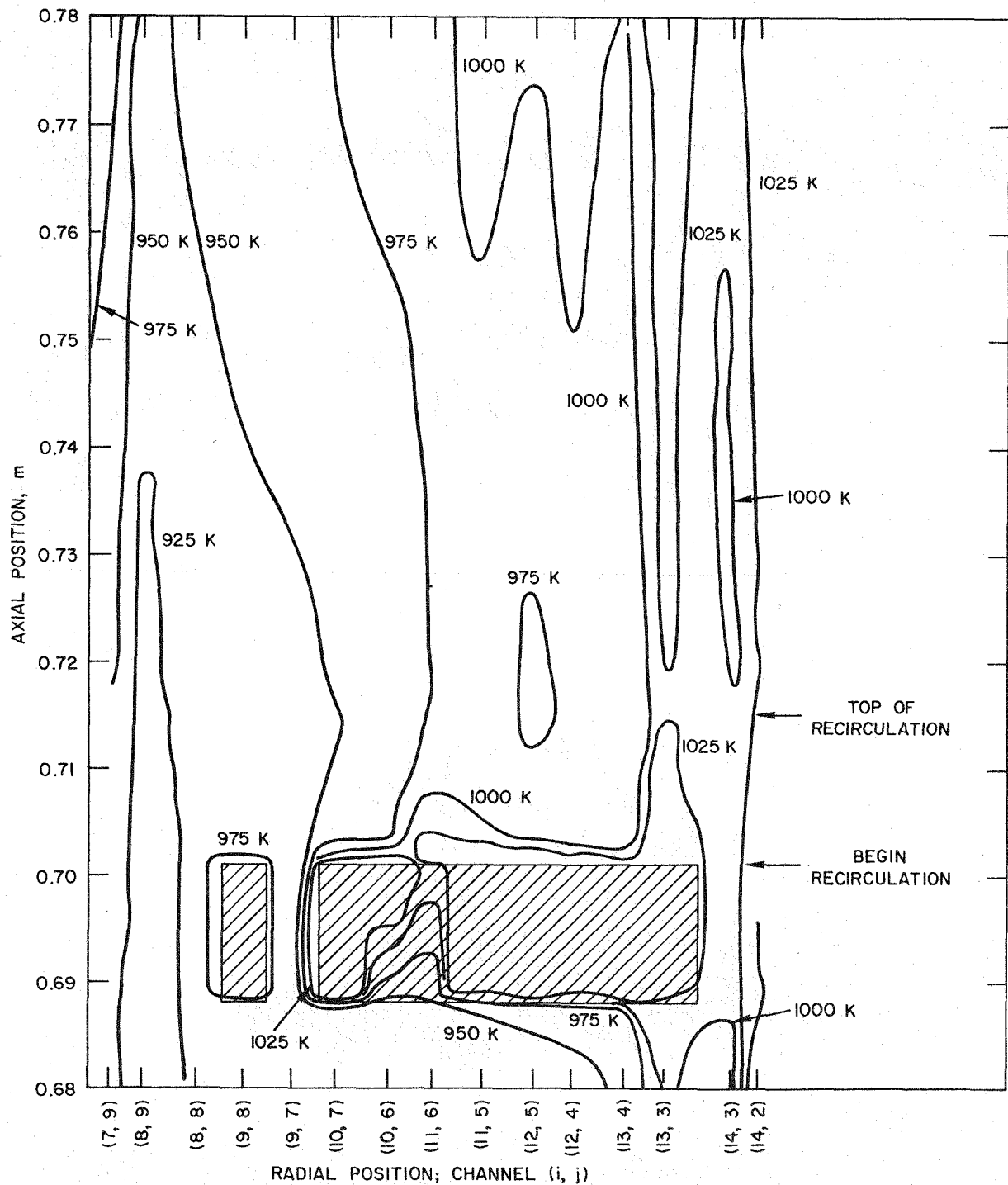


Fig. 65. Temperature Field in the Wake of the P4 Partial Flow Blockage (Sect. B-B) Prior to Local Failure of the Hex-Can Wall

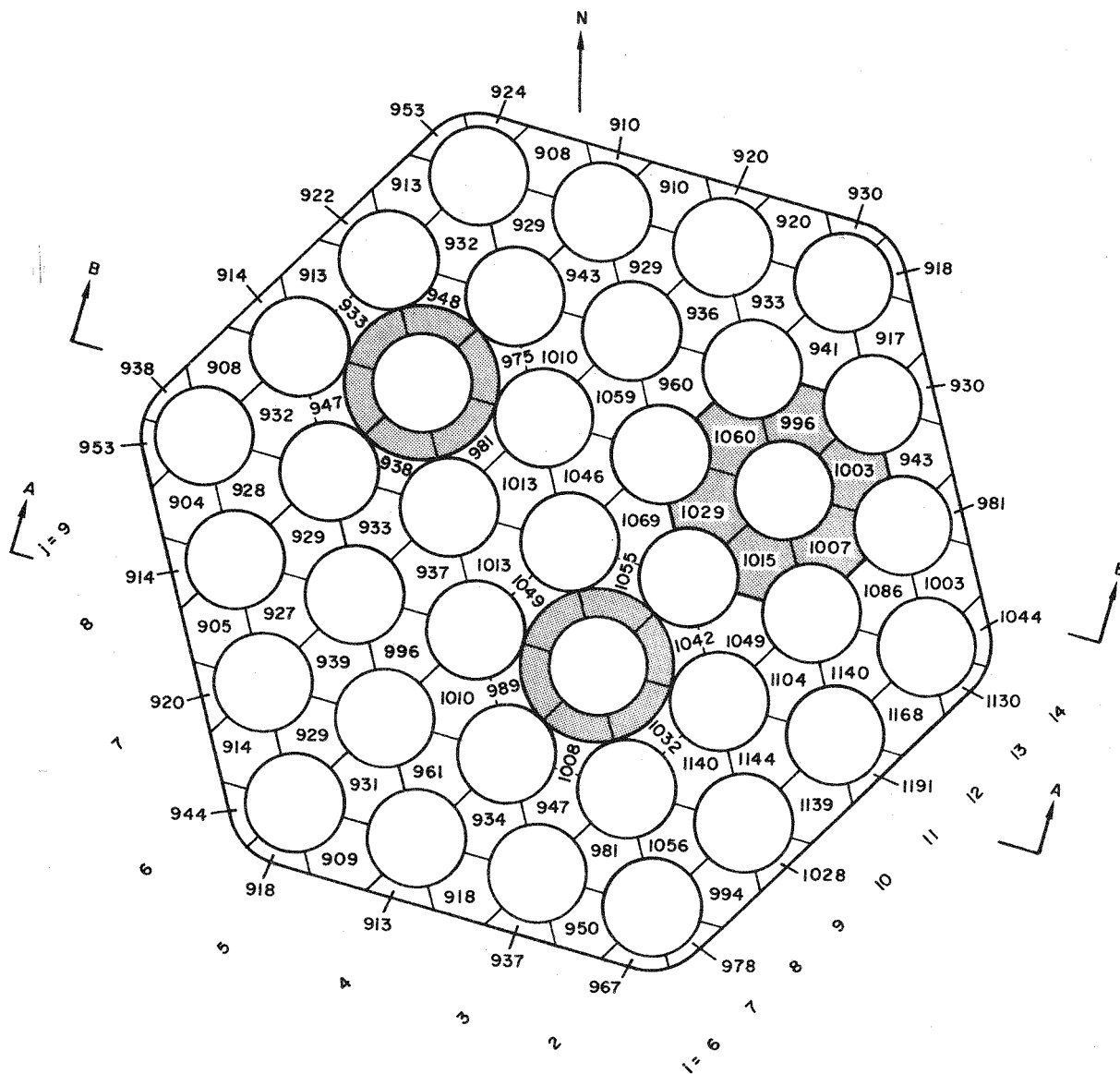


Fig. 66. Calculated Sodium Temperatures Downstream of the P4 Partial Flow Blockage (Temperatures in K)

D. Influence of Unplanned System Temperature Increase on Pump Performance and Inlet Flowmeter Signal

During the power transient, the sodium temperature at the inlet to the P4 fuel bundle was predicted to increase to a maximum of 742 K before the heat exchange system caught up to the power increase and restored the inlet temperature to 695 K. An operational error in the manual control of the helium circulator speed caused a shortfall in the helium flow rate through the loop heat exchanger. As a result of the deficiency in loop cooling during the power transient, the inlet temperature recovery to 695 K was not realized. Instead, an average temperature increase of  $\sim 0.6$  K/s produced elevated sodium temperatures around the loop; the system sodium temperature increased by  $\sim 55$  K ( $100^\circ\text{F}$ ) before reactor scram.

The pump was operated at a fixed voltage through variable transformers from line power. The signal from the inlet flowmeters to the PPS PF-H channel, like the pump power input, was not compensated for the change in bundle inlet sodium temperature. Both the pump performance and the flowmeter signal were sensitive to changes in sodium temperature. The influence of the 55 K sodium temperature increase on the performance of the pump and in the uncompensated flowmeter signal during the power transient are evaluated below.

1. Pump

Performance characteristics for the pump over a range of temperature, pressure, and flow had been measured in a test rig by the manufacturer, MSA Research Corporation. The test data were used to develop a pump model that was incorporated in the SLSF loop steady-state hydraulics code FEFPSTY to estimate pump power and voltage for a variety of experiment operating conditions studied over the years. This pump model had as its nucleus the power and voltage required by the pump, as a function of sodium temperature, to reach the pump reference (150 gpm, 150 psi) operating conditions.

The FEFPSTY code was used to evaluate the influence of the unplanned 55 K sodium temperature increase on the performance of the pump. This was a three-step process. First, the pump power and voltage were estimated for the P4 design operating conditions. For the P4 design operating conditions, with the sodium in the pump at a average temperature of 679 K ( $762^\circ\text{F}$ ), the pump power and voltage were 79.1 kW and 279.8 volts, respectively.

The second step in the process was to estimate the pump power and voltage for the P4 design flow rates, but at a pump sodium temperature of 734 K ( $762^\circ\text{F} + 100^\circ\text{F}$ ). For this case, the pump power and voltage were 84.4 kW and 292.5 volts, respectively.

The third step was to reduce the sodium flow rate in step two until the pump power and voltage reached step-one values. For the initial calculation in step three, sodium flow was reduced to  $(\text{pump power step one} / \text{pump power step two})^{1/2}$ , or 96.8%, of that in step one. Other variations in loop flow around 7.73 kg/s (17.00 lb/s) were also studied, as shown in Table 21. The change in sodium flow rate predicted by the FEFPSTY code for the 55 K over-temperature is  $-5.2\%/100$  K ( $-2.9\%/100^\circ\text{F}$ ) based on fixed pump power input and  $-6.5\%/100$  K ( $-3.6\%/100^\circ\text{F}$ ) based on fixed pump voltage.

Table 21. Influence of Sodium Flow Rate and Temperature on Pump Input

| Step | Description                                              | Sodium<br>Temperature<br>in Pump | At Pump Reference*<br>Operating Conditions |                  | Sodium Flow Rate<br>kg/s (1b/s) |              | Pump<br>Power | Pump<br>Voltage |
|------|----------------------------------------------------------|----------------------------------|--------------------------------------------|------------------|---------------------------------|--------------|---------------|-----------------|
|      |                                                          | K (°F)                           | Power<br>kW                                | Voltage<br>volts | fuel bundle                     | loop         | kW            | volts           |
| 1    | P4 design operating conditions                           | 679 (762)                        | 119.2                                      | 339.2            | 3.38 (7.45)                     | 7.98 (17.56) | 79.1          | 279.8           |
| 2    | 55 K (100°F) over-temperature; P4 design flow conditions | 734 (862)                        | 125.0                                      | 351.5            | 3.38 (7.45)                     | 7.98 (17.56) | 84.4          | 292.5           |
|      |                                                          |                                  |                                            |                  | 3.30 (7.25)                     | 7.77 (17.10) | 79.6          | 283.4           |
|      |                                                          |                                  |                                            |                  | 3.29 (7.23)                     | 7.75 (17.05) | 79.1          | 282.5           |
| 3    | 55 K (100°F) over-temperature; reduced flow conditions   | 734 (862)                        | 125.0                                      | 351.5            | 3.28 (7.21)                     | 7.73 (17.00) | 78.6          | 281.5           |
|      |                                                          |                                  |                                            |                  | 3.27 (7.19)                     | 7.70 (16.95) | 78.1          | 280.5           |
|      |                                                          |                                  |                                            |                  | 3.26 (7.17)                     | 7.68 (16.90) | 77.6          | 279.5           |

---

\* 150 gpm, 150 psi

This calculated sensitivity is in excellent agreement with a measured flow change of 6.45%/100 K (3.5%/100°F) during low-power testing on September 4. Over a nine-hour period on Sept. 4, during which the pump voltage was held constant, bundle-inlet sodium flow rate increased from 2.9 kg/s to 3.1 kg/s while the inlet sodium temperature decreased from 696 K to 595 K. Later, after the reconfiguration, a sensitivity of 4.9%/100 K (2.7%/100°F) was measured during low-flow testing on October 28. Over a nine-h period on Oct. 28, during which the pump voltage was held constant, bundle-inlet sodium flow rate increased from 2.02 kg/s to 2.18 kg/s while the sodium temperature dropped from 696 K to 535 K.

Both the pump model and the experiment data indicate a flow sensitivity of about -3% per 100°F sodium temperature increase for the pump at constant power/voltage conditions. This downward drift in bundle-inlet flow unrelated to partial blockage or local fault events during the power transient was about 15% of the margin between full flow and the 80.3% time-averaged, PF-H inlet flow setpoint. The downward drift in pump performance due to the increasing sodium temperature was not the sole contributor to the PF-H reactor scram at 118.4 s in the power transient. However, it may have either advanced the scram or been a sufficiently large contributor that the final contributor to reactor scram could not be overridden.

## 2. Inlet Flowmeter

All flowmeters were calibrated in an out-of-pile pumped-sodium loop prior to test train assembly. The calibration equation for permanent-magnet inlet flowmeter FE 1-3<sup>14,46</sup> was

$$\dot{m} = 3.08(10^{-5})\left(\frac{V_{DAS}}{0.508}\right)^2 + [0.563 + 5.843(10^{-5})T] \frac{V_{DAS}}{0.508} - 2.831(10^{-5})T + 0.0156 \quad (9)$$

where

$\dot{m}$  = mass flow rate, lb/s  
 $V_{DAS}$  = flowmeter signal to the DAS, volts  
 $T$  = sodium temperature, °F

Sodium flow rates were calculated for experiment operation and evaluation using the flowmeter calibration equations and the corresponding flowmeter voltage signals and sodium temperatures, but the flowmeter signals to the PPS were not processed this way. Instead, a preset voltage level that corresponded to an 80.3% bundle flow rate at an inlet temperature of 695 K (792°F) was calculated for the PF-H trip setpoint.

For sodium at 695 K, Eq. 9 reduces to

$$\dot{m} = 1.194 (10^{-4}) V_{DAS}^2 + 1.199 V_{DAS} - 0.007 \quad (10)$$

At the initial inlet flow rate of 3.45 kg/s (7.61 lb/s) and bundle inlet temperature of 695 K, the flowmeter FE 1-3 DAS-channel voltage was 6.35 volts. Because the first term in Eq. 10 is very small at P4 full-flow conditions and partially cancels the third term, Eq. 10 can be further simplified to

$$\dot{m} = 1.199 V_{DAS} \quad (11)$$

Equation 11 and its counterparts for the other bundle-inlet flowmeters were used to calculate the PF-H trip setpoint voltage.

To determine the sensitivity of the flowmeter FE 1-3 DAS-channel voltage to the sodium temperature, the center term of Eq. 9 was equated at the two temperatures of interest, 695 K (792°F) and 751 K (892°F):

$$[0.563 + 5.843(10^{-5})(792)] \frac{V_{792}}{0.508} = [0.563 + 5.843(10^{-5})(892)] \frac{V_{892}}{0.508}$$

This reduces to  $V_{892} = 0.99 V_{792}$ , which indicates that the inlet flowmeter FE 1-3 had a output sensitivity to sodium temperature of -1.8%/100 K (-1%/100°F). Thus, the flowmeter voltage signal to the PPF PF-H indicated a 1% lower bundle-inlet flow rate at the height of the inlet over-temperature condition than the measured flow rate processed through the flowmeter calibration equations.

## E. Hex-Can Failure

### 1. Evaluation of Hex-Can Meltthrough by Impinging Fuel Jet

The potential for breaching of a stainless steel hex-can wall by direct meltthrough as a result of heat transfer from a normally impinging jet of molten  $UO_2$  was analyzed. Conditions were representative of fuel impingement upon the hex-can in the P4 experiment. The objective was to calculate the maximum thickness of steel eroded away during the jet impingement phase.

The impinging  $UO_2$  stream was assumed to have a temperature of 3500 K while the hex-can wall temperature was equal to 935 K at the onset of impingement. For this temperature combination, the interface contact temperature<sup>47,48</sup> achieved following the sudden and perfect contact of semi-infinite slabs of  $UO_2$  and stainless steel was calculated to equal 1677 K. This is only 23 K below the assumed stainless steel melting temperature (1700 K) such that steel melting and erosion would be expected to begin very shortly following the onset of impingement. Consequently, the analysis assumes that ablation commenced immediately after the fuel first contacted the hex-can wall.

Assuming a quasi-steady ablation process, the steel erosion rate is given by

$$\frac{d\delta}{dt} = \frac{h_{conv} (T_{f,o} - T_{f,freeze})}{\rho_{s,o} (e_{s,liq} - e_{s,o})} \quad (12)$$



where

|                       |                                                                     |
|-----------------------|---------------------------------------------------------------------|
| $\delta$              | = thickness of steel eroded,                                        |
| $t$                   | = time,                                                             |
| $h_{\text{conv}}$     | = forced convection heat transfer coefficient for impingement flow, |
| $T_{f,0}$             | = temperature of impinging fuel,                                    |
| $T_{f,\text{freeze}}$ | = fuel freezing temperature,                                        |
| $\rho_{s,0}$          | = initial stainless steel density,                                  |
| $e_{s,\text{liq}}$    | = specific enthalpy of stainless steel at its liquidus,             |
| $e_{s,0}$             | = initial stainless steel specific enthalpy.                        |

Equation 12 assumes that a layer of solid fuel, or crust, is present between the flowing fuel and the stainless steel tending to insulate the steel from the hot molten fuel. The heat flux from the impinging jet is therefore proportional to the difference between the temperature of the impinging fuel and the fuel freezing temperature. For the assumed conditions, this temperature difference is equal to  $\sim 360$  K. Crust formation in impingement flow was studied with simulant material experiments<sup>49</sup> in which a freezing liquid was impinged upon the end of a meltable rod. The simulant tests exhibited ablation rates consistent with the existence of a protective crust. More recently, reactor material experiments<sup>50</sup> have been performed in which jets of predominantly molten oxide fuel at  $\sim 160$  K above its freezing temperature were impinged normally upon stainless steel plates initially at 300 and 385 K. Comparison of model calculations with the response of thermocouples imbedded immediately beneath the steel surface indicated that a protective crust of frozen oxide fuel was present over nearly all of the plate surface area throughout the impingement process.<sup>50</sup> However, in the experiments, no major melting of the stainless steel plates was observed. Future reactor material tests will investigate crust formation as ablation is taking place by heating the steel to higher initial temperatures where melting is expected immediately following fuel-steel contact. Equation 12 represents an energy balance in which the heat flux from the jet goes into raising the enthalpy of each mass of steel eroded from its initial value to its liquidus. The melted steel may be envisioned to flow as a film between the fuel crust and the remaining solid steel substrate. The film may undergo some heating above the steel melting temperature as it flows out of the impingement region. This would tend to decrease the erosion rate such that Eq. 12 may somewhat overestimate the actual erosion in the presence of a protective crust.

The ablation rate, Eq. 12, applies over the total impingement time,

$$t_{\text{imp}} = \frac{m_f}{\rho_f U_f A_f} \quad (13)$$

where

|          |                             |
|----------|-----------------------------|
| $m_f$    | = total fuel mass,          |
| $\rho_f$ | = fuel density,             |
| $U_f$    | = jet impingement velocity, |
| $A_f$    | = jet cross-sectional area. |

The total erosion after the completion of the impingement phase is thus equal to

$$\delta_{\text{tot}} = \frac{d\delta}{dt} t_{\text{imp}} \quad (14)$$

Two general cases of jet geometry are considered. The first is a slot jet of width, B, assumed small compared to the slot length, L (i.e.,  $A_f = BL$ ). The forced convection heat transfer coefficient for impingement flow is taken equal to that recommended by Martin<sup>51</sup> for normal impingement of a slot jet,

$$h_{\text{conv}} = \frac{1.53}{\frac{x}{S} + \frac{H}{S} + 1.39} \text{Re}^m \text{Pr}^{0.42} \frac{k_f}{S} \quad (15)$$

where

$$m = 0.695 - \frac{1}{\frac{\frac{x}{S} + \frac{H}{S}}{1.33} + 3.06} \quad (16)$$

$x$  = normal distance from jet centerplane,  
 $S$  =  $2B$  = jet hydraulic diameter,  
 $H$  = height of nozzle above surface,  
 $\text{Re}$  = jet Reynolds number,

$\text{Pr} = \frac{c_f \mu_f}{k_f}$  = jet Prandtl number,

$c_f$  = jet specific heat,  
 $\mu_f$  = jet viscosity,  
 $k_f$  = jet thermal conductivity.

Equation 15 gives the heat transfer coefficient averaged over the plate surface up to a distance,  $x$ , from the centerplane. Since the maximum erosion directly beneath the impinging jet is of interest here,  $x$  is set equal to zero. Because Eq. 15 describes the results of experiments carried out with gas jets,  $h_{\text{conv}}$  depends upon the nozzle-to-plate distance,  $H$ . For the present application,  $h_{\text{conv}}$  exhibits a maximum approximately when  $\frac{H}{S} = 5$  and this value is used to further maximize the erosion. With  $x = 0$  and  $\frac{H}{S} = 5$ , Eq. 15 reduces to

$$h_{\text{conv}} = 0.239 \text{Re}^{0.609} \text{Pr}^{0.42} \frac{k_f}{S} \quad (17)$$

Substituting Eqs. 12, 13, and 17 into Eq. 14 provides the total erosion,

$$\delta_{\text{tot}} = \frac{0.239 (\rho_f / \mu_f)^{0.609} \text{Pr}^{0.42} (T_{f,o} - T_{f,\text{freeze}}) k_f 2m_f}{\rho_{s,o} (e_{s,\text{liq}} - e_{s,o}) \rho_f U_f^{0.391} S^{1.391} L} \quad (18)$$

It is observed that the total erosion decreases as each of the jet velocity, slot width, or slot length parameters increase.

The second jet configuration is that of a circular impinging stream. The following fit was made to the Nusselt number data presented graphically in Ref. 51 for the heat transfer coefficient averaged over a disc of vanishingly small diameter and a nozzle-to-plate distance equal to 7.5 times the jet diameter (again corresponding to an approximate maximum):

$$h_{\text{conv}} = 1.02 \text{ Re}^{0.513} \text{ Pr}^{0.42} \frac{k_f}{D} \quad (19)$$

where  $D$  is the jet diameter. For  $h_{\text{conv}}$  given by Eq. 19, the total erosion equals

$$\delta_{\text{tot}} = \frac{1.02 (\rho_f / \mu_f)^{0.513} \text{ Pr}^{0.42} (T_{f,o} - T_{f,\text{freeze}}) k_f^{4m_f}}{\rho_{s,o} (e_{s,\text{liq}} - e_{s,o}) \rho_f \pi U_f^{0.487} D^{2.487}} \quad (20)$$

The total erosion is strongly sensitive to the jet diameter.

Calculations were carried out assuming that a 10-g mass of fuel impinged normally upon the hex-can wall. The fuel velocity at impingement was taken equal to 1 m/s. Because the total erosion decreases as the velocity increases, this value bounds the erosions which would be predicted for higher velocities. Table 22 shows the thermophysical properties<sup>38,52</sup> assumed for the  $\text{UO}_2$  and stainless steel. The results obtained for a slot jet are shown in Table 23 for slot widths equal to the nominal interpin spacing (0.15 cm) and the clearance between the outermost row of pins and the hex-can (0.07 cm). The slot length was taken equal to 2.86 cm representative of the disposition of fuel observed in posttest examination. This value may also be compared with the greater 10.16-cm length of the original fuel canister. For the nominal interpin spacing case, the predicted maximum total erosion amounts to 0.0854 mm, which is only 2.8% of the full hex-can wall thickness. Halving the slot width to the pin/wall clearance increases the maximum total erosion to 7.7% of the hex-can wall thickness.

For a circular jet geometry, the diameter was first set equal to the value (1.92 cm) providing the same jet area as the cross-sectional area of the fuel found solidified upon the hex-can wall. The predicted erosion is very small (0.009 mm), corresponding to only 0.29% of the wall thickness (Table 24). To indicate the type of jet dimensions required for the calculation of complete meltthrough, the diameter was decreased to the interpin spacing (0.15 cm). In this case, the impinging jet could erode a thickness of steel equal to 1.78 times that of the hex-can. However, the degree of confinement necessary to produce such a jet is judged to be unrealistic in view of the length of the fuel canister, the continuity of open space between the pins in the axial direction, and the distance which the fuel must travel to reach the wall. For the other three cases, the predicted maximum erosions are more than an order of magnitude smaller than the wall thickness, suggesting that ablation of steel during the impingement phase was not by itself responsible for the local hex-can failure observed in the experiment.

Table 22. Thermophysical Properties Assumed in the Hex-Can Wall Ablation Calculations

|                |   |                                               |
|----------------|---|-----------------------------------------------|
| $T_{f,freeze}$ | = | 3138 K                                        |
| $T_{f,o}$      | = | 3500 K                                        |
| $\rho_f$       | = | $8.44 \times 10^3 \text{ kg/m}^3$             |
| $\mu$          | = | $3.70 \times 10^{-3} \text{ Pa}\cdot\text{s}$ |
| $c$            | = | $0.503 \text{ kJ}/(\text{kg}\cdot\text{K})$   |
| $k$            | = | $3.69 \text{ W}/(\text{m}\cdot\text{K})$      |
| $Pr$           | = | 0.505                                         |
| $T_{s,liq}$    | = | 1700 K                                        |
| $e_{s,liq}$    | = | 1098 KJ/kg                                    |
| $T_{s,o}$      | = | 935 K                                         |
| $\rho_{s,o}$   | = | $7.66 \times 10^3 \text{ kg/m}^3$             |
| $e_{s,o}$      | = | 344 kJ/kg                                     |

Table 23. Predicted Erosion of the Hex-Can Wall for Slot Jet Configuration

| Jet Width<br>cm | Jet Length<br>cm | Heat Transfer<br>Coefficient<br>$\text{kW}/(\text{m}^2\cdot\text{K})$ | Erosion<br>Rate<br>cm/s | Impingement<br>Time<br>s | Wall<br>Thickness<br>Eroded<br>cm | Fraction of<br>Full Hex-Can<br>Thickness<br>Eroded |
|-----------------|------------------|-----------------------------------------------------------------------|-------------------------|--------------------------|-----------------------------------|----------------------------------------------------|
| 0.147           | 2.858            | 48.1                                                                  | 0.302                   | 0.0283                   | 0.0085                            | 0.028                                              |
| 0.071           | 2.858            | 63.8                                                                  | 0.400                   | 0.0583                   | 0.0233                            | 0.077                                              |

Table 24. Predicted Erosion of the Hex-Can Wall for Circular Jet Configuration

| Jet<br>Diameter<br>cm | Heat Transfer<br>Coefficient<br>$\text{kW}/(\text{m}^2\cdot\text{K})$ | Erosion<br>Rate<br>cm/s | Impingement<br>Time<br>s | Wall<br>Thickness<br>Eroded<br>cm | Fraction of<br>Full Hex-Can<br>Thickness<br>Eroded |
|-----------------------|-----------------------------------------------------------------------|-------------------------|--------------------------|-----------------------------------|----------------------------------------------------|
| 1.923                 | 35.0                                                                  | 0.220                   | 0.004                    | 0.0009                            | 0.003                                              |
| 0.147                 | 123                                                                   | 0.771                   | 0.703                    | 0.541                             | 1.78                                               |

In summary, calculations have been carried out estimating the maximum erosion of a stainless steel hex-can wall initially at 935 K by a normally impinging jet consisting of 10 g of  $\text{UO}_2$  at 3500 K, assuming that a crust layer of frozen  $\text{UO}_2$  is present interstitial to the hot molten  $\text{UO}_2$  and the ablating steel. For a jet having a width corresponding to the interpin spacing and a length consistent with the observed fuel posttest disposition, the predicted maximum erosion amounts to, at most, a few percent of the hex-can wall thickness. The calculations further show that the prediction of complete meltthrough of the wall requires an unrealistic degree of jet confinement (eg., a circular jet with a diameter equal to the interpin spacing). The analysis thus suggests that ablation during the impingement process was not, by itself, responsible for local failure of the hex-can wall during the experiment.

## 2. THTB Analysis of Fuel/Hex-Can Temperature Field Prior to Local Hex-Can Failure

A THTB analysis was done to study the local meltthrough of the P4 hex-can by a mass of fuel stuck to the inner surface of the wall. The model reflected the conditions near the hex-can wall at the time of fuel release as accurately as could be obtained from thermal-hydraulic data and fuel-bundle thermal-hydraulic analyses. Certain key dimensions of the THTB model were adjusted to fit the known meltthrough of the hex-can 76 s after impact by the fuel. The model also needed to be consistent with the physical extent of the meltthrough. Because of the long axial dimension of the meltthrough relative to its lateral dimension, a two-dimensional x-y THTB geometry was chosen. The lateral dimension of the meltthrough in the experiment was, more or less, a distance of one pin pitch or about 0.725 cm. The successful description of the timing and extent of the P4 meltthrough demonstrates the consistency of this event with a general understanding of P4 test conditions and provides, perhaps, some insight into underlying meltthrough mechanisms. Owing to high sensitivity to parameters which are not really well-known, this analysis should be considered principally illustrative.

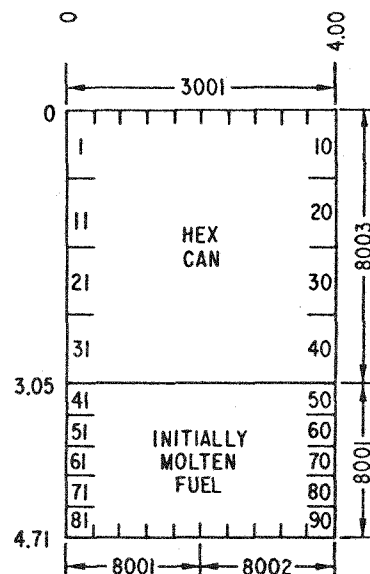
In order to use the input generator for THTB, a simple slab geometry model, shown in Fig. 67, was chosen. This approach required specifying the right-hand lateral boundary condition (code 8003 in Fig. 67) of the hex-can as an input parameter to THTB. The 8003 boundary simulated the heat sink which would be afforded by the wall of the hex-can that extended beyond the fuel mass. The temperature history assumed for this boundary was determined iteratively as the average of the calculated temperature for the hex-can wall about to experience meltthrough and the SABRE-1-calculated temperature of the surrounding sodium coolant. The boundary labeled as 3001 in Fig. 67 represented an inward-flowing surface heat flux of  $6 \text{ W/cm}^2$  due to gamma heating of the materials exterior to the hex-can. The left-hand boundary of the model was made a plane of symmetry and was, therefore, adiabatic.

The two boundaries for the fuel mass are 8001 - a convection boundary to the sodium coolant, and 8002 - a temperature boundary that simulates the presence of a fuel pin. These input boundary conditions are shown in the Appendix C listing of the THTB input for the P4 case. The calculations were performed with an upgraded (THTM) version of the THTB code that uses metric units for the input and does a heat balance for the complete system of nodes at each temperature printout.

Two cases were run - one simulated P4 and the other a large plant. Input and results for the P4 case are described below; the large-plant case is discussed in Ch. VII.

For the P4 case, the power density in the 93%-enriched  $UO_2$  fuel ejected from the cylindrical canister on Pin 5 doubled, to 480 W/g, as it was displaced outward beyond the 58%-enriched fuel of the outer-row pins to the hex-can wall. At 156-MW ETR power, gamma heating in the hex-can, tungsten chevrons, and zirconia insulators was 7.8 W/g at the core midplane.<sup>53</sup> The sodium temperature adjacent to the fuel mass, from SABRE-1 calculations, was 245 K higher in the downstream blockage wake than the bulk sodium temperature at that elevation. Fuel density was assumed to be unchanged from that in the canister.

THTB results for some selected nodes in the P4 case are plotted in Fig. 68. The calculated time to hex-can meltthrough was most sensitive to: (1) the temperature history for the lateral boundary of the hex-can (8003 of Fig. 67), (2) the x-dimension of the model, and (3) the thickness (y dimension) of the fuel mass. The observed interval of 76 s between initial contact by molten fuel and local hex-can failure was duplicated in the THTB analysis for a fuel mass 0.8-cm wide (x dimension) and 0.17-cm thick (y dimension). First melting of the inner hex-can surface occurred 45 s after contact by the fuel. For the 2.86-cm axial length of fuel observed in the posttest examination, this fuel cross section corresponds to a fuel mass of 4 g. Considering that the simplified model did not address axial (z direction) heat transfer and that the model, and input, were conservative in other regards, it is reasonable to conclude that the mass of fuel on the hex-can wall at the failure site was ~5 g. Total power in this fuel mass would have been ~2.4 kW. All contributors to the local hex-can failure at 110 s into the power transient are shown in Table 25.

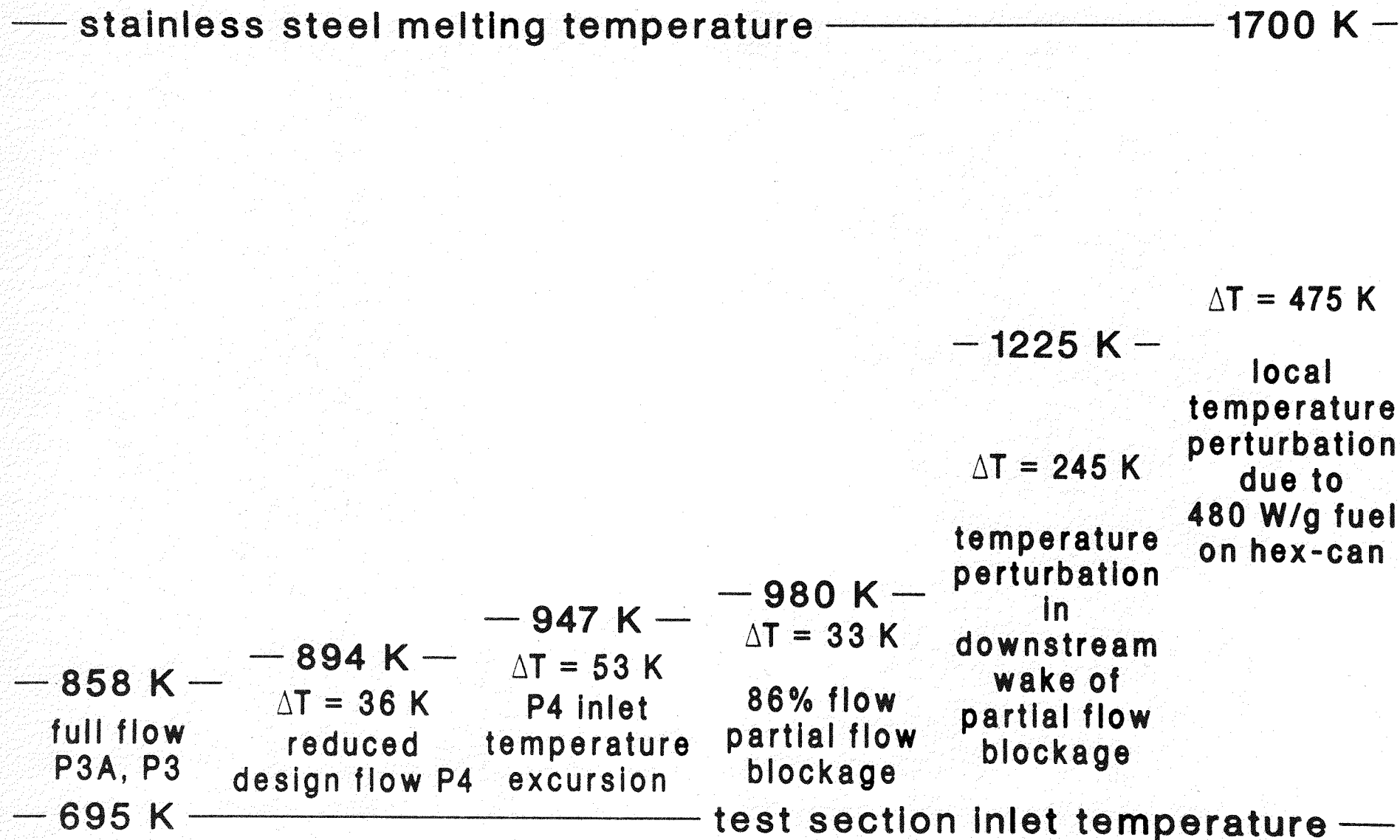


NOTE:  
ALL DIMENSIONS IN mm

BOUNDARY CONDITIONS:  
3001 INWARD FLOWING SURFACE HEAT FLUX  
8001 CONVECTION TO SODIUM COOLANT  
8002 PRESENCE OF FUEL PIN  
8003 OUTWARD HEAT FLOW THROUGH EXTENDED HEX-CAN WALL

Fig. 67. THTB Model of Fuel on the P4 Hex-Can Wall

Table 25. Contributors to Local Failure of the P4 Hex-Can Wall



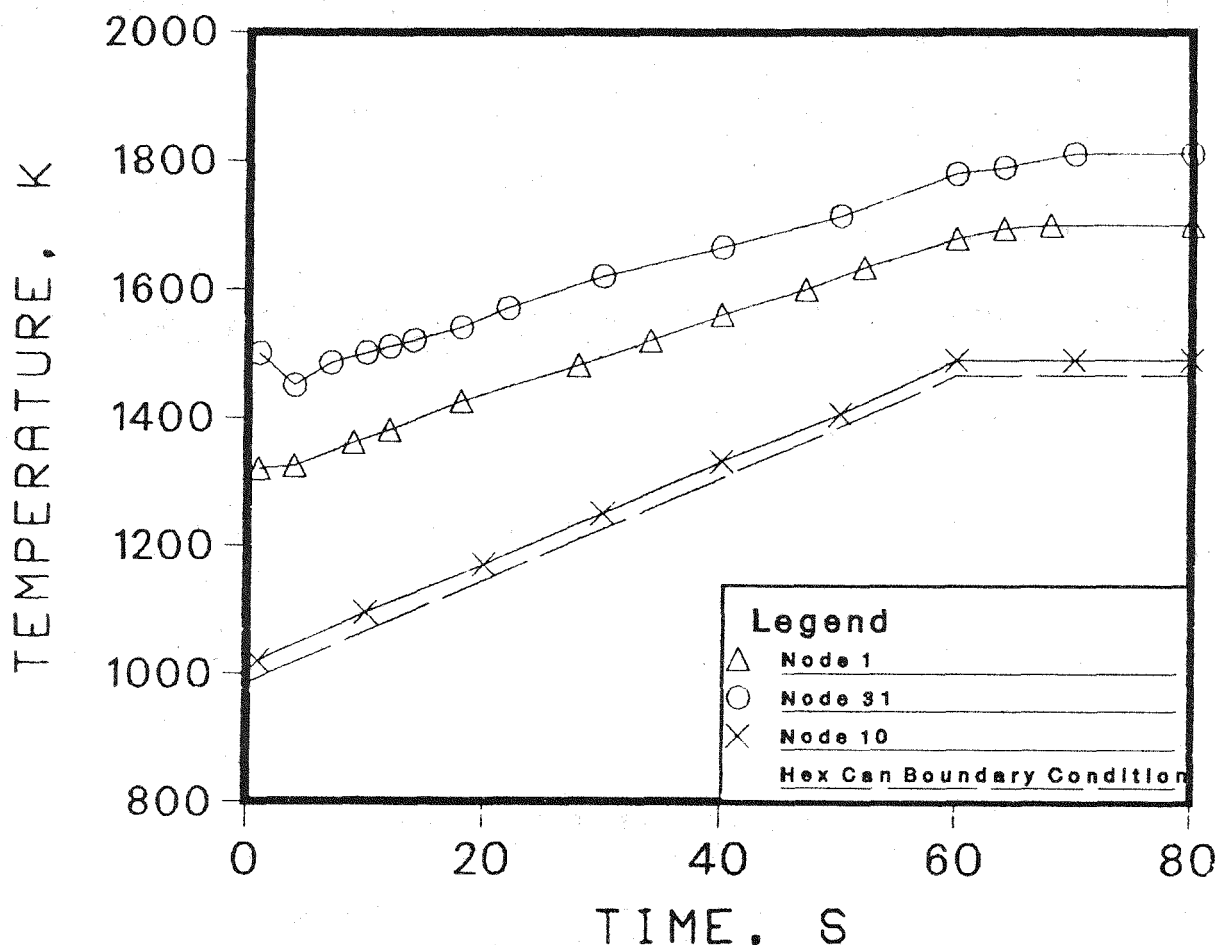


Fig. 68. Selected Hex-Can Node Temperatures for the P4 Case

## F. Gas Release

### 1. Quantitative Evaluation

A qualitative evaluation of the changing loop gas-plenum pressure during the power transient is contained in Ch. IV. Because the OLCS provided a once-through flow of gas in the loop plenum and worked to maintain the loop plenum gas pressure at a preset, regulated value, the measured loop gas-plenum pressure did not represent a closed-volume response. Corrections to the measured loop gas-plenum pressure were made to remove the effects of OLCS operation for quantitative evaluation of the changing plenum pressure.

The rate of change of pressure in the loop gas plenum is described by

$$dP/dt = C (W_i - W_o) \quad (21)$$



where

- P = loop plenum pressure, MPa
- t = time, s
- C = constant, MPa/sccm·s [sccm=cm<sup>3</sup>/min at STP]
- W<sub>i</sub> = OLCS plenum-inlet gas flow, sccm
- W<sub>o</sub> = OLCS plenum-outlet gas flow, sccm.

The constant C was determined as follows:

$$C = \frac{(0.0821 \text{ l atm/mole} \cdot \text{K})(699.7 \text{ K})(14.7 \text{ psi/atm})(7.44 \times 10^{-4} \text{ moles/s per 1000 sccm})}{(32.3 \text{ l plenum volume})(145 \text{ psi/MPa})}$$

$$= 1.341 (10^{-4}) \text{ MPa/s per 1000 sccm}$$

$$= 1.341 (10^{-7}) \text{ MPa/sccm} \cdot \text{s}$$

The corrected pressure for net zero change in the mass of gas flowing in the OLCS is given by

$$P_C = P_m + \int_0^t 1.341 (10^{-7}) (W_i - W_o) dt \quad (22)$$

where

- P<sub>m</sub> = measured OLCS plenum-inlet gas pressure, MPa
- P<sub>C</sub> = corrected loop gas-plenum pressure for zero OLCS flow, MPa.

The loop gas-plenum pressure, both as measured and corrected to remove the influence of OLCS operation, is shown in Fig. 18. The leveling of the corrected pressure beyond 125 s indicates that changes in measured pressure at that time were due solely to the OLCS gas bleed from the loop plenum to regain the setpoint pressure. The OLCS plenum-inlet gas pressure was measured by pressure transducers PE-378 and PE-379; gas flow rates at the inlet and outlet were measured by OLCS flowmeters FM 30 and FM 9, respectively.

From the corrected pressure, the gas evolution contributing to the rate of change of pressure was estimated. Some of the pressure change might have been caused by loop or sodium thermal expansion, which is addressed later. Based on an isothermal plenum-gas temperature (691 K < T<sub>gas</sub> < 696 K) and the assumption that loop thermal expansion was not of major importance, an estimate was made of gas transport during the power transient. The rate of gas release to the loop plenum is given by

$$W_e = (1/C) dP_C/dt \quad (23)$$

where

- W<sub>e</sub> = net gas evolution rate, sccm.

Point-by-point differentiation of the corrected digital pressure data was subject to digital noise. Therefore, preaveraging before differentiation was necessary. Results for no averaging, averaging of three data points, and

averaging of six data points in determining gas flow into the loop plenum are shown in Fig. 69. The preaveraging reduced peak values, but brought out the fine structure in the gas release from the fuel bundle. Figure 69(c) clearly indicates gas release from the fuel bundle following melting of the fluted canister cladding at 15-16 s, molten fuel release from the canister on Pin 5 at 34 s, and molten fuel release from the fluted canister at 86 s. The largest gas inflow to the loop plenum resulted from local failure of the hex-can wall at 110 s. This was followed by a subsequent gas release into the coolant stream from both previously failed fuel pins and the fuel bundle insulator region during transfer of the pump to an emergency-power supply following reactor scram at 118.4 s. System pressure was reduced during the 40-ms interval that pump power was off during the transfer, permitting the additional gas to enter the coolant stream and reach the loop gas plenum.

The volume of evolved gas was determined from the integral

$$V_e = \int_0^t W_e dt \quad (24)$$

where

$V_e$  = volume of gas,  $\text{cm}^3$  (STP).

Were the change in loop plenum pressure due solely to release of gas from failed pins and from the pressurized insulator region through the local hex-can failure, the pressure change corresponded to a total gas inflow of  $2820 \text{ cm}^3$  (STP), as shown in Fig. 70. Local hex-can failure at 110 s released  $1280 \text{ cm}^3$  (STP) of gas, on top of a prior  $1540 \text{ cm}^3$  equivalent gas release. The prompt volumetric gas release at each fuel failure event, from Fig. 70, and the corresponding number of pins failed are summarized in Table 26. It was assumed that all six fuel pins adjoining the fluted canister failed when cladding melting temperatures were reached at 15 to 16 s into the power transient. Estimates of the number of pin failures for the other events were drawn from the  $75 \text{ cm}^3$  (STP) release at 15 to 16 s equalling failure of the six adjacent pins.

Table 26. Gas Release and Failed-Pin Equivalent

| Event                                                        | Time<br>s | Gas Release<br>$\text{cm}^3$ (STP) | Number of<br>Pins Failed |
|--------------------------------------------------------------|-----------|------------------------------------|--------------------------|
| Cladding melting<br>on fluted canister                       | 15-16     | 75                                 | 6                        |
| Molten fuel release<br>from cylindrical<br>canister on Pin 5 | 34        | 140                                | ~11                      |
| Molten fuel release<br>from fluted canister                  | 86        | 60                                 | ~5                       |
|                                                              |           | <hr/> 275                          | <hr/> 22                 |

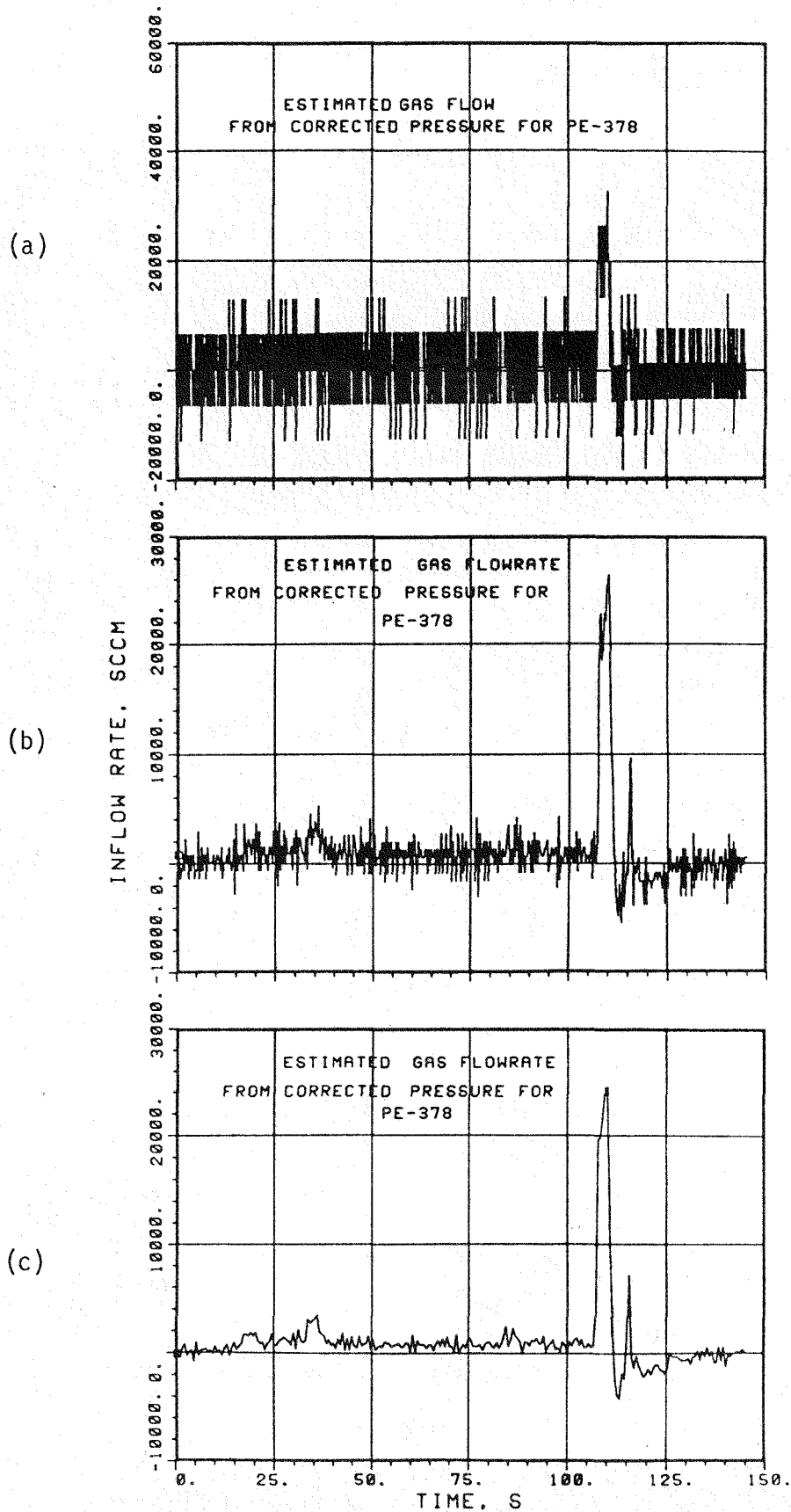


Fig. 69. Rate of Gas Release to the Loop Plenum Determined by Differentiating Plenum Pressure: (a) no averaging, (b) three data points averaged, (c) six data points averaged.

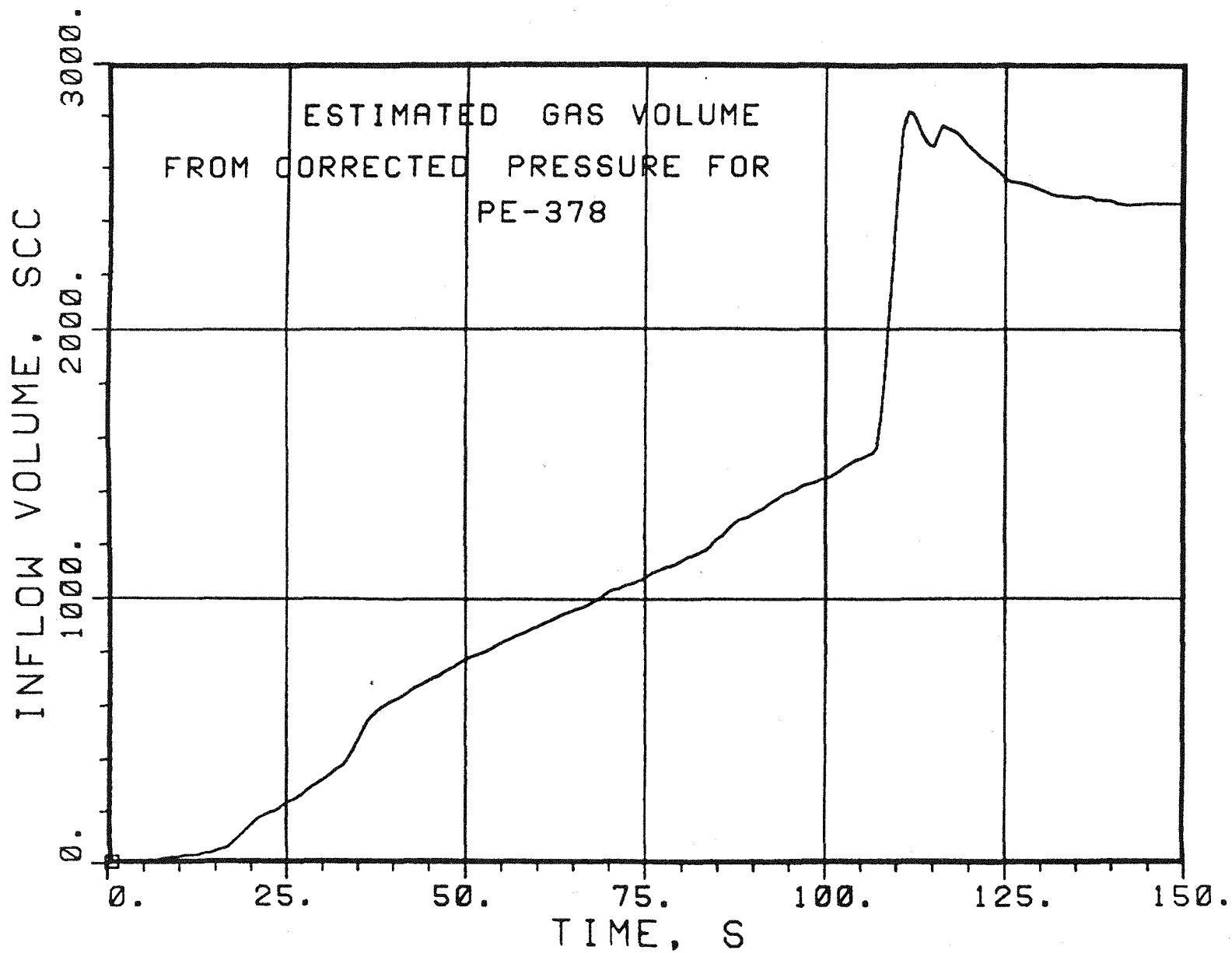


Fig. 70. Estimated Inflow Volume Causing Pressurization of the Loop Gas Plenum

The slow and uniform increase in plenum pressure prior to local hex-can failure at 110 s might have been slow gas leakage from failed pins. It might also be related to thermal expansion of the sodium and the loop geometry due to the unplanned 55 K increase in the bundle inlet and the system sodium temperature during the power transient. For the 85-kg sodium inventory, a 55 K increase in temperature above the planned operating levels produced a 1.6% increase in sodium volume. Loop volume increased by 0.3%, based on a linear thermal expansion coefficient of  $18 (10^{-6}) K^{-1}$ . The net volumetric increase of sodium in the loop, then, was  $1.6\% - 0.3\%$ , or 1.3%. A 1.3% increase in the 100-liter loop sodium volume produced a  $1300 \text{ cm}^3$  expansion of sodium into the loop plenum volume. This agrees well with a  $1265 \text{ cm}^3$  volume change obtained by subtracting the  $275 \text{ cm}^3$  prompt gas release (Table 26) from the estimated  $1540 \text{ cm}^3$  total inflow volume at 110 s. What appears to be a slow and uniform increase in gas flow to the loop plenum, in Fig. 70, was sodium expansion into the loop plenum volume broken by prompt gas release from pins failed at 15-16 s, 34 s, and 86 s, followed by gas release through a local hex-can failure at 110 s.

## 2. Coolant Temperature Perturbations

Considerable attention has been devoted to the investigation and evaluation of the effects of fission gas release on fuel pin performance and reactor safety. Based upon all the analytical and experimental results, it has been concluded that, for liquid metal reactors employing mixed-oxide fuel and under normal steady-state operating conditions, fission gas release by itself is not a potential cause for pin-to-pin failure propagation.<sup>54</sup>

One of the postulated modes for pin-to-pin failure propagation is local overheating of adjacent pins by gas blanketing. Gas release behavior in gas-jet impingement tests has encompassed any condition that might be encountered in operation. For a high-pressure gas release ( $P_{\text{gas}}/P_{\text{sodium}} > 3$ ), test results indicate that sodium entrainment in the gas jet produces a spray cooling which does not materially change the cladding temperatures. For low gas pressure, a liquid film remains on the cladding surface and governs the heat transfer process. Maximum cladding temperature perturbations in the range 60 to 120 K, and up to 240 K for the most severe combination of test parameters, have been measured in gas impingement tests at a gas-to-sodium pressure ratio of  $\sim 2$ . At local hex-can failure in P4, the pressure ratio was 1.7 ( $P_{\text{gas}} = 520 \text{ kPa (75 psia)}$ ,  $P_{\text{sodium}} = 310 \text{ kPa (45 psia)}$ ). This condition had the potential for producing the maximum or near-maximum increase in cladding temperature due to gas impingement. Although the P4 operating conditions were intentionally severe, no cladding failures attributed to this unplanned gas release were identified during the detailed macroexamination.

In tests to study two-phase flow effects, it was found that the rise of local coolant temperature was highest for large cladding breach sizes at relatively low pressure differences, which confines the gas to a limited number of subchannels. For the worst conditions, the maximum coolant temperature increase, extrapolated to full LMFBR power, was less than 100 K for gas release rates up to  $\sim 1000 \text{ cm}^3 (\text{STP})/\text{s}$  ( $61 \text{ in}^3 (\text{STP})/\text{s}$ ) under steady-state conditions.<sup>54</sup>

These results were confirmed by the large volume, relatively low  $\Delta P$  gas release into the P4 fuel bundle through the local hex-can failure. About

1280 cm<sup>3</sup> (STP) of argon was released into the fuel bundle from the pressurized insulator region. The average release rate, over a three-s interval, was 425 cm<sup>3</sup> (STP)/s. Two thermocouples in the instrumented fuel bundle at the elevation of the gas release indicated local temperature perturbations of ~90 K followed by recovery to pre-gas-release levels. Smaller temperature perturbations were measured elsewhere in the bundle, also followed by recovery. The two-phase flow conditions reduced the bundle coolant flow rate from 2.95 kg/s to 2.8 kg/s for the three-s duration of the gas release.

In summary, the characteristics of the P4 gas release fitted the worst-case conditions for cladding overheating due to gas jet impingement and for coolant temperature increase due to two-phase flow effects. No cladding failure attributed to the gas release was identified. Excellent agreement was found between the peak coolant temperature perturbations measured in P4 and the ~100 K rise projected for worst-case (large breach, low  $\Delta P$ ) two-phase flow effects.

### 3. Gas Transit Time to the Pump

Early in the SLSF experiment program, model testing was performed to determine the gas deentrainment characteristics of the loop upper plenum. Based on simple physical models, the void fraction in the return flow to the pump and the reduction in pump efficiency were mapped out for various sodium flow rates and gas release rates. The results of this study indicated that no reduction in pump efficiency would result for conceivable gas release rates in the test section during whole-core loss-of-flow accident simulations because of complete deentrainment, but that for "[loop] flow rates of ~5 liters/s or greater, the injection of any amount of gas always results in some gas recirculation."<sup>55</sup>

The gas release following local failure of the hex-can wall was evaluated using the methods described in Ref. 55. The total gas release was ~1280 cm<sup>3</sup> (STP) or about 5000 cm<sup>3</sup> at the 83 kPa (12 psia) sodium-pressure and 880 K sodium-temperature conditions in the loop sodium plenum. Total loop flow varied between 7.4 and 8.1 kg/s at that time. Assuming that the gas release was uniform at 1.67 liter/s over the three-s interval and that the sodium flow rate was an average 9.5 liter/s, the void fraction in the coolant stream entering the plenum was ~15%. Experimental results indicate that the corresponding void fraction in the sodium entering the pump would be ~7.5%. This translates to an estimated 16.5% reduction in pumping efficiency, which indicates the sensitivity of the annular linear induction pump to the presence of gas in its flow stream.

Reactor scram at 118.4 s into the power transient followed a tailing off in both fuel-bundle inlet pressure and flow, which was atypical of either an increasing partial flow-channel blockage or debris sweepout. This appeared symptomatic of reduced pump performance, possibly due to gas entrainment in the pumped sodium. If the transit time for gas to reach the pump from the local hex-can failure approximated the measured 8.4 s from hex-can failure to scram, reduced pump performance due to gas entrainment in the pumped sodium could also be considered a contributor to reduced inlet flow and reactor scram.

The transit time for gas to reach the pump was estimated for three flow conditions, described as follows:

- Case 1 - coolant flow rate unperturbed by released gas, gas velocity = sodium velocity, gas mixed through sodium plenum volume before entering the heat exchanger
- Case 2 - measured coolant flow rate (perturbed by released gas), gas velocity = sodium velocity, gas mixed through sodium plenum volume before entering the heat exchanger
- Case 3 - measured coolant flow rate (perturbed by released gas), gas velocity = sodium velocity, gas carried directly from test train flow diverter to nearest heat exchanger tubes

These conditions, the regional coolant volumes in the flow stream between the fuel bundle and the pump, and the resulting transit times are shown in Table 27. Gas velocity equalled sodium velocity in all cases. For unperturbed coolant flow rates and gas mixing throughout the loop sodium plenum volume before entering the heat exchanger (Case 1), the transit time is 10 s. The gas release did perturb the coolant flow rates; upward flow in the test train and combined flow sections increased by ~0.6 to 0.7 kg/s. For the perturbed

Table 27. Transit Time for Gas from the Local Hex-Can Failure to the Pump

|                                           |                                            | Case 1            | Case 2 | Case 3 |
|-------------------------------------------|--------------------------------------------|-------------------|--------|--------|
| sodium flow rates, kg/s                   |                                            |                   |        |        |
| test train                                |                                            | 2.95              | 3.55   | 3.55   |
| combined flow                             |                                            | 7.4               | 8.1    | 8.1    |
| heat exchanger                            |                                            | 7.4               | 7.4    | 7.4    |
| region                                    | sodium volume<br>liters (in <sup>3</sup> ) | transit time<br>s |        |        |
| local hex-can failure<br>to bundle outlet | 2.3 (139)                                  | 0.58              | 0.48   | 0.48   |
| combined flow section                     | 14.3 (875)                                 | 1.57              | 1.43   | 1.43   |
| loop sodium plenum                        | 14.6 (889)                                 | 1.59              | 1.45   | 0.22   |
| sodium plenum to<br>pump inlet            | 55.6 (3390)                                | 6.24              | 6.24   | 6.24   |
| total transit time:                       |                                            | 9.98              | 9.74   | 8.37   |

coolant flow rates (Case 2), the transit time is 9.7 s. Case 3 illustrates a different behavior of gas entering the loop sodium plenum. The distance from the test train flow diverter, where sodium exited the test train into the sodium plenum, to the innermost row of heat exchanger tubes was ~45.7 mm (1.8 in). If gas separation was incomplete and some gas was carried directly from the flow diverter into the nearest heat exchanger tubes in the upper tube sheet, the transit time through the sodium plenum would be markedly reduced. At a sodium flow rate of 8.1 kg/s into the loop sodium plenum, the average coolant velocity between the flow diverter and the nearest heat exchanger tubes would be 0.21 m/s (0.68 ft/s); the shortest transit time in the plenum would be 0.22 s. The 8.37-s transit time for Case 3 is in excellent agreement with the measured time from local hex-can failure to scram. It is quite possible, then, that deentrainment of the gas from the sodium flowing through loop plenum was incomplete, that gas entrained in the sodium was carried down through the heat exchanger and entered the pump, which contributed to reduced pump performance, a decrease in flow and pressure, and reactor scram.

#### G. Sodium Urano-Plutonate Dissociation

Sodium urano-plutonate in all the failed, sodium-logged fuel pins dissociated out to 0.75  $R_{fuel}$  during the approach to power that was terminated by blockage reconfiguration at 100-MW ETR power (see Ch. V). Likewise, the isotherm of sodium urano-plutonate dissociation in the partial flow blockage must have moved from the interior of the fuel-debris volume toward the sodium-cooled blockage surface as reactor power increased. The depths from the blockage surface to temperature isotherms for sodium urano-plutonate dissociation (1520 K at 4 atm)<sup>56</sup> and for fuel melting (3000 K) were calculated for slab and spherical blockage geometries as a function of reactor power. The dissociation radius within the failed fuel pins was also calculated.

For uniform heat generation in a slab geometry, the temperature change with position in the body is described by

$$T(x) - T_o = \frac{P}{2k} (x - x_o)^2 \quad (25)$$

Likewise, the relationship for temperature change within a spherical body experiencing uniform internal heat generation is given by

$$T(r) - T_o = \frac{P (r - r_o)^2}{6k} \quad (26)$$

where

$P$  = volumetric power generation rate,  
 $k$  = material thermal conductivity,

$x - x_o$   
 $r - r_o$  = distance from surface.



Estimates of the distance in slab and spherical flow blockage geometries to isotherms for sodium urano-plutonate dissociation and for fuel melting during the return-to-full-power operation are shown in Table 28.

Table 28. Distance to Isotherms for Sodium Urano-Plutonate Dissociation and Fuel Melting in a Fuel Blockage

| Reactor Power<br>MW | Bundle Midplane Na Temperature<br>K | Peak Volumetric Heat Generation Rate<br>W/cm <sup>3</sup> | Distance from Surface to Isotherm<br>mm |        |              |        |
|---------------------|-------------------------------------|-----------------------------------------------------------|-----------------------------------------|--------|--------------|--------|
|                     |                                     |                                                           | Dissociation                            |        | Fuel Melting |        |
|                     |                                     |                                                           | Slab                                    | Sphere | Slab         | Sphere |
| 20                  | 720                                 | 280                                                       | 3.3                                     | 5.7    | 5.1          | 8.8    |
| 40                  | 743                                 | 570                                                       | 2.3                                     | 4.0    | 3.6          | 6.2    |
| 60                  | 767                                 | 850                                                       | 1.8                                     | 3.1    | 2.9          | 5.0    |
| 80                  | 790                                 | 1140                                                      | 1.6                                     | 2.7    | 2.5          | 4.3    |
| 100                 | 815                                 | 1420                                                      | 1.4                                     | 2.3    | 2.2          | 3.8    |

For purposes of these calculations, the blockage fuel was considered to be 85%-dense ( $U_{0.75}Pu_{0.25}$ )  $O_{2.00}$ , the same as in the fuel pins. Thermal conductivity values for 100%-dense ( $U_{0.8}Pu_{0.2}$ )  $O_{2.00}$  fuel from Fink<sup>57</sup> were reduced by 15% to represent 100%-dense ( $U_{0.75}Pu_{0.25}$ )  $O_{2.00}$  fuel; the thermal conductivity of the 85%-dense fuel was obtained from the thermal conductivity of 100%-dense fuel by use of the Maxwell-Eucken equation

$$k = k_o \left( \frac{1 - P}{1 + 2 P} \right) \quad (27)$$

where

$k_o$  = thermal conductivity at the theoretical density,  
 $P$  = fractional porosity.

Table 28 results indicate that much of the sodium urano-plutonate in the partial flow-channel blockage had dissociated upon reaching 100-MW ETR power. The isotherm for dissociation of sodium urano-plutonate was estimated to be within 1 to 2 mm of the blockage surface. These values may be very conservative. They do not take into account the reduction in fuel thermal conductivity caused by presence of the sodium urano-plutonate or the effect of a higher fuel porosity (lower fuel thermal conductivity) in the blockage than in the fuel pins. Either or both of these effects would further reduce the thickness of the sodium urano-plutonate remaining at power. For the blockage-fuel conditions described above, fuel melting at 100-MW ETR power is estimated to reach within 2 to 4 mm of the blockage surface. It, too, would be influenced by the fuel density in the debris blockage.

The radial location of the 1520 K isotherm for dissociation of sodium urano-plutonate within the failed, sodium-logged fuel pins was estimated for the following operating conditions:

|                              |                        |
|------------------------------|------------------------|
| ETR power                    | 100 MW                 |
| Fuel pin linear power*       | 28.7 kW/m (8.75 kW/ft) |
| Bulk sodium temperature*     | 815 K (1010°F)         |
| Fuel surface temperature*    | 855 K (1080°F)         |
| Coolant pressure at blockage | 4.05 atm               |
| Coolant flow rate            | 2.95 kg/s (6.50 lb/s)  |

---

\* at P4 fuel-bundle axial midplane

The fuel-pin temperature analysis routine and the various mixed-oxide thermal conductivity correlations in the computer code MARGE<sup>58</sup> were used for the calculations. Two cases were considered:

1. fuel in as-fabricated condition,  $OD_{fuel} = 4.88$  mm (0.1945 in.), fuel density = 90.4%, fuel thermal conductivity unaffected by presence of sodium urano-plutonate
2. fuel swelled to cladding ID,  $OD_{fuel} = 5.08$  mm (0.200 in.), fuel density = 85.5%, fuel thermal conductivity unaffected by presence of sodium urano-plutonate

For Case 1, fuel in the as-fabricated condition, the radial location ( $R_{diss}/R_{fuel}$ ) for the dissociation isotherm was 0.5. This was an average from five mixed-oxide thermal conductivity correlations. For Case 2, using the same thermal conductivity correlations, the average of five calculations for the dissociation radius was  $R_{diss}/R_{fuel} = 0.6$ . Neither set of calculations yielded a dissociation radius as large as the 75% measured in the posttest microexamination.

The above calculations were repeated to account for the effect of sodium urano-plutonate on the thermal conductivity of the sodium-wetted fuel. A multiplier was applied to the fuel conductivity for each thermal-conductivity correlation; the calculations for both Case 1 and Case 2 were repeated until  $R_{diss}/R_{fuel}$  equalled 0.75. The resulting multipliers, or correction factors, to account for the diminished thermal conductivity of mixed-oxide fuel filled with sodium urano-plutonate were 0.57 and 0.66 for Cases 1 and 2, respectively. The results for all cases are shown in Table 29.

Table 29. Sodium Urano-Plutonate Dissociation in Failed Fuel Pins

| Case | Description                                                                                   | $R_{diss}/R_{fuel}$ |
|------|-----------------------------------------------------------------------------------------------|---------------------|
| 1.   | as-manufactured pellet diameter, initial fuel density,<br>no sodium urano-plutonate in fuel   | 0.5                 |
| 2.   | fuel swelled to cladding ID, smeared fuel density,<br>no sodium urano-plutonate in fuel       | 0.6                 |
| 3.   | Case-1 fuel containing sodium urano-plutonate,<br>$k_{fuel+uranate} = 0.57 k_{fuel}$ (Case 1) | 0.75                |
| 4.   | Case-2 fuel containing sodium urano-plutonate,<br>$k_{fuel+uranate} = 0.66 k_{fuel}$ (Case 2) | 0.75                |

At the time of the blockage reconfiguration, the P4 fuel had accumulated 2.2 full-power days of irradiation. This is equivalent to about 0.1 a/o burnup. At such a low burnup, it is unlikely the fuel had completely swelled to the cladding ID due solely to burnup effects. The thermal conductivity of the P4 mixed-oxide fuel, diminished by the presence of sodium urano-plutonate to between 57 and 66% of Case 1 and Case 2 values, respectively, as estimated above, is in excellent agreement with the 60% conductivity value for trisodium uranate obtained in EBR-II out-of-pile tests.<sup>56,59</sup>

## VII. EXTRAPOLATION TO LARGE-PLANT CONDITIONS

Through test train design and selection of operating conditions, SLSF experiments have closely simulated steady-state operation and transient response in LMFBR systems. The achievement of relatively uniform radial temperature and coolant flow fields typical of the center pins and coolant subchannels in a large-plant core-center subassembly was a design goal. This was accomplished by using a row-by-row grading of the fuel  $^{235}\text{U}$  enrichment, in conjunction with a cadmium neutron filter on the loop secondary vessel, to produce a flat radial power profile across the fuel bundle. The spacing between the outer-row pins and the hex-can was half the distance between pins in the bundle to decrease flow streaming at the bundle periphery. This improved the uniformity of coolant distribution and the radial coolant temperature profile within the bundle. An insulator region exterior to the hex-can contained zirconia insulators in an argon gas atmosphere and served as a thermal barrier to radial heat transfer from the test section.

Relatively flat radial-power and coolant-temperature profiles across the fuel bundle resulting from the use of an uncooled hex-can outer surface and a graded fuel enrichment coupled with a cadmium neutron filter yielded nearly equal vulnerability to failure in the test fuel pins. However, these design features also introduced atypicalities in the response of P4 coolant and hex-can wall temperatures to fuel release that are not prototypic of a large plant. The lack of external sodium cooling or the heat-sink effect of adjacent subassemblies limited the capability of the P4 hex-can to tolerate contact with molten fuel or fuel debris. Within the graded-enrichment P4 fuel bundle, the power density of disrupted fuel had a radial-position dependency. Fuel ejected toward the bundle periphery increased in power and amplified the temperature increase in the wake of a partial flow blockage.

Calculations were performed to determine the coolant and hex-can wall temperatures in the absence of the P4 atypicalities identified above. These calculations and an evaluation of a large-plant DN response to a P4-type fuel exposure are discussed below.

### A. Thermal-Hydraulic Characteristics/Hex-Can Integrity

The blockage-wake temperature and the temperature of the hex-can wall in contact with once-molten fuel during the P4 power transient were evaluated using the SABRE-1 and THTB computer codes, respectively, as described in Ch. VI. Similar calculations were performed to extrapolate the P4 results to large-plant conditions.

Ideally, estimates of the blockage-wake temperature field in a large-plant core subassembly should be based on a P4-sized partial flow blockage in a subassembly containing 217 or 271 pins. However, SABRE-1 pretest calculations indicated that differences between the velocity, temperature, and pressure fields surrounding a fluted canister residing in a 37-pin subassembly and one residing in a 217-pin subassembly were quite minimal.<sup>44</sup> The estimate of the blockage-wake temperature field for a P4-sized partial flow blockage in a large-plant subassembly was made by repeating the calculations for the 37-pin P4 case, but with a uniform radial power density in the fuel pins and throughout the partial flow blockage.

The blockage-wake temperature field for the large-plant case is shown in Fig. 71. It is shown for the same "slice" through the bundle as that of Fig. 65 for the P4 case. The corresponding subchannel-by-subchannel description of sodium temperature is shown in Fig. 72. For this large-plant case, the temperature of the sodium near the hex-can wall where failure occurred in P4 was 55 K higher than the temperature of the sodium in free-stream flow.

To simulate a large plant, the THTB model used to analyze the temperature field in the P4 hex-can wall resulting from the presence of once-molten fuel was revised. The geometry of initially molten fuel and the thickness of the hex-can were unchanged from P4, but changes were made in the model input and boundary conditions. Power density in the initially molten fuel was reduced from 480 W/g in the P4 case to the 240-W/g level typical of the FTR core midplane at a total pin power of 33 kW. The boundary condition that described an inward-flowing surface heat flux on the uncooled surface of the P4 hex-can was replaced by a boundary condition describing convection to inter-subassembly sodium.

Once-molten fuel on the hex-can wall of a subassembly in the large plant was considered to be downstream from a local blockage, as in P4. The fuel-bundle inlet temperature was assumed to be 695 K. The temperature of sodium bathing the initially molten fuel on the hex-can in the large plant was 950 K, based on a flowing, free-stream sodium temperature of 895 K, including minor flow reduction effects due to the local blockage, and a 55 K temperature perturbation in the blockage wake, from SABRE-1 calculations. Sodium in the inter-subassembly gap at the elevation of the fuel adhering to the hex-can was assumed to be at 900 K; this was then increased by 50 K to account for perturbations due to the local heat source. In addition to the conservatism inherent in the model and in the selection of the heat sink temperatures, the boundary condition for outward heat flow along the extended hex-can wall (boundary condition 8003 in Fig. 67) was unchanged from that for the P4 case.

THTB results for selected nodes in the large-plant analysis are shown in Fig. 73. For comparable conditions in a large plant - five g of initially molten, low-burnup fuel adhering to the hex-can in the wake of a partial flow blockage - the margin to hex-can melting at Node 1 is 650 K. These results indicate that the increased power density in the fuel displaced outward in the P4 fuel bundle and the uncooled hex-can outer wall, both atypicalities not found in a large plant, were the major contributors to the local hex-can failure in P4. The radial-position-dependent variable power density of fuel in the graded-enrichment, cadmium-filtered P4 fuel bundle influenced the hex-can temperature field in two ways. The increased fuel power density accentuated the temperature perturbations in the blockage wake and doubled the heat load on the hex-can where it was in contact with initially molten fuel.

In summary, the conditions that elevated the local temperature of the P4 hex-can to melting are not present in a large plant. The margin to hex-can melting for five grams of fuel adhering to the inside surface of a large-plant subassembly downstream from a partial flow blockage at the core midplane is conservatively estimated to be 650 K.

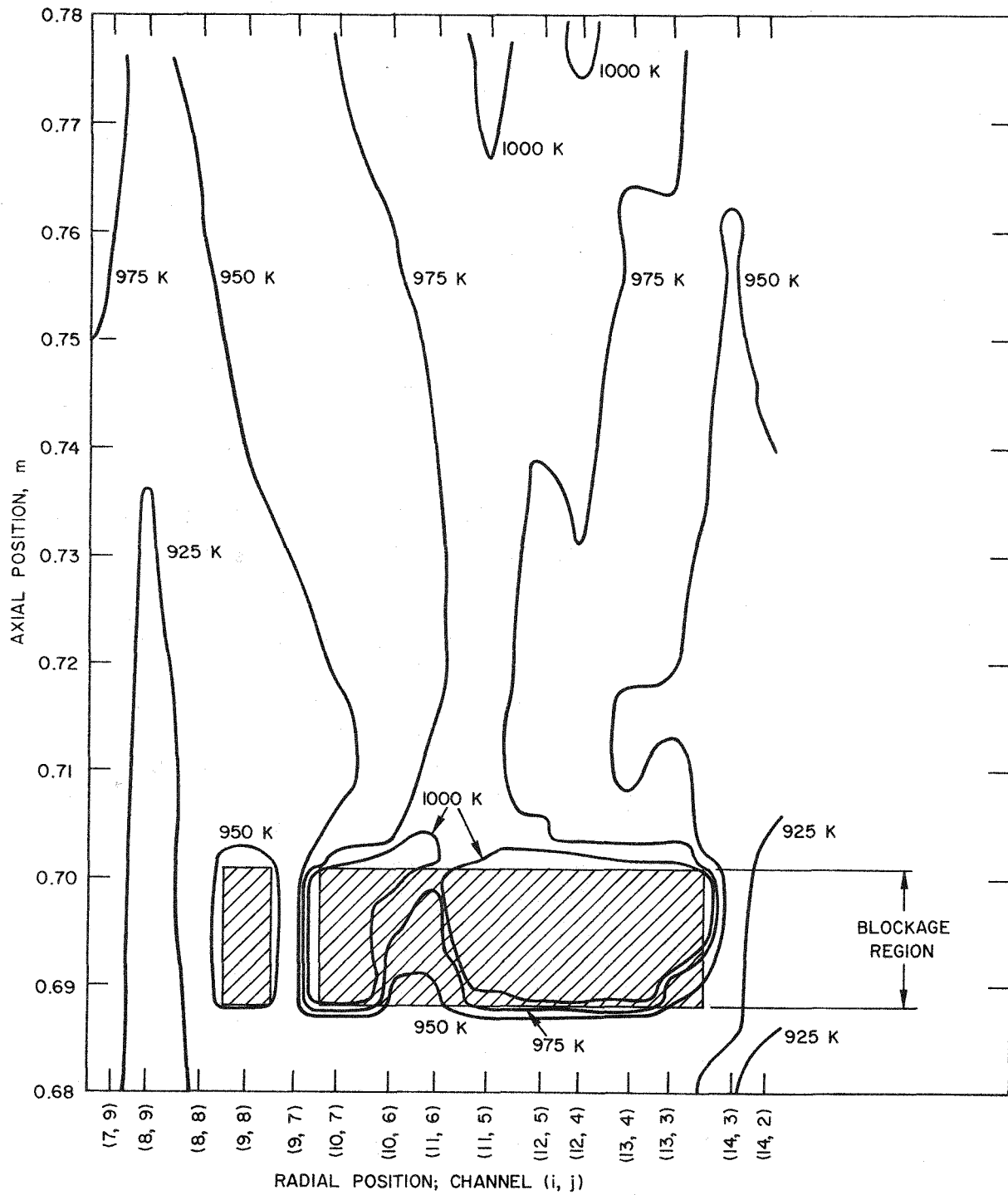


Fig. 71. Temperature Field in the Wake of a Partial Flow Blockage (Large-Plant Case)

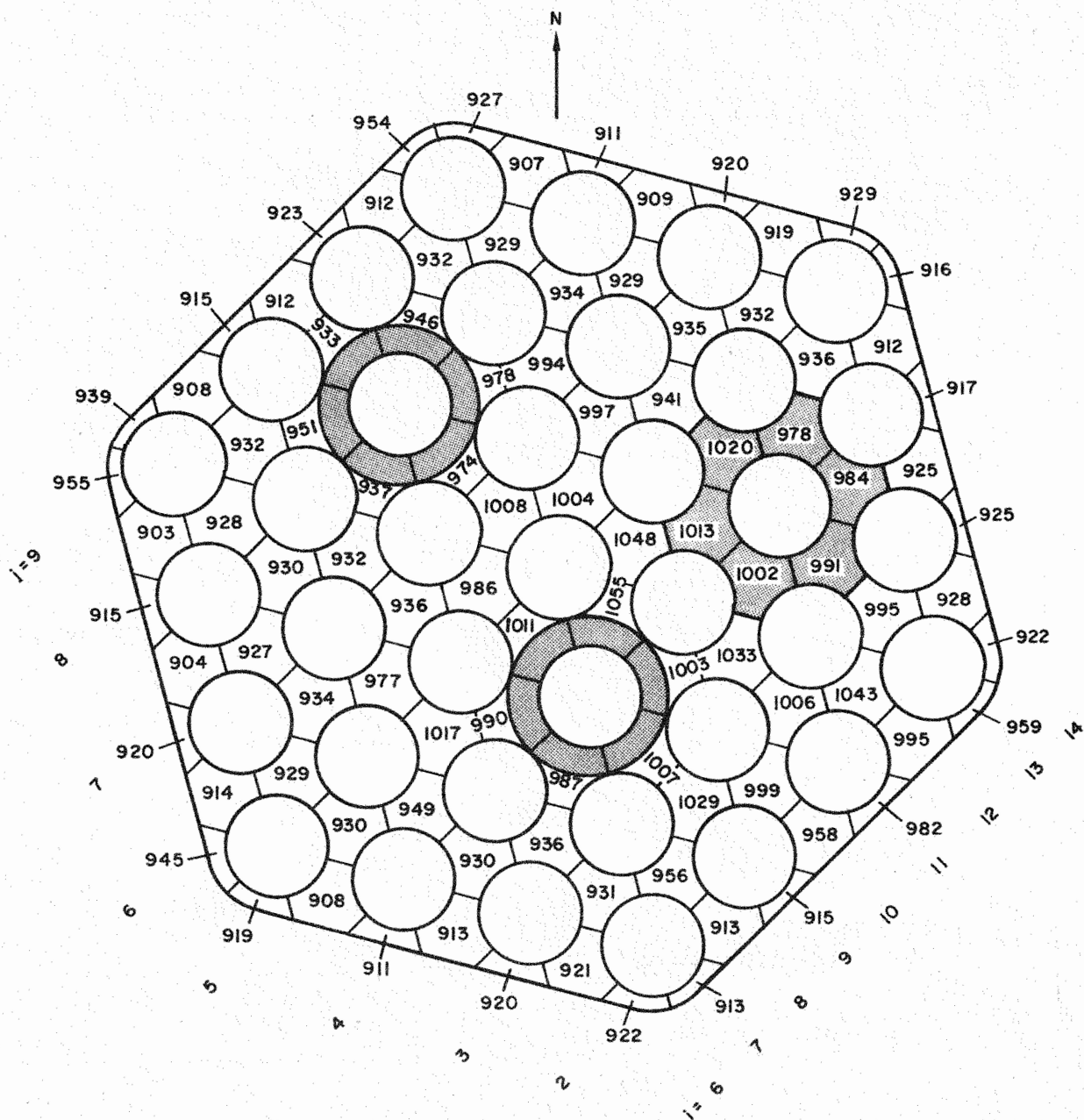


Fig. 72. Calculated Sodium Temperatures Downstream of a Partial Flow Blockage (Large-Plant Case) (Temperatures in K)

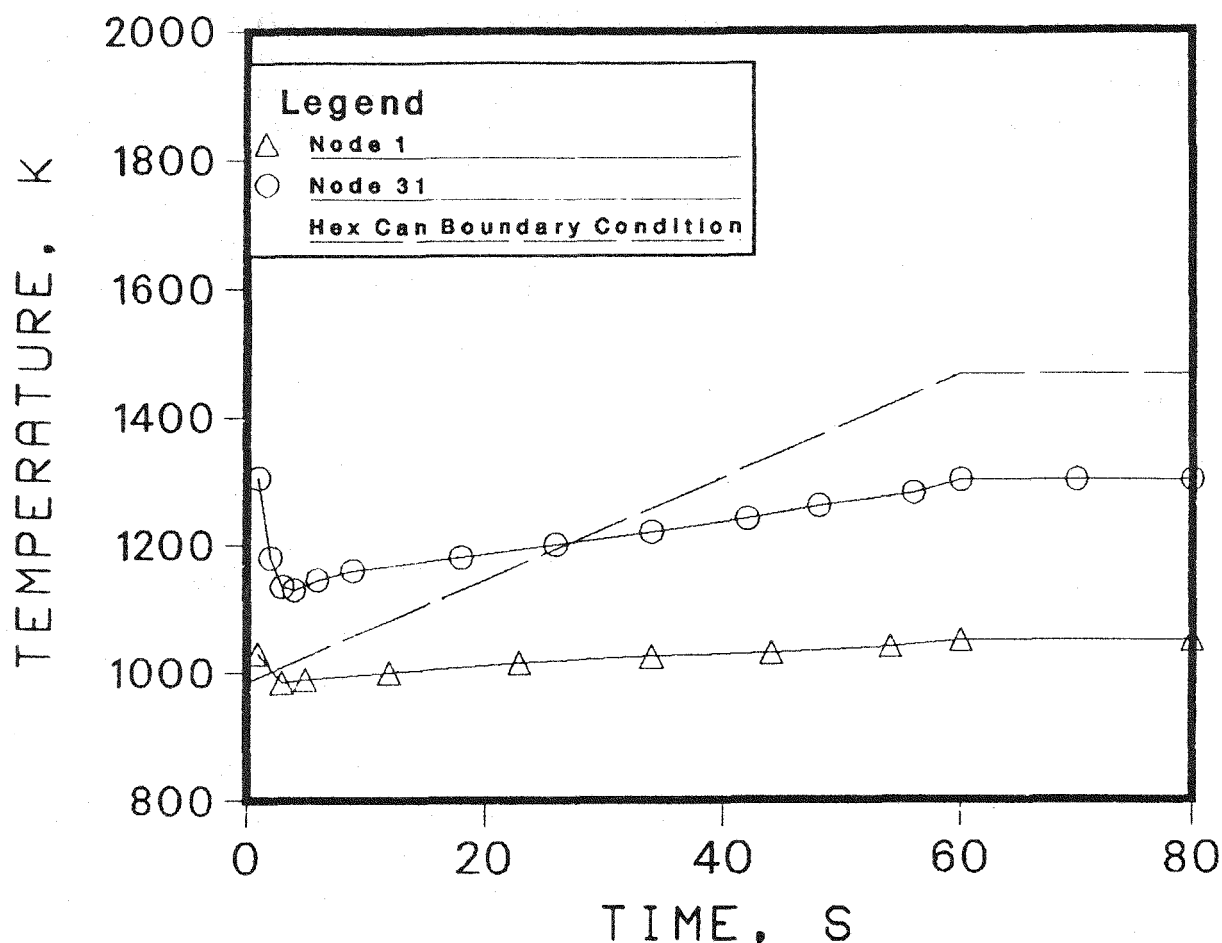


Fig. 73. Selected Hex-Can Node Temperatures for the Large-Plant Case

#### B. Failed-Fuel Response

In extrapolating the DN response from the P4 loop, with the DND and the DNM, to a large plant and its delayed neutron detectors, there are essentially two major areas to be considered: those that arise from the increased sodium inventory of the large plant, and are essentially numerical questions of scale and sensitivity, and those that are matters of time delay. The two are, of course, interconnected because of the decay of the DN emitters.

In a large reactor, a postulated P4-sized failure would release the same DN source to the coolant. This source would be carried through a flow system with the same essential features as the SLSF loop except for scale. The total volume of sodium would be enormously greater, and the upper plenum proportionately larger than in P4, so that the source material must become more dispersed after it leaves the original fuel channel. The residence time in the plenum depends on the ratio of recirculation flow to volume, and would



presumably be longer. In any case, the mixing time in the plenum would not be expected to be so fast or complete as we have assumed in the P4 case. Changing mixing patterns might introduce non-uniform concentrations and time delays, introducing, in effect, "noise" into the system response, but this would be a minor effect in the case of a major failure event, and will be neglected for the moment.

The general effect of these larger volumes will be, by diluting the sample, to reduce the sensitivity of detection compared with P4. However, this presents no problem, since in P4 the DN detectors were deliberately made extremely insensitive so as not to be overloaded. For the least sensitive P4 fission counters, the combination of detector sensitivity and geometric efficiency was  $1.5(10^{-7})$  counts per delayed neutron emitted in the sodium plenum volume. In operating reactors, DN detectors of much higher sensitivity are used; quite small pin failures can be detected.<sup>23,24,25</sup>

A more significant consequence of the larger size is that all of the transport times will be longer. There are two effects. The delay from the source to the detector is of the greater importance, since it determines the earliest time that information is available to the operator or alarm system. The relative decay of the shorter-lived components is also significant, since it affects the sharpness of the rise of the leading edge of the signal. The most significant transport delays are from the source to the upper plenum, and to the DN detector itself. The recirculation time of the reactor coolant is of lesser importance, but it does influence the rate of approach to equilibrium of the DN concentration in the plenum.

The detector system installed on a sodium sampling system, much as the OLSS/DNM in P4, is typical of a conventional LMFBR design. Only with innovative design will it be possible to have detectors that view the plenum directly as the DND did in P4. Therefore, a major part of the transit delay will be that of the sample stream itself.

For any reactor system, a computer simulation model can be constructed incorporating the particular details of the coolant system and the detector loop; such a model would enable detailed predictions to be made. In the P4 CSMP model, almost instantaneous mixing in the plenum was assumed. This is clearly unrealistic for a large volume in which nonuniform effects will be expected due to the flow patterns. These effects may include incomplete mixing, so that the sampling fraction may vary according to the location of the failure (in extreme cases some parts of the core might be invisible to a detector drawing its sample from a different location, so that several detectors should be employed). A more subtle effect would be fluctuations in the delay times to the sampling point, with consequent changes in the ratios of the DN isotopes. Both of these effects might, in turn, depend on the magnitude of the flow.

Such detailed effects cannot be considered here, but to give a general indication, the CSMP model used to study the DN response in P4 can be adapted, without major changes, to simulate a large reactor plant by altering a few key parameters or their ratios. The ratio of the primary system flow to the reactor plenum volume and the ratio of the OLSS sample flow to the plenum volume are such parameters. The main flow governs the transit time to the plenum, the residence time in the plenum, and the recirculation time. The

general effect of a long transit and recirculation time in a large plant is to reduce the contribution of the short-lived DN isotopes to the DN signal.

Predictions of such an adapted model are shown in Fig. 74 for fuel releases typical of the P4 power transient. The flow rate was adjusted so as to give a recirculation time of 40 s. An 8-s transport delay to the plenum from the core midplane was selected, which includes time spent in plenum mixing as well as transport to the sampling location. The sample flow concentration was reduced to 0.1% of that in the P4 case. This seems a reasonable value; in EBR-II, the fractional flow to the DN detector is about 1/80 of the whole. In addition, the large-plant calculations were performed with a constant 100-cm<sup>2</sup> equivalent area of exposed fuel to demonstrate the detectability of fuel release events above a large background signal.

The various cases illustrated in Fig. 74 are for values of the transport delay from the upper plenum to the detector of 0, 10, 20, and 30 s. Zero delay corresponds to a detector viewing a part of the plenum directly, as in the case of the DND. However, the transit time from the core to the plenum sample point adds 8 s, increasing the effective delays to 8, 18, 28, and 38 s. These calculations demonstrate that though the longer transport times do reduce the contribution of the short-lived isotopes, and though the effect is to reduce the sharpness of the leading edge of the signal, the effect is not so great as to reduce the effectiveness of the detection system. In general, short delay times are better, and allow earlier detection and action, but they are not critical to the operation.

One aim of a DN system, as particularly exemplified by the P4 blockage reconfiguration, would be to distinguish a slow increase that is significant from one which is due to non-threatening change, such as a slow widening of a minor crack in an already breached pin, a change in the power level, a change in a rod position, or a short-term fluctuation due to incomplete mixing, as mentioned earlier. Obviously it would be necessary to make a decision based on more than the DN level alone; the DN level compensated for reactor power is a more meaningful parameter and would have been a significant indicator in P4. Ideally, each large plant might have a computer to process on-line the relevant signals, including the power level, the DN rate, flow rates, temperatures and any other relevant data, and would process these data using a simulation model similar to that employed for P4, but programmed in detail for the flow patterns of the particular plant. Such a treatment would be able to provide a continuous output that would translate the DN signal into the "significance" of the signal at any given time, and at any stage of plant operation (start-up, full power, shut-down, etc.) expressed, perhaps, as equivalent area or equivalent mass of exposed fuel. In this way a rapid extension of a minor failure, which poses no immediate threat, could be distinguished from a significant change in a larger mass of exposed fuel. Finally, adjustment for "noise" in the system would require some type of smoothing or correlation; in the case of a multiple-loop reactor, a correlation between a DN detector in each loop would remove such fluctuations from the picture. One example of a system to provide quantitative diagnostic information on an on-going basis is the series DN-detector configuration planned to monitor future run-beyond-cladding-breach tests in EBR-II.<sup>60</sup>

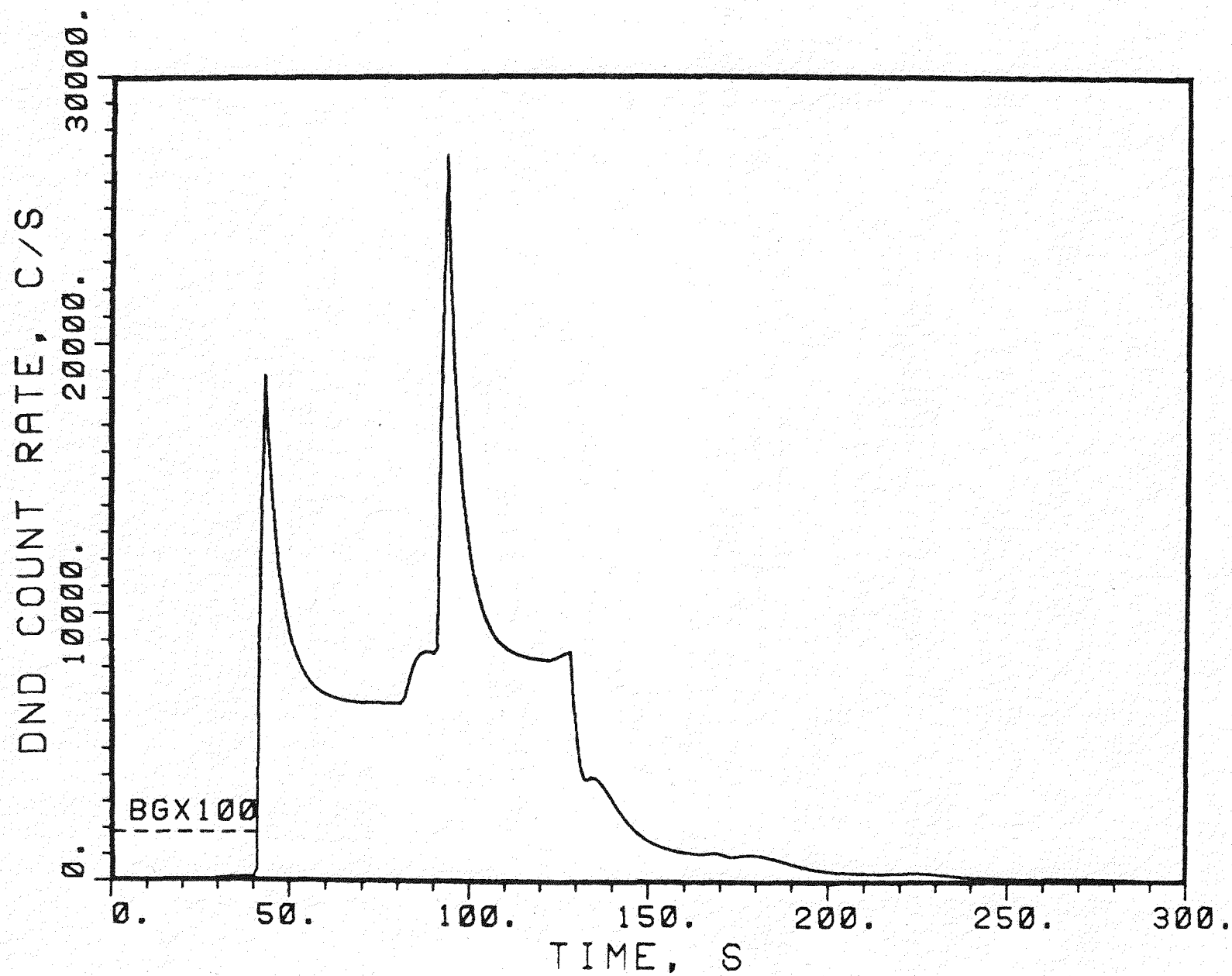


Fig. 74. Calculated Large-Plant DN Response to P4 Power Transient Sequence  
(a) Transport Delay = 0 s.

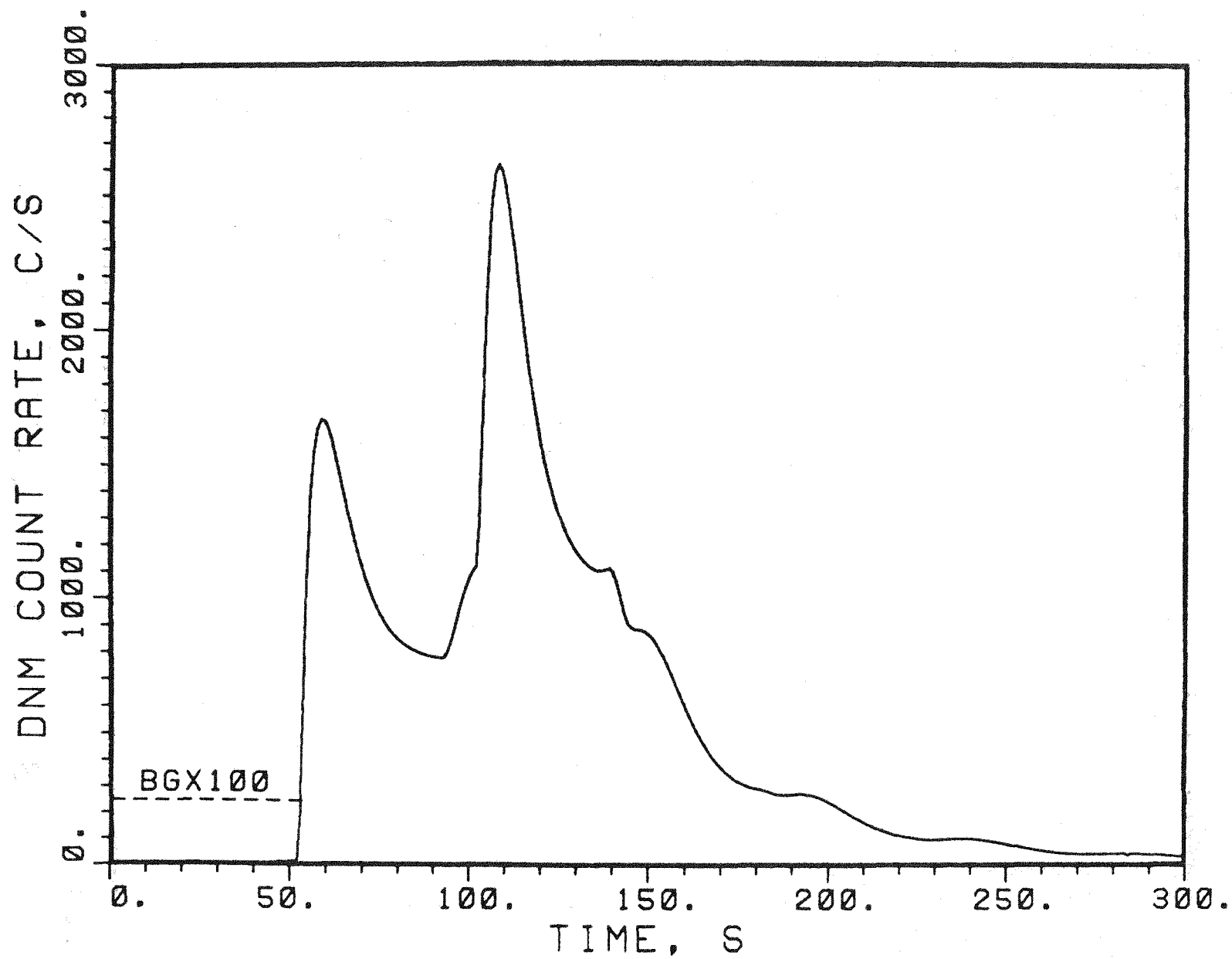


Fig. 74b. Transport Delay = 10 s.

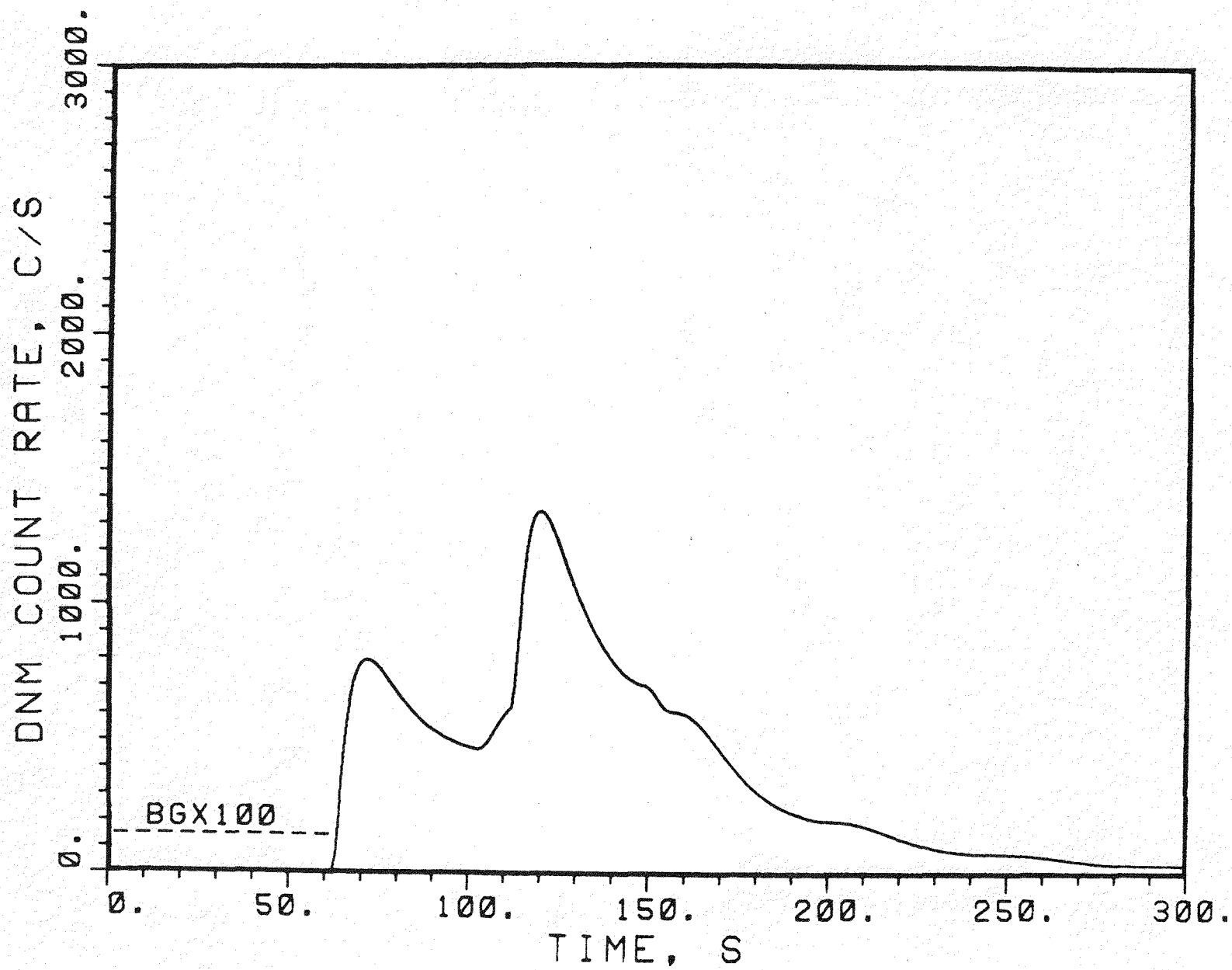


Fig. 74c. Transport Delay = 20 s.

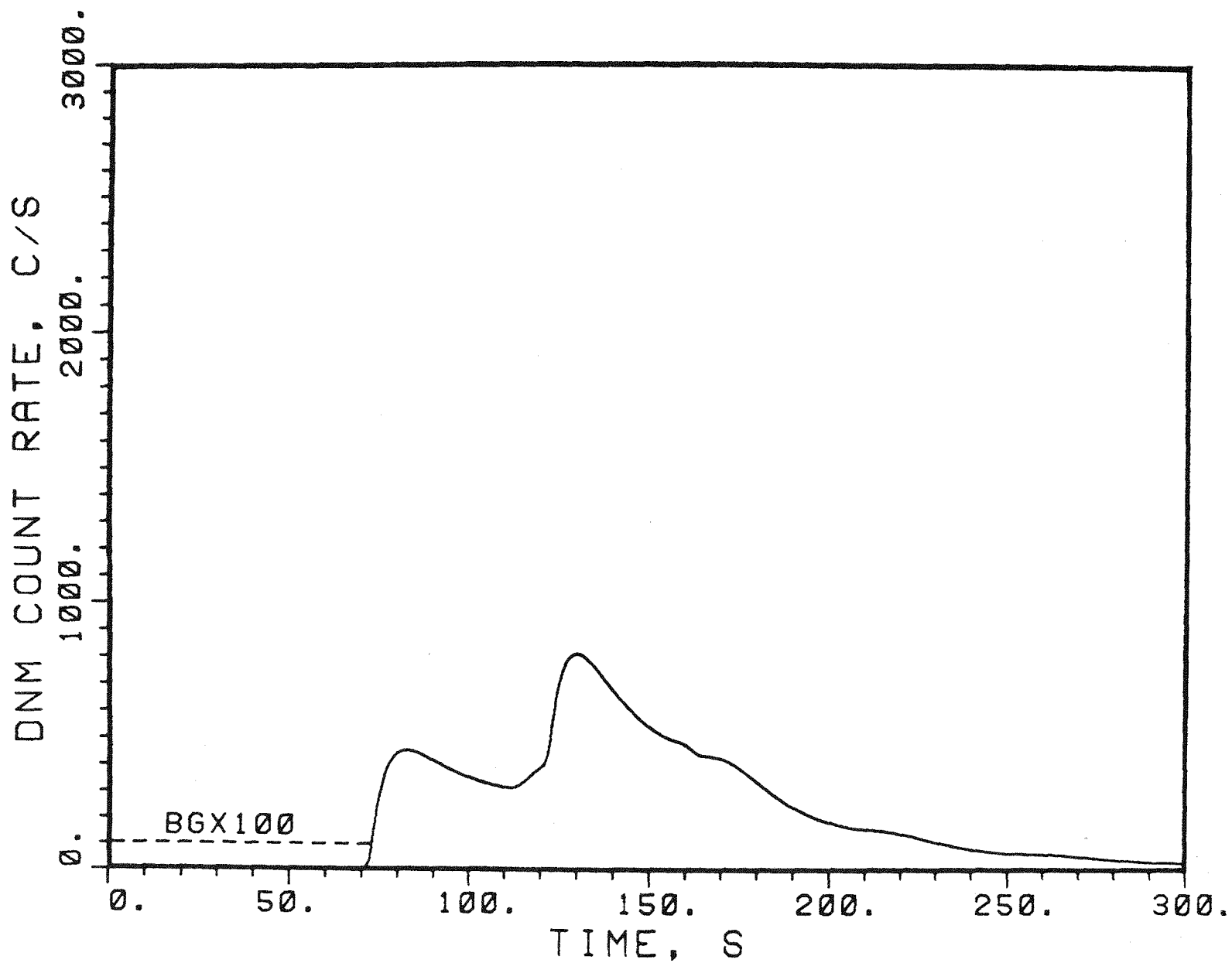


Fig. 74d. Transport Delay = 30 s.

The real conclusion to be drawn from P4 with respect to DN detection is its sensitivity and reliability when compared to other instruments. It is striking that the changes in the fuel which led to the reconfiguration were seen by the DN systems even when there was no indication from other sensors, temperature, acoustic, etc. With a well-designed detection system which has a very wide range of sensitivity, it is possible to observe, without saturation, a sequence of major events, such as those during the P4 power transient, and to interpret them in time, magnitude, and nature of fuel release.

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Appendix A

Data Acquisition System

Instrument Summary



Table A.1 Instrument Summary

| Sensor No.** | DAS Channel | Conversion Factor | Elevation    | Cable Dia., mm | Conduit No. | Description                                                                          |
|--------------|-------------|-------------------|--------------|----------------|-------------|--------------------------------------------------------------------------------------|
| Test Train   |             |                   |              |                |             |                                                                                      |
| FE 1-1       | 0           | A                 | 77'-4 1/8"   | 3.18           | 1           | Sodium flow rate at fuel bundle inlet<br>(permanent magnet flowmeters, 0-100 gpm)    |
| FE 1-2       | 1           | A                 | 77'-4 1/8"   |                |             |                                                                                      |
| FE 1-3       | 2           | A                 | 77'-6 5/16"  |                |             |                                                                                      |
| FE 1-4       | 3           | A                 | 77'-6 5/16"  | 3.18 (2)       | 2           | Sodium flow rate at fuel bundle outlet<br>(eddy current probe-type meter, 0-100 gpm) |
| FE 2-1       | 4           | A                 | 87'-10 5/16" |                | 7           |                                                                                      |
| FE 3-1       | 5           | A                 | 90'-2 11/32" | 3.18 (2)       | 8           | Total loop sodium flow rate<br>(eddy current probe-type meters, 0-150 gpm)           |
| FE 3-2       | 6           | A                 | 90'-6 5/16"  | 3.18 (2)       | 8           |                                                                                      |
| TE 1-1       | 15          | D                 | 77'-9 3/16"  | 1.57           | 1           | Sodium temperature at fuel bundle inlet                                              |
| TE 1-2       | 16          | D                 | 77'-9 19/32" | 1.57           | 1           |                                                                                      |
| TE 1-3       | 17          | D                 | 77'-9 3/16"  | 1.57           | 1           |                                                                                      |
| TE 1-4       | 18          | B                 | 77'-9 19/32" | 1.57           | 1           |                                                                                      |
| TE 2-1       | 19          | C                 | 86'-0 5/8"   | 1.57           | 7           | Sodium temperature at fuel bundle outlet                                             |
| TE 2-2       | 20          | C                 | 86'-3"       | 1.57           | 7           |                                                                                      |
| TE 2-3       | 21          | C                 | 86'-6 5/16"  | 1.57           | 7           |                                                                                      |
| TE 3-1       | 22          | K                 | 81'-8 5/16"  | 1.42           | 3           | Pin 1                                                                                |
| TE 3-2       | 23          | K                 | 79'-4 5/16"  | 1.42           | 6           | Pin 3                                                                                |
| TE 3-3       | 24          | K                 | 80'-6 5/16"  | 1.42           | 3           | Pin 4                                                                                |
| TE 3-4       | 25          | K                 | 81'-0 5/16"  |                | 3           | Pin 4                                                                                |
| TE 3-5       | 26          | K                 | 80'-6 5/16"  | 1.42           | 3           | Pin 5                                                                                |
| TE 3-6       | 27          | K                 | 81'-0 5/16"  |                | 3           | Pin 5                                                                                |
| TE 3-7       | 28          | K                 | 80'-6 5/16"  | 1.42           | 4           | Pin 6                                                                                |
| TE 3-8       | 29          | G                 | 81'-10 5/16" | 1.42           | 4           | Pin 7                                                                                |
| TE 3-9       | 30          | K                 | 80'-6 5/16"  | 1.42           | 6           | Pin 8                                                                                |
| TE 3-10      | 31          | K                 | 80'-8 5/16"  |                | 6           | Pin 8                                                                                |
| TE 3-11      | 32          | K                 | 82'-10 5/16" | 1.42           | 6           | Pin 10                                                                               |
| TE 3-12      | 33          | K                 | 80'-6 5/16"  | 1.42           | 6           | Pin 12                                                                               |
| TE 3-13      | 34          | K                 | 81'-0 5/16"  |                | 6           | Pin 12                                                                               |
| TE 3-14      | 35          | K                 | 79'-0 5/16"  | 1.42           | 3           | Pin 14                                                                               |
| TE 3-15      | 36          | K                 | 79'-8 5/16"  | 1.42           | 4           | Pin 18                                                                               |
| TE 4-1       | 37          | K                 | 83'-6 5/16"  | 0.71           | 6           | Pin 21                                                                               |
| TE 4-2       | 38          | K                 | 84'-0 5/16"  | 0.71           | 6           | Pin 22                                                                               |
| TE 4-3       | 39          | K                 | 81'-10 5/16" | 0.71           | 6           | Pin 25                                                                               |
| TE 4-4       | 40          | K                 | 81'-8 5/16"  | 0.71           | 6           | Pin 26                                                                               |
| TE 4-5       | 41          | G                 | 84'-6 5/16"  | 0.71           | 3           | Pin 30                                                                               |
| TE 4-6       | 42          | K                 | 78'-8 5/16"  | 0.71           | 3           | Pin 32                                                                               |
| TE 4-7       | 43          | K                 | 79'-6 5/16"  | 0.71           | 4           | Pin 34                                                                               |
| TE 4-8       | 44          | K                 | 78'-2 5/16"  | 0.71           | 4           | Pin 35                                                                               |
| TE 4-9       | 45          | K                 | 85'-6 5/16"  | 0.71           | 4           | Pin 37                                                                               |

Table A.1 Instrument Summary (cont'd)

| Sensor No. | DAS Channel | Conversion Factor | Elevation    | Cable Dia., mm | Conduit No. | Description                                                                        |
|------------|-------------|-------------------|--------------|----------------|-------------|------------------------------------------------------------------------------------|
| TE 5-1     | 46          | K                 | -76'-7 1/4"  | 0.71           | 2           | Sodium temperature in meltdown cup region                                          |
| TE 5-2     | 47          | K                 | 76'-7 1/4"   | 0.71           | 2           |                                                                                    |
| TE 5-3     | 48          | K                 | 76'-7 1/4"   | 0.71           | 2           |                                                                                    |
| TE 5-4     | 49          | K                 | 76'-7 1/4"   | 0.71           | 2           |                                                                                    |
| TE 6-1     | 50          | G                 | 81'-8 5/16"  | 0.71           | 6           | Sodium temperature at inner surface of hex-can wall 5 11/16 in above fuel midplane |
| TE 6-2     | 52          | G                 | 80'-8 5/16"  | 0.71           | 3           |                                                                                    |
| TE 6-3     | 53          | G                 | 80'-8 5/16"  | 0.71           | 3           |                                                                                    |
| TE 6-4     | 51          | G                 | 80'-8 5/16"  | 0.71           | 4           |                                                                                    |
| TE 6-5     | 54          | G                 | 80'-8 5/16"  | 0.71           | 4           |                                                                                    |
| TE 6-6     | 55          | G                 | 80'-8 5/16"  | 0.71           | 6           |                                                                                    |
| TE 7-1     | 56          | B                 | 78'-6 5/8"   | 0.71           | 1           | Bypass sodium temperature                                                          |
| TE 7-2     |             | B                 | 86'-1 7/8"   | 0.71           | 1           |                                                                                    |
| TE 7-3     |             | B                 | 78'-6 5/8"   | 0.71           | 1           |                                                                                    |
| TE 7-4     |             | B                 | 83'-11 5/16" | 0.71           | 1           |                                                                                    |
| TE 7-5     | 57          | B                 | 86'-1 7/8"   | 0.71           | 1           |                                                                                    |
| TE 7-6     | 58          | B                 | 82'-0 5/8"   | 0.71           | 1           |                                                                                    |
| TE 7-7     | 59          | B                 | 86'-1 7/8"   | 0.71           | 2           |                                                                                    |
| TE 7-8     |             | B                 | 78'-6 5/8"   | 0.71           | 2           |                                                                                    |
| TE 7-9     | 60          | B                 | 82'-0 5/8"   | 0.71           | 2           |                                                                                    |
| TE 8-1     | 61          | B                 | 89'-2 11/16" | 1.57           | 8           | Mixed mean sodium temperature                                                      |
| TE 8-2     | 62          | B                 | 91'-5 7/16"  | 1.57           | 8           |                                                                                    |
| TE 8-3     | 63          | B                 | 95'-11 5/16" | 1.57           | 9           |                                                                                    |
| TE 8-4     | 64          | B                 | 99'-6 1/16"  | 1.57           | 10          |                                                                                    |
| TE 9-1     | 65          | B                 | 101'-6 5/8"  | 1.57           | 10          | Loop plenum gas temperature                                                        |
| TE 9-2     | 66          | B                 | 101'-6 5/8"  | 1.57           | 10          |                                                                                    |
| TE 19-1    | 167         | A                 | 102'-10 1/2" | ---            | --          | Connector temperature                                                              |
| TE 22-1    | 97          | K                 | 86'-3 15/16" | 1.57           | 5           | Sodium temperature at pump outlet                                                  |
| TE 22-2    | 98          | K                 | 86'-3 15/16" | 1.57           | 5           |                                                                                    |
| TE 22-3    |             | K                 | 86'-3 15/16" | 1.57           | 5           |                                                                                    |
| PE 1-1     | 7           | A                 | 77'-10 7/16" | 1.57           | 2           | Fuel bundle inlet sodium pressure, 0-2000 psia, type II fast response              |
| PE 1-2     | 8           | A                 | 77'-10 7/16" | 1.57           | 1           | 0-400 psia, type IA wide response                                                  |
| PE 2-1     | 105         | A                 | 77'-10 7/16" | 1.57           | 2           | Fuel bundle inlet sodium pressure, 0-200 psia, type I slow response                |
| PE 2-2     | 106         | A                 | 77'-10 7/16" | 1.57           | 2           |                                                                                    |
| PE 3-1     | 9           | A                 | 86'-4 13/16" | 1.57           | 7           | Fuel bundle outlet sodium pressure, 0-2000 psia, type II fast response             |
| PE 3-2     |             | A                 | 86'-4 13/16" | 1.57           | 7           |                                                                                    |
| PE 4-1     | 107         | A                 | 86'-4 13/16" | 1.57           | 7           | Fuel bundle outlet sodium pressure, 0-100 psia, type I slow response               |
| PE 4-2     |             | A                 | 86'-4 13/16" | 1.57           | 7           | 0-200 psia, type IA wide response                                                  |
| PE 5-1     | 10          | A                 | 88'-6 1/2"   | 1.57           | 5           | Pump sodium outlet pressure, 0-200 psia, type I                                    |

Table A.1 Instrument Summary (cont'd)

| Sensor No. | DAS Channel | Conversion Factor | Elevation      | Cable Dia.,<br>mm | Conduit No. | Description                                                       |
|------------|-------------|-------------------|----------------|-------------------|-------------|-------------------------------------------------------------------|
| PE 6-1     | 11          | A                 | 92'-2"         | 1.57              | 9           | Mixed mean sodium pressure,<br>0-2000 psia, type II fast response |
| PE 6-2     | 12          | A                 | 92'-2"         | 1.57              | 9           | 0-100 psia, type IA wide response                                 |
| PE 7-1     | 112         | A                 | 101'-7 5/8"    | 3.18              | --          | Loop plenum-gas pressure, 0-100 psia,<br>type I slow response     |
| PE 7-2     | 113         | A                 | 101'-7 5/8"    | 3.18              | --          |                                                                   |
| PE 7-3     | 114         | A                 | 101'-7 5/8"    | 3.18              | --          |                                                                   |
| PE 7-4     | 115         | A                 | 101'-7 5/8"    | 3.18              | --          |                                                                   |
| PE 9-1     | 13          | A                 | 86'-5 7/16"    | 3.18              | 5           | Acoustic detector (bypass flow)                                   |
| PE 9-2     | 14          | A                 | 95'-11 3/4"    | 3.18              | 9           | Acoustic detector (total flow)                                    |
| PE 10-1    | 116         | A                 | 85'-10 1/8"    | 1.57              | 4           | Pin 1                                                             |
| PE 10-2    | 117         | A                 | 85'-10 1/8"    | 1.57              | 4           | Pin 2                                                             |
| PE 10-3    | 118         | A                 | 85'-10 1/8"    | 1.57              | 3           | Pin 3                                                             |
| PE 10-4    | 119         | A                 | 85'-10 1/8"    | 1.57              | 3           | Pin 4                                                             |
| PE 10-5    | 120         | A                 | 85'-10 1/8"    | 1.57              | 4           | Pin 6                                                             |
| PE 10-6    | 121         | A                 | 85'-10 1/8"    | 1.57              | 4           | Pin 7                                                             |
| PE 10-7    | 122         | A                 | 85'-10 1/8"    | 1.57              | 6           | Pin 9                                                             |
| PE 10-8    |             | A                 | 85'-10 1/8"    | 1.57              | 6           | Pin 10                                                            |
| PE 10-9    | 123         | A                 | 85'-10 1/8"    | 1.57              | 3           | Pin 11                                                            |
| PE 10-10   | 124         | A                 | 85'-10 1/8"    | 1.57              | 3           | Pin 13                                                            |
| PE 10-11   | 125         | A                 | 85'-10 1/8"    | 1.57              | 3           | Pin 14                                                            |
| PE 10-12   |             | A                 | 85'-10 1/8"    | 1.57              | 3           | Pin 15                                                            |
| PE 10-13   |             | A                 | 85'-10 1/8"    | 1.57              | 4           | Pin 16                                                            |
| PE 10-14   | 126         | A                 | 85'-10 1/8"    | 1.57              | 4           | Pin 19                                                            |
| LE 1-1     | 127         | A                 | 100'-1 3/4" to | 3.18              | 10          | Sodium level in loop plenum                                       |
| LE 1-2     |             | A                 | 100'-7 3/8"    | 3.18              | 10          |                                                                   |

## Loop

|         |    |   |            |                                                                                                  |                                    |  |
|---------|----|---|------------|--------------------------------------------------------------------------------------------------|------------------------------------|--|
| TE 11-1 | 67 | B | 98'-10"    | Heat exchanger helium inlet temperature<br>outside the helium divider near the top<br>tube sheet |                                    |  |
| TE 11-2 | 68 | B | 98'-10"    |                                                                                                  |                                    |  |
| TE 12-1 | 69 | B | 99'-0 7/8" | Heat exchanger helium outlet temperature<br>near the top tube sheet                              |                                    |  |
| TE 12-2 | 70 | B | 99'-0 7/8" |                                                                                                  |                                    |  |
| TE 13-1 | 71 | D | 93'-8 1/2" | first row                                                                                        | Heat exchanger tube<br>temperature |  |
| TE 13-2 | 72 | D | 93'-8 1/2" | second row                                                                                       |                                    |  |
| TE 13-3 | 73 | D | 93'-8 1/2" | third row                                                                                        |                                    |  |
| TE 13-4 | 74 | B | 93'-8 1/2" | fifth row                                                                                        |                                    |  |

Table A.1 Instrument Summary (cont'd)

| Sensor No. | DAS Channel | Conversion Factor | Elevation    | Cable Dia.,<br>mm | Conduit No. | Description                                                                    |
|------------|-------------|-------------------|--------------|-------------------|-------------|--------------------------------------------------------------------------------|
| TE 14-1    | 75          | F                 | 80'-1"       |                   |             | Temperature on outside of primary vessel at the axial center of the ETR core   |
| TE 14-2    |             | F                 | 80'-1"       |                   |             |                                                                                |
| TE 14-3    | 76          | F                 | 80'-1"       |                   |             |                                                                                |
| TE 14-4    |             | F                 | 80'-1"       |                   |             |                                                                                |
| TE 14-5    | 77          | F                 | 80'-1"       |                   |             |                                                                                |
| TE 14-6    |             | F                 | 80'-1"       |                   |             |                                                                                |
| TE 14-7    | 78          | F                 | 80'-1"       |                   |             |                                                                                |
| TE 14-8    |             | F                 | 80'-1"       |                   |             |                                                                                |
| TE 15-1    | 79          | H                 | 76'-9 1/2"   |                   |             | Temperature on outside of primary vessel in meltdown cup region                |
| TE 15-2    | 80          | H                 | 76'-9 1/2"   |                   |             |                                                                                |
| TE 15-3    | 81          | H                 | 76'-9 1/2"   |                   |             |                                                                                |
| TE 15-4    | 82          | H                 | 76'-9 1/2"   |                   |             |                                                                                |
| TE 15-5    | 83          | B                 | 81'-8 5/8"   |                   |             | Temperature on outside of primary vessel at top of fuel                        |
| TE 15-6    | 84          | B                 | 81'-8 5/8"   |                   |             | Temperature on inside of secondary vessel, at the axial center of the ETR core |
| TE 16-1    | 85          | B                 | 80'-1"       |                   |             |                                                                                |
| TE 16-2    | 86          | B                 | 80'-1"       |                   |             |                                                                                |
| TE 16-3    | 87          | B                 | 80'-1"       |                   |             |                                                                                |
| TE 16-4    | 88          | B                 | 80'-1"       |                   |             |                                                                                |
| TE 17-1    | 89          | B                 | 83'-4"       |                   |             | Temperature on outside of primary vessel 29 in below pump outlet               |
| TE 17-2    | 90          | B                 | 83'-4"       |                   |             |                                                                                |
| TE 18-1    | 91          | B                 | 91'-6 1/16"  |                   |             | Coil #7                                                                        |
| TE 18-2    | 92          | B                 | 91'-1 5/16"  |                   |             | Coil #11                                                                       |
| TE 18-3    | 93          | B                 | 90'-1 7/16"  |                   |             | Coil #21                                                                       |
| TE 18-4    | 94          | B                 | 89'-7 1/2"   |                   |             | Coil #26                                                                       |
| TE 18-5    | 95          | B                 | 88'-5 1/4"   |                   |             | Coil #38                                                                       |
| TE 18-6    | 96          | B                 | 87'-1 13/16" |                   |             | Coil #51                                                                       |
| TE C-1     | 193         | B                 | 102'-5 3/4"  |                   |             | Loop connector temperature                                                     |
| TE C-2     | 194         | B                 | 102'-5 3/4"  |                   |             |                                                                                |
| TE C-3     | 195         | B                 | 102'-5 3/4"  |                   |             |                                                                                |
| TE C-4     | 196         | B                 | 102'-5 3/4"  |                   |             |                                                                                |
| PE 17-1    | 108         | A                 |              |                   |             | Loop plenum-gas inlet pressure<br>0-100 psia, type I slow response             |
| PE 17-2    | 109         | A                 |              |                   |             |                                                                                |
| PE 17-3    | 110         | A                 |              |                   |             |                                                                                |
| PE 17-4    | 111         | A                 |              |                   |             |                                                                                |
| JT-11      | 102         | L                 |              |                   |             | Pump power, section 1, kW                                                      |
| JT-12      | 103         | M                 |              |                   |             | Pump power, section 2, kW                                                      |
| JT-13      | 104         | M                 |              |                   |             | Pump power, section 3, kW                                                      |
| ALIP KW    | 100         | M                 |              |                   |             | Pump real power                                                                |
| ALIP KVAR  | 101         | M                 |              |                   |             | Pump reactive power                                                            |
| ALIP VOLT  | 99          | J                 |              |                   |             | Pump voltage                                                                   |

Table A.1 Instrument Summary (cont'd)

| Sensor No.    | DAS Channel | Conversion Factor | Elevation | Cable Dia., mm | Conduit No. | Description                                  |
|---------------|-------------|-------------------|-----------|----------------|-------------|----------------------------------------------|
| FE 4-1        | 165         | A                 |           |                |             | Annulus gas make-up flow                     |
| FE 4-2        | 166         | A                 |           |                |             |                                              |
| Helium System |             |                   |           |                |             |                                              |
| TT-19         | 148         | A                 |           |                |             | RTC outlet piping helium temperature         |
| TT-20         | 149         | A                 |           |                |             | RTC inlet piping helium temperature          |
| PT-12         | 150         | A                 |           |                |             | M-18 outlet mix pressure                     |
| PT-13         | 151         | A                 |           |                |             | IPL HX inlet-leg helium pressure             |
| PT-14         | 152         | A                 |           |                |             | Helium circulator outlet pressure            |
| PT-15         | 153         | A                 |           |                |             | IPL HX outlet-leg helium pressure            |
| PT-16         | 154         | A                 |           |                |             | Helium circulator inlet pressure             |
| dPR-6         | 155         | A                 |           |                |             | Helium gas circulator ΔP                     |
| TRC-8         | 156         | A                 |           |                |             | Circulator inlet temperature                 |
| FRC-1         | 157         | A                 |           |                |             | Helium system total flow rate                |
| FR-2          | 158         | A                 |           |                |             | Helium system flow rate to IPL HX            |
| TRC-4         | 159         | A                 |           |                |             | Aftercooler mix temperature                  |
| PR-5          | 160         | A                 |           |                |             | IPL helium gas flow inlet pressure           |
| dPR-18        | 161         | A                 |           |                |             | Loop HX helium flow ΔP                       |
| FCV-2         | 162         | A                 |           |                |             | Position of flow control valve 2             |
| dPT-19        | 163         | A                 |           |                |             | FCV-2 ΔP                                     |
| dPT-20        | 164         | A                 |           |                |             |                                              |
| Reactor       |             |                   |           |                |             |                                              |
| PTC 1-0       | 128         | A                 |           |                |             | PTC 1                                        |
| PTC 2-0       | 132         | A                 |           |                |             | PTC 2 Time, power transient controller (PTC) |
| PTC 3-0       | 136         | A                 |           |                |             |                                              |
| PTC 1-1       | 129         | A                 |           |                |             | PTC 1                                        |
| PTC 2-1       | 133         | A                 |           |                |             | PTC 2 Lin-N average, PTC                     |
| PTC 3-1       | 137         | A                 |           |                |             |                                              |
| PTC 1-2       | 130         | A                 |           |                |             | PTC 1                                        |
| PTC 2-2       | 134         | A                 |           |                |             | PTC 2 Reference power, PTC                   |
| PTC 3-2       | 138         | A                 |           |                |             |                                              |
| PTC 1-3       | 131         | A                 |           |                |             | PTC 1                                        |
| PTC 2-3       | 135         | A                 |           |                |             | PTC 2 Controller state, PTC                  |
| PTC 3-3       | 139         | A                 |           |                |             |                                              |
| ROD 5         | 140         | A                 |           |                |             | Rod 5                                        |
| ROD 9         | 141         | A                 |           |                |             | Rod 9 Rod position                           |
| PTC 4         | 142         | A                 |           |                |             | Rod insert/withdrawal status                 |

Table A.1 Instrument Summary (cont'd)

| Sensor No.                        | DAS Channel | Conversion Factor | Elevation | Cable Dia.,<br>mm | Conduit No. | Description                        |
|-----------------------------------|-------------|-------------------|-----------|-------------------|-------------|------------------------------------|
| LIN A                             | 143         | A                 |           |                   | A           |                                    |
| LIN B                             | 144         | A                 |           |                   | B           | Raw Lin-N signal                   |
| LIN C                             | 145         | A                 |           |                   | C           |                                    |
| ETR PWR-A                         | 184         | A                 |           |                   | A           |                                    |
| ETR PWR-B                         | 185         | A                 |           |                   | B           | ETR neutron level                  |
| ETR PWR-C                         | 186         | A                 |           |                   | C           |                                    |
| ETR PWR                           | 187         | A                 |           |                   |             | ETR water power                    |
| ETR T-OUT                         | 188         | A                 |           |                   |             | ETR core outlet temperature        |
| ETR DEL-T                         | 189         | A                 |           |                   |             | ETR core $\Delta T$                |
| ETR DEL-P                         | 190         | A                 |           |                   |             | ETR core $\Delta P$                |
| ETR P-INL                         | 191         | A                 |           |                   |             | ETR core inlet pressure            |
| FISS BRK                          | 192         | A                 |           |                   |             | Fission break                      |
| SPNDs                             | 208-241     | A                 |           |                   |             | ETR self-powered neutron detectors |
| On-Line Cover Gas Sampling System |             |                   |           |                   |             |                                    |
| PE-378                            | 197         | A                 |           |                   |             | OLCS loop inlet pressure           |
| PE-379                            | 198         | A                 |           |                   |             | OLCS loop outlet pressure          |
| FM-9                              | 199         | A                 |           |                   |             | OLCS loop inlet flow rate          |
| FM-30                             | 200         | A                 |           |                   |             | OLCS sample flow rate              |
| On-Line Sodium Sampling System    |             |                   |           |                   |             |                                    |
| FE-084                            | 201         | A                 |           |                   |             | OLSS sodium flow rate              |
| PE-808                            | 202         | A                 |           |                   |             | OLSS suction pressure at pump      |

Conversion factors

- C Transmitter range 0 to 2000°F
- D Transmitter range 250°F to 1250°F range
- F Transmitter range 250°F to 1800°F
- G Transmitter range 500°F to 2500°F
- H Transmitter range 200°F to 2000°F

Appendix B

Chronology of P4 Preparation and Operation

## Chronology of P4 Preparation and Operation

| Date<br>(1981)         | Event(s)                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                              |
|------------------------|---------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------|
| February 14            | P4 test train shipped to EG&G Idaho, Inc. from ANL-Illinois.                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                          |
| April 1                | P4 loop filled with sodium.                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                           |
| April 20 to<br>May 2   | <p>B-3 series of tests with the P4 loop in the FS&amp;R.</p> <p>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, and to demonstrate the flow reduction for the decay-heat test. The measured ratio of fuel-bundle flow to loop flow was within 3% of pretest predictions, based on inlet flowmeter FE 1-3 and combined-flow flowmeter FE 3-2 measurements. 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). Tests of a digital-function generator to control Variac power to the pump produced the desired 200-s flow reduction to 10% of initial flow for the decay heat test.</p>                                                                                                                                                                                                                                                                                                                                          |
| May 20                 | P4 loop transferred to the ETR.                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                                       |
| June 24 to<br>August 3 | <p>B-1 series of tests with the P4 loop in the ETR.</p> <p>The objective of the B-1 testing was to functionally test, demonstrate and/or characterize, prior to ETR startup, all subsystems of the SLSF system not previously verified to the extent possible at zero reactor power. Testing included secondary cooling system startup, balancing of the helium system loop and bypass leg impedance, a repeat of the pump performance and loop hydraulic characteristic mapping, and demonstration of the functioning of the plant protection system and experiment safety assurance system.</p> <p>B-1 also included pre-operational testing of the OLCS and OLSS. OLCS operations were demonstrated over a range of sampling rates and dilution ratios. The time required to stabilize flow at 0.07 MPa (10.5 psia) was determined. Leak check procedures to verify OLCS containment integrity following subatmospheric operation were demonstrated. OLSS operation included cold trapping to maintain a sodium oxide level below 2 ppm and tests to determine the pump voltage - OLSS sodium flow characteristics in the pressure range 0.07 MPa (10.5 psia) to 0.17 MPa (24.8 psia), the flow rate [<math>&gt;3.2</math> kg/m (1.0 gpm)] at which cavitation began to reduce</p> |



sodium flow at 0.07 MPa cover gas pressure, and the adequacy of the OLSS heat exchanger to maintain the OLSS sodium at a maximum of 110 K (200°F) below the loop sodium temperature. In addition, the OLSS testing identified an OLCS gas pressure of 0.16 MPa (22.8 psia) as necessary to maintain atmospheric pressure at the OLSS pump inlet during operation at high flow in the OLSS.

August 8 to  
August 15

#### First cycle of P4 power operation

Operation during the first 10-day cycle was limited to a maximum ETR power of 40 MW. This allowed loop checkout and buildup of fission products for failure detection without premature canister failure. The first cycle also included a benchmark test at 40 MW to establish a reference pre-failure data base, power, flow, and loop plenum pressure perturbations to determine system response to nonadverse changes in operating conditions, and a low-power practice transient to verify the operation of the ETR PTC. The practice transient did not yield the correct final/initial power ratio and was repeated in the second cycle of operation.

August 19 to  
August 21

#### Second cycle of P4 power operation.

The second low-power practice transient was performed. Although the ratio of final power to initial power was improved over that of the first transient it was still lower than planned. This was compensated for by starting the planned power transient on the high side of the reference initial power. The power transient was initiated after the loop plenum-gas pressure was reduced from 0.16 MPa (22.8 psia) to 0.07 MPa (10.0 psia). Reactor power was on target during the planned power transient, but an ETR scram due to low fuel bundle inlet flow (PPS PF-H) occurred about two minutes after the start of the power transient. The 10 to 30 gram target for molten fuel release was exceeded and partial flow channel blockage reduced the fuel bundle flow to 86% of the initial level. A scram recovery was not attempted because the test section flowrate was below the administrative limit of 90% for continued steady state operation.

September 3  
to September 5

#### Third cycle of P4 power operation.

A delayed neutron mechanisms test was performed at a fuel bundle power of 18 kW (~3 MW ETR power) to measure the dependence of the DN signal on the temperature of the sodium coolant in the range 695 K (792°F) to 589 K (600°F). DN data were recorded during the DN mechanisms test and also during the ETR power increase to 40 MW. The reactor power was increased to 40 MW in 5 MW steps, each step followed by a four-hr hold to ensure that equilibrium conditions were reached. The 40 MW benchmark test was repeated, first at

the 86% fuel bundle flow produced by the pump operating at its benchmark voltage and then at the 100% fuel bundle flow of the pre-failure benchmark test. This cycle was completed with a 107 K reduction in the fuel bundle inlet temperature to study the influence of the sodium coolant temperature on the DN signal at an ETR power of 40 MW.

October 1 to  
October 2

Fourth cycle of P4 power operation.

Following the completion of additional safety analyses that justified a lower (68%) administrative limit on steady-state fuel bundle sodium flow rate, power operation resumed. ETR power was increased in 10 and 20 MW steps in a step-and-hold manner toward full power. A blockage reconfiguration occurred while the ETR power was being trimmed at ~100 MW. The fuel bundle coolant flow rate dropped from 86% to 60% of the initial value; an ETR scram resulted from a low fuel bundle inlet flow (PPS PF-H) signal.

October 27 to  
October 29

Fifth cycle of P4 power operation.

ETR was brought up to less than 2% of full power solely to do the final delayed neutron mechanisms test. This test was performed to study the coolant-temperature dependence of the DN signal from the larger partial flow channel blockage present after the blockage reconfiguration. It was conducted over a wider temperature range, 695 K (792°F) to 458 K (365°F), than the earlier DN mechanisms test.

November 4 to  
November 5

Controlled freeze of sodium in the P4 loop was completed in 10 hrs.

November 20

The P4 loop was removed from the ETR.

## Appendix C

### Computer Code Input

- THTB
- SABRE-1

Input (in Metric Units) for THTM Thermal Analysis  
of Fuel on the P4 Hex-Can

```

27. 2-D SLAB GEOMETRY
28. P4 MOLTEN FUEL ON HEX CAN
29. CONV. BOUND. ON GLOB BACKSIDE
30. MAY. 6, 1985
31. -1 0 0 1 0 0 90 1 10 0 00
32. 100.00 500.0 0.0 0.0 0.0 0.0
33. 0 0 1 1 0 2 3 0 1 1 0 0 0
34. 1
35. 2
36. 10 11 2 2 8
37. 0.00000 0.07621 0.15240 0.22860 0.30480 0.33800
38. 0.37120 0.40440 0.43760 0.47080
39. 0.00000 0.04000 0.08000 0.12000 0.16000 0.20000
40. 0.24000 0.28000 0.32000 0.36000 0.40000
41. 0.00000 1.00000
42. 1.0 5.0 1.0 11.0 1.0 2.0 7003. 980.0
43. 5.0 10.0 1.0 11.0 1.0 2.0 7001. 3505.
44. 1.0 2.0 1.0 11.0 1.0 2.0 1.0 0.0
45. 1.0 10.0 1.0 2.0 1.0 2.0 2.0 0.0
46. 9.0 10.0 1.0 6.0 1.0 2.0 3.0 8001.
47. 9.0 10.0 6.0 11.0 1.0 2.0 3.0 8002.
48. 1.0 5.0 10.0 11.0 1.0 2.0 4.0 8003.
49. 5.0 10.0 10.0 11.0 1.0 2.0 4.0 8001.
50. 1.0 10.0 1.0 11.0 1.0 2.0 5.0 0.0
51. 1.0 10.0 1.0 11.0 1.0 2.0 6.0 0.0
52. 3001.
53. 1. 2.0 1.0 11.0 1.0 2.0
54. 11000. 1. 6.00
55. 4001.
56. 1.0 5.0 1.0 11.0 1.0 2.0 0.898E+00
57. 5.0 10.0 1.0 11.0 1.0 2.0 482.000E+00
58. 7001 FUEL GLOB 10.00 281.33044.3 9.0
59. 672.0 866.51072.01472.02272.02672.03044.33044.36922.0
60. 0.04500.03700.02990.02410.02200.02700.03503.03503.0350
61. 0.30960.31380.33470.34730.41840.62760.64020.50210.5021
62. 7003 STAINLESS HEX CAN 7.96 283.61699.8 12.0
63. 538.7 699.8 810.9 922.01033.21144.31255.41366.51533.21699.81699.83033.2
64. 0.18500.20250.21980.23730.25480.27220.28970.30720.33320.35950.35950.3595
65. 0.56900.58160.59830.61500.62760.64430.66110.67780.69870.72380.81170.8117
66. 8001 NA FACE 2.0 3.0
67. 0.00 1.00 100.0
68. 1035.0 1234. 1234.
69. 10.0 10.0 10.0
70. 8002 PIN FACE 1.0 5.0
71. 0.00 2.00 4.00 6.00 100.
72. 1035.0 1755. 2550. 2700. 2700.
73. 8003 HEX BOUND. 1.0 3.0
74. 0.00 60.0 100.
75. 980.0 1466. 1466.
76. 11001 SURF FLUX 2.0
77. 0.0 9999.
78. 1.0 1.0
79. 12001 HEAT MULT 2.0
80. 0.0 9999.
81. 1.0 1.0
82. 0 0 .01 1000. .0011 100. LH 100
83. 0 0 .01 7000. .0011 200. LH 200

```

Input (in Metric Units) for THTM Thermal Analysis  
of Fuel on a Large-Plant Hex-Can

```

27. 2-D SLAB GEOMETRY
28. P4 MOLTEN FUEL ON HEX CAN
29. CONV. BOUND. ON GLOB BACKSIDE
30. NA CONV. OUTSIDE HEXCAN
31. -1 0 0 1 0 0 90 1 10 0 00
32. 100.00 500.0 0.0 0.0 0.0 0.0
33. 0 0 1 1 0 2 4 0 1 1 0 0 0
34. 1
35. 2
36. 10 11 2 2 8
37. 0.00000 0.07621 0.15240 0.22860 0.30480 0.33800
38. 0.37120 0.40440 0.43760 0.47080
39. 0.00000 0.04000 0.08000 0.12000 0.16000 0.20000
40. 0.24000 0.28000 0.32000 0.36000 0.40000
41. 0.00000 1.00000
42. 1.0 5.0 1.0 11.0 1.0 2.0 7003. 980.0
43. 5.0 10.0 1.0 11.0 1.0 2.0 7001. 3505.
44. 1.0 2.0 1.0 11.0 1.0 2.0 1.0 8004.
45. 1.0 10.0 1.0 2.0 1.0 2.0 2.0 0.0
46. 9.0 10.0 1.0 6.0 1.0 2.0 3.0 8001.
47. 9.0 10.0 6.0 11.0 1.0 2.0 3.0 8002.
48. 1.0 5.0 10.0 11.0 1.0 2.0 4.0 8003.
49. 5.0 10.0 10.0 11.0 1.0 2.0 4.0 8001.
50. 1.0 10.0 1.0 11.0 1.0 2.0 5.0 0.0
51. 1.0 10.0 1.0 11.0 1.0 2.0 6.0 0.0
52. 3001.
53. 1. 2.0 1.0 11.0 1.0 2.0
54. 11000. 1. 6.00
55. 4001.
56. 1.0 5.0 1.0 11.0 1.0 2.0 0.898E+00
57. 5.0 10.0 1.0 11.0 1.0 2.0 240.000E+00
58. 7001 FUEL GLOB 10.00 281.33044.3 9.0
59. 672.0 866.51072.01472.02272.02672.03044.33044.36922.0
60. 0.04500.03700.02990.02410.02200.02700.03503.03503.0350
61. 0.30960.31380.33470.34730.41840.62760.64020.50210.5021
62. 7003 STAINLESS HEX CAN 7.96 283.61699.8 12.0
63. 588.7 699.8 810.9 922.01033.21144.31255.41366.51533.21699.81699.83033.2
64. 0.18500.20250.21980.23730.25480.27220.28970.30720.33320.35950.35950.3595
65. 0.56900.58160.59830.61500.62760.64430.66110.67780.69870.72380.81170.8117
66. 8001 NA FACE 2.0 3.0
67. 0.00 1.00 100.0
68. 980.0 950. 950.
69. 10.0 10.0 10.0
70. 8002 PIN FACE 1.0 5.0
71. 0.00 2.00 4.00 6.00 100.
72. 1035.0 1755. 2550. 2700. 2700.
73. 8003 HEX BOUND. 1.0 3.0
74. 0.00 60.0 100.
75. 980.0 1466. 1466.
76. 8004 NA OUTS 2.0 3.0
77. 0.00 1.00 100.0
78. 950.0 950. 950.
79. 10.0 10.0 10.0
80. 11001 SURF FLUX 2.0
81. 0.0 9999.
82. 1.0 1.0
83. 12001 HEAT MULT 2.0
84. 0.0 9999.
85. 1.0 1.0
86. 0 0 .01 1000. .0011 100. LH 100
87. 0 0 .01 7000. .0011 200. LH 200

```

Input for SABRE-1 Analysis of the Temperature Field  
in the Wake of the P4 Partial Flow Blockage

```
ITER 100 25 25 TOL 1E-10 URF 0.7 0.10
ISH 50
HUN 5
MESH 17 09 72 MPRINT 14 4 10 MPRESS 10 5 72
PRINT 0
DIA .005842 PITCH 0.0072644 EDGAP 0.0007112 IJPIN 0
FLOW 4.600 TIN 741.7
TPRINT .02 .02
RESTART
GUESS
CHAN TYPE 1
RPOWER 33084.3
NA
BLCON 3.
CHANNELS 00 00 00 00 36 31 21 31 21 31 21 31 32 00 00 00
189571 STORES REQUIRED
TO SAVE STORAGE, DIMENSION OF ARRAYS CAN BE DECREASED BY UP TO 10429
00 00 00 36 26 10 10 10 10 10 10 10 22 32 00 00
00 00 36 26 10 10 10 10 10 10 10 10 10 22 32 00
00 36 26 10 10 10 10 10 10 10 10 10 10 22 32
00 35 25 10 10 10 10 10 10 10 10 10 10 23 33
00 00 35 25 10 10 10 10 10 10 10 10 10 23 33 00
00 00 00 35 25 10 10 10 10 10 10 10 23 33 00 00
00 00 00 00 35 34 24 34 24 34 24 34 33 00 00 00
BLOCKAGE 31 36 00 00 00 00 0 0 0 0 0 0 0 0 00 00 00
00 00 00 0 0 0 0 0 1 0 1 0 0 0 0 00 00
00 00 0 0 0 0 0 0 0 1 1 1 1 0 0 0 00
00 0 0 0 1 1 0 1 1 1 1 1 1 0 0 0 0
00 0 0 0 1 1 1 1 1 1 1 1 1 1 0 0 0
00 00 0 0 0 0 1 1 1 1 1 1 1 1 0 0 00
00 00 00 0 0 0 1 1 1 1 1 1 1 0 0 00 00
00 00 00 00 0 0 0 0 1 1 1 1 1 0 00 00 00
ZMESH 0. .03327 .07137 .10947 .14757 .18567 .223770 .26187 .29997
.33807 .37617 .41427 .45237 .49047 .50952 .52857 .54762 .56667
.57937 .59207 .60477 .61747 .63017 .64287 .65557 .66827 .68097
.683 .685 .687 .689 .691
.69367 .696 .698 .700 .702 .704 .70637 .709 .711 .713 .715 .717
.71907 .73177 .74447 .75717 .77622 .79527 .83337
.87147 .90957 .94767 .98577 1.02387 1.10007 1.17627 1.25247
1.32867 1.40487 1.48107 1.55727 1.63347 1.70967 1.78587 1.86207
1.93827 2.01447 2.09067 2.16687 2.24307 2.31927
AXPOWER 0. .04 .1 .2 .3 1.235 1.7675 2.30 2.67
3.035 3.293 3.55 3.703 3.855 3.899 3.943 3.986 4.03
4.044 4.058 4.071 4.085 4.099 4.1125 4.12 4.12 4.12
4.12 4.12 4.12 4.12 4.12 4.12 4.12 4.12 4.12 4.12 4.12
4.12 4.12
4.12 4.12 4.12 4.1111 4.102 4.093 4.084 4.075 3.868
3.66 3.435 3.21 2.73 2.25 1.03 .3 .23
.18 .14 .10 .08 .06 .04 .02 .01
0. 0. 0. 0. 0. 0. 0.
POWER 00 00 00 00 5955.2 5955.2 11469.2 5514.05
11028.1 5514.05 11469.2 5955.2 5955.2 00 00 00
00 00 00 5514.05 11469.2 16983.3 16983.3 17093.6
17093.6 17093.6 16983.3 16983.3 11469.2 5514.05 00 00
00 00 5514.05 11028.1 16542.2 16542.2 16542.2 17093.6
17093.6 17093.6 16542.2 16542.2 16542.2 11028.1 5514.05 00
00 5955.2 11469.2 16983.3 16542.2 16542.2 16542.2 16542.2
16542.2 16542.2 16542.2 16542.2 16983.3 11469.2 5955.2
00 5955.2 11469.2 16983.3 16542.2 16542.2 16762.7 16762.7
16983.3 16762.7 17314.1 17093.6 17093.6 16983.3 11469.2 5955.2
00 00 5514.05 11028.1 16542.2 16542.2 16762.7 16762.7
16983.3 16762.7 17314.1 17093.6 17093.6 11028.1 5514.05 00
00 00 00 5514.05 11469.2 16983.3 23776.2 23159.1
23159.1 23159.1 23776.2 23776.2 11469.2 5514.05 00 00
00 00 00 00 5955.2 5955.2 11469.2 5514.05
22056.1 11028.0 22940.2 11469. 5955.2 00 00 00
CHANDATA 310 1.E-06 2.E-02 1.422E-03 4.9175E-06
END
```

Input for SABRE-1 Analysis of the Temperature Field in the  
Wake of a P4 Partial Flow Blockage with Uniform Power Density  
(Large-Plant Case)

```
ITER 100 25 25 TOL 1E-10 URF 0.7 0.10
ISW 50
NUN 5
NESH 17 09 72 MPRINT 14 4 10 MPRESS 10 5 72
PRINT 1
DIA .005842 PITCH 0.0072644 EDGAP 0.0007112 IJPIN 0
FLOW 4.600 TIN 741.7
TFRINT .02 .02
RESTART
GUESS
CHANTYPE 1
RPOWER 33084.3
NA
BLCON 3.
CHANNELS 00 00 00 00 36 31 21 31 21 31 21 31 32 00 00 00
169571 STORES REQUIRED
TO SAVE STORAGE, DIMENSION OF ARRAYS CAN BE DECREASED BY UP TO 10429
00 00 00 36 26 10 10 10 10 10 10 10 22 32 00 00
00 00 36 26 10 10 10 10 10 10 10 10 22 32 00
00 36 26 10 10 10 10 10 10 10 10 10 10 22 32
00 35 25 10 10 10 10 10 10 10 10 10 10 23 33
00 00 35 25 10 10 10 10 10 10 10 10 23 33 00
00 00 00 35 25 10 10 10 10 10 10 10 23 33 00 00
00 00 00 00 35 34 24 34 24 34 24 34 33 00 00 00
BLOCKAGE 31 36 00 00 00 00 0 0 0 0 0 0 0 0 00 00 00
00 00 00 0 0 0 0 1 0 1 0 0 0 0 0 00 00
00 00 0 0 0 0 0 0 0 1 1 1 1 1 0 0 00
00 0 0 0 1 1 0 1 1 1 1 1 1 0 0 0 0
00 0 0 0 1 1 1 1 1 1 1 1 1 1 0 0 0
00 00 0 0 0 0 1 1 1 1 1 1 1 1 0 0 00
00 00 00 0 0 0 1 1 1 1 1 1 1 0 0 00 00
00 00 00 00 0 0 0 0 1 1 1 1 1 0 00 00 00
ZMESH 0. .03327 .07137 .10947 .14757 .18567 .223770 .26187 .29997
.33807 .37617 .41427 .45237 .49047 .50952 .52857 .54762 .56667
.57937 .59207 .60477 .61747 .63017 .64287 .65557 .66827 .68097
.633 .685 .687 .689 .691
.69367 .696 .698 .700 .702 .704 .70637 .709 .711 .713 .715 .717
.71907 .73177 .74447 .75717 .77622 .79527 .83337
.87147 .90957 .94767 .98577 1.02387 1.10007 1.17627 1.25247
1.32867 1.40487 1.48107 1.55727 1.63347 1.70967 1.78537 1.86207
1.93827 2.01447 2.09067 2.16687 2.24307 2.31927
AXPOWER 0. .04 .1 .2 .3 1.235 1.7675 2.30 2.67
3.035 3.293 3.55 3.703 3.855 3.899 3.943 3.986 4.03
4.044 4.058 4.071 4.085 4.099 4.1125 4.12 4.12 4.12
4.12 4.12 4.12 4.12 4.12 4.12 4.12 4.12 4.12 4.12 4.12 4.12
4.12 4.12
4.12 4.12 4.12 4.1111 4.102 4.093 4.084 4.075 3.868
3.66 3.435 3.21 2.73 2.25 1.03 .3 .23
.18 .14 .10 .08 .06 .04 .02 .01
0. 0. 0. 0. 0. 0.
POWER 00 00 00 5955.2 5955.2 11469.2 5514.05
11028.1 5514.05 11469.2 5955.2 5955.2 00 00 00
00 00 00 5514.05 11469.2 16983.3 16983.3 17093.6
17093.6 17093.6 16983.3 16983.3 11469.2 5514.05 00 00
00 00 5514.05 11028.1 16542.2 16542.2 16542.2 17093.6
17093.6 17093.6 16542.2 16542.2 16542.2 11028.1 5514.05 00
00 5955.2 11469.2 16983.3 16542.2 16542.2 16542.2 16542.2
16542.2 16542.2 16542.2 16542.2 16983.3 11469.2 5955.2
00 5955.2 11469.2 16983.3 16542.2 16542.2 16762.7 16762.7
16983.3 16762.7 17314.1 17093.6 17093.6 16983.3 11469.2 5955.2
00 00 5514.05 11028.1 16542.2 16542.2 16762.7 16762.7
16983.3 16762.7 17314.1 17093.6 17093.6 11028.1 5514.05 00
00 00 00 5514.05 11469.2 16983.3 16983.3 16542.2
16542.2 16542.2 16983.3 16983.3 11469.2 5514.05 00 00
00 00 00 5955.2 5955.2 11469.2 5514.05
11028.1 5514.05 11469.2 5955.2 5955.2 00 00 00
CHANDATA 310 1.E-06 2.E-02 1.422E-03 4.9175E-06
END
```