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RIA 1-4

EXPERIMENT SPECIFICATION DOCUMENT

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1. INTRODUCTION

The Reactivity Initiated Accident (RIA) test series is being performed in the Power Burst Facility (PBF) to provide data for verifying analytical codes capable of predicting light water reactor fuel performance during hypothetical control rod drop (or ejection) accidents[1]. The most important aspect of an RIA or an RIA test is the magnitude of energy deposited into the fuel, therefore the primary purpose of the RIA test series is to better define the relationship between energy deposition and fuel rod behavior. In particular, the RIA research objectives are to determine failure thresholds, modes, and consequences with respect to total energy deposition, irradiation history, and fuel design. The most severe RIA is the postulated Boiling Water Reactor (BWR) control rod drop during reactor startup, therefore all the RIA tests will be conducted at BWR startup coolant conditions.

Test RIA 1-4 will consist of a 4 x 4 array of rods positioned in a coolant flow shroud. Fourteen of the rods are unirradiated BWR/6-type fuel rods and two are water-filled rods. Two water rods will be interior rods positioned along one diagonal. The primary objective of Test RIA 1-4 will be to obtain data on clustered fuel rod behavior during a rapid power transient simulating a BWR control rod drop.

Section 2 is intended to provide a basis for understanding the test plan for Test RIA 1-4. Past RIA test experience and a review of the PBF-RIA tests which are scheduled for completion prior to Test RIA 1-4 are summarized in this section. Section 3 describes the fuel rod, water rod, grid spacer, flow channel, and bundle support structure specifications. The instrumentation required for this test is discussed in Section 4. Sections 5 and 6 present the preliminary reactor operation and posttest operation requirements.

2. SUMMARY OF RIA TEST EXPERIENCE

Prior to the RIA test series being performed in PBF, most of the experimental data on fuel behavior during an RIA have been obtained from tests conducted at the SPERT, TREAT, and NSRR facilities [a]. All of these tests were single rod tests conducted in an atmospheric pressure, room temperature capsule environment without forced coolant flow. The experimental results indicated that when unirradiated light water reactor (LWR) fuel rods were subjected to peak volumetric energy depositions near 2960 cal/cm^3 (near the fuel pellet surface), failures of the cladding due to melting, oxidation, and embrittlement occurred. For pellet surface energy depositions greater than about 4060 cal/cm^3, brittle fractures of the cladding and fuel fragmentation occurred prior to cladding melting. Not enough data have been obtained yet to qualify failure of irradiated rods subjected to RIA energy depositions, however they have failed at energy depositions as low as 1500 cal/cm^3 (radially averaged energy deposition at axial flux peak).

The SPERT, TREAT, and NSRR results are limited largely because the test coolant conditions during the capsule tests were much different than those in a typical LWR. The PBF-RIA test series will provide data on RIA fuel behavior with the more typical LWR coolant conditions of high pressure, high temperature, and forced coolant flow.

The PBF-RIA tests will be performed at BWR hot-startup coolant conditions; the conditions postulated for the most severe RIA. The RIA Scoping test was performed first to: (a) identify the energy deposition failure threshold for hot-startup conditions, (b) evaluate calorimetry techniques for RIA transients, and (c) determine the magnitudes of possible pressure pulses that can result from RIA-induced rod failure.

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[a] SPERT - Special Power Excursion Reactor Tests
TREAT - Transient Reactor Tests
NSRR - Nuclear Safety Research Reactor

Tests RIA 1-1, RIA 1-2, and RIA-1-6 will be performed to provide data on the relative effects of pre-irradiation, internal pressure, and fuel rod design on cladding failure and failure consequences during a RIA. These tests each consist of four separately shrouded fuel rods of SAXTON reactor design.

Test RIA 1-3 will be comprised of four unirradiated BWR/6-type fuel rods, each surrounded by a separate flow shroud. The main objectives of Test RIA 1-3 are to: (a) determine the response of unirradiated BWR/6 fuel rods to a power transient at BWR hot startup conditions, (b) provide data for defining the energy threshold for incipient failure of unirradiated BWR/6 fuel rods, and (c) determine the effect of internal fuel rod pressure on the transient response of unirradiated BWR/6-type fuel rods.

Test RIA 1-4 will be comprised of 14 unirradiated BWR/6-type fuel rods of the same specifications as the fuel rods used in RIA 1-3. The results of Test RIA 1-4 will be compared with those from Test RIA 1-3 to evaluate the effects of rod clustering on the cladding failure threshold and rod failure mode for unirradiated BWR/6- type fuel rods.

3. EXPERIMENT CONFIGURATION SPECIFICATIONS

Test RIA 1-4 will consist of a 16 rod bundle made up of 14 BWR/6-type fuel rods and 2 water-filled rods. The bundle will be contained within a square flow channel where BWR/6-type grid spacers will maintain a 4 x 4 rod array. The two water rods will be positioned in the center of the bundle along one diagonal and will lock the grid spacers in the required axial positions. A support structure will center the rod bundle and flow channel inside the PBF in-pile tube and provide for test train instrumentation attachment. Specifications for the fuel and water rods, grid spacers, flow channel, and support structure follow.

3.1 Fuel Rod Specifications

The fourteen BWR/6-type fuel rods will be previously unirradiated and will consist of 2 interior rods, four corner rods, and eight side rods[a]. All fourteen fuel rods will be backfilled with commercially pure helium (99.995%) to a cold internal rod pressure of 0.310 $\pm$ 0.007 MPa. The nominal design characteristics of the fuel, cladding, and rod assembly are listed in Table I. Table II lists the post fabrication measurements which will be required before testing to describe the fuel rod design characteristics. The as-built dimensions of the fuel rod pitch and rod-to-shroud spacing will be measured at the upper and lower ends after assembly.

In addition to the measurements described in Table II, care must be taken to insure that the position of each pellet in the fuel stacks is known, maintained, and recorded during fuel rod assembly. The cladding tubes must be free of any scratches or defects. A tube will

[a] Fuel enrichments will be specified after reactor physics calculations are completed for the 14 fuel rod and two water rod geometry. Calculations for a 16 fuel rod cluster of BWR/6-type rods indicates fuel enrichments of 5.8 wt% for the interior rods, 4.2 wt% for the side rods, and 3.6 wt% for the corner rods. The fuel enrichments for the 14 fuel rod clusters are anticipated to be about the same.

TABLE I

TEST RIA 1-4 FUEL ROD DESIGN CHARACTERISTICS

Characteristic	Value
Fuel	
Pellet OD (mm)	10.566 ± 0.013
Pellet Length (mm)	10.566 ± 0.3
Pellet Density (%TD)	95 ± 0.2
Enriched Pellet Stack Length (mm)	914 ± 1
Length of Al_2O_3 Pellet Stack (mm)	39.8 ± 0.2
UO_2 Pellet End Configuration	Flat, 45 degree chamfer
Pellet Centerhole Diameter (mm)	1.88 ± 0.05
Burnup	0
O/U Ratio	2.002 ± 0.002
Drilled Pellet Region[a] (mm)	740 to 914
Cladding	
Material	Zircaloy-2
Tube OD (mm)	12.522 ± 0.190
Tube ID (mm)	10.795 ± 0.038
Tube Wall Thickness (mm)	0.8635 ± 0.076
Fuel Rod	
Overall Length (mm)	[b]
Gas Plenum Length (mm)	62.8 ± 0.5
Gas Backfill Volume (cm^3)	8.5 to 9.5
Plenum Spring Free Length (mm)	66.9 ± 0.5
Plenum Spring Compressed Length (mm)	14.4 ± 0.5
Plenum Spring Coil Diameter (mm)	9.75 ± 0.5
Plenum Spring Wire Diameter (mm)	1.57 ± 0.1
Spring Rate (N/mm)	2.63 to 3.85
Plenum Spring Material	Stainless Steel - 302

[a] In rods drilled for centerline thermocouple or SPND.
[b] As required to accomodate instrumentation.

TABLE II

TEST RIA 1-4 FUEL ROD CHARACTERIZATION REQUIREMENTS

Measurement	Fuel Rod Serial No.	Frequency	Accuracy	Comments
Pellet diameter	804-2, -4, -6, -11 -13, -15,	100%	±0.005 mm	Measure at pellet midplane.
Pellet length	804-2, -4, -6, -11 -13, -15,	100%	±0.005 mm	
Pellet weight	804-2, -4, -6, -11 -13, -15,	100%	±0.001 g	Dry pellets prior to weighing.
Pellet immersion density	804-2, -4, -6, -11 -13, -15,	100%	±0.001 g/cm^3	100% of drilled pellets, 10% of undrilled pellets.
Pellet center hole diameter	804-2, -11, -13	[a]	±0.005 mm	Top, bottom, and approximate midplane of drilled pellets.
Fuel stack length	All Fuel Rods	-	± 1 mm	Use radiographic technique or equivalent.
Plenum length	All Fuel Rods	-	±0.5 mm	
Cladding inside diameter	All Fuel Rods	Continuous	±0.005 mm	Measurements at 0 and 90 degrees orientation.
Cladding outside diameter	All Fuel Rods and Water Rods	Continuous	±0.005 mm	Measurements at 0 and 90 degrees corresponding to ID measurements.
Fuel rod overall length	All Fuel Rods	-	±1 mm	
Radiograph	All Fuel Rods	-	-	
Internal gas pressure	All Fuel Rods	-	±0.007 MPa	Full length of Rod.
Leak check	All Fuel Rods	-		
Photographs	All Fuel Rods	-		
Void volume	All Fuel Rods	-	±0.2 cm^3	At 45, 135, 225, and 315 degrees
Fuel rod weight	All Fuel Rods	-	±0.1 g	
Plenum spring free and compressed length	All Fuel Rods	-	±0.1 mm	Cladding, bottom end cap, fuel only.
Plenum spring constant	All Fuel Rods	-	±0.1 N/mm	

[a] Four pellets above and four pellets below fuel thermocouple or SPND tip location on applicable rods.

be rejected if the cladding wall thickness varies more than 5% from nominal. After backfilling, the rods will be sealed using a laser puncture/sealing technique through the upper end of the cladding. Standard fuel rod cleanliness procedures must be followed in all fabricating operations.

3.2 Water Rod Specifications

The two water-filled rods of the bundle will be fabricated from Zr-2 tubing having the same inner and outer diameter dimensions as the cladding of the 14 fuel rods. The two water rods will flatten the radial power profile of the bundle and position the bundle grid spacers vertically in the bundle. Four 1.6 $\pm$0.1 mm diameter holes should be drilled around the circumference of the water rod at both ends to allow coolant water flow. End plugs will be installed on both ends of the water rods.

3.3 Grid Spacers Specifications

The bundle grid spacers will be similar in design to BWR/6 spacers but sized for the 4 x 4 bundle array. They will be fabricated from zircaloy-4 and equipped with Inconel springs to maintain rod spacing. The grid spacers will be centered at 152 $\pm$1, 457 $\pm$1, 762 $\pm$1, and 914 $\pm$1 mm from the bottom of the fuel stack. The lower ends of the fuel rod should be fastened rigidly to the test train assembly. The as-built dimensions of the grid spacer blades will be measured.

3.4 Coolant Flow Channel

The coolant flow channel (or flow shroud) surrounding the 4 x 4 array of fuel rods will be similar to a BWR/6 flow channel regarding material, wall thickness, and corner radii. As in a BWR/6 reactor, the channel will be fabricated of zircaloy-4 with a wall thickness of 3.05 mm. The length of the flow channel will be as required to be compatible with the support structure. Figure 1 provides the radial dimensions of the flow channel and the required fuel rod positioning within the channel.

The inlet and outlet to the flow channel will be designed to accommodate the required instrumentation and an inlet orifice plate[a]. The lower part of the flow channel will provide a radially uniform flow through the bundle. The fuel rods will be supported at the bottom of the assembly as in a BWR/6 bundle. Provision will be made for mixing the coolant flow between the fueled region and the location of the shroud outlet ΔT thermocouples.

3.5 Bundle Support Structure Specifications

The bundle support structure will hold the bundle of rods and flow channel in position in the radial center of the PBF in-pile tube and provide for attaching the appropriate test train instrumentation. The fuel rod active length will correspond to the active fuel region of the PBF core to within ±1 mm. The percentage of bypass flow to test bundle flow should be designed for a nominal 50% with a tolerance of ±10%.

A fine mesh fragment screen between the test fuel and the PBF in-pile tube outlet will be required. The screen design will provide a trap for fuel fragments for posttest removal from the in-pile tube. The screen will be located as far as possible from the bundle to reduce the probability of thermal burnout. A screen mesh will be selected on the basis of pressure drop, particle retention, thermal considerations, and structural considerations.

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[a] Orifice hole diameter will be specified later.

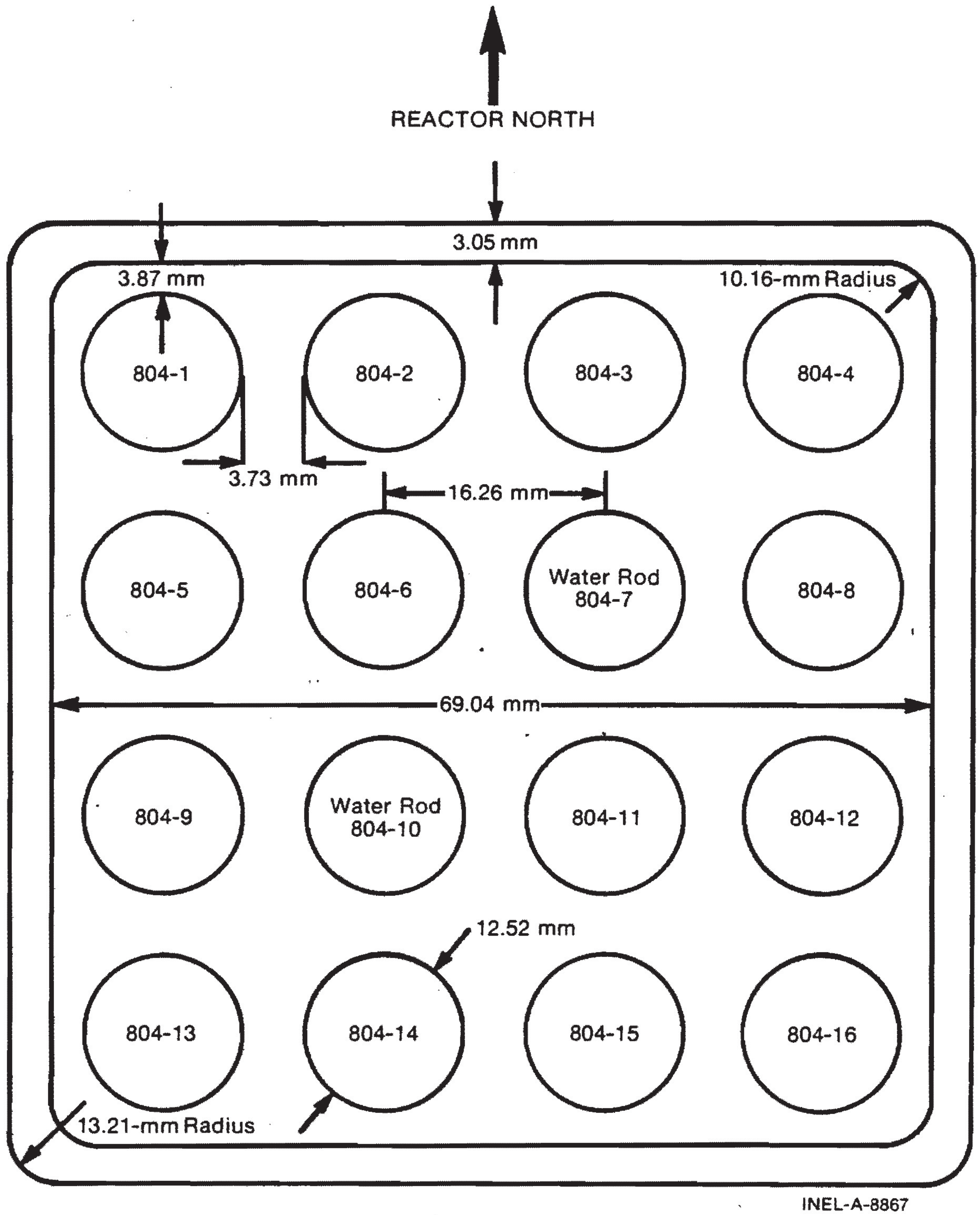

Figure 1 Flow channel dimensions and rod positions.

The upper section of the bundle support structure separates the inlet and outlet coolant flow of the PBF in-pile tube and supports the test train.

Consideration should be given to ease of disassembly of the test train in the hot cell or under water. The support structure design should be such that any disassembly which may become necessary in the PBF canal can be performed without significant schedule impact. Consideration should also be given to insure that the irradiated fuel rod assembly will fit into available shipping casks for transport to the hot cell.

4. INSTRUMENTATION

The instrumentation for this test will be designed to aid in determining fuel rod characteristics and failure mechanisms during an RIA transient. Table III lists the instrumentation requirements for the fuel rods, and Table IV lists the test train hardware instrumentation requirements. Figures 2 and 3 show the relative positions of the instrumentation for the test.

All fuel centerline and plenum thermocouple leads will have a gas tight seal. The cable below the seal will be leak checked prior to testing. Care should be taken to insure that attachment of the cladding thermocouples does not result in the loss of cladding integrity. It should be noted that the fast response time specified in Table III for the cladding thermocouples can be achieved using small diameter thermocouples with spaded junctions.

A reversible flow meter capable of measuring flow from 0.284 to 28.4 ℓ/s (4.5 to 450 gpm) in either direction should be employed at the inlet of the fuel bundle. A standard flow meter capable of measuring flow from 0.284 to 28.4 ℓ/s should also be installed at the bundle outlet. Two-phase flow is expected at the channel outlet following the transient, therefore, an alternate means of mass flow measurement will be necessary. A combination of a drag screen densitometer and the outlet flowmeter or an ultrasonic densitometer and the outlet flow meter should be capable of providing the mass flow measurement at the bundle outlet.

It will be necessary to fix the cladding LVDT cores to the upper end caps to insure that contact is not lost between the LVDT core and cladding. It is possible that the large acceleration of the cladding elongation will impart sufficient velocity to the LVDT core to create an over reading if it is not attached to the cladding. Separation of the fuel rod into two or more pieces will also be indicated more clearly if the LVDT core is attached directly to the upper end cap. Eight (0.51% cobalt - 99.49% aluminum) flux wires, each enclosed in a

TABLE III

MEASUREMENT REQUIREMENTS FOR TEST RIA 1-4 FUEL ROD INSTRUMENTATION

Measurement	Instrument	Instruments per Rod	Instrument Location [a]	Rod to be Instrumented	Required Instrument Range	Required Accuracy	Required Response Time(s)	Comments
Fuel centerline temperature	Thermocouple	1	790 $\pm$1 mm	804-2, 804-11, 804-13	500 to 2800 K	$\pm$ 100 K	0.05	Presently, centerline TCs have response times of 200 ms. This is development item.
Rod internal pressure	Pressure transducer	1	Top of fuel rod	804-2, 804-11, 804-13	0 to 6.9 MPa	$\pm$0.2 MPa	0.01	Should be calibrated at test coolant temperature.
Cladding surface temperature	Thermocouple	2	460 $\pm$ 1 mm or 790 $\pm$ 1 mm	804-2, 804-4, 804-6, 804-7, 804-11, 804-13, 804-15	300 to 2150 K	$\pm$25 K	0.01	Response time designates this as a development item. Suggest consideration of coaxial moly-zirc. TC with spaded junction or small diameter tungsten-rhenium TC with spaded junction. Electron beam welding attachment should be evaluated.
Plenum Temperature	Thermocouple	1	Upper Plenum of fuel rod	804-11, 804-2, 804-13	300 to 1000 K	$\pm$10 K	0.02	Will require development to meet response time requirement.
Pellet stack elongation	LVDT	1	Upper end of fuel rod	804-4, 804-6, 804-15	0 to 38 mm	$\pm$0.1 mm	0.003	Should be calibrated at 533 K

[a] All elevations are relative to the bottom of the fuel stack.

TABLE IV

MEASUREMENT REQUIREMENTS FOR TEST RIA 1-4 TEST TRAIN HARDWARE INSTRUMENTATION

Measurement	Instrument	Number Required	Instrument Location [a]	Required Instrument Range	Required Accuracy	Required Response Time (s)	Comments
Upper Support Structure							
Coolant Pressure	Pressure Transducers	1	Above Flow Channel Outlet	0 to 17.2 MPa	±0.03 MPa	---	To measure normal system pressure.
Support Structure							
Coolant Pressure	Pressure Transducers	2	Near Test Bundle Flow Inlet and Outlet	0 to 69 MPa	±3.5 MPa	1×10^{-4}	To measure large pressure pulses.
Coolant Pressure	Pressure Transducer	1	Bottom of Support Structure	0 to 17.2 MPa	±0.03 MPa	---	To measure normal system pressure.
Coolant Outlet Density	Drag Screen or UDO	1	Flow Channel Outlet	0 to 100% Quality	±5%	0.010	To measure bundle outlet void fraction.
Coolant Flowrate	Turbine Flowmeter	1	Flow Channel Outlet	0.28 to 28.4 ℓ/s	±2%	0.003	Should be calibrated over its range in vertical position.
Coolant Flowrate	Turbine Flowmeter, Reversible	1	Flow Channel Inlet	0.28 to 28.4 ℓ/s	±2%	0.003	Should be calibrated over its range in either direction with the orifice plate.
Coolant Inlet Temperature	Thermocouple	2	Rod Bundle Flow Inlet	300 to 600 K	±3 K	N/A	---
Coolant Outlet Temperature	Thermocouple	2	Rod Bundle Flow Outlet	300 to 600 K	±3 K	N/A	---
Flow Channel Temperature	Thermocouple	2	460 ± 1 mm	300 to 600 K	±3 K	0.20	Attach to flow channel by welding as with clad thermocouples.
Cladding Axial Strain	LVDT	3	Bottom End of Rods 804-4, 804-6, 804-15	-50 to 20 mm	±0.1 mm	0.0003	Should be calibrated at 300 and 550 K coolant temperature.
Neutron Flux	Flux Wires	8(100% Cobalt) 8(0.51% Cobalt)	Vertically Along Flow Channel	1.2×10^{15} nvt	---	---	0.51% cobalt wires for steady state operation; replaced with 100% cobalt wires for transient.
Relative Neutron Flux	Cobalt SPNDS	10	2 Vertical Columns of 5 Along Flow Channel at 0 and 180 degrees at 91 ±1, 274 ±1, 457 ±1, 640 ±1, and 823 ±1 mm.	1.6×10^{17} n/cm²·s	$\pm 3 \times 10^{15}$ n/cm²·s	0.002	Gamma and neutron sensitivity of each SPND must be measured. Must be at least 2 mm away from SPGDs.
Coolant Differential Temperature	Thermocouple Pairs	8 Pairs	At Inlet and Outlet of Rod Bundle	0 to 20 K	±0.1 K	N/A	To measure temperature rise of coolant in flow channel.
Relative Gamma Flux	Inconel or Zirconium SPGD	10	2 Vertical Columns of 5 Along Flow Channel at 0 and 180 degrees.	1×10^{12} R/hr (Transient)	$\pm 5 \times 10^{10}$ R/hr	0.002	Gamma sensitivity of each must be measured using reactor generated gamma spectrum. Must be at least 2 mm away from SPNDs. SPND should be 0.914 m long. Gamma sensitivity should at least 1×10^{-15} a/R/hr. Neutron sensitivity should be less than 1×10^{-22} a/nv.

[a] All elevations are relative to the bottom elevation of fuel stacks.

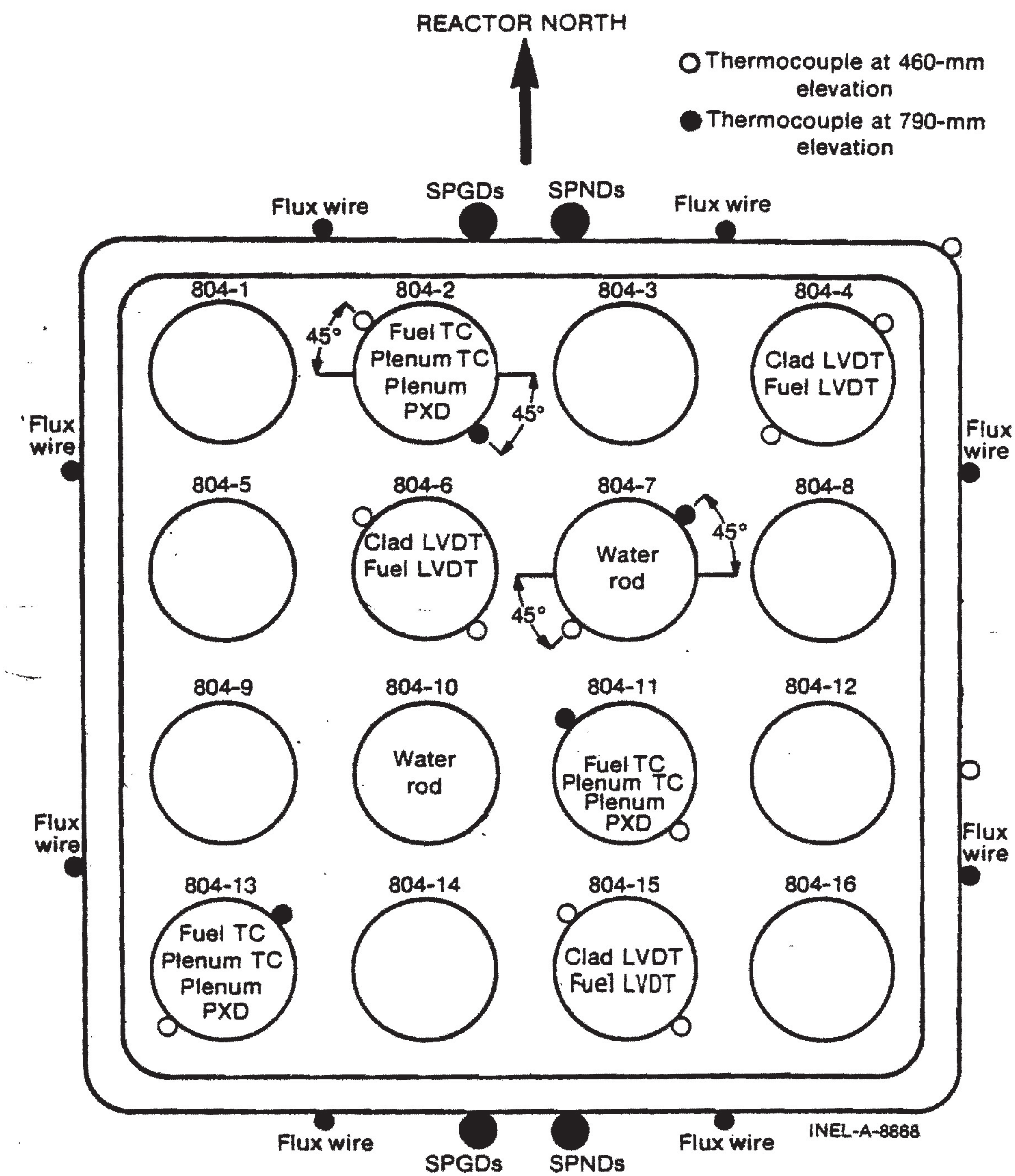

Figure 2 Fuel rod bundle instrumentation locations.

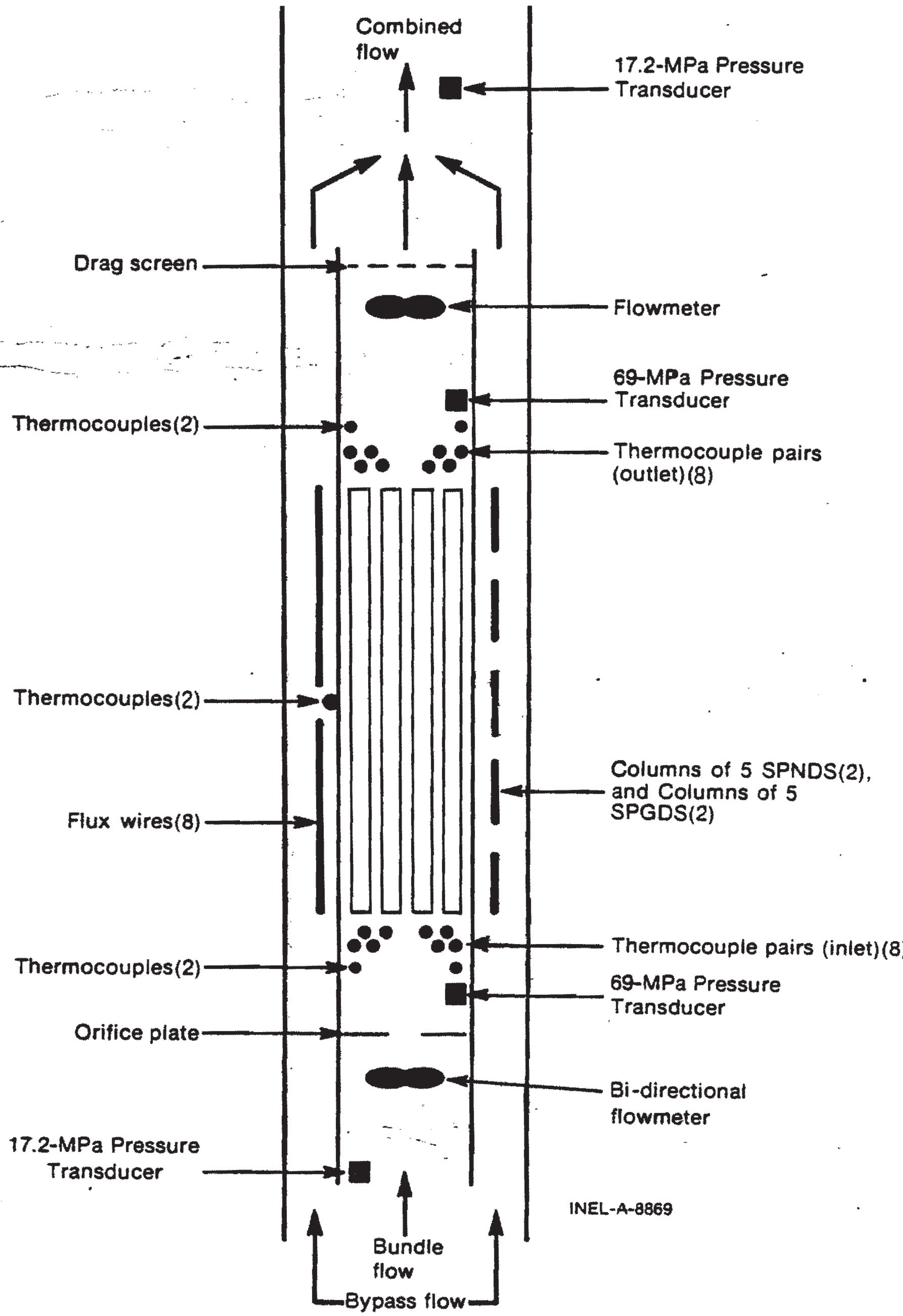

Figure 3 Schematic representation of the RIA 1-4 test train assembly showing relative instrumentation.

small diameter zircaloy tube, will be attached to the outer wall of the flow shroud as shown in Figure 2. The flux wires and zircaloy tubes will be removed after the steady state portion of the test is completed. Eight 100% cobalt wires similarly enclosed in zircaloy tubing will be installed in their place prior to the power burst. Attention should be given to making this flux wire exchange possible in the PBF canal. The as-built axial and lateral position of each flux wire holder will be measured. The flux wires will extend over the active fuel length of the rods; the bottom of the flux wires aligned with the bottom of the active fuel stack.

The eight upper differential thermocouples used to measure the coolant temperature rise should be routed from the top of the flow channel rather than penetrating the flow channel wall in order to minimize measurement errors.

5. REACTOR OPERATION

Detailed operation specifications will be provided in the Test RIA 1-4 Experiment Operation Specification Document. The test sequence will begin with about 24 hours of steady state power operation to condition the fuel rods and increase the test rod power. The steady state operation will be done with a shroud coolant flow of 15.8 ℓ/s, coolant inlet temperature of 553 K, and coolant pressure of 7.29 MPa. These conditions, as calculated, will result in single-phase coolant at the shroud outlet. In addition, the coolant density at the active fuel midplane will be about the same as the initial coolant density for the power burst. Since the power burst energy deposition is essentially an adiabatic process; i.e. nearly all of the energy is deposited in the fuel before heat is transferred to the coolant, the power calibration will be conducted with single-phase coolant conditions. The reactor will be shut down for replacement of the flux wire installed on the flow shroud and the 100% cobalt flux wires located in the core. After the flux wire exchange, the test rods will be exposed to the minimum possible power operation before the power burst. The power burst will be conducted from BWR hot-startup conditions (6.45 MPa, 553 K and 1360 cm^3/s). The reactor transient period required to perform Test RIA 1-4 will be determined on the basis of PBF Lead Rod and previous RIA test results.

6. POSTTEST OPERATION

The cladding thermocouple and internal pressure transducer leads should be cut about 25 to 50 mm above the upper end plug. The fuel and plenum thermocouple leads should be cut 25 to 50 mm above the gas seal in each lead.

Closure plugs should be installed on the upper and lower ends of the flow shroud after removal from the test assembly to prevent further loss of material from failed fuel rods during handling and shipment to the hot cell. It is anticipated that the fuel rod cladding will be heavily-oxidized and in an embrittled condition. Therefore, posttest handling, shipment, and storage should be performed carefully to minimize the possibility of further fuel rod damage.