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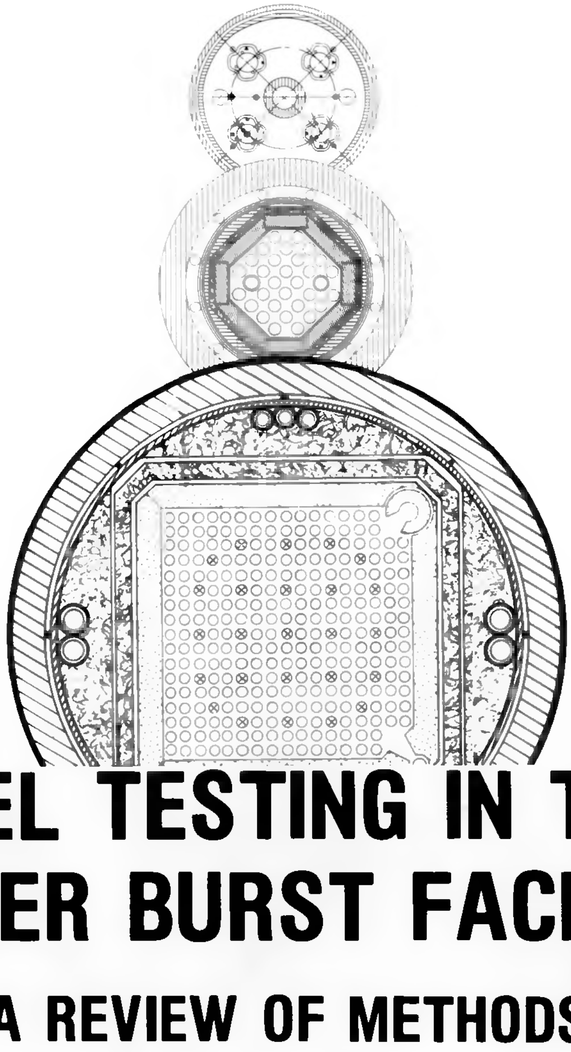

FUEL TESTING IN THE POWER BURST FACILITY

A REVIEW OF METHODS

MASTER

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ABSTRACT

This document briefly reviews the operating parameters of the Power Burst Facility at the Idaho National Engineering Laboratory in Idaho, USA. Discussed are the capabilities of the reactor and the pressurized water reactor loop system which controls the fluid environment of experiments. The document principally addresses the testing methods employed during past programs and some of those planned for future tests as an aid for potential new users of the facility.

SUMMARY

This document presents a description of the Power Burst Facility (PBF) at the Idaho National Engineering Laboratory and briefly describes a number of experimental methods and designs of test trains used in past programs and planned for new programs to study the behavior of light water reactor fuel under conditions ranging from normal to severe accident. The PBF is a thermal reactor with a central 28-cm-diameter, 0.914-m-long through-hole flux-trap for testing purposes. Steady state and transient neutron flux capabilities are described.

The reactor has been used in studies of severe fuel damage, operational transients, pellet-cladding interaction, power-cooling-mismatch (burnout and vapor blanketing), loss-of-coolant accident, and extremely fast transients representing reactivity initiated accidents. Test trains developed for these programs range from single-, four-, and nine-rod assemblies, up to the current 32-rod bundles being used for severe fuel damage tests. Test trains and typical instruments are described, as are new test trains and instruments capable of operating near the melting point of UO_2 (3130 K).

CONTENTS

FUEL TESTING IN THE POWER BURST FACILITY A REVIEW OF METHODS

This report provides a brief description of the Power Burst Facility (PBF), located at the Idaho National Engineering Laboratory (INEL) in the USA, and discussions of the types of nuclear fuel testing which can be performed.

THE POWER BURST FACILITY

The Power Burst Facility (Figure 1) is a versatile flux-trap reactor facility designed to study the behavior of nuclear fuel rods under normal, off-normal, and accident conditions in light water reactors. The PBF has already been used for a wide-ranging and very productive program of safety-related studies of reactor fuel behavior, in which it has subjected fuel rod specimens to such off-normal and accident conditions as reactivity initiated accidents, loss-of-coolant accidents, anticipated transients with and without scram, pellet-cladding interaction, burnout, severe fuel damage accidents, and others.

The PBF can be operated in a number of modes, as illustrated in Figure 2, including natural burst (in which power levels up to 270 GW are attained); shaped power bursts (achieving very high powers but with user-programmed power shape); steady state at any level up to 28 MW; and a wide variety of arbitrary power shapes, such as ramps within or near 28 MW. Table 1 provides a summary of the capabilities of this reactor.

The PBF consists of a driver core with a 0.91-m active length; a central flux trap region containing an in-pile tube (IPT) in which the test fuel is located; a pressurized water flow loop that permits control of the test fuel rod coolant environment conditions of flow rate, temperature, and pressure; an open tank reactor vessel; and a canal. Figure 3 presents a schematic drawing of the loop coolant system, and Figure 4 presents a schematic of the IPT. The open top reactor vessel provides access for installation and removal of test hardware. Figure 5 shows the in-pile tube hanging over the canal which is used for transfer and temporary storage of PBF reactor fuel and test fuel assemblies. The loop coolant system is able to provide temperature, pressure, and flow conditions typical of pressurized or boiling water reactor systems. The in-pile tube portion of the loop can be rapidly depressurized, through the use of high-speed valves, in a manner similar to a light water reactor during a postulated loss-of-coolant accident.

The PBF core (Figure 6) is approximately a right-circular cylinder, 1.3 m in diameter and 0.9 m high, enclosing a centrally located, vertical test space. The maximum diameter available for experiments in the PBF is 0.56 m, some of which must, of course, be used for containment of the test assembly. In its current configuration, test space in the PBF is considerably reduced so that the in-pile tube can be used. The IPT is designed to withstand pressures in excess of 200 MPa and affords the user great latitude in experiment design and conduct, since pressure disturbances (e.g., from steam explosions) are completely contained. The usable diameter inside the flow tube within the IPT is nominally 147 mm, a space which has been used successfully to contain and execute tests of up to 32 fuel rods surrounded by a high-temperature insulating shroud.

The linear power at which fuel rods can be driven in the PBF is, of course, strongly dependent on enrichment, bundle size, and the presence of moderating and poison materials. An example of the ability of this facility to power test fuel can be seen in the Severe Fuel Damage program. During those tests, the PBF is used to drive a poorly moderated, highly insulated bundle of 30 high-burnup pressurized water reactor rods (residual enrichment ∿2.5 at.%), together with two Ag-In-Cd control rods, at an axially averaged rod power of 5.3 kW/m. During the Power-Cooling-Mismatch program, the PBF was able to drive each rod in a nine-rod bundle (with graded enrichments averaging 35 at.%) at up to 58 kW/m steady state.

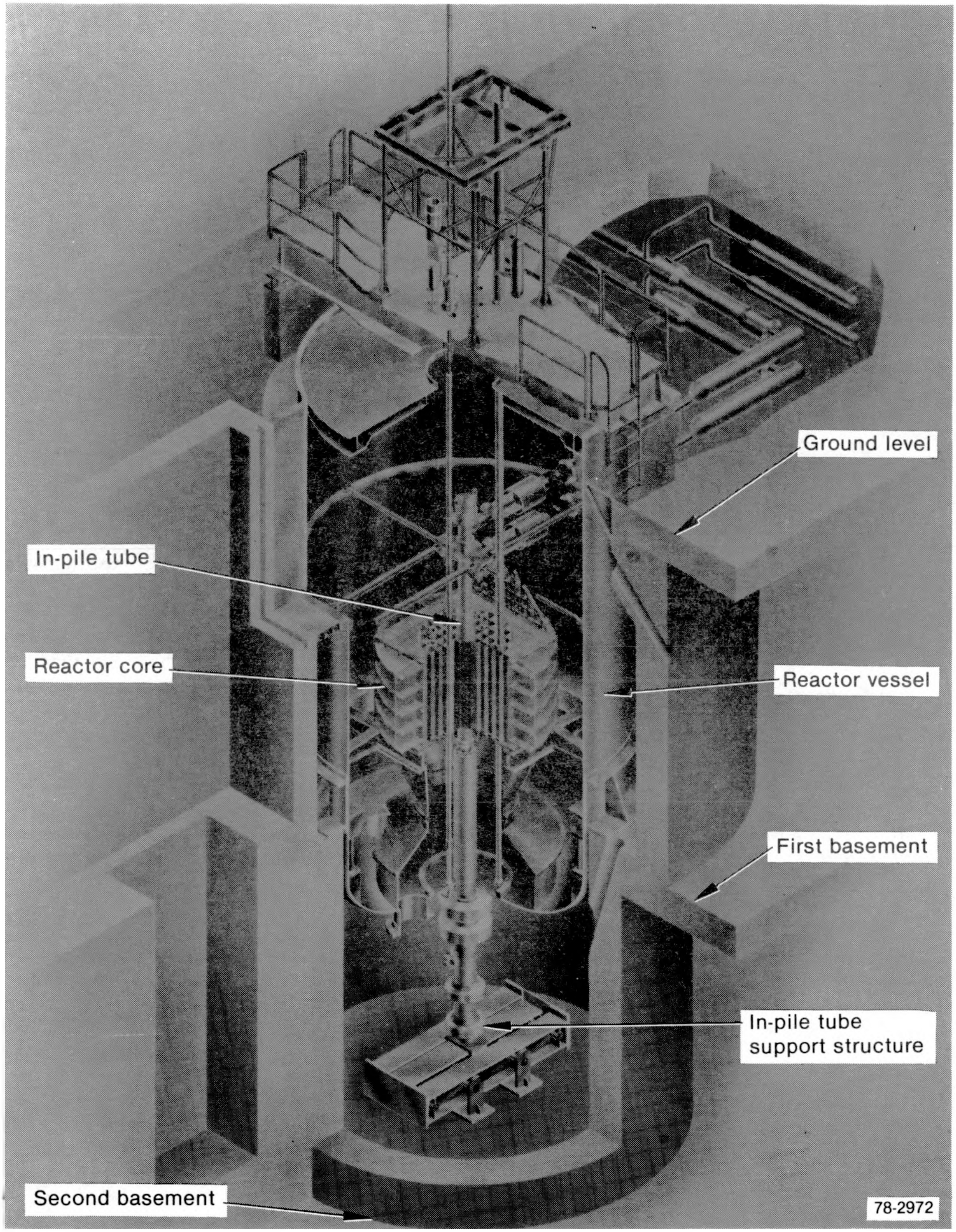

Figure 1. Power Burst Facility reactor cutaway view.

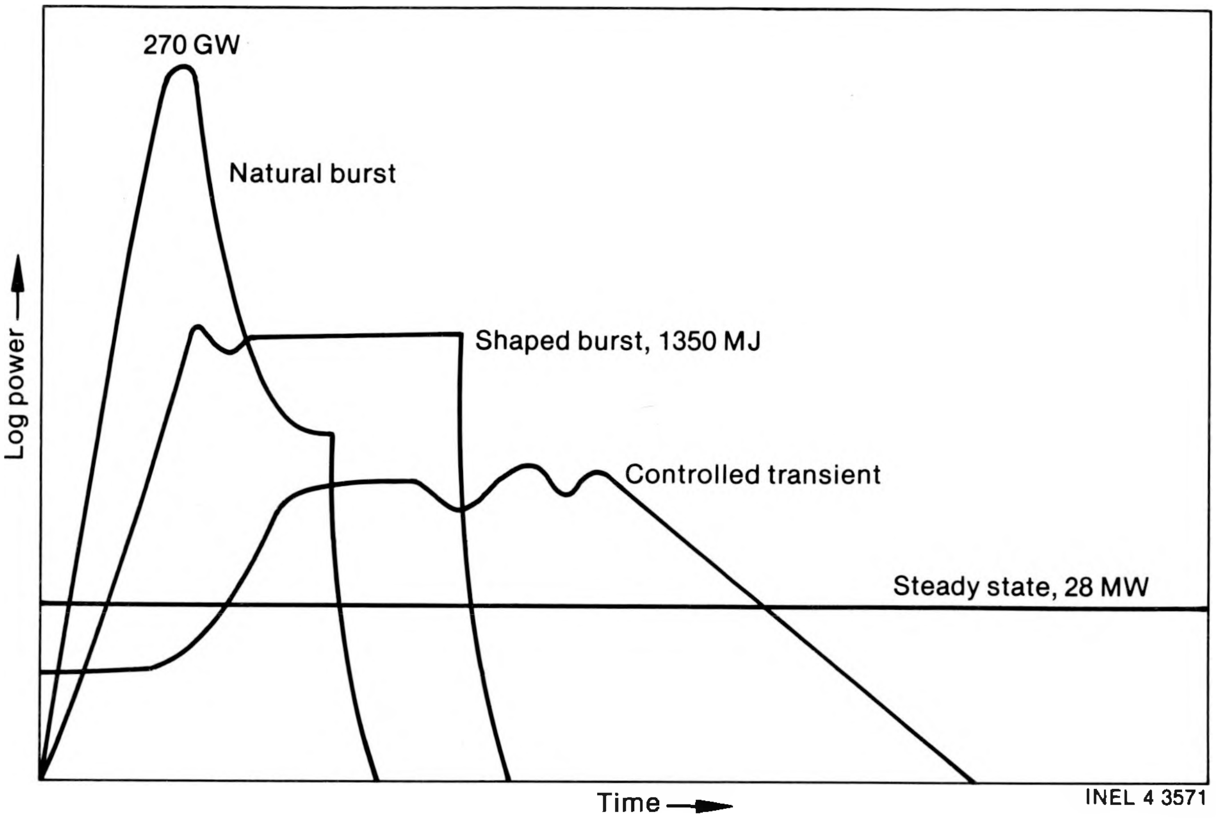

Figure 2. PBF modes of operation.

Table 1. Experimental capabilities of the Power Burst Facility

Parameter	PBF Capability
Maximum core steady state power	28 MW
Neutron flux in water-filled IPT at 28 MW (83% thermal <0.51 eV)	1.2×10^{14} n/cm^2·s
Maximum core power in shaped burst	1000 MW
Maximum core power for initiation of shaped burst	100 kW
Minimum asymptotic period for shaped burst	20 ms
Maximum core power for natural burst	270 GW
Neutron flux in water-filled IPT at 270 GW (83% thermal <0.51 eV)	8.3×10^{17} n/cm^2·s
Neutron flux in voided IPT at 270 GW	1.4×10^{18} n/cm^2·s
Gamma flux in voided IPT at 270 GW	7.0×10^{17} γ/cm^2·s
Minimum asymptotic period for natural burst	1.0 ms
Maximum power for initiation of natural burst or controlled transient	28 MW

Table 1. (continued)

Parameter	PBF Capability
Maximum energy release for natural or shaped burst	1350 MJ
Active core height	0.91 m
Inside diameter of IPT	15.5 cm
Inside length of IPT	4.47 m
Central test space inside diameter without IPT, gas jacket, and filler blocks	28 cm
Range of loop operating temperatures	340 to 616 K
Range of loop operating pressures	0.69 to 15.3 MPa
Range of loop coolant flow rates	1 to 50.3 L/s

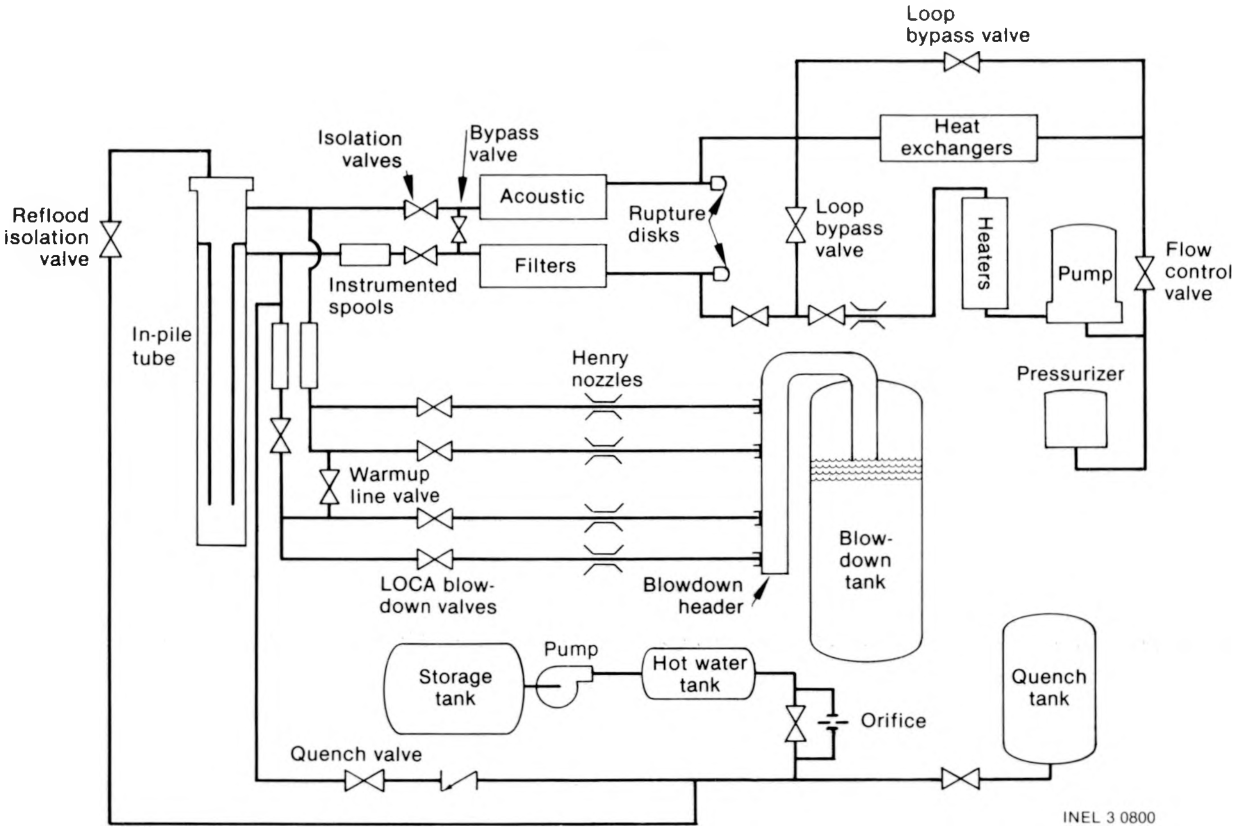

Figure 3. The PBF loop cooling system (top) as modified to include a blowdown system (center) and a reflood system (bottom).

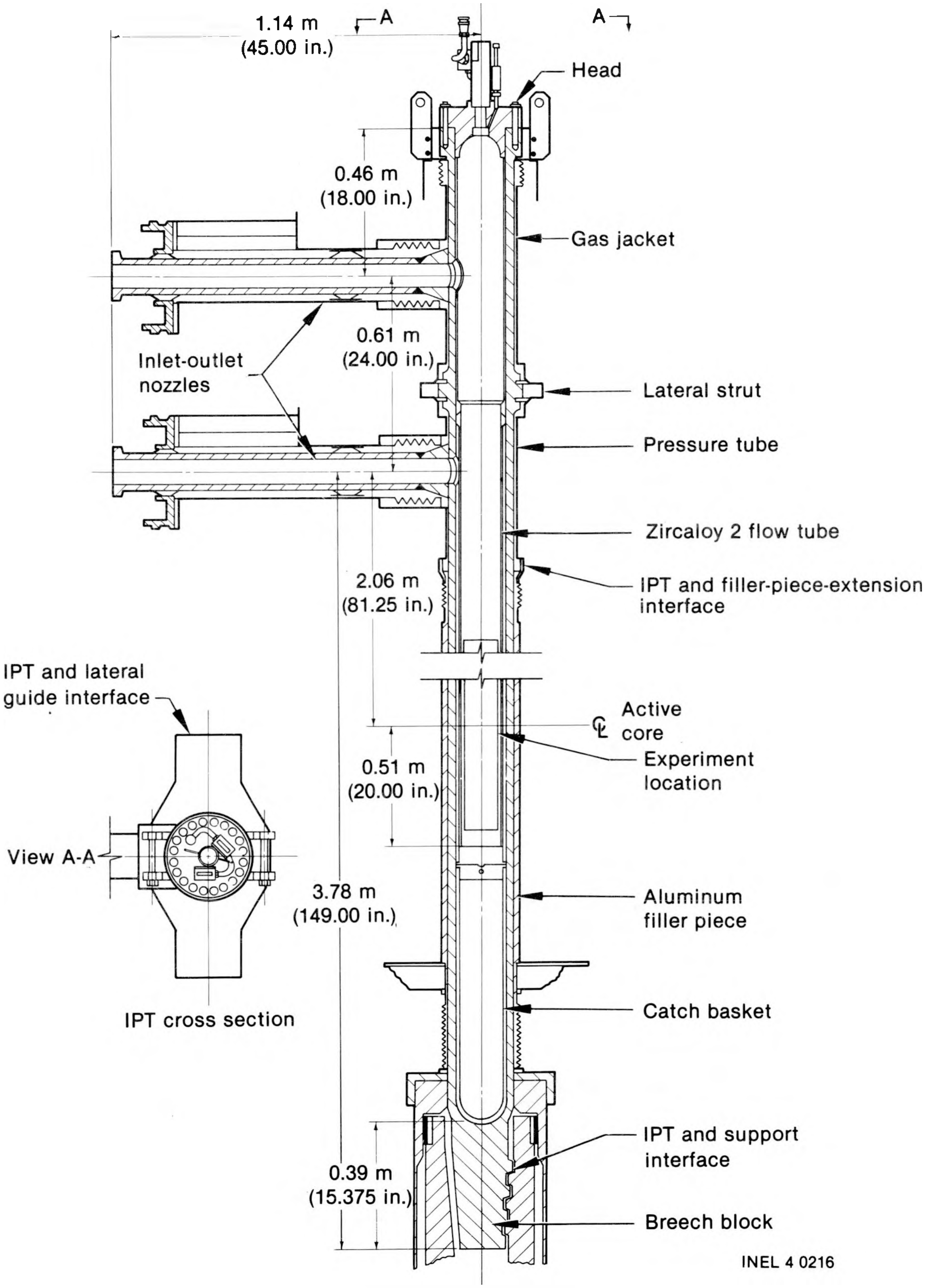

Figure 4. Axial cross section of in-pile tube.

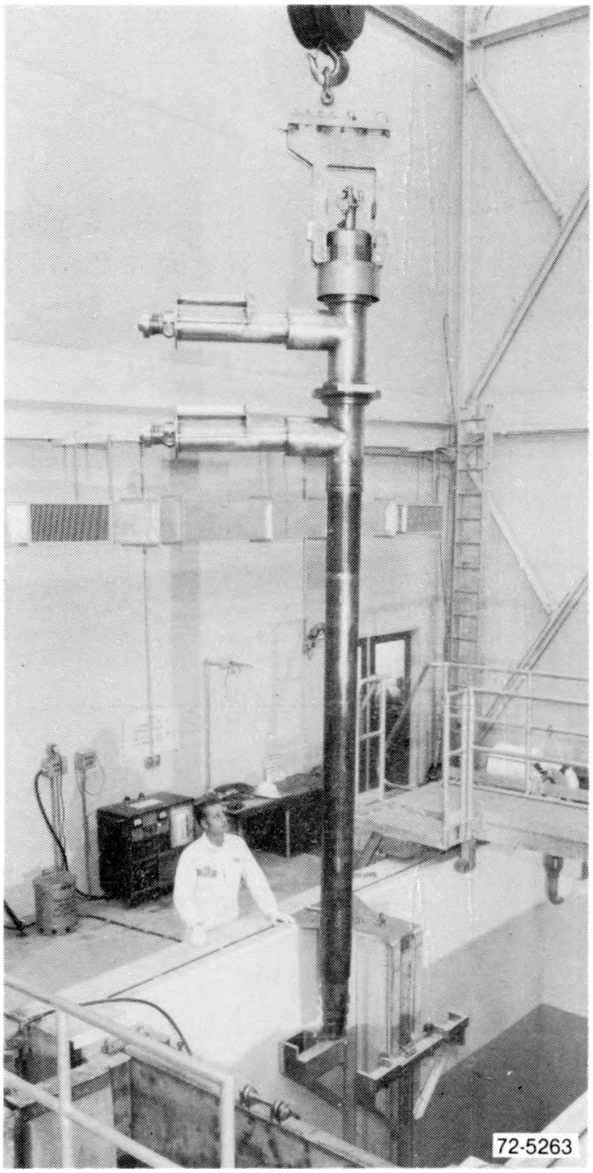

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Figure 5. PBF in-pile tube removed from the core and hanging above storage canal.

Other Features of the PBF

Data Acquisition and Reduction. A computer-controlled data acquisition and reduction system (the DARS), located in the PBF control room, is used to record test instrumentation measurements. This system allows acquisition of up to 256 narrowband (10 to 100 Hz) and 66 wideband (5 to 20 kHz) data channels. All channels are acquired in digital form and are recorded in pulse code modulation format on magnetic tape. All narrowband parameters are available on-line on a cathode ray tube (CRT) display.

The DARS produces hard-copy plots on-line as needed for preliminary data examination during a test, and will reduce most detailed test data within three days following a test. Further processing of the data is performed on a larger, dual CDC 7600 central computer installed at the INEL Computer Science Center in Idaho Falls. The INEL Scientific Data Management System and other programs are used to produce the data in final form and to perform detailed analyses required for test results reports. A graphics software package produces report-quality data and comparison plots. Detailed measurement uncertainty analyses are performed and the results are included in experimental data reports with the formally qualified data and other supporting information.

Test Train Assembly Facility. In support of the PBF tests, the Test Train Assembly Facility (TTAF) was designed to assemble both irradiated and unirradiated experimental test assemblies and to store fuel. The general work areas within the TTAF consist of a fuel rod assembly area, a test train assembly area, a storage canal, a machine shop, and a metal cleaning facility.

The TTAF provides the capability to characterize and assemble instrumented, unirradiated fuel rods; assemble instrument assemblies and test train structures; refurbish irradiated test train structures for reuse; load irradiated fuel elements into test train structures; and prepare completed test trains for installation in the PBF reactor. The Test Reactor Area (TRA) Hot Cell, which is also used in the assembly of test trains, is located adjacent to the TTAF.

The TRA Hot Cell is a multicell facility designed to characterize irradiated fuel rods by gamma scan and dimensional inspection prior to experimentation; to instrument irradiated fuel rods with pressure transducers, thermocouples, strain gauges, and other measurement devices; and to determine gas volumes and repressurize the fuel rods when required. The TTAF and Hot Cell facilities form a comprehensive capability to fabricate and assemble test trains.

Postirradiation Examination. The single-rod and bundle assemblies tested in the PBF are examined at hot cell facilities on the INEL site. Nondestructive examination techniques include photography, gross and spectral gamma scanning, and neutron radiography of individual fuel rods and bundles as large as 32 rods. Computer-generated tomographic reconstructions from the neutron radiography allow

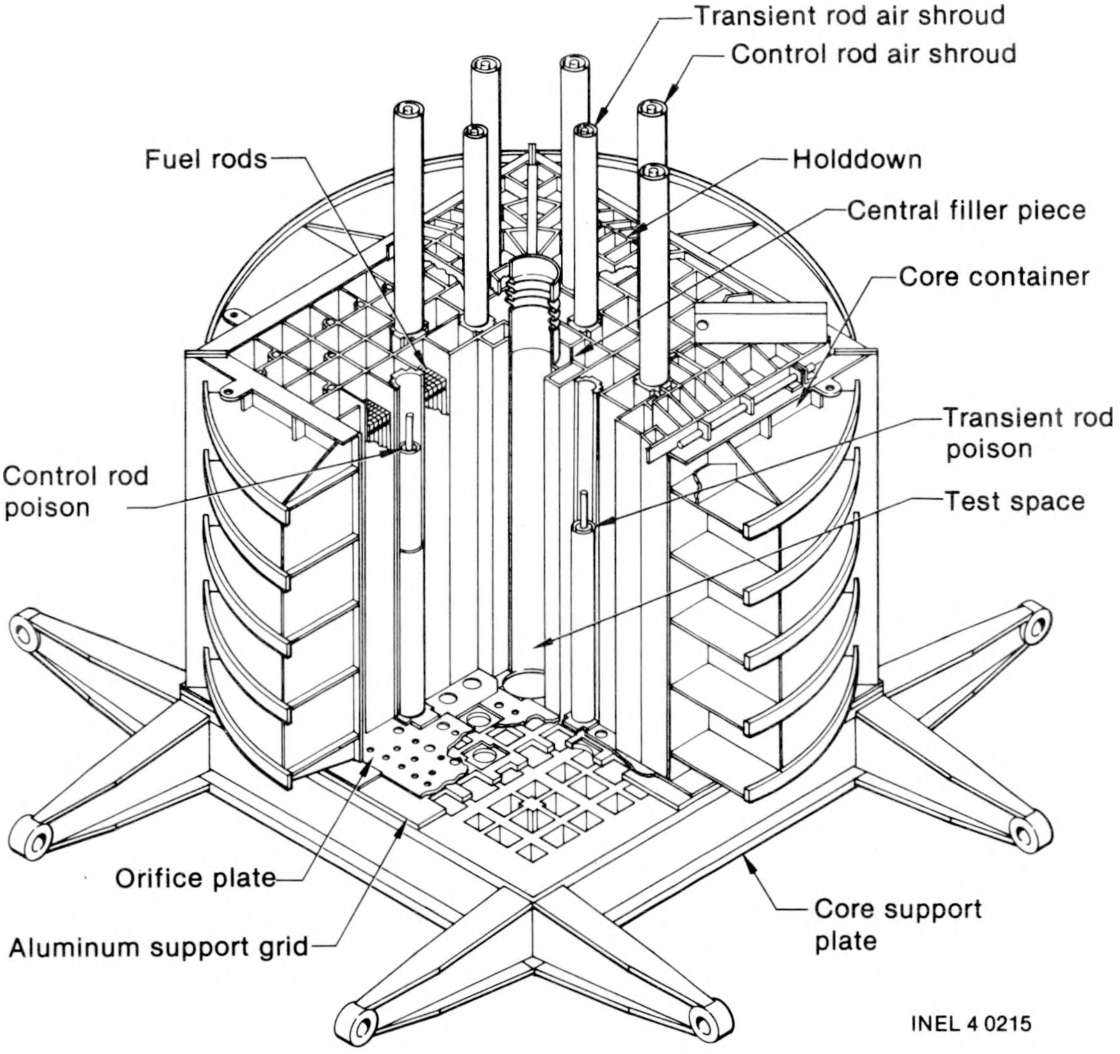

Figure 6. Cutaway view of PBF core.

nondestructive examination of bundle cross sections. Other examination techniques include diameter and length measurements, gas analysis, and particle size analysis. Epoxy encapsulation and sectioning of single-rod and bundle test assemblies can be performed, and metallographic examination of samples up to 13 cm (~5 in.) in diameter is possible. Many analysis techniques are available to determine the posttest physical and chemical characteristics of the experiments, including scanning electron microscopy-energy dispersive spectroscopy (SEM-EDS), molecular optical laser examination (MOLE), burnup analysis, Auger analysis, X-ray diffraction, emission spectroscopy, mass spectrometry, neutron activation analysis, beta-spectroscopy, and delayed neutron counting.

TEST METHODS OF THE PBF

Potential users of the PBF will find it useful to know of experimental methods used successfully during previous programs. The following sections present brief reviews of several past programs, with emphasis on test hardware and instrumentation. Further information about any of the tests can be obtained from the open literature or by contacting the Office of Reactor Operations and Programs in the U.S. Department of Energy, Idaho Operations Office.

SEVERE FUEL DAMAGE

In response to the Three Mile Island (TMI-2) accident, the USNRC initiated several research programs to improve the data base on severe accidents and to guide changes in regulatory policies. Principal among these programs is the Severe Fuel Damage (SFD) program in the PBF, in which data pertaining to fuel behavior under severely degraded conditions is sought.

The primary objectives of the PBF Severe Fuel Damage Test Program are to (a) characterize fuel rod damage resulting from severe cladding oxidation and melting, UO_2 dissolution and relocation, and fuel rod fragmentation; (b) measure the release rates, transport, and deposition of fission products; (c) measure the magnitude and timing of hydrogen generation; (d) measure the coolability characteristics of test bundles with various types and degrees of damage; and (e) determine the effects of irradiated fuel rods and control rods.

It should be noted that melting of UO_2, per se, is not an objective of this SFD program. In recognition that the SFD tests present severe experimental demands and that much is to be learned at intermediate temperatures, the USNRC Severe Fuel Damage Program in the PBF is constrained to temperatures below about 2400 K in its study of accident behavior; UO_2 dissolution by molten zirconium begins at about 2150 K.

Upon examination, it was clear that the SFD objectives could not be met with tests of single fuel rods, nor even with small (i.e., nine-rod) bundles, since the behavior of the fuel rods, the transport of fission products, and the configurations determining coolability are all sensitive to the presence of experiment walls which confine fuel motion and conduct heat away. For the SFD tests, therefore, a bundle of 32 rods on a 6 x 6 array, minus the corner rods, was selected as a practical compromise between the need for a large bundle size and the cost benefit of confining the experiment within the existing in-pile tube. Larger bundles are possible in the PBF, but were not chosen for this initial entry into the study of severe accidents to avoid the costs of developing an alternate in-pile tube. The SFD fuel bundle and test train cross section are illustrated in Figure 7, showing features of the shroud (wall) and coolant channels, all of which reside in the in-pile tube. Of principal interest in the shroud is the inner liner of zircaloy backed by high-temperature insulation. The inner zircaloy wall serves (a) as a water barrier during early preconditioning of the fuel (viz, several days at full power) and (b) as a sacrificial heat source during the dry heatup, to reduce wall losses when it is oxidized by steam.

The four tests planned for the SFD program each conduct a 32-rod bundle through a temperature history representing one of many accident sequences as discussed in The Reactor Safety Study (WASH-1400) and subsequent studies. Since the test train illustrated in Figure 7 is designed for temperatures up to 2400 K, the accident sequences being replicated by PBF heating are terminated at this point by controlled reactor shutdown.

Instrumentation for the experiments consists of a wide variety of state-of-the-art devices, including refractory metal-clad thermocouples for fuel and steam temperature measurements, pressure transducers mounted inside fuel rods, neutron detector arrays, and a complement of other instruments to monitor flow rates, reactor power, coolant pressure, hydrogen concentration, and others. Instruments of perhaps greatest importance, however, are those that constitute the sampling and monitoring system illustrated in Figures 8 and 9. This sophisticated system is now a permanent part of the PBF capability, vital to the acquisition of fission product behavior data, and available to PBF users for future programs.

As the effluent leaves the test train region, on-line aerosol and gamma spectral measurements are taken continuously. Just downstream of these on-line measurements are two banks of six grab samplers each; one bank employs particle filters to remove aerosol samples, while the other bank collects discrete samples of the entire effluent. All twelve samplers are operated remotely and can be activated at any time during the experiment.

Further downstream, the effluent passes through a condenser, gas and liquid flowmeters, and a gas and liquid separator. The noncondensable gases are monitored continuously with a hydrogen analyzer and a gamma spectrometer. The condensed liquid is monitored by a gamma spectrometer and then filtered to remove solid particles. Both the gas and liquid are then collected in a single tank, which is

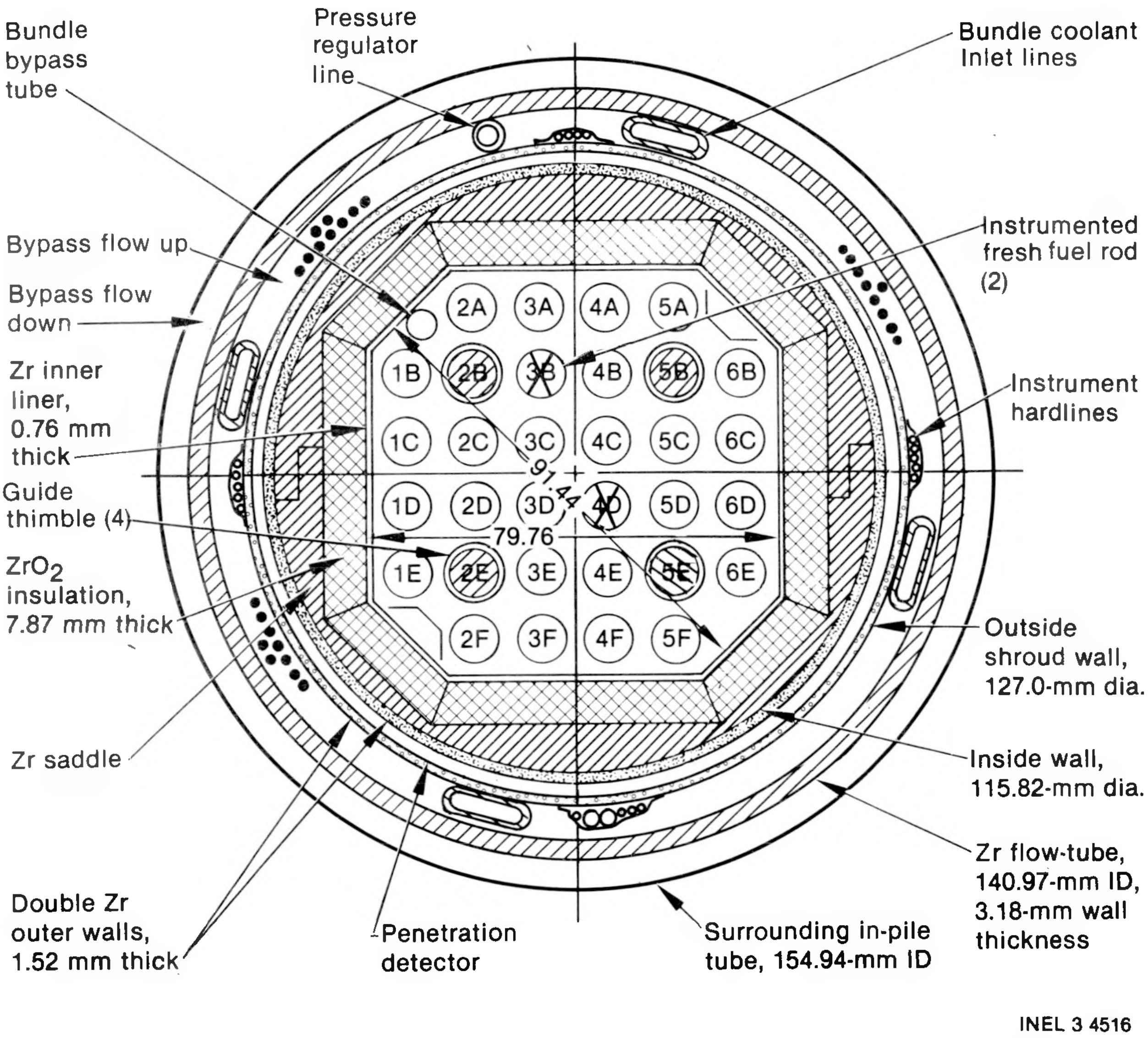

Figure 7. Severe Fuel Damage test train core region.

equipped with posttest gas and liquid circulation systems for on-site gamma spectral measurements and posttest grab sampling.

This combination of measurements allows quantification of fission product release rates and release fractions, aerosol release rates, deposition fractions in the simulated plenum and piping, and hydrogen release rates. The posttest analyses conducted on the grab samples define the fission product and aerosol composition, aerosol size characteristics, the relative timing of fission product isotope release and aerosol release, and the distribution of fission products throughout the system, which ultimately defines the mass balance of experiment constituents.

The Severe Fuel Damage Program in the PBF has (at this writing) executed two tests in which designs of the test train, instruments, and sampling and monitoring system have been proven.

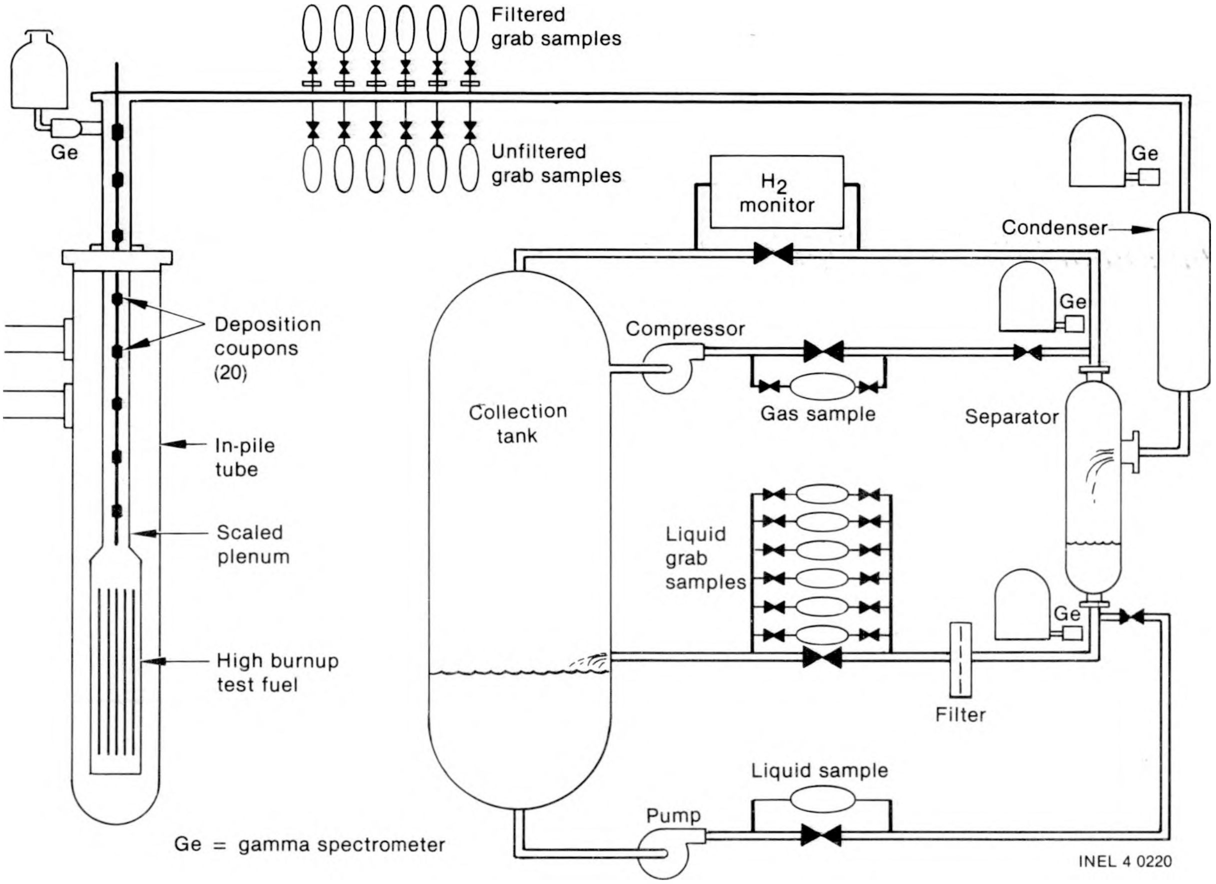

Figure 8. PBF sampling and monitoring system schematic.

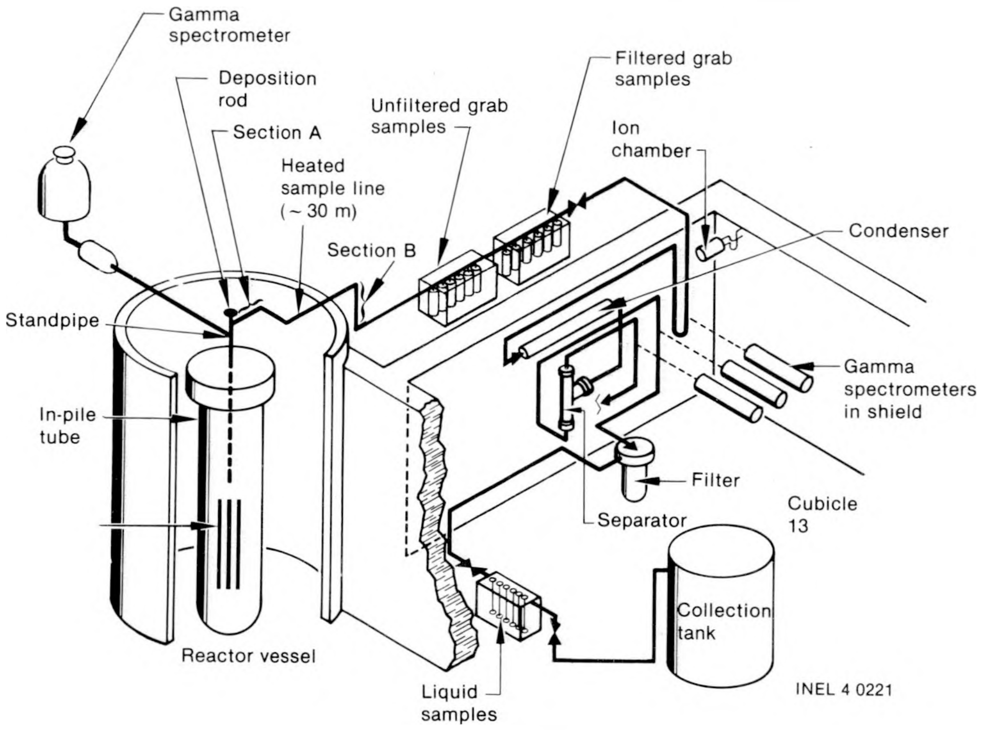

Figure 9. Isometric illustration of sampling and monitoring system.

CORE-MELT TESTING (TO 3100 K)

The need to study fuel and fission product behavior during accidents exceeding the 2400-K limit of the SFD Program is recognized by the reactor community and the USNRC. During 1982 and 1983, the USNRC funded the development of a test method for "Series 2" of the Severe Fuel Damage Program which would permit testing of 32-rod bundles up to and including melting of UO_2 fuel at about 3100 K. The test method included not only the test train itself, capable of sustaining temperatures up to 3100 K in the PBF reactor, but also several instruments essential to retrieve vital data from the hostile environment produced. These instruments include a refractory video probe imaging system; an optical pyrometry system; an on-line system to continuously sample gases, fission products, and aerosols near the fuel; and an on-line system to characterize fission product and aerosol content of the effluent at a point downstream, roughly equivalent to the upper plenum of a light water reactor. Additionally, research was completed on coating systems for refractory metal-clad thermocouples which extend their useful range in a steam environment above 2500 K. These systems, together with the sampling and monitoring system (Figures 8 and 9), are meant to yield the most accurate, representative data available in the world today on the behavior of fuel and fission products during accidents leading to fuel melting.

Test Train. Figure 10 depicts the fuel bundle and surrounding features, which contain a core-melt experiment and protect the PBF reactor. Noteworthy is the use of thoria as a crucible, a structural barrier to prevent radial movement of liquid materials into the insulator region and beyond. Thoria is the most stable of refractory materials against the high temperature steam environment. Its melting point is more than 500 K above that of UO_2, and it is not rapidly attacked by the metals and oxides present, thus making it the most suitable material known for this application. The crucible also serves as an insulator, preventing the temperature at its outer surface from rising above the 2477-K service ceiling of the fibrous zirconia used as the principal heat barrier for the experiment.

Above the fuel bundle, the test train is structured, as in Figure 11, to conduct gases, aerosols, and fission products through a path and temperature letdown approximating that of a light water reactor upper plenum. Data on fission product retention in the primary system by processes of plateout, settling, and thermophoresis will be obtained by the sequential sampler and on coupons located throughout the system.

Fuel. Both the SFD and the core-melt test programs require high-burnup fuel to achieve the best replication of fission product chemistry and fuel behavior. A sufficient supply of fuel rods with high burnups and 1-m active lengths has been secured from the BR-3 reactor in Belgium to conduct approximately five tests. Additional fuel rods from this source are also available. The fuel rods, nominally the same as pressurized water reactor 17 x 17 designs with 9.5-mm outside diameter, have burnups ranging from 33 to 40 GWd/t, and residual effective ^{235}U enrichments up to 5.2 at.%.

Achievable Power. A question of vital importance when using high-burnup fuel is whether the residual fissile content is adequate to achieve the necessary power levels in the PBF. Calculations confirm that a steam-filled fuel bundle of 30 high-burnup fuel rods of the BR-3 type and two control rods will achieve a total power of about 90 kW when the PBF reaches its maximum steady power of 28 MW. At this level, the individual rods average about 3 kW/m linear heat generation rate, which is well above the approximate 0.26-kW/m linear heat generation rate of a typical pressurized water reactor 1 h into an accident. The extra power available easily accommodates expected heat losses from the experiment.

Status. At the time that development of the Series 2 SFD program was terminated by the USNRC, the test train (viz, that portion residing in the PBF) and the video probe/optical pyrometer had progressed into the final design stage. Effluent monitoring systems were still in preliminary design phases, but only a few confirmatory tests remain before final design could commence. All systems had received sufficient examination, both theoretical and experimental, to ensure that the core-melt program could proceed with a high degree of confidence. Potential users of the PBF for meltdown studies will have full access to the designs and supporting studies, either for continuing these designs or for use in alternate meltdown studies.

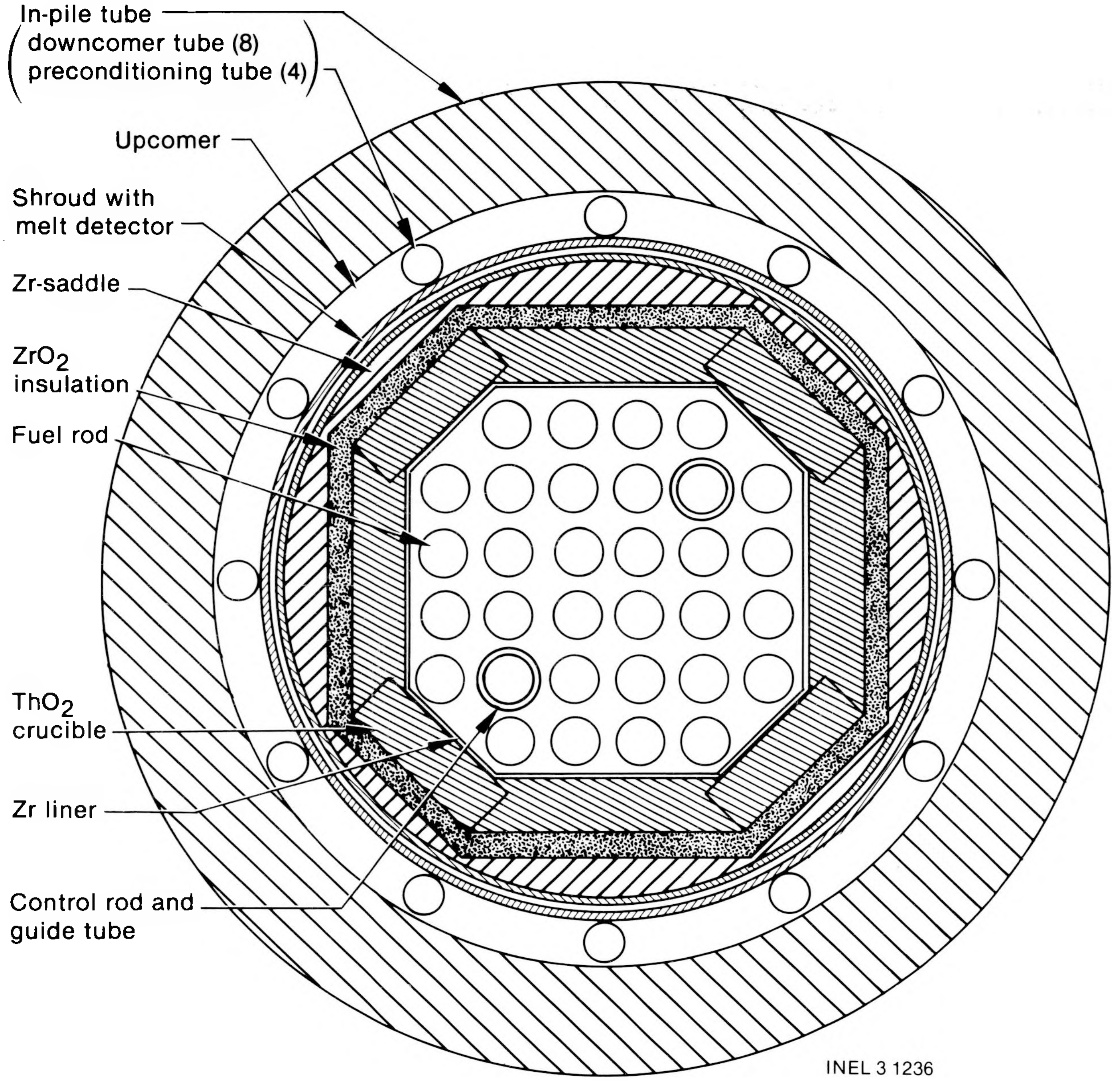

Figure 10. Cross section of the fueled region of a test train design to be used for tests to 3100 K; UO_2 fuel melting.

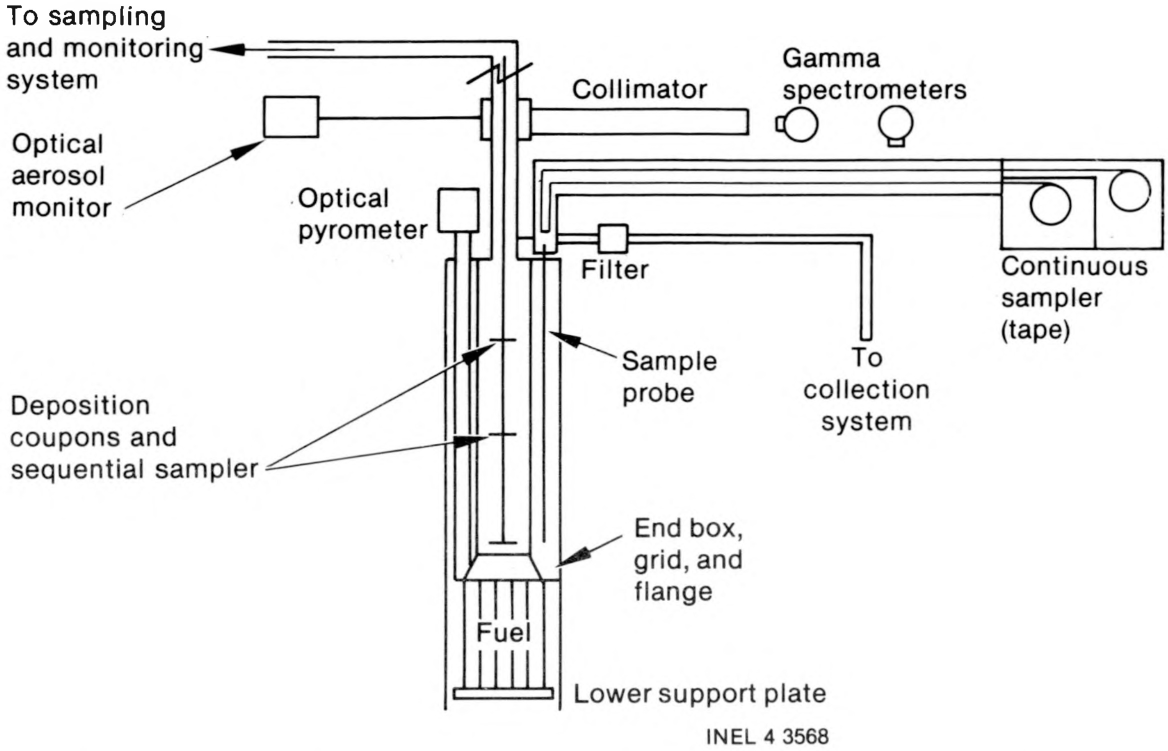

Figure 11. Illustration of test train features above the fuel bundle to be used in tests achieving $U0_2$ fuel melting.

MELT-PROGRESSION TESTING

Beyond the SFD tests (<2400 K) and the core-melt tests ($\leq UO_2$ melting), important questions still remain regarding the consequences of core relocation. Once a fraction of the core becomes liquefied, its relocation toward lower core support structures, the lower plenum, the reactor pressure vessel, and eventually the concrete basemat is possible, depending on accident parameters and mitigative measures taken. Study of the long-term interactions between molten core materials and these structures is particularly difficult out-of-pile, and in some cases impossible, because of the need for continuous volumetric heating unaltered by geometric and chemical changes.

The PBF can not only conduct such tests of melt progression, but can also obtain valuable data on fission products and aerosols. Preliminary feasibility studies have shown that a variety of tests are possible in the PBF to reproduce individual steps in the melt-progression sequence. These include: melt impinging on water-covered support structures; large melt drop through water and attack of the reactor pressure vessel lower head; and continuous heating of melt in contact with a variety of concrete types. Figures 12, 13, and 14 illustrate concepts of these three categories of melt progression. Note in Figure 13 that the concept includes a sudden depressurization (blowdown) as might occur during certain accidents when the lower vessel head is breached.

At blowdown, volatile fission products in the high-burnup fuel are expected to nucleate and grow rapidly, foaming and relocating the fuel and releasing a new charge of aerosols and fission products to the effluent. Although a large, 44-cm-diameter test space is assumed in Figures 13 and 14, smaller experiments, (as in Figure 12), are also possible, which will fit the existing PBF configuration. The 44-cm-diameter test space, which requires extensive modifications to the PBF core, was investigated to determine whether very large tests could be conducted and if adequate power was available. The test concept was found to be feasible. As shown in Figure 15, a test space diameter of 44 cm will accommodate a conventional light water reactor 17 x 17 fuel bundle, together with all necessary protective materials.

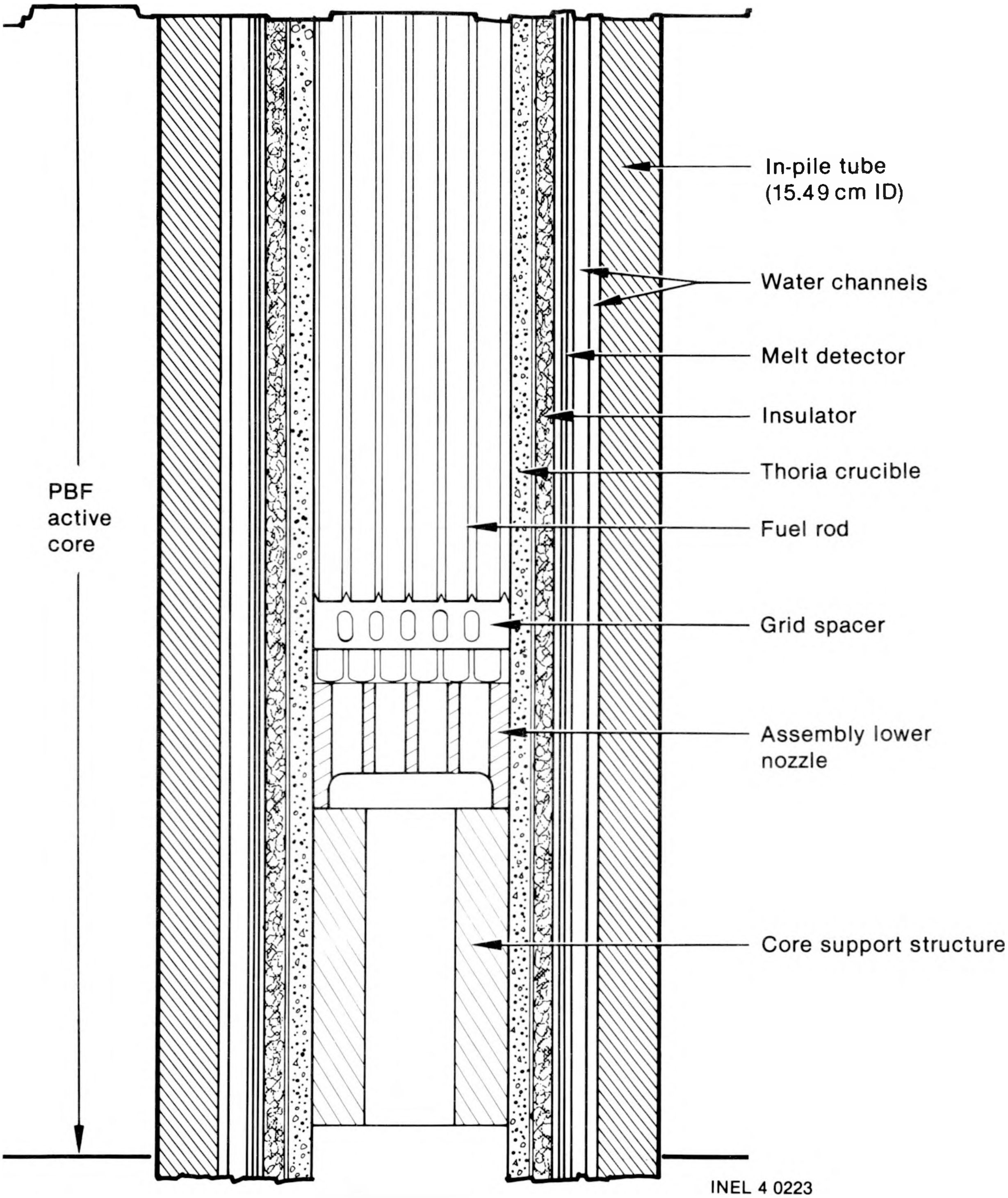

Figure 12. Concept for study of melt progression through core support structures (32-rod bundle in in-pile tube).

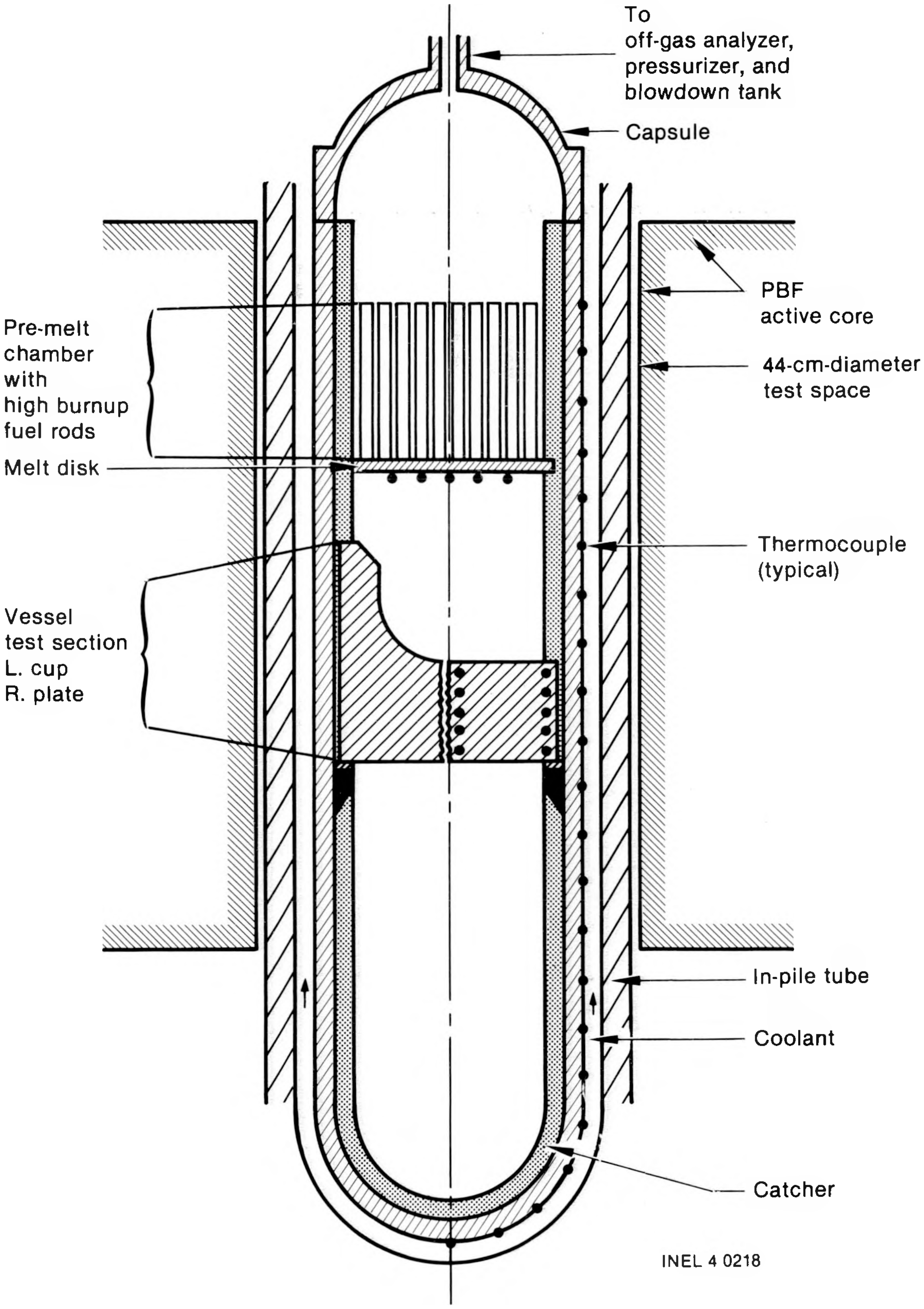

Figure 13. Concept for large-diameter test of melt progression to and through the lower head of the reactor pressure vessel.

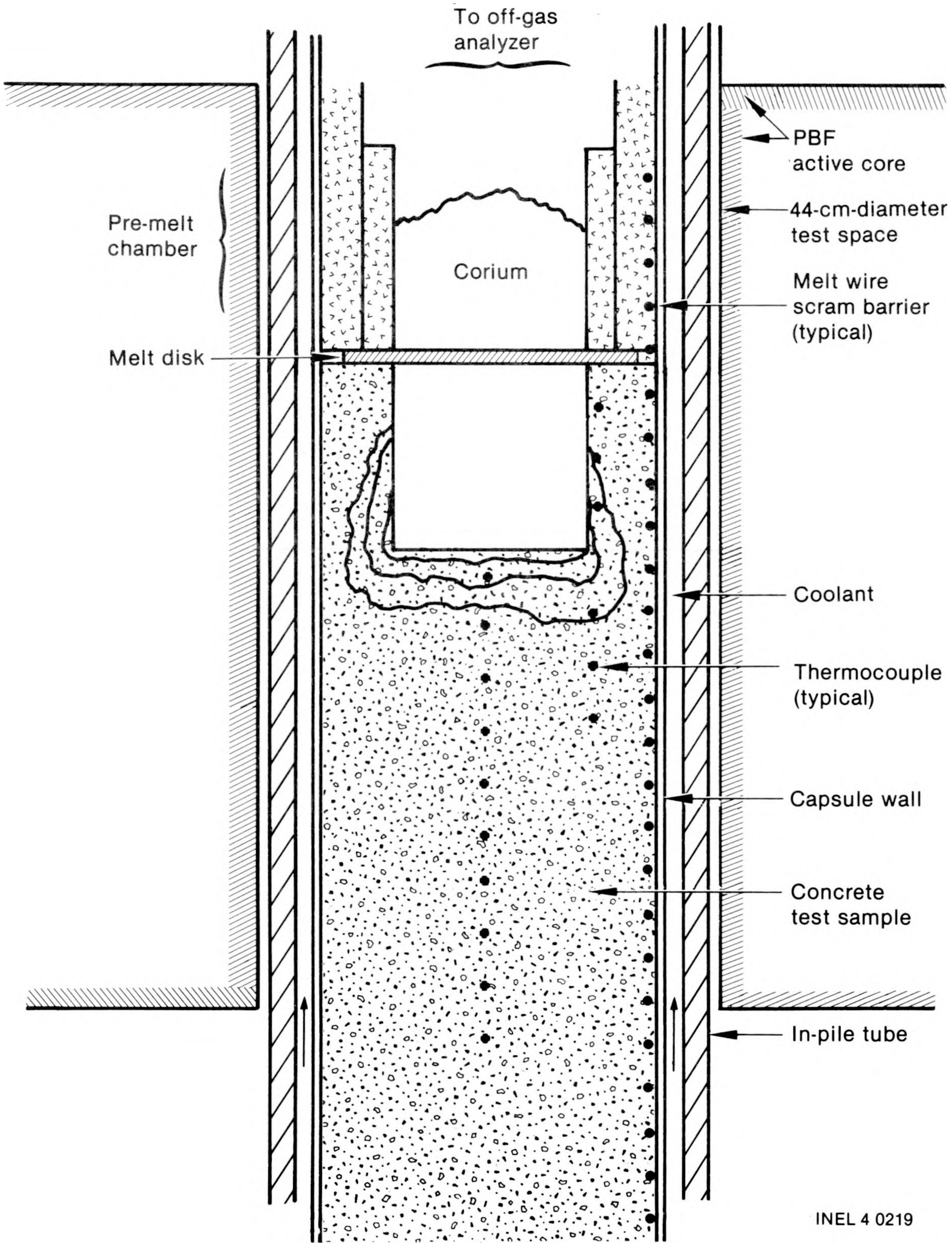

Figure 14. Concept of a large-diameter test of core/concrete interactions.

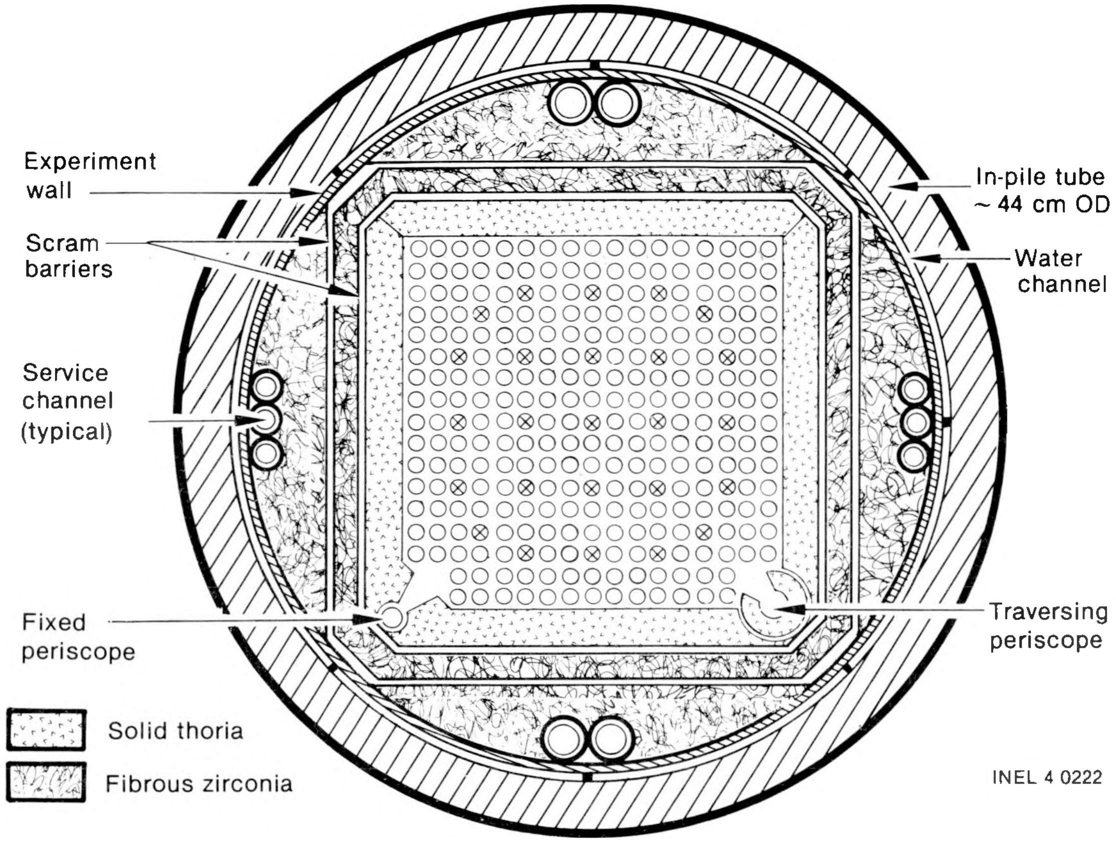

Figure 15. Concept of large-diameter test to enclose a full 17 x 17 pressurized water reactor fuel assembly.

Potential users of the PBF who wish to pursue melt-progression testing will find that significant preparations for such testing (i.e., feasibility studies) have already been undertaken which show, as in the core-melt and SFD studies, that sufficient fission heat can be generated in each case to overcome expected heat losses and maintain required liquid temperatures. Loop equipment is available to establish either boiling water reactor or pressurized water reactor conditions within the test train, as well as controlled depressurization if needed. The sampling and monitoring system (Figures 8 and 9) is available and, as discussed above, instruments to monitor fuel and aerosol behavior have been designed.

PELLET-CLADDING INTERACTION TESTING

A need to perform pellet-cladding interaction (PCI) tests on standard-enriched light water reactor fuel rods which have undergone at least 20 GWd/t burnup has been expressed a number of times in recent years. Although a program to conduct such tests has never taken place, the PBF staff and the EG&G Idaho Physics Department have reviewed requirements for PCI testing and special solutions to the problem of low effective enrichment.

Typically, rods to be tested in a PCI program will begin with approximately 2.5% enriched fuel. After 20 GWd/t burnup, these rods will have only about 0.95 at.% ^{235}U and about 0.4 at.% ^{239}Pu, for an effective enrichment of about 1.5 at.%. A fuel rod with this enrichment in a conventional single-rod test train in the in-pile tube would only achieve about 35.7 kW/m linear heat generation rate at 27 MW PBF power, whereas PCI ramp tests would require power levels nearer 59 kW/m.

To attain higher power levels, the EG&G Idaho Physics Department completed conceptual design of a "flux multiplier" (a thin annulus of 35% enriched UO_2 sandwiched in zircaloy), which would surround the test rods in the PBF and increase the neutron flux. Calculations show that fuel rods with the above depletion could be ramped to over 67 kW/m with the flux multiplier. Even higher powers would be possible if the in-pile tube were replaced. Alternately, two or more fuel rods could be ramped simultaneously. Ramp rates and intermediate holding levels are completely free for specification by the user.

LOSS-OF-COOLANT ACCIDENT TESTING

The loss-of-coolant accident (LOCA) has long been recognized as important among hypothetical reactor accidents due to the number of components in the primary system which can fail, and the possible consequence that emergency coolant might also be lost, thus preventing it from cooling the core. During a LOCA resulting from a large break such as a double-ended hot or cold leg break, system depressurization occurs so rapidly that the additional hazards of fuel rod burnout, ballooning, and rupture may occur long before removal of decay heat becomes a problem. A number of tests were conducted in the PBF to study fuel rod behavior during rapid LOCA depressurizations.

To execute blowdown in the PBF, extensive modifications to the experiment cooling loop were required. As shown in Figure 3, a blowdown header and tank were added and connected to both the hot and cold legs of the loop coolant system. Converging-diverging nozzles were placed in both blowdown pathways downstream of quick-opening blowdown valves. Throats in the nozzles can be replaced and sized to produce specified depressurization rates. In this configuration, blowdown from either or both legs of the coolant system can be selected. Important parts of the blowdown system are the instrumentation spools at the juncture of each blowndown leg. These spools each provide three measurements of coolant void fraction via three-beam gamma densitometry. In addition, the spools provide measurements of volumetric flow rate, momentum flux, pressure, and temperature.

The blowdown system remains a permanent part of the PBF and is available for use by all PBF program participants.

The principal objective of the PBF LOCA tests was to evaluate the thermal and mechanical behavior of fuel rods during blowdown. Since initial conditions of fuel rods vary over wide ranges of several parameters (including prepressurization, burnup, power history, and design, all of which can significantly affect ballooning and rupture), a test train design was chosen to simultaneously test four fuel rods independently, but under identical coolant blowdown conditions. Figure 16 presents a cross section of a typical LOCA test train, illustrating rod isolation and instrumentation. Special "fluted" shrouds were formed to channel coolant for each fuel rod and simultaneously produce constraints against rod bowing and ballooning as would be caused by neighboring fuel rods. Figure 17 illustrates the axial design of the same test train and shows the placement of a flux shaper used to flatten the typical cosine power distribution produced by the PBF.

Specifications varied with each LOCA test, but a test would typically consist of four separate phases; loop heatup, preconditioning, blowdown, and quench. After the primary coolant loop was brought up to the desired temperature and flow rate, the rods would be power-cycled several times and then operated at steady power for approximately 10 h to build up the desired fission product inventory. Blowdown would follow in a predetermined and automatically sequenced manner, followed by a specified delay and quench.

The LOCA test program studied a wide variety of fuel rod types including those with burnups to 16 GWd/t and prepressurizations from 0.1 to 4.8 MPa. In an effort to determine the behavior of the highest power rods (hot spot) in typical pressurized water reactors, preconditioning would typically take place near 55 kW/m linear heat generation rate. Additional details of these tests are available in the open literature.

Quadrant location	Fuel element
1	Element 1
2	Element 3
3	Element 5
4	Element 7

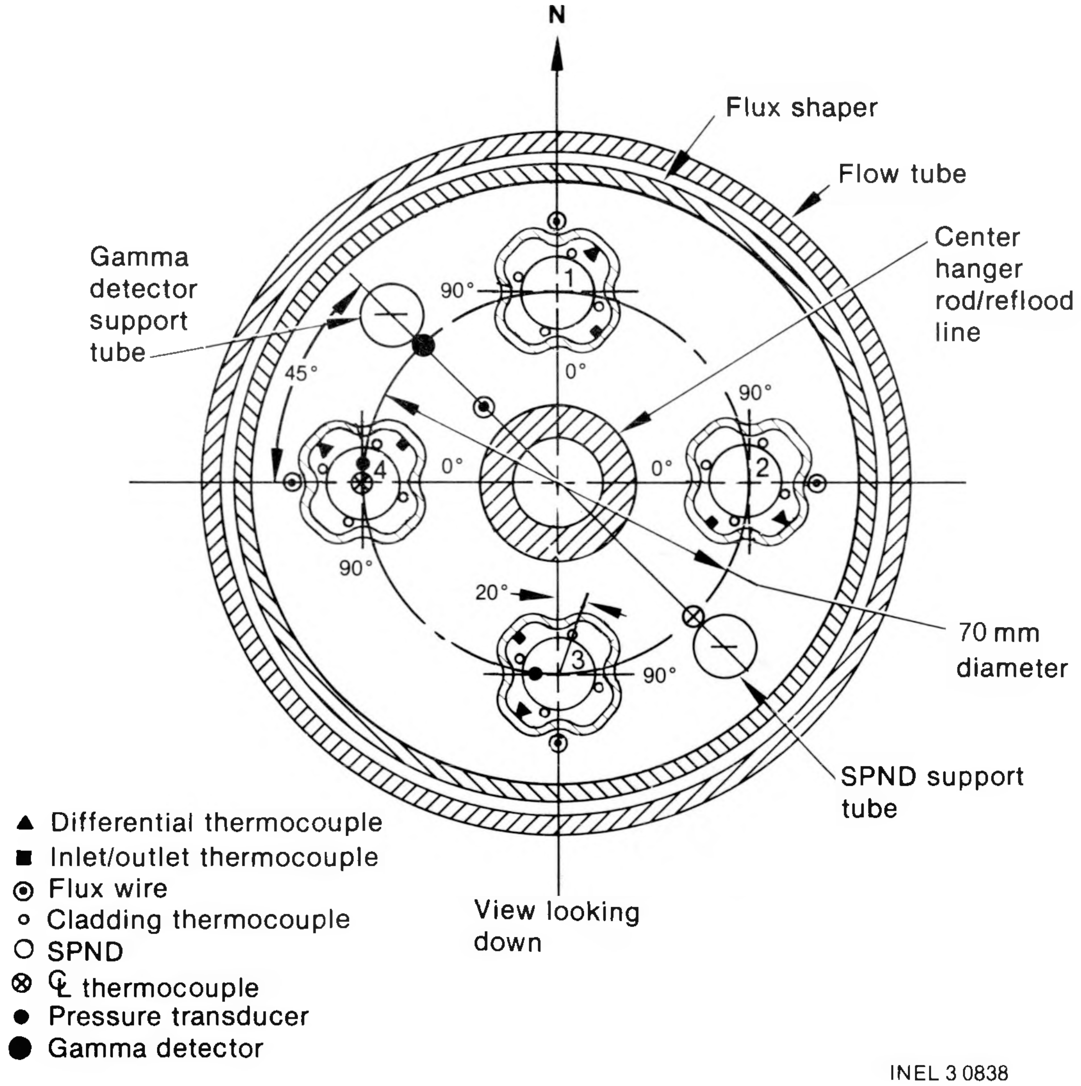

Figure 16. A typical four-rod LOCA test train and instruments.

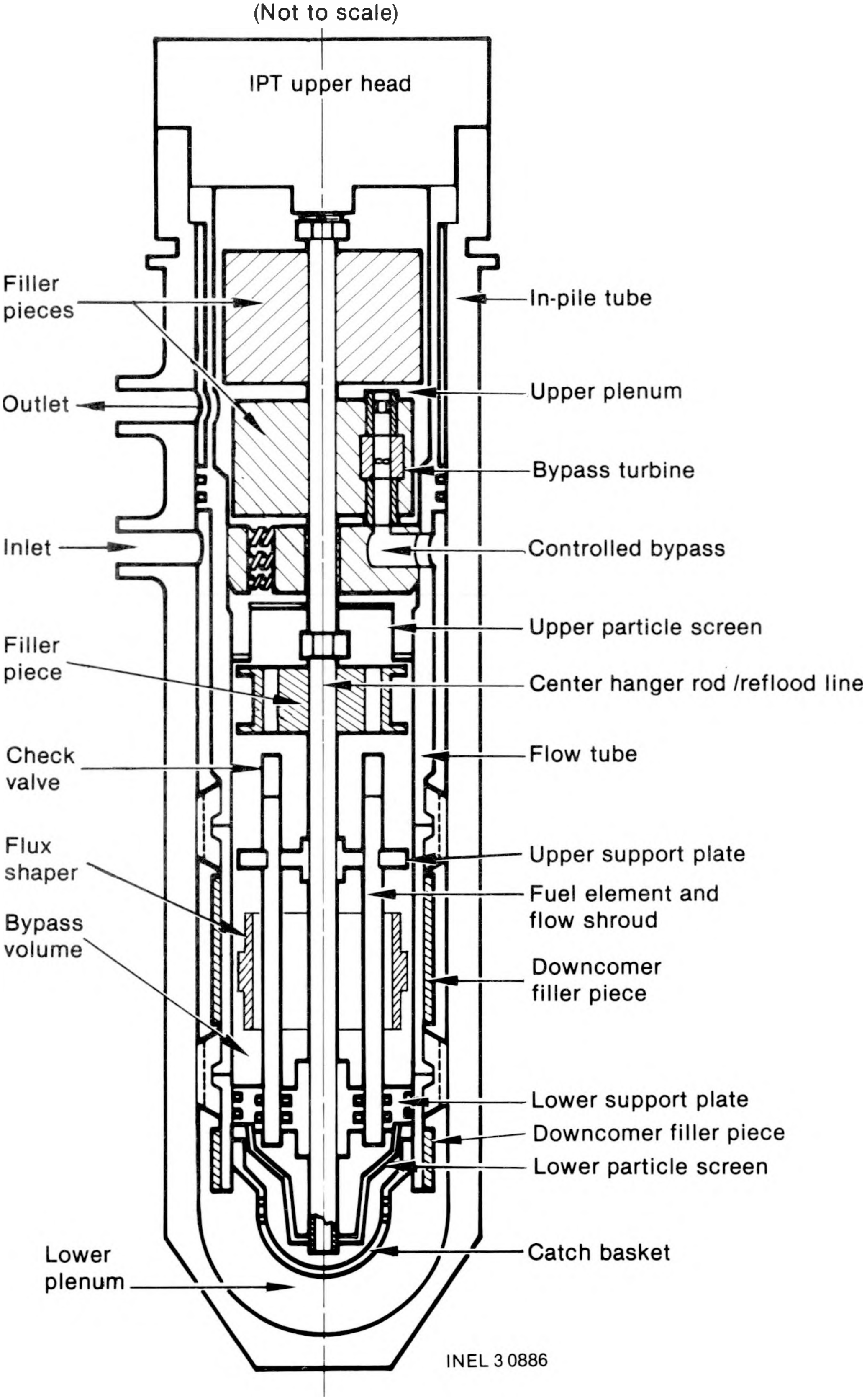

Figure 17. Axial schematic of a typical four-rod LOCA test train.

OPERATIONAL TRANSIENT TESTING

The PBF has been used successfully in a study of fuel rod responses to operational transients, defined as anticipated events in either boiling or pressurized water reactors (BWRs or PWRs) in which the reactor power undergoes an uncorrected excursion above normal levels in response to an unplanned event such as a loss of secondary heat sink. Among the most severe of these transients is the BWR main steamline isolation valve closure without scram, which results in void collapse and positive reactivity feedback to the core. This event was result in a transient power increase of about 745%. Possible consequences of such an event include pellet-cladding interaction (PCI), burnout with attendant oxidation and embrittlement of the cladding, and cladding collapse onto the fuel. The USNRC sponsored several tests of operational transients in the PBF to determine whether new regulations and/or operational procedures were required.

A test conducted in 1982, OPT 1-2, is described here to illustrate the techniques used, which are available to other PBF users. Specifications for the test included BWR rods with sufficient burnup that prior PCI damage and cladding embrittlement were in place. These rods, obtained from a BWR vendor, were preirradiated to between 8 and 10 GWd/t under BWR conditions. A special requirement for the tests, which also had to be conducted under BWR coolant conditions, was for two-phase flow. To accomplish this, a four-rod test train similar in design to that shown in Figures 16 and 17 was used. Two positions were occupied by the test rods, and the remaining two positions were occupied by 10% enriched fresh fuel rods. As shown in Figure 18, slightly subcooled coolant was first ducted over the fresh fuel rods, used to preheat and raise the quality of the coolant. The resulting two-phase coolant was then returned to the inlet of the two preirradiated test rods.

Nuclear operation for the OPT 1-2 test required several successive power ramps of the two test rods to 23, 27, and finally to 28.5 kW/m, which was held for a period of 12 h to build up the short-half-life fission product inventory necessary to measure fission product releases. To obtain the proper BWR operational transient, the PBF transient rods were programmed to initiate a transient from a PBF power level of about 26 MW. The resulting power transient, which met the test specifications, is illustrated in Figure 19. Test rod power increased to 300 kW/m in about of 1.5 s, then dropped to about 10 kW/m during the next 20 s, where it was held for about 20 min. In addition to the very high power levels, the test rods also experienced a planned large reduction in flow rate, effected by opening the variable orifice valve.

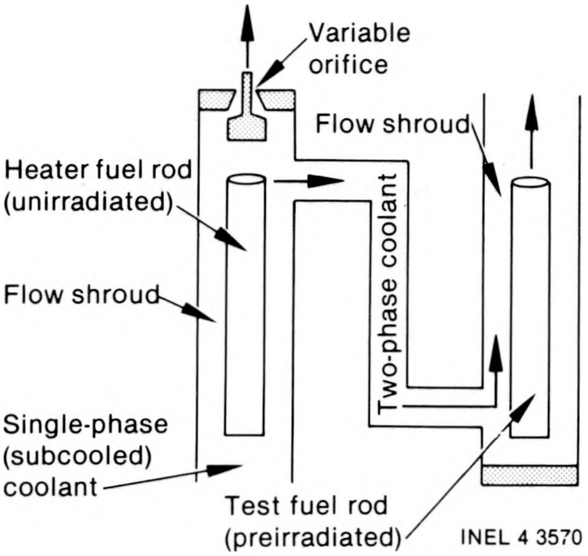

Figure 18. Schematic of fuel rod shroud pair showing flow path.

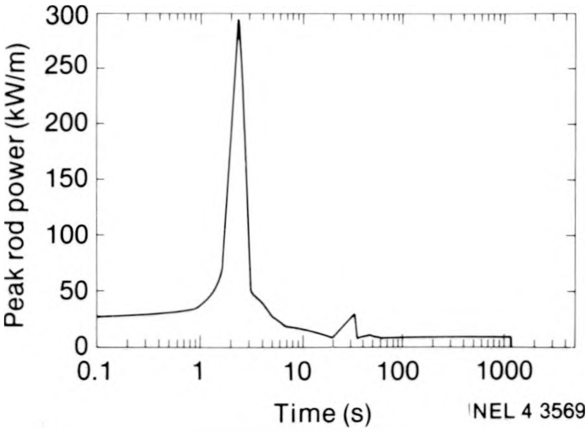

Figure 19. Test rod peak power versus time during Test OPT 1-2 transient.

Further details of the PBF Operational Transient tests are available in the literature. The test described here illustrates the versatility of the PBF to produce specific power histories, as well as the experimental techniques used to achieve coolant quality and flow rates which replicate hypothetical accidents.

REACTIVITY INITIATED ACCIDENT TESTING

The PBF was originally designed to produce fast, high-energy power excursions to continue the study of reactivity initiated accidents (RIA) begun earlier with the SPERT (Special Power Excursion Reactor Test) reactors and the Capsule Drive Core at the INEL. Important capabilities brought to this program by the PBF included not only the short exponential period (1 ms) and high peak power (270 GW) possible, but also the large space available for fuel rod bundles (up to 64 rods in the IPT) and the loop facility which could establish full pressurized or boiling water reactor coolant conditions on the rods during each test.

The principal objectives of the RIA tests in the PBF included the determination of failure modes and failure thresholds as a function of exponential period, total energy deposited, fuel type, and burnup. Of great importance during the tests was the measurement of pressure pulses derived from "steam explosions" (or molten fuel-coolant interactions) and determination of the efficiency with which thermal energy in the fuel was converted to mechanical energy.

Typically, the RIA tests in the PBF were conducted on four fuel rods simultaneously, using a four-rod test train design as illustrated in Figure 20; an axial cross section of the test train in the in-pile tube is shown in Figure 21. As indicated in the figures, instrumentation consisted a number of cladding-mounted, fast-response thermocouples, flux wires, self-powered neutron detectors, centerline fuel thermocouples or ultrasonic transducers (not shown), individual flow turbines, and LVDTs (linear variable displacement transducers) to monitor elongation of the rods responding to power bursts. Figure 22 illustrates the data obtained on fuel rod linear power and fuel enthalpy during one such test.

Potential users of the PBF should note that the facility is equipped with an extremely versatile data acquisition and reduction system, as described earlier, capable of recording all measured parameters during such rapid tests with great fidelity. This computer-controlled system will record 66 data channels with a frequency bandwidth of 5 to 20 kHz, and 256 channels at 10 to 100 Hz.

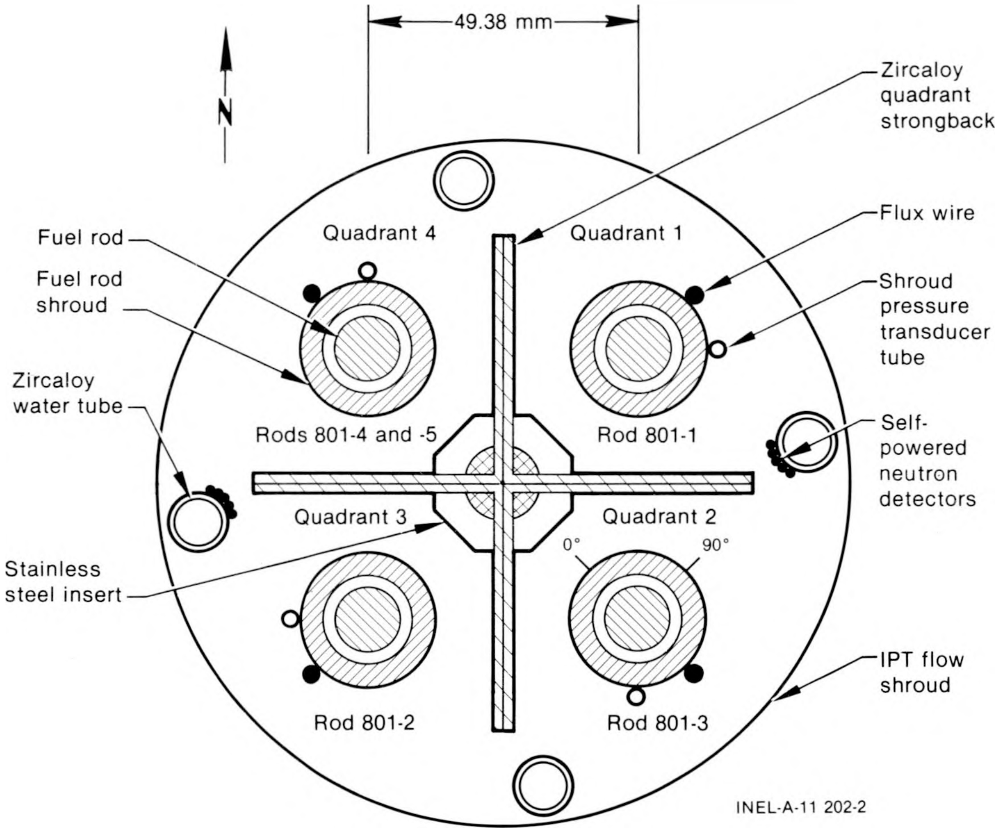

Figure 20. Fuel rod arrangement and instrumentation in the Test RIA 1-1 four-rod test assembly.

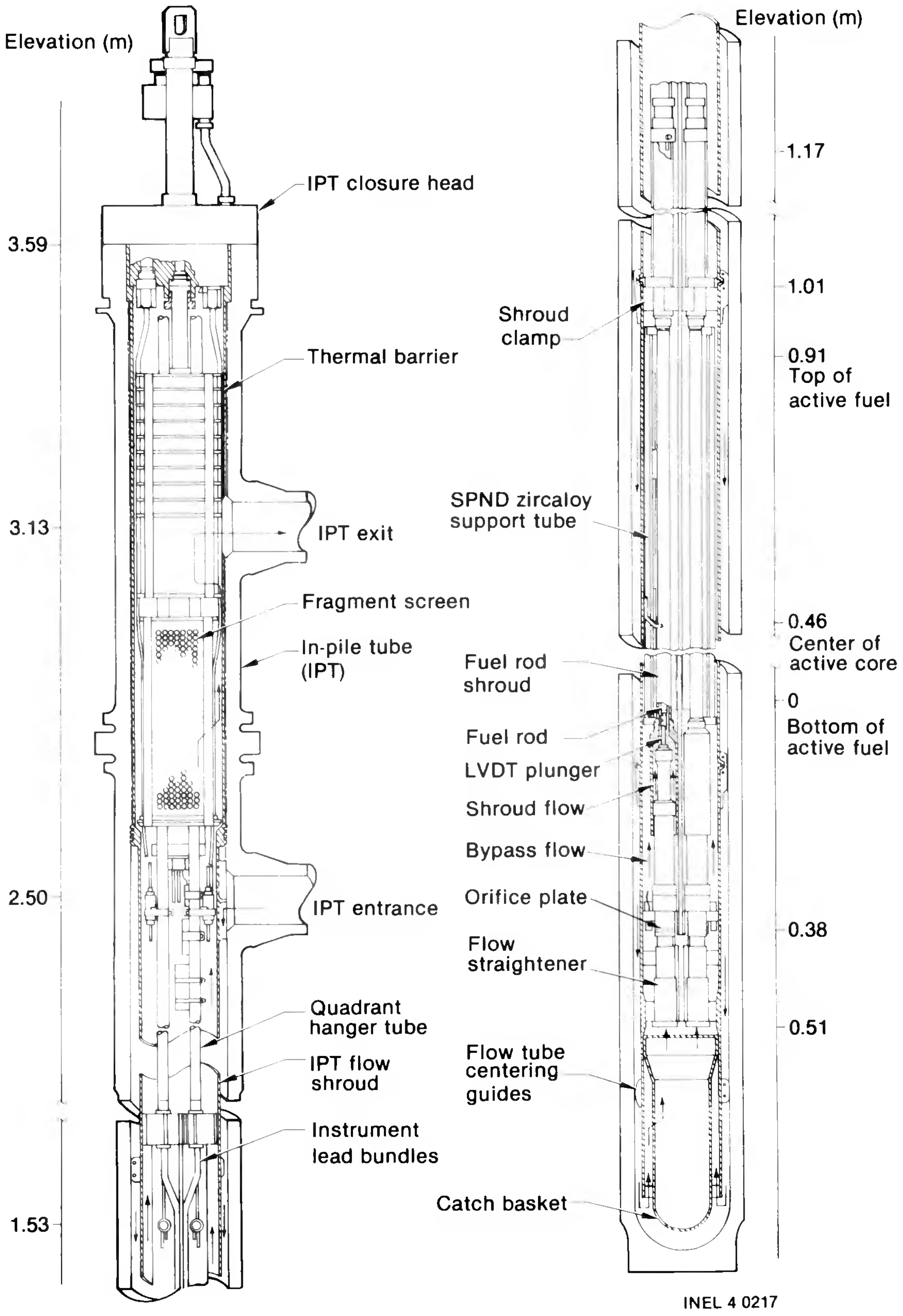

Figure 21. Axial cross section of RIA 1-1 test train assembly.

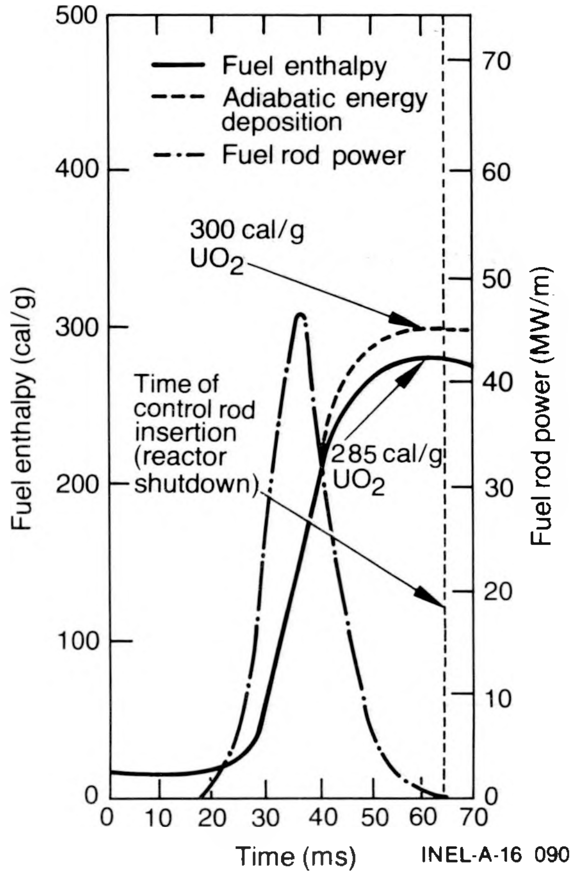

Figure 22. Test RIA 1-1 power burst and increase in adiabatic energy deposition and peak fuel enthalpy up to the time of reactor scram.

POWER-COOLING-MISMATCH TESTING

The result of many hypothesized accidents is an imbalance between the heat generation rate of the nuclear core and the heat removal rate of the coolant. Two extreme cases have commonly been designated the loss-of-coolant accident, in which all or part of the core coolant is rapidly lost, and the reactivity initiated accident, in which a sudden power increase is initiated within the nuclear core. Between these two extremes lies a wide range of off-normal power-cooling conditions, commonly referred to as power-cooling-mismatch (PCM) accidents.

An extensive Power-Cooling-Mismatch Test Series was conducted in the PBF as part of the USNRC's Fuel Behavior Program. This test series was designed to study the behavior of pressurized-water-reactor-type fuel rods operated under power-coolant imbalance conditions, and the results were used to provide modeling and damage information. In addition, data from the test series were used to help assess the potential for rod-to-rod interactions during high-temperature film boiling operation.

The PCM test series in the PBF covered a wide range of individual test objectives and, as a result, utilized several test train designs. The first test used a single-fuel-rod test train, as much to develop experimental methods as to study the effects of operating a fuel rod for extended time beyond critical heat flux under vapor blanket heat transfer conditions.

During the second-generation tests, PCM studies were conducted with four-rod test trains similar to that shown in Figure 16. The objectives included determination of the power at which critical heat flux (CHF) occurred, how the onset of CHF was affected by flow rate, how the onset of CHF was altered by prior operation in CHF; how CHF propagated along a fuel rod; how the cladding responded to CHF; the extent of cladding oxidation under vapor blanketing conditions; cladding collapse as a function of cladding temperature beyond CHF; and the occurrence and nature of rod failure.

The PCM tests included incorporation of a wide variety of metal-clad thermocouples and methods for attaching them to the cladding to avoid perturbing heat transfer regimes. Many of the fuel rods contained centerline thermocouples which incorporated state-of-the-art designs and seals. Ultrasonic drilling techniques were used to drill 1-mm-diameter holes through the fuel pellets for the centerline thermocouples. Several tests utilized ultrasonic thermometers for centerline temperature measurement and this technology was consequently highly advanced at the Idaho National Engineering Laboratory. Other instruments developed partly for the PCM tests include turbine flowmeters and linear variable displacement transducers used to monitor fuel rod elongation under CHF conditions.

One objective of the PCM tests which could not be met with single- or four-rod test trains was to determine whether critical heat flux could be propagated from rod-to-rod by any mechanism, particularly by rod bowing. Rod bowing could lead, in fact, to the initiation as well as the propagation of CHF. To study multi-rod phenomena, a test train was designed to hold a square, 3 x 3, array of nine fuel rods in typical pressurized water reactor pitch. This test train design, illustrated in Figure 23, was used successfully for the two final PCM tests (PCM-5 and PCM-7). Figure 24 illustrates one of many data sets obtained through the PCM tests, which confirmed that CHF quench occurs under the same power and thermal-hydraulic conditions as departure from nucleate boiling (DNB). The nine-rod test trains were also able to confirm that neither DNB itself nor propagation of DNB from rod-to-rod can be expected. Other important conclusions are discussed in the open literature.

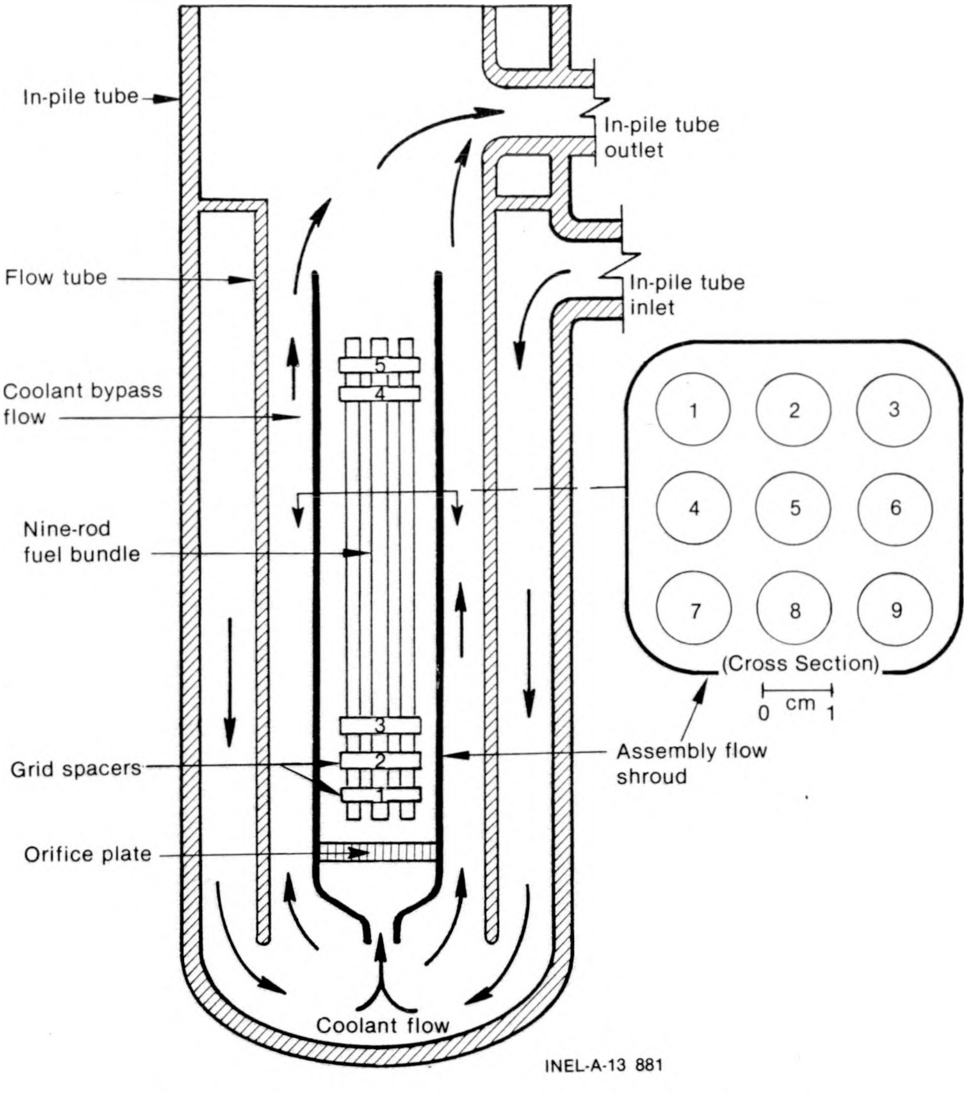

Figure 23. Schematic representation of the PCM-7 test assembly.

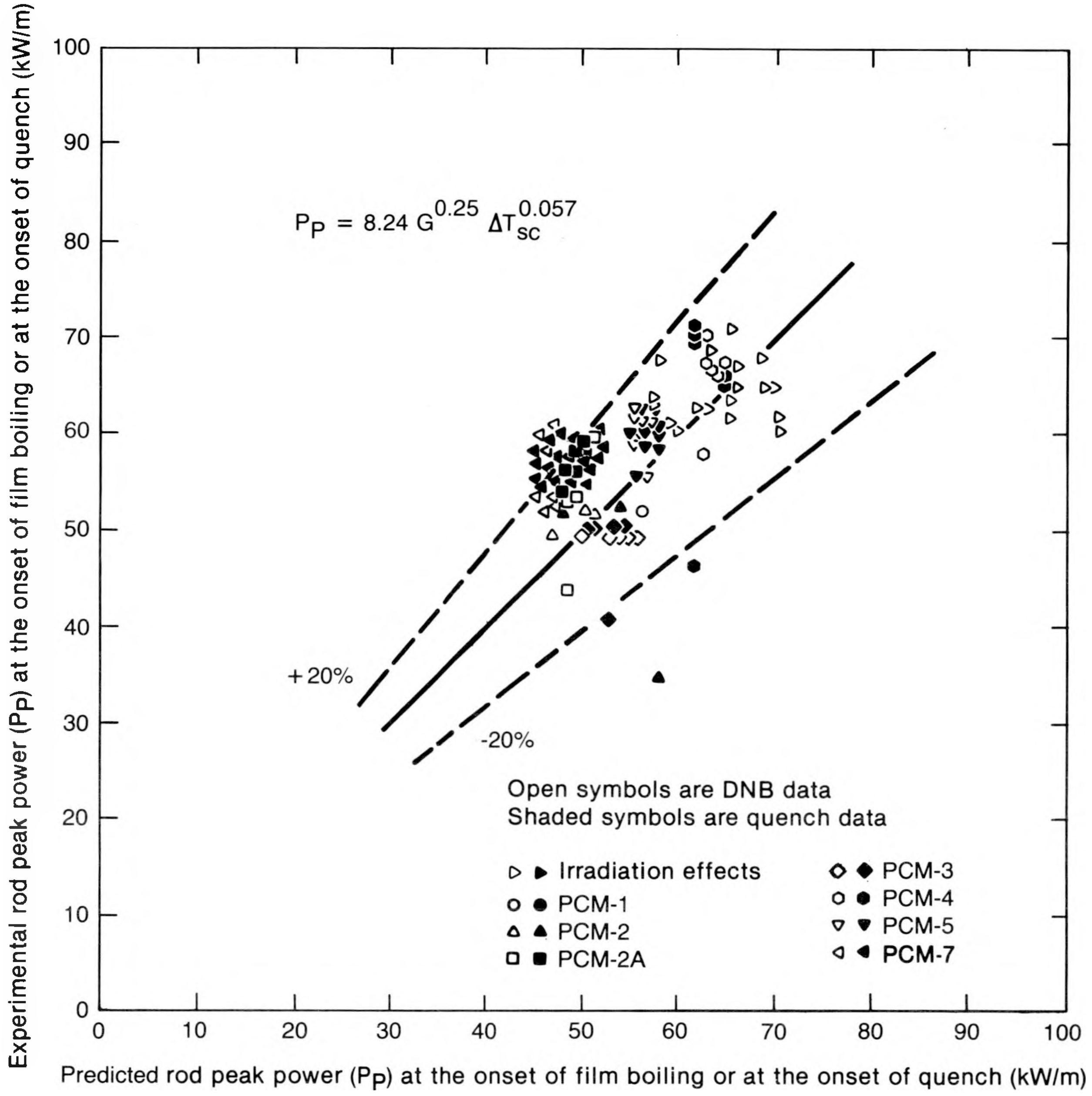

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Figure 24. Comparison of the conditions at the onset of film boiling and quenching.

AVAILABILITY OF EQUIPMENT

In-Pile Tubes - Two fully qualified Inconel in-pile tubes are available. One has been used for all tests in the PBF program to date; the other is new, cold, and clean.

Test Trains - A number of test trains are available for reuse, but would require various expenditures for cleanup, reinstrumentation, and repairs, as indicated in Table 2.

Table 2. Description and status of available test trains

Number of Trains Available	Number of Rods Held	Prior Use	Condition
2	4[a]	LOFT Lead Rod, LOCA, TC, and CANDU tests (reflood capable)	Intact with IPT closure heads installed; both trains are still loaded with fuel rods; about 50% instrument replacement required.
3	4[b]	RIA 1-1, 1-2; OPT 1-1, 1-2 (no reflood capability)	Intact; about 30% instrument replacement necessary; no closure head.
1	9	Never used; designed for RIA and OPTRAN	Excellent; no closure head; not instrumented.
1	32	Severe Fuel Damage Test 1-1; not completed due to development of leak	Excellent; activated and slightly contaminated; requires repair/replacement of preconditioning lines; fuel rods still loaded.

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a. EG&G Idaho design.

b. Battelle Pacific Northwest Laboratories design.