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ATOMIC ENERGY
OF CANADA LIMITED

L'ÉNERGIE ATOMIQUE
DU CANADA LIMITÉE

AN OVERVIEW OF THERMALHYDRAULICS R&D
FOR SLOWPOKE HEATING REACTORS

UN APERÇU DE LA RD EN THERMOHYDRAULIQUE
POUR LES RÉACTEURS DE CHAUFFAGE SLOWPOKE

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Chalk River, Ontario

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AN OVERVIEW OF THERMALHYDRAULICS R&D FOR SLOWPOKE HEATING REACTORS

by

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RÉSUMÉ

L'EACL est en train de démontrer l'utilisation des réacteurs piscines d'une puissance de 10 MW pour la production d'eau chaude à environ 90°C. Le but central initial de la mise au point est la prévision d'une source d'eau chaude pour les réseaux de chauffage d'établissements et de localités. Les mises au point en cours sont destinées à étendre les applications à la production de l'électricité et aux procédés industriels tels que la désalination et les besoins agricoles. Le concept de réacteur est basé sur le réacteur de recherche Slowpoke-2 dont huit unités sont exploitées avec succès au Canada et à l'étranger.

L'écoulement du circuit primaire est entraîné par convection naturelle, l'eau chauffée produite par le coeur du réacteur situé près du fond de la piscine étant envoyée par des tuyaux à des échangeurs de chaleur à faible perte de charge situés dans la partie supérieure de la piscine. Comme le volume de la piscine est relativement grand, le temps de transit du fluide autour du circuit est long et assure ainsi que la réponse du réacteur à toutes les transitoires normales soit très lente.

Pour étudier les aspects thermohydrauliques de ce type de réacteur, dont son comportement dans des conditions extrêmes, on a conçu et construit une boucle à écoulement par convection naturelle chauffée électriquement. Le coeur de la boucle comporte une grappe de barres qui reproduit avec précision un quart du coeur du Réacteur de Démonstration SLOWPOKE de 2 MW actuellement éprouvé à l'Établissement de Recherches Nucléaires de Whiteshell.

Avec cette boucle, on a effectué des mesures de distribution de pression, de température, de vitesse et de vide à l'état sous-saturé dans le coeur simulé par diverses techniques intrusives et non intrusives. En outre, on a étudié le comportement de l'écoulement monophasique et diphasique du circuit.

Dans la présente communication, on donne des exemples de diverses mesures effectuées dans le coeur et on fait des comparaisons entre le comportement mesuré du circuit et celui prédit par les divers programmes de calcul de l'écoulement en régime stationnaire et transitoire.

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Abstract

AECL is currently demonstrating the use of pool-type reactors of up to 10 MW output to produce hot water at about $90^0 C$. The initial focus for the development is the provision of a source of hot water for institutional and municipal heating networks. Ongoing developments are designed to broaden the applications to electricity generation and industrial processes such as desalination and agricultural needs. The reactor concept is based on the Slowpoke-2 research reactor, eight of which are successfully operating in Canada and abroad.

The primary-circuit flow is driven by natural convection, with the heated water, produced by the reactor core near the bottom of the pool, being ducted to low-pressure-drop heat exchangers in the upper part of the pool. As the pool volume is relatively large, the fluid transit time around the circuit is long, ensuring that the reactor response to all normal transients is extremely slow.

To investigate thermalhydraulics aspects of the reactor design, including its behaviour under extreme conditions, an electrically heated, natural-convection loop was designed and constructed. The core of the loop consists of a rod bundle that is a precise reproduction of one quarter of the core of the 2-MW SLOWPOKE Demonstration Reactor presently being tested at the Whiteshell Nuclear Research Establishment.

With this loop, measurements of the distribution of pressure, temperature, velocity and subcooled void have been made in the simulated core, via a variety of intrusive and non-intrusive techniques. In addition, both the single- and two-phase behaviour of the system have been studied.

This paper gives examples of the various in-core measurements made and also makes comparisons between the measured system behaviour and that predicted by the various steady-state and transient computer codes.

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AN OVERVIEW OF THERMALHYDRAULICS R&D FOR SLOWPOKE HEATING REACTORS

1.   INTRODUCTION

AECL is currently developing a 10-MW nuclear heat source, called the SLOWPOKE Energy System, suitable for heating large buildings or groups of buildings [1,2]. The design is patterned after the 20-kW SLOWPOKE-2 research reactor, eight of which currently operate within Canadian cities and abroad [3]. The SLOWPOKE research reactors have a high degree of inherent safety arising from the negative temperature and void coefficients of the core and the limited maximum excess reactivity, and they are licenced for unattended operation.

As a preliminary step to the construction of a commercial 10-MW unit, a 2-MW prototype, which embodies all of the salient features of the 10-MW design, has been built and is presently being tested. The basic reactor design is shown in Figure 1. The reactor core is located near the bottom of a pool, and the hot water that is produced is ducted to two low-pressure-drop heat exchangers situated in the upper part of the pool. The outlet water from the heat exchangers is returned to the pool. The use of a low-pressure-drop design for the heat exchangers and the core enable the heat to be removed from the core by natural convection. This eliminates the need for pumps in the primary circuit.

To study the various thermalhydraulic aspects of the reactor design, a large electrically heated, natural-convection loop was designed and constructed. The electrically heated core of the loop is geometrically identical to one of the four nuclear-fuelled rod bundles used in the 2-MW prototype reactor.

This report discusses the overall thermalhydraulic program associated with the primary circuit of the reactor. It concentrates on the experimental measurements taken with the large loop and the application of these measurements to the development and validation of computer codes. For convenience of presentation, the report has been broadly split into the thermalhydraulic behaviour of the core and the overall system thermalhydraulics. Currently the program involves measurements on both the electrically heated loop and the SLOWPOKE Demonstration Reactor. These measurements will be used for the final validation of the thermalhydraulic computer codes and will allow confident extrapolation to a larger 10-MW design.

2.   TEST FACILITIES

The thermalhydraulic test program has followed traditional evolution from bench-top scale experiments through large-scale experiments to prototype demonstration. The early bench-top tests were concerned primarily with instrumentation development and initial measurements of subcooled void and will not be discussed here.

The majority of the thermalhydraulic work to date has been done with the large test loop which is shown schematically in Figure 2. This loop was designed to simulate the main thermalhydraulic features of the SLOWPOKE Energy System heating reactor. To do this, the major pressure-drop components in the circuit, the rod bundle test section and the heat exchanger, and the elevations that determine the gravity term in the momentum equation, were scaled 1:1. As a result, the circuit temperature and void profiles, and core flow velocities in the test loop and the reactor, are thermalhydraulically similar.

The nominal full power of the test loop was 500 kW. A 1.2-MW, continuously variable, DC electrical power supply was used that allowed tests to be done at the equivalent of more than twice reactor full power. Heat rejection was through a low-pressure-drop heat exchanger of the same design as that used in the SLOWPOKE Demonstration Reactor, with the secondary side connected to the building water supply. Adjustment of the secondary-side cooling-water temperature or flow was used to vary the loop operating temperature.

The test bundle was a 7X7 array of heater pins, 50-cm long, with the electrical power being distributed to the pins through an end plate at each end of the bundle. These end plates have the same flow cross section and shape as the end plates used in the nuclear fuel bundles. Two bundle-flux profiles have been used, uniform and truncated cosine. The uniform bundle has a uniform heat flux both axially and radially while the cosine bundle has a cosine profile axially, obtained by varying the tube wall thickness along the length, and a depressed radial flux simulating the neutron flux in the prototype reactor. The profiles for the truncated cosine and the radial depression were obtained from reactor physics calculations made for the core of the SLOWPOKE Demonstration Reactor. The heater bundle is contained in a thick-wall, transparent, polycarbonate test section which allows visual observation of the two-phase phenomena within the bundle. The inside walls of the test section are machined to accurately represent the detailed geometry surrounding the bundles in the prototype reactor.

Above the test section, the hot riser duct is constructed of 15-cm diameter Pyrex glass, which allows visual observation of two-phase phenomena occurring beyond the normal operating limits. The remainder of the circuit is made of 20-cm diameter, stainless-steel tubing. The surge tank next to the heat exchanger provides a free water surface that simulates the reactor pool surface.

Temperatures at the various locations around the loop, as shown in Figure 2, were measured by thermocouples and platinum resistance thermometers. Friction and obstruction pressure drops were measured across major sections of the loop by sensitive pressure transducers. Due to the low liquid velocities, and the need for negligible pressure drop, conventional flow measuring equipment could not be used. For steady-state and slowly varying transients, the flow was calculated by a heat balance across the test section, which in turn had been verified by laser Doppler anemometry and time-of-flight measurements. For rapidly varying transients, clamp-on ultrasonic flowmeters, which had been calibrated against heat balance flows, were used.

Signals from the loop instrumentation were monitored on a Perkin Elmer 7/32 computer. This computer performed the calculations required to output the data in engineering units and also stored the scan data on magnetic tape when required. For transient tests the computer scanned the instrument signals automatically every 3.4 seconds and stored all data on magnetic tape.

3.   COMPUTER CODES

Besides obtaining an overall understanding of the thermalhydraulic phenomena, a major reason for taking experimental measurements is to provide data for the development and validation of computer models. As these models will be referred to frequently later in   ? paper, a brief description of the main thermalhydraulic computer codes   given here.

3.1   ASSERT: An advanced subchannel code used to model steady-state and transient single- and two-phase flow through rod bundles [4]. It is based on the advanced drift flux which allows for unequal velocity and unequal temperature for the two phases. It was developed primarily for flow through horizontal bundles but can be used for vertical bundles. To be able to calculate the effect of gravity on phase distribution in horizontal bundles, the buoyancy effects on the lateral or radial velocities are modelled by means of drift flux relations.

3.2   STRETCH: An unpublished steady-state program developed to provide engineering data for loop and reactor design. It divides the circuit into sequential nodes with the absolute pressure at the first node calculated from the pool head. Using an initially assumed flow, the pressure and temperature at all subsequent nodes are calculated from the elevation change, various pressure drops through the node, and the heat input. The pressure at the outlet of the last node is compared to the original inlet pressure. Any difference results in a flow adjustment with iteration continuing until the pressure difference is less than one pascal.

3.3   SPORTS: A one-dimensional, non-linear code that calculates steady-state and transient response of a single- or two-phase system [5]. It can simulate the effects of subcooled boiling and neutron kinetics. Its principal advantages are: it is not restricted to small time steps, the full equation of state is satisfied at each point in the computational grid, and it is efficient and economical to run.

3.4   RIGSIM: A transient code intended for the simulation of slow transients, i.e. those with time constants of 10 seconds or more [6]. It is a real-time, dynamic simulation of the test loop and has been implemented on a hybrid (analogue/digital) computer and operates in real time.

4.   CORE THERMALHYDRAULIC MEASUREMENTS

4.1   Core Pressure Drop

Accurate knowledge of the core axial pressure-drop is essential for modelling
both the system behaviour and the core local phenomena. To obtain these
data, a one-quarter section of the core was tested in a hydraulic loop and
detailed axial-pressure measurements taken with a travelling probe. An
example of these measurements is given in Figure 3. There is a pressure
reduction and partial recovery at the core end plates, and the unrecovered
loss is used to calculate the obstruction loss factors required for the
system models. Also shown in Figure 3 is the predicted pressure gradient
from the ASSERT code. The profile is well predicted with only slight
deviation in the frictional loss along the core length.

4.2   Radial Temperature Distribution

The coolant-temperature profile within the core will depend on the core-power
profiles and the details of the flow within the core. The temperature
profiles were measured in one experiment by using three transverse,
travelling thermocouple probes in the subchannels between the rows of rods,
as shown in Figure 4. The thermocouples were 0.5 mm in diameter and the tips
were bent downwards to point into the flow, so they caused no flow disturbance
upstream of the measurement location. A typical traverse across the bundle
is given in Figure 5. It shows that the temperatures are generally higher at
the sides of the bundle, which would be expected as there are higher power
rods adjacent to the walls, and also because the wall friction will give a
lower fluid velocity.

The thermocouple probe could also be rotated and thus the tip could be swung
across the subchannel row and into contact with the heater pins. This
enabled very detailed measurements to be made close to the pins, as is shown
in the local contour map in Figure 6. It is apparent from the contours that
the thermal boundary layer adjacent to the heater pins is thick, which would
be expected in a relatively low-velocity, natural-convection system.

4.3   Local Liquid Velocities

Local liquid velocities within the bundle were measured by laser Doppler
anemometry (LDA). In this method, a laser beam is split into two, and
particles moving in the fluid are illuminated at the intersection of these
two beams. From the time taken for the particle to cross a given number of
interference fringes at the intersection point, the velocity is calculated.
At each location the velocity was sampled 256 times and a mean and standard
deviation calculated. A typical sample of the data is shown in Figure 7. A
total of about 190 separate measurements within the bundle were made.

To obtain the overall bundle profiles, measurements were taken between each
row of rods at the subchannel center and at the location of the gaps between
the rods. These velocity data, when plotted as a function of position,
allowed interpolation to obtain the velocities at intermediate points along
the lines, and also the near-wall velocities where measurement was difficult.

To obtain detailed velocity profiles within the subchannels, detailed measurements were made in three subchannels. Again, plotting them as a function of position allowed the interpolation of intermediate velocities. These point velocities were then converted into a contour velocity map as is shown for a section of the bundle in Figure 8.

As Figure 8 shows, the wall exerts a strong influence on the velocities in the wall subchannels and also has a significant effect on the first row of full subchannels. The wall effect is still measurable in the second row of full subchannels.

Initial velocity measurements were made with the bundle at power. A problem that arose was that meaningful data could not be obtained from the central subchannels of the bundle due to diffraction effects in the heated water, which caused drastic reductions in the data acceptance rate. This was solved by taking measurements isothermally under pumped flow conditions rather than natural convection.

4.4   Subcooled Void

At high powers and temperatures, subcooled void can be visually observed in the bundle. Two methods that we have used for quantifying the void are a fiber-optic probe for local measurements and a scanning γ-ray densitometer for chordal and bundle average measurements.

The fiber-optic probe method depends on the change in refractive index between steam and water at the tip of a 100-μm optical fiber. A 15-mW helium-neon laser provides the light source, and the intensity of the light reflected back from the probe tip depends on whether the tip is in steam or water. A photodiode converts the light intensity into an electrical signal which is then processed to give the void fraction.

In one experiment, this probe was traversed across the bundle between two rows of rods, as is shown in Figure 9. The rods at the beginning and end of the traverse line were at a higher power than the rods in the center, and this has the effect of producing significantly more void at the two ends of the traverse. Immediately adjacent to the walls of the test section, the void drops sharply. The general void concentration at the ends of the traverse line was also visually obvious during the tests.

The second method of measuring void was with a movable γ-ray densitometer. This system consists of a 20-Ci Cs137 source, a detector, and a multi-channel analyser. The source and detector and the associated collimating shielding are mounted on worm-driven slides which in turn are mounted on a vertically travelling table, as is shown in Figure 10. The source and detector are collimated so that the beam between them is only 0.5 mm wide. This beam is alternately swept back and forth across the bundle by the worm gear with the table being moved vertically after each pass.

The output signal from the detector goes to the multi-channel analyser (MCA). There, the signal pulses are stored in bins, with the bin in use being incremented sequentially by a clock in the MCA unit. As the source/detector is scanning across the core, each bin will therefore represent a particular

narrow section of the core. A typical scanning arrangement used consists : ¯ horizontal passes across the core at 6 different axial elevations.

The number of counts stored in each bin during the experimental run is compared to the counts in the same bin when the test section is full of water (0% void) and empty (100% void) and the chordal average void for that bin location calculated. A typical example of these data is shown in Figure 11 where all the bin void values have been used to construct a contour map describing the axial and azimuthal void. In the particular example in Figure 11, the bundle had a cosine axial-heat-flux profile which results in the maximum void occurring between the mid-plane and the outlet of the bundle. In addition, because of the velocity profile across the bundle, the fluid temperature is higher near the walls of the test section, giving higher void in these areas.

5.   SYSTEM THERMALHYDRAULICS MEASUREMENTS

5.1   Steady-State, One- and Two-phase

Single-phase, steady-state runs at various powers were done to obtain measurements of the various obstruction pressure-loss coefficients (K-factors) around the circuit. The initial values for the K-factors in the computer codes were estimated from literature values for various fittings. These K-factors were then adjusted to fit the measured pressure drops from an experimental run. With these final K-factor values, the single-phase flow vs power characteristics of the loop were predicted using the STRETCH computer code. The predictions from STRETCH are compared to experimental data and to the SPORTS predictions of the experimental data points in Figure 12. The agreement between the codes and the experiments is excellent.

To check the code agreement under two-phase conditions, loop measurements were made where the temperature at the top of the riser pipe was allowed to exceed saturation, resulting in two-phase flow at that point. The vapour was allowed to bleed off through the non-condensible gas vent, which prevented vapour locking in the heat exchanger. Although two-phase flow was present in the circuit, it did not extend significantly into the vertical part of the riser pipe and thus had no effect on the gravity head. The resultant flow was steady. In Figure 13 the experimental data are compared to the SPORTS and STRETCH codes. It shows that the codes predict the behaviour of the loop under steady-state, two-phase conditions adequately.

5.2   Single-phase Transients

Two types of single-phase transient tests were done to provide data for the validation of our system computer codes. In the first, a step change was made to the heater bundle power, and in the second a step change was made in the temperature on the secondary-side cooling water, with, in each case, the loop response being observed. The procedure used was to allow the loop to stabilize for 30 to 40 minutes and then put the data aquisition on automatic 15-second scans. The step change was then done and the loop allowed to come to its new equilibrium.

Examples of two runs are given in Figures 14a and 14b [6]. In 14a the

response to a step up in power from 400 to 600 kW is shown. The test-section outlet temperature initially increases rapidly, which, as this provides more gravity-driving head, causes the flow to increase, and thus causes a decrease in the outlet temperature. The mass flow continues to increase for about 40 seconds until the warmer water enters the heat exchanger, and the water density in the downcomer side decreases.

The experimental transients shown in Figure 14 were predicted by the RIGSIM model, and it can be seen that the agreement is excellent. In Figure 14b the response to a change in the temperature of the secondary-side cooling water is shown. Because of the thermal inertia of the secondary-side water preheater, it was not possible to get such a sharp step as with the power change. However, the agreement is still very good.

5.3   Two-phase Transients

During two-phase transient testing, two types of two-phase instabilities were observed [7]. The first type of instability was obtained during a direct extension of the two-phase, steady-state tests described above. A small increase in the loop power increases the core outlet temperature slightly, which drives the onset of flashing to below the horizontal section of the riser pipe and into the vertical part. When this occurs, there is a decrease in the fluid density in the top part of the vertical riser and a consequent increase in the total flow rate. As the inlet temperature to the core is essentially fixed (by the surge tank outlet temperature), the flow increase during flashing in the vertical portion of the pipe leads to a reduction in core outlet temperature. This slightly cooler water continues to fill the riser pipe until it displaces the flashing liquid at the top. The temperature of the water column in the riser pipe is now lower than that required to maintain equilibrium flow and the flow rate reaches a minimum. As there is a fixed power input to the core, the outlet temperature from the core is now well above the time-averaged value. When this hot water now nears the top of the pipe, flashing occurs again with a corresponding flow increase, thus repeating the cycle. Except for the initial development stage, this oscillatory behaviour is non-divergent and reaches a limit cycle.

The experimental data for one flashing run are shown, superimposed on the SPORTS predictions for that run, in Figure 15. The SPORTS prediction was produced by initially obtaining a steady-state solution and then switching on the transient terms in the code. SPORTS predicts that the oscillations are indeed a limit cycle, which agrees with the experimentally observed behaviour. The shapes of the curves from the simulation and the experiment are remarkably similar, both having a period of about 175 seconds.

A second type of two-phase instability can occur at high powers. High core power combined with high inlet temperatures results in a significant amount of subcooled void being generated in the core while the remainder of the circuit is still operating in single phase. Under these conditions the flow will be stable, providing the average core void remains below 6 to 8%. If this value is exceeded, the flow becomes unstable and oscillates with a frequency of about 2 Hz. The oscillations diverge rapidly and can result in flow reversal and large voids being generated in the core. Under these conditions, the core power was tripped to prevent dryout in the core or

breakage of the glass pipe from the violent flow surges. When these high
power conditions were simulated using SPORTS, the code predicted that the
system goes unstable at similar core-void values.

5.4   Application of Experimental Data

The two major applications of these data are in licensing and operations.
Licensing is primarily addressed through computer codes which are validated
against the experimental data. For operations, a reactor operator requires
to know the safe operating regions and thermalhydraulic limits for the
reactor.

The normal operating region for the Slowpoke Demonstration Reactor is bounded
by a core outlet temperature of 90^0C and a reactor power of 2 MW, with reactor
operation being allowed, providing neither of these limits is exceeded. The
safe operating envelope is considerably larger than the normal operating
envelope, exceeding these boundaries by a considerable amount. If the 90^0C
core-outlet temperature is exceeded at constant power, a change in reactor
characteristics occurs at 103^0C when coolant flashing starts at the inlet to
the heat exchangers. This phenomenon is described in section 5.3 above, with
limit-cycle flow oscillations occurring. Its occurrence in the reactor is
prevented by core-outlet coolant overtemperature trips.

A second limit of safe operation is reached for a combination of high core-
outlet temperature and a reactor power considerably above 2 MW. This is the
second type of two-phase instability described in section 5.3 and results in
rapidly diverging flow oscillations. Reactor operation to this limit is
prevented by both the overtemperature and the overpower trips. In addition,
as this limit is approached, the increase in the subcooled void present in
the core provides a strong, negative, reactivity-feedback effect.

The third limit is not thermalhydraulic but the formation of fuel defects
caused by gross overpowering of the fuel. This is protected against by
overpower trips.

6.   CONCLUSIONS

The thermalhydraulics program in support of the SLOWPOKE heating reactors
entailed the construction and operation of a large, electrically heated loop
simulating the reactors and the development and validation of both steady-
state and transient computer codes.

Examples of the many types of experimental measurements made on this test
loop have been given. These measurements ensure a complete understanding of
the thermalhydraulics and in addition provide design and code validation
data. They cover both the details of the in-core thermalhydraulics and also
the system behaviour.

The experimental measurements required the use of a variety of conventional
and unconventional measuring techniques. System measurements were normally
made with standard temperature, pressure and flow instrumentation. Detailed
measurements within the heater bundle required the use of less conventional
instrumentation. Examples of these are a scanning γ-ray densitometer and

traversing fiber-optic probes for measuring subcooled void and its
distribution, and traversing laser Doppler anemometry for local and
integrated flow velocity.

The computer codes used to design and predict the behaviour of the SLOWPOKE
heating reactors were discussed as were their validation against experimental
measurements for both steady-state and transient single- and two-phase flow.

The current stage in the test program involves the use of the 2-MW SLOWPOKE
Demonstration Reactor presently undergoing testing at the Whiteshell Nuclear
Research Establishment. Experimental data on the reactor behaviour during
steady operation and transients will be used for further validation of our
computer codes. This final validation step will allow us to confidently
extrapolate to commercial-size units.

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LID

HEAT
EXCHANGER

10 metres

WATER

NATURAL
CIRCULATION

CORE
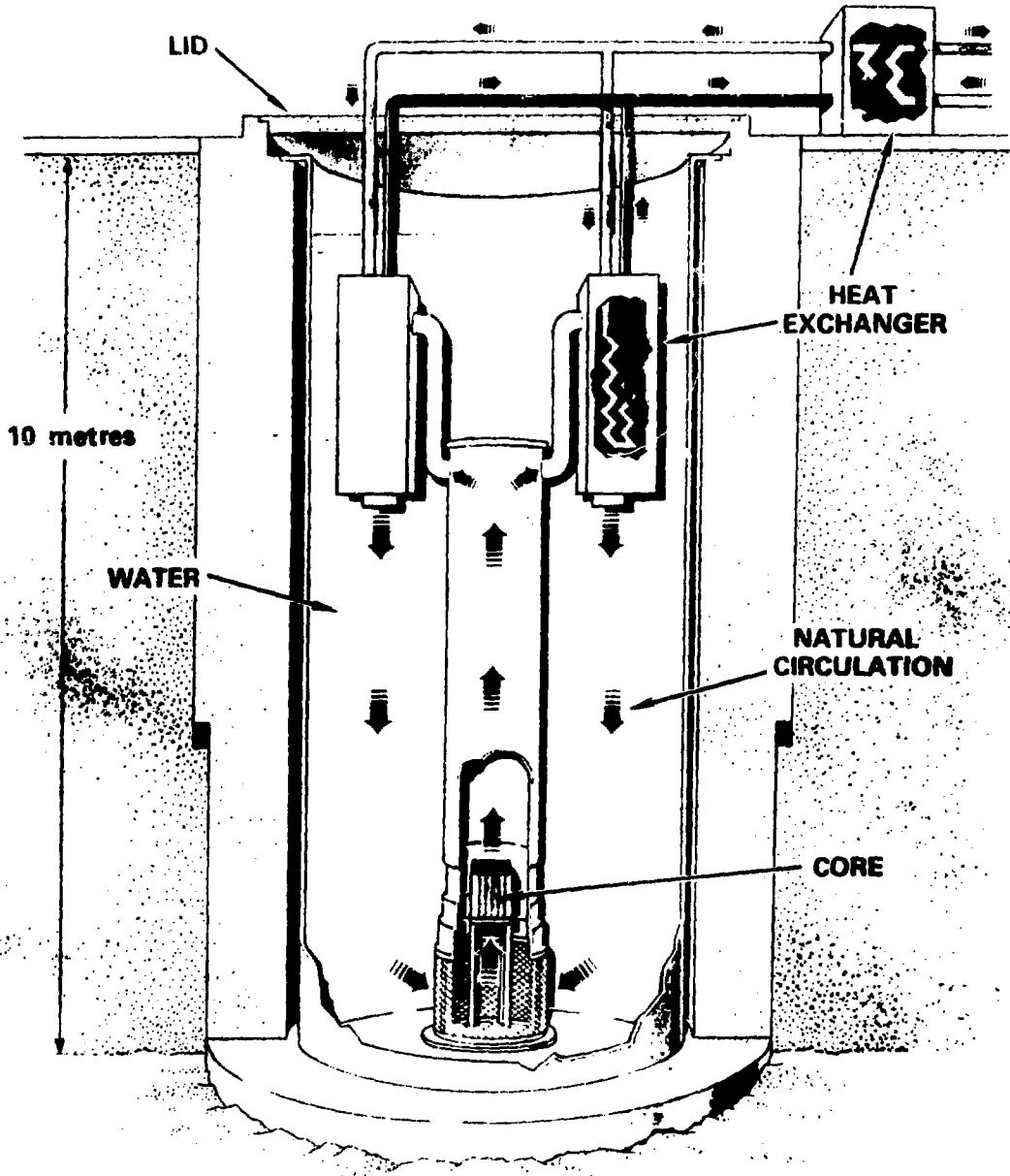
FIGURE I

SLOWPOKE ENERGY SYSTEM
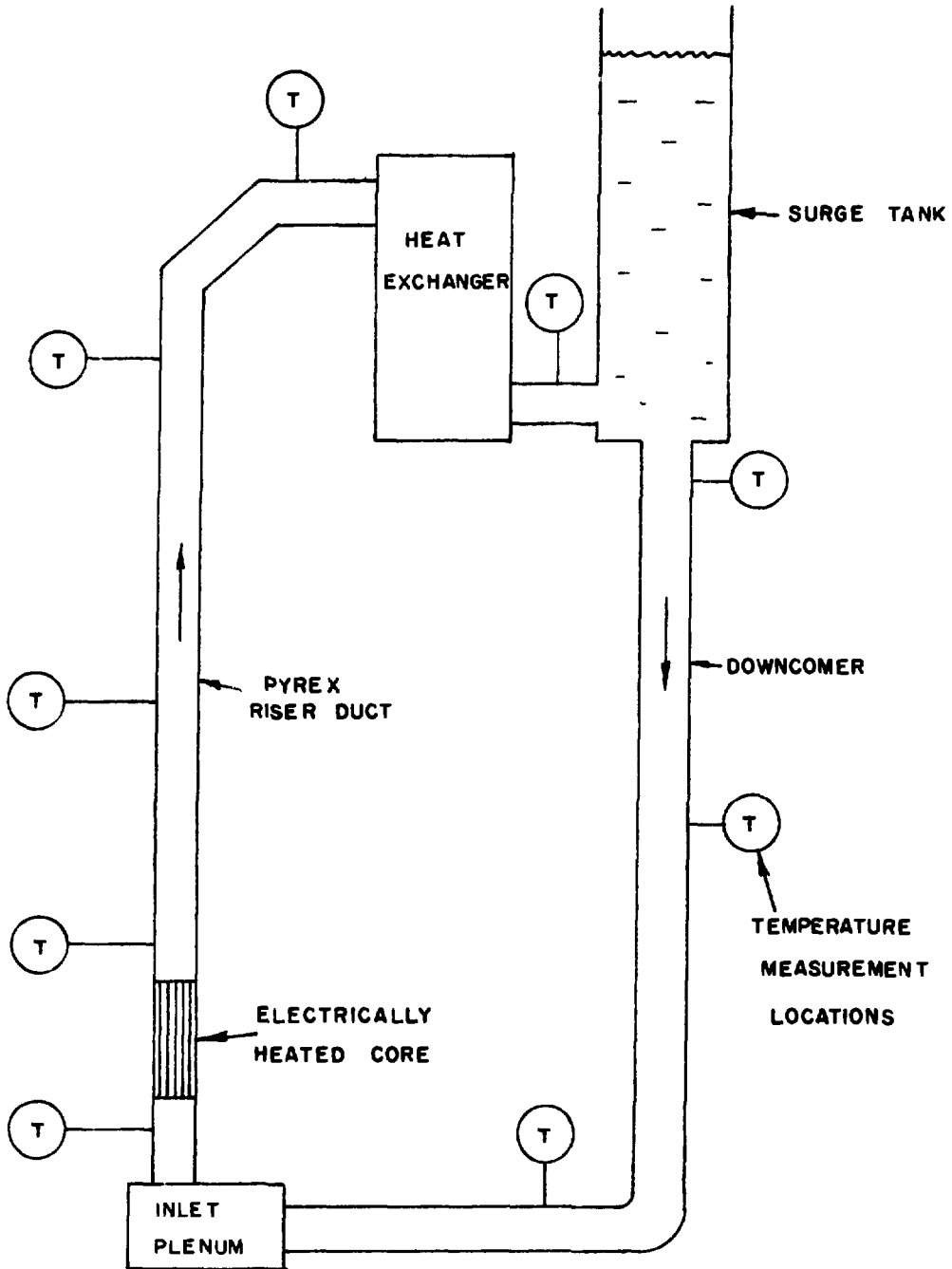
FIGURE 2

LOOP SCHEMATIC
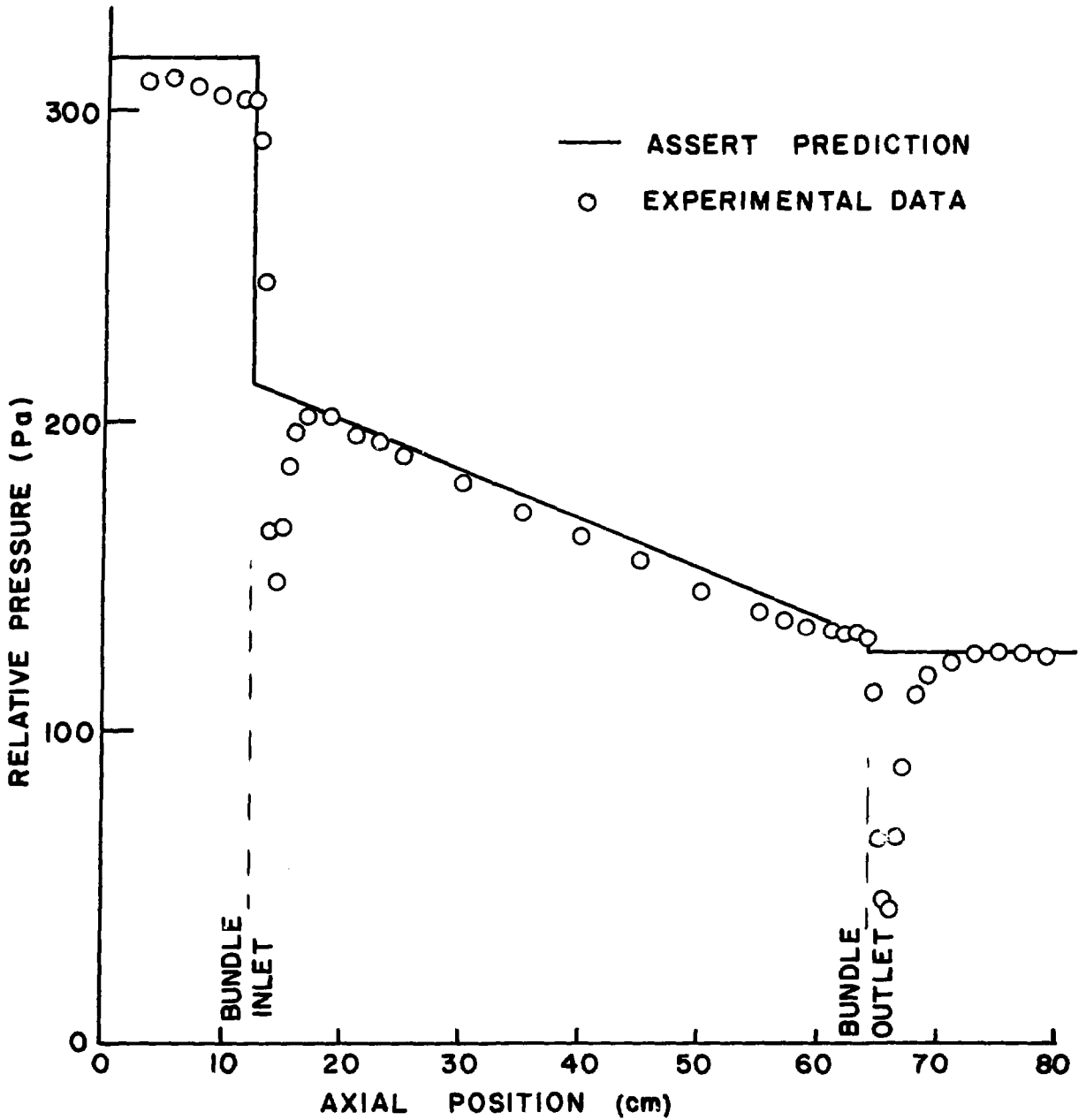
FIGURE 3

COMPARISON OF PREDICTED AND MEASURED
AXIAL PRESSURE PROFILES

THERMOCOUPLE

TRAVERSE LINES

50mm

7mm

0·5mm DIAMETER
Cr/Al THERMOCOUPLE

TEST SECTION
WALL

FLOW
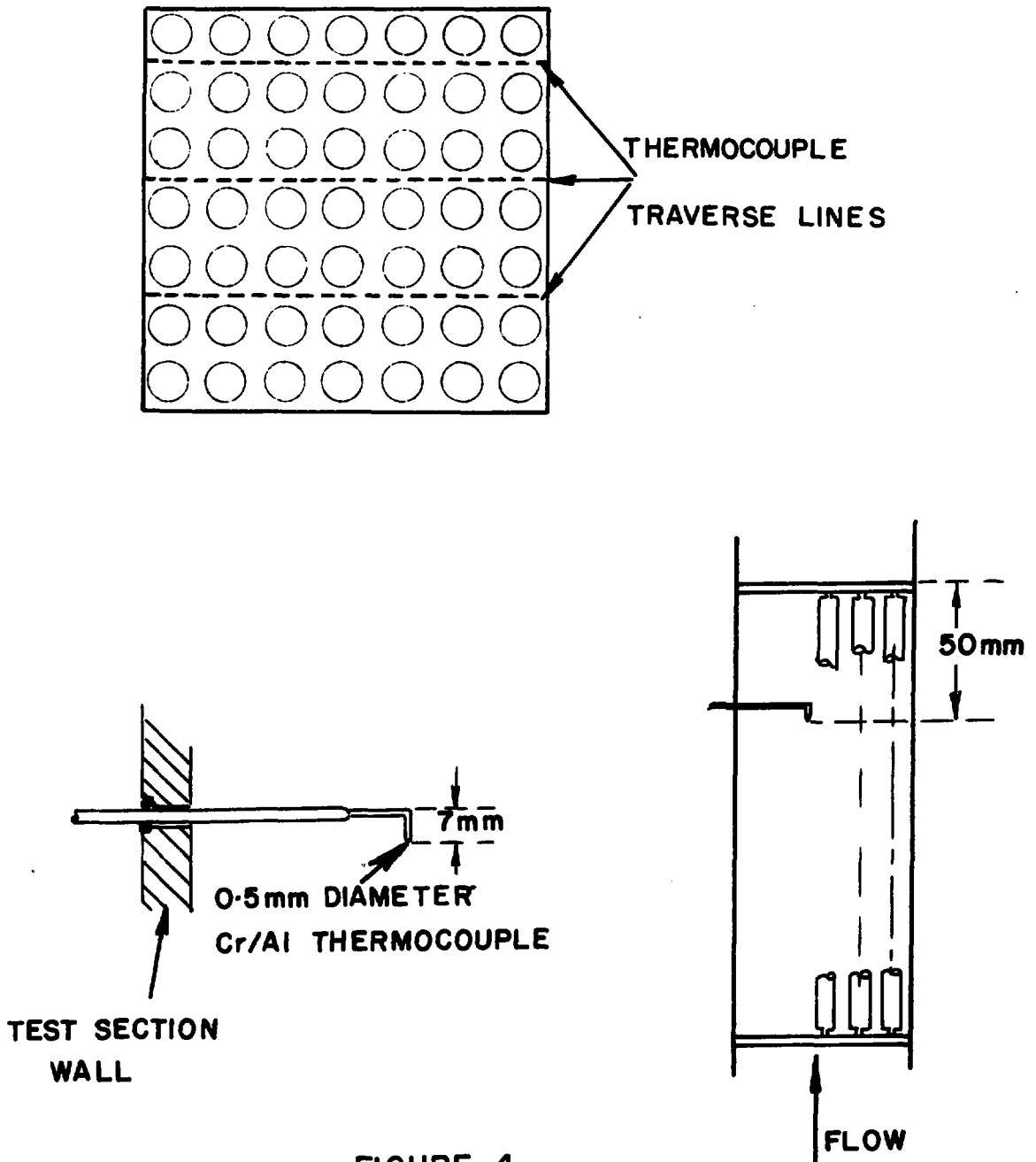
FIGURE 4

TRAVELLING THERMOCOUPLE
DESIGN AND LOCATIONS

TRAVERSE LINE
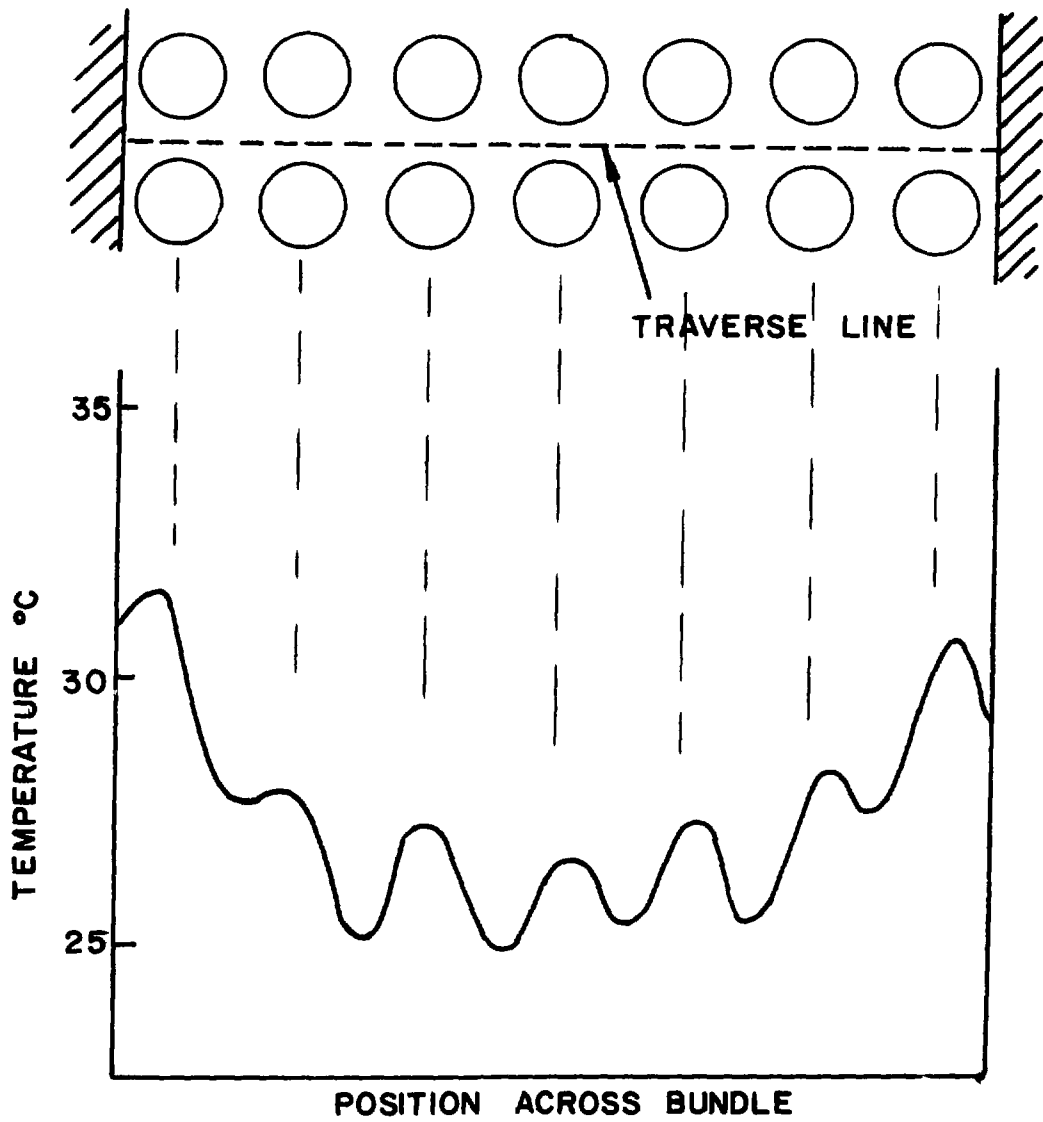
FIGURE 5

SINGLE PHASE COOLANT

TEMPERATURE PROFILE

HEATER RODS

TEST SECTION
WALL

TEMPERATURES °C
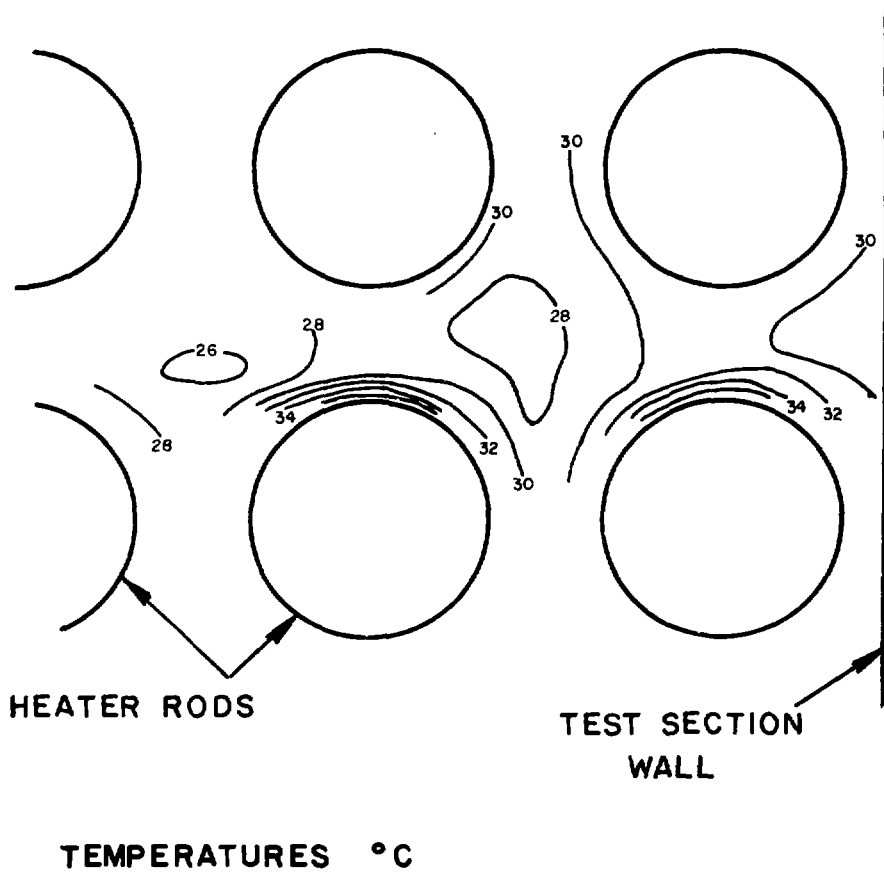
FIGURE 6

COOLANT ISOTHERMS WITHIN

TEST BUNDLE
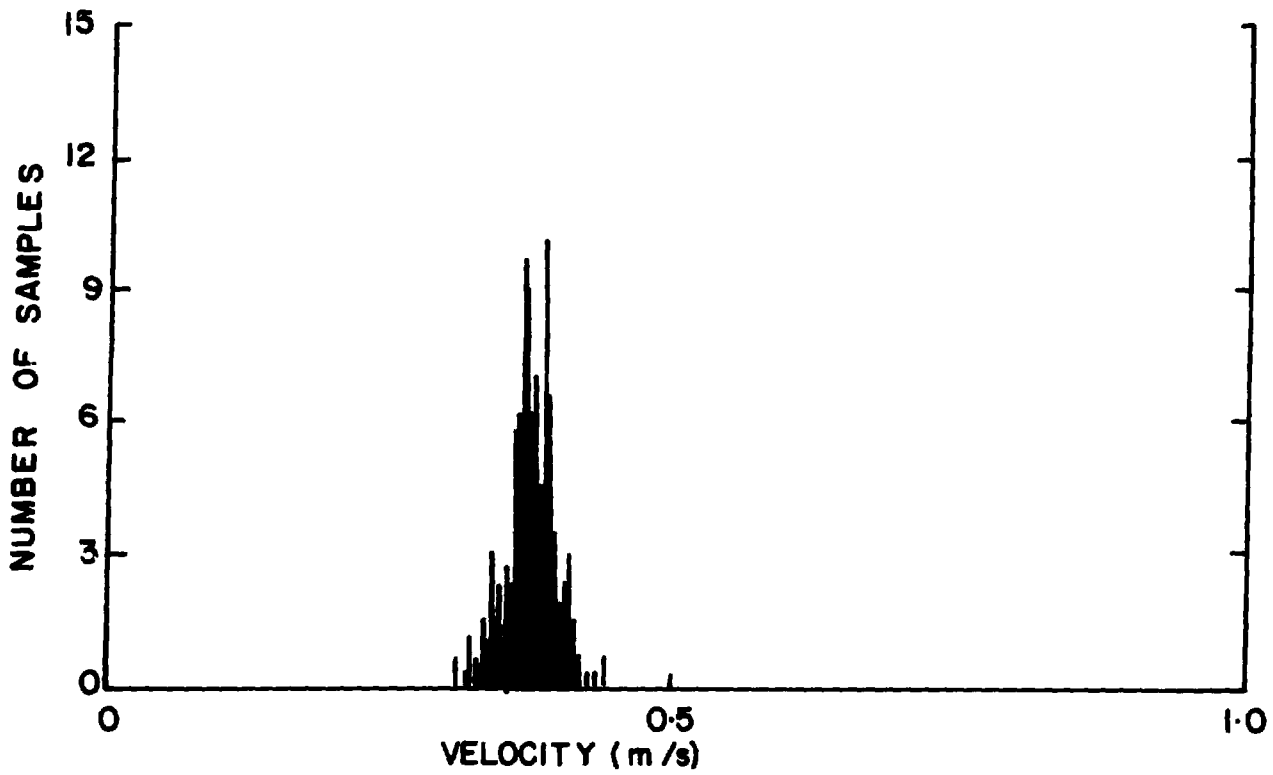
FIGURE 7

LASER DOPPLER
VELOCITY MEASUREMENTS

INSIDE SUBCHANNELS OF

49-ELEMENT BUNDLE
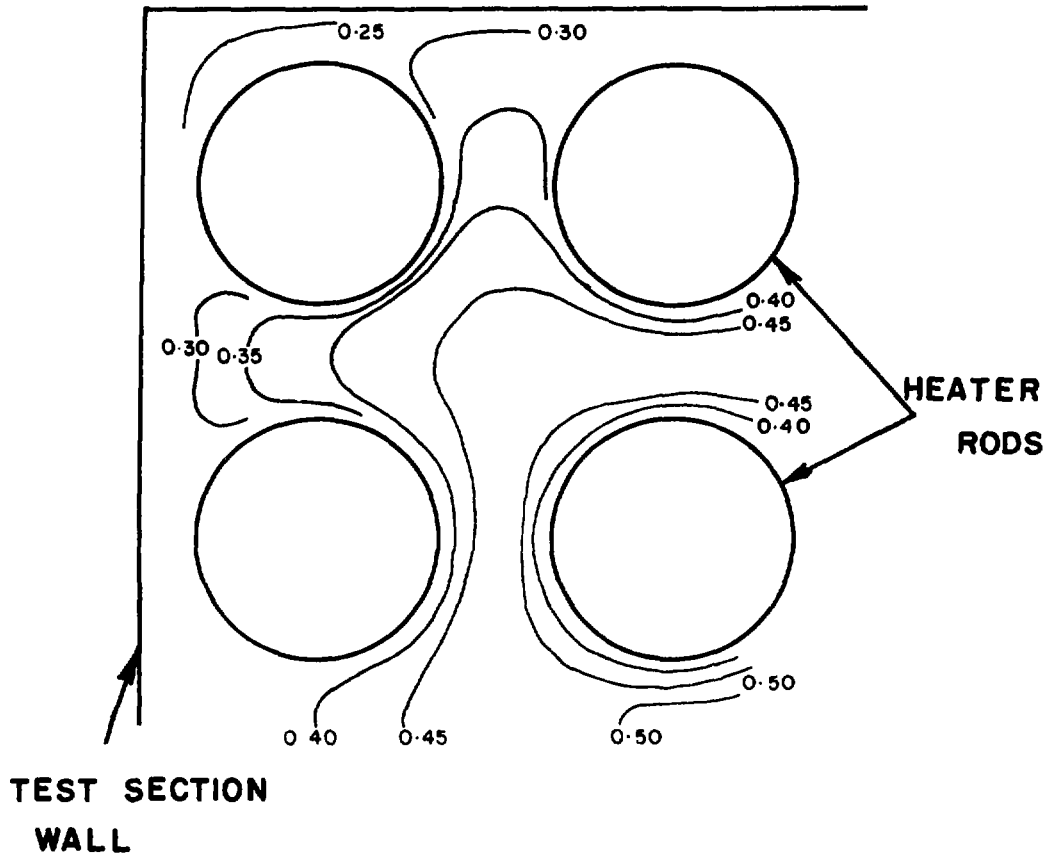
FIGURE 8

VELOCITIES WITHIN TEST BUNDLE (m/s)

TRAVERSE LINE
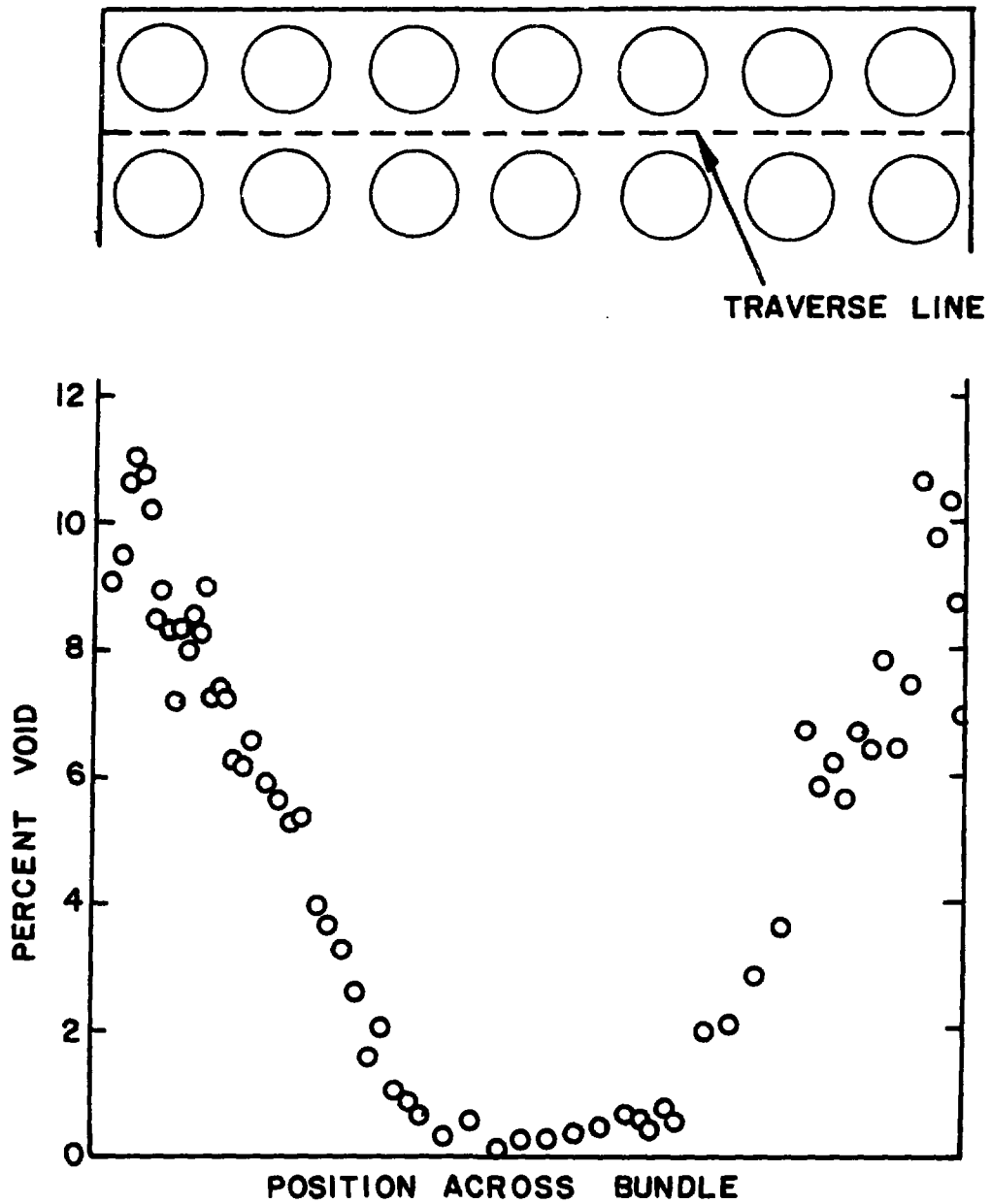
FIGURE 9

LOCAL SUBCOOLED VOID
MEASURED WITH AN OPTICAL PROBE
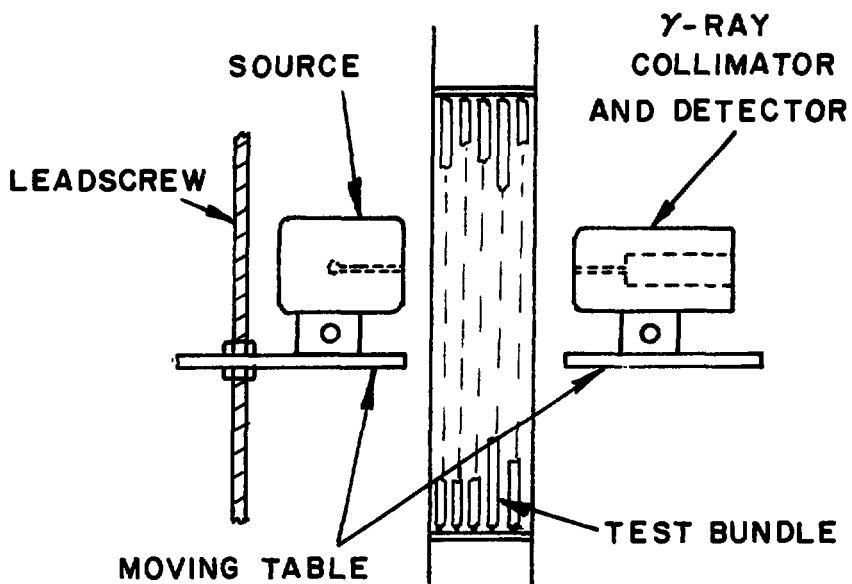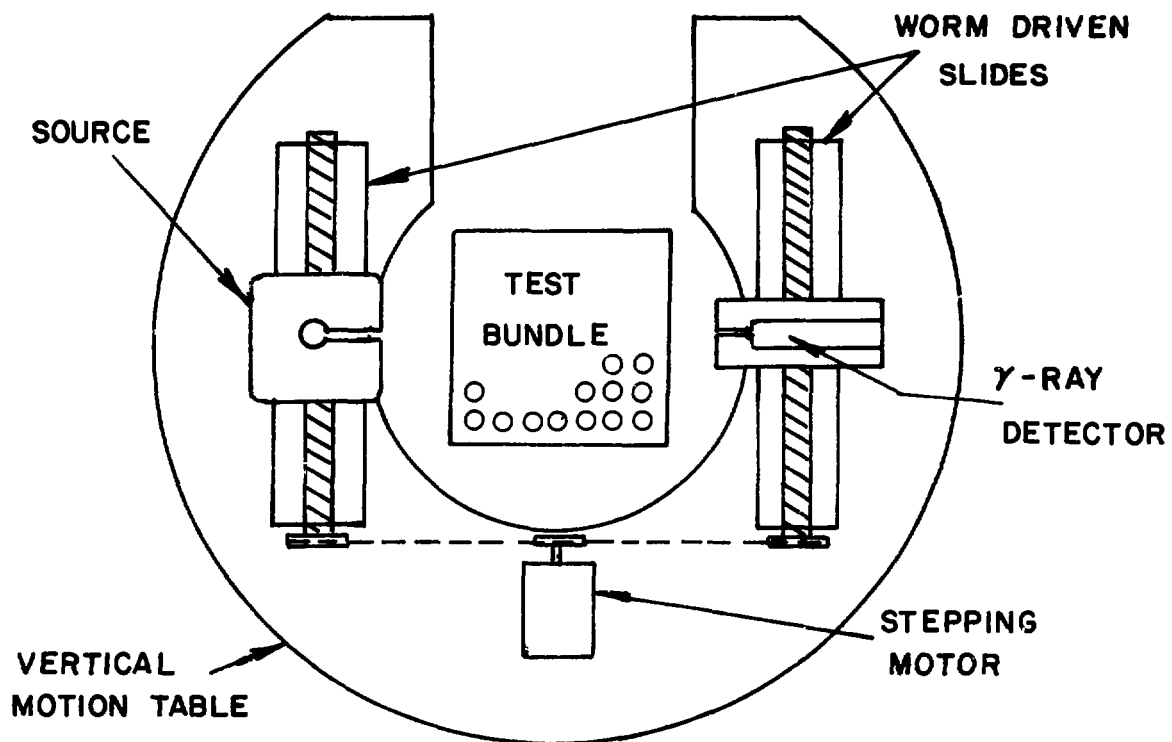
FIGURE 10

γ-RAY DENSITOMETER HARDWARE
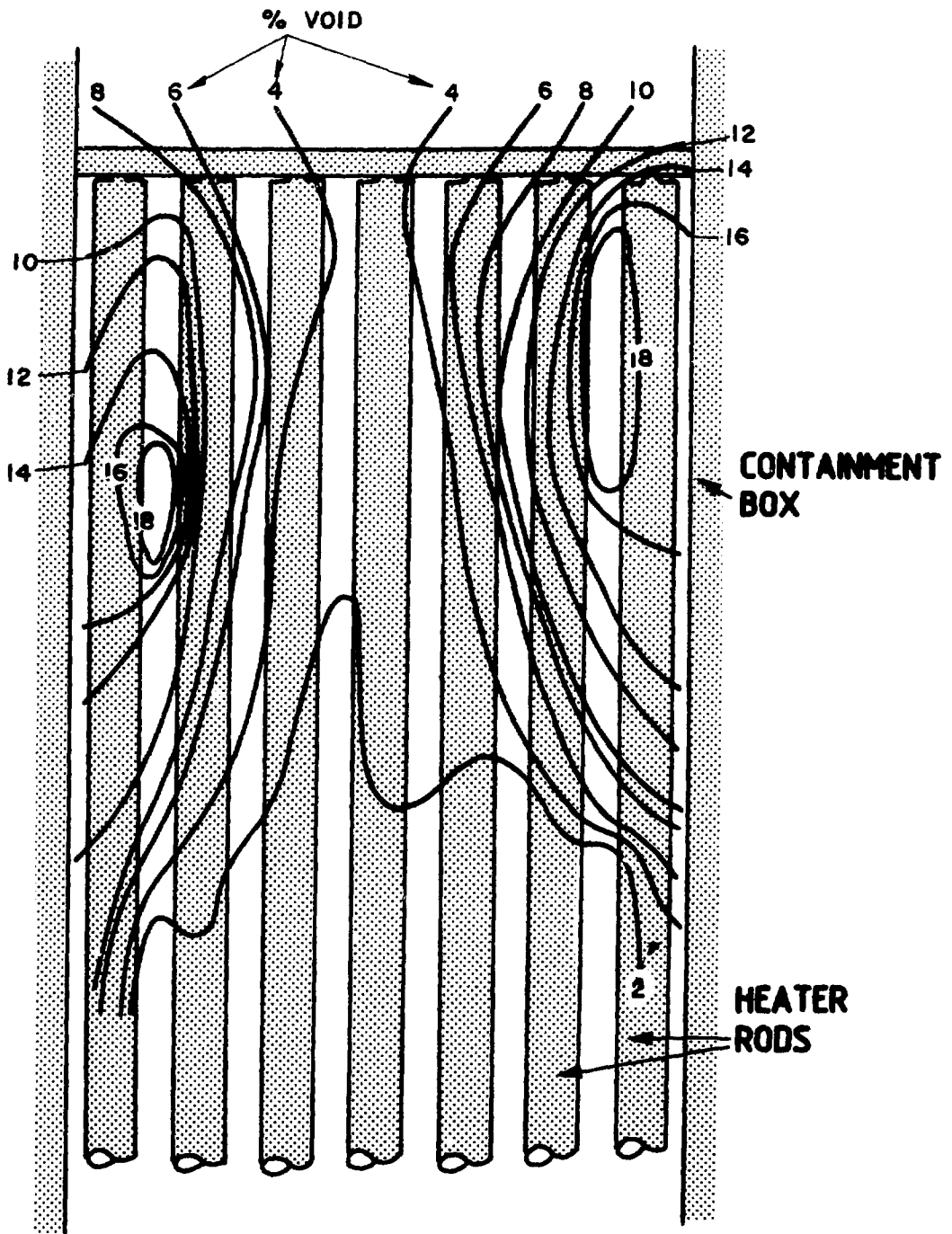
FIGURE 11

SUBCOOLED VOID CONTOUR MAP
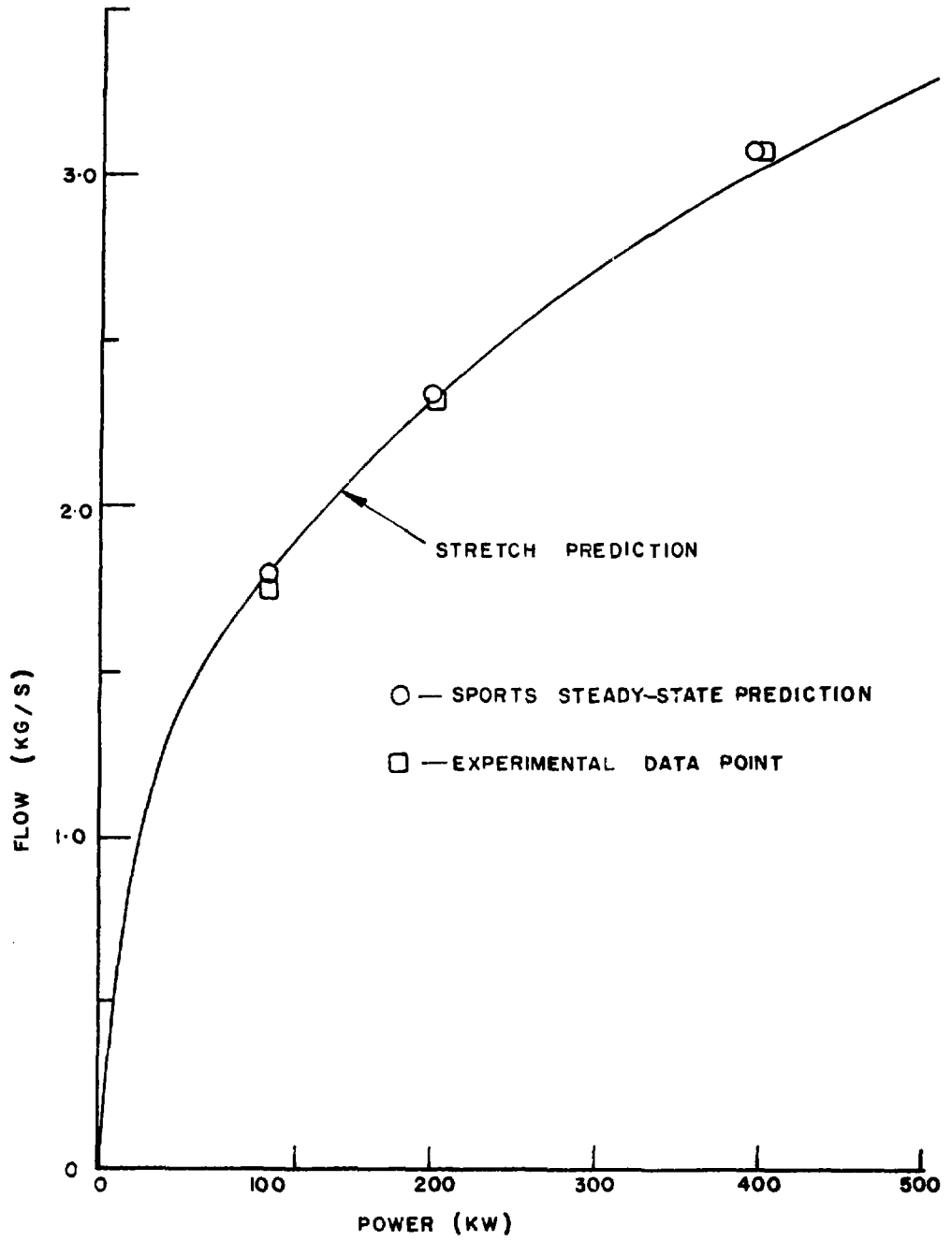
FIGURE 12   FROM [7]

SINGLE-PHASE FLOW CHARACTERISTICS
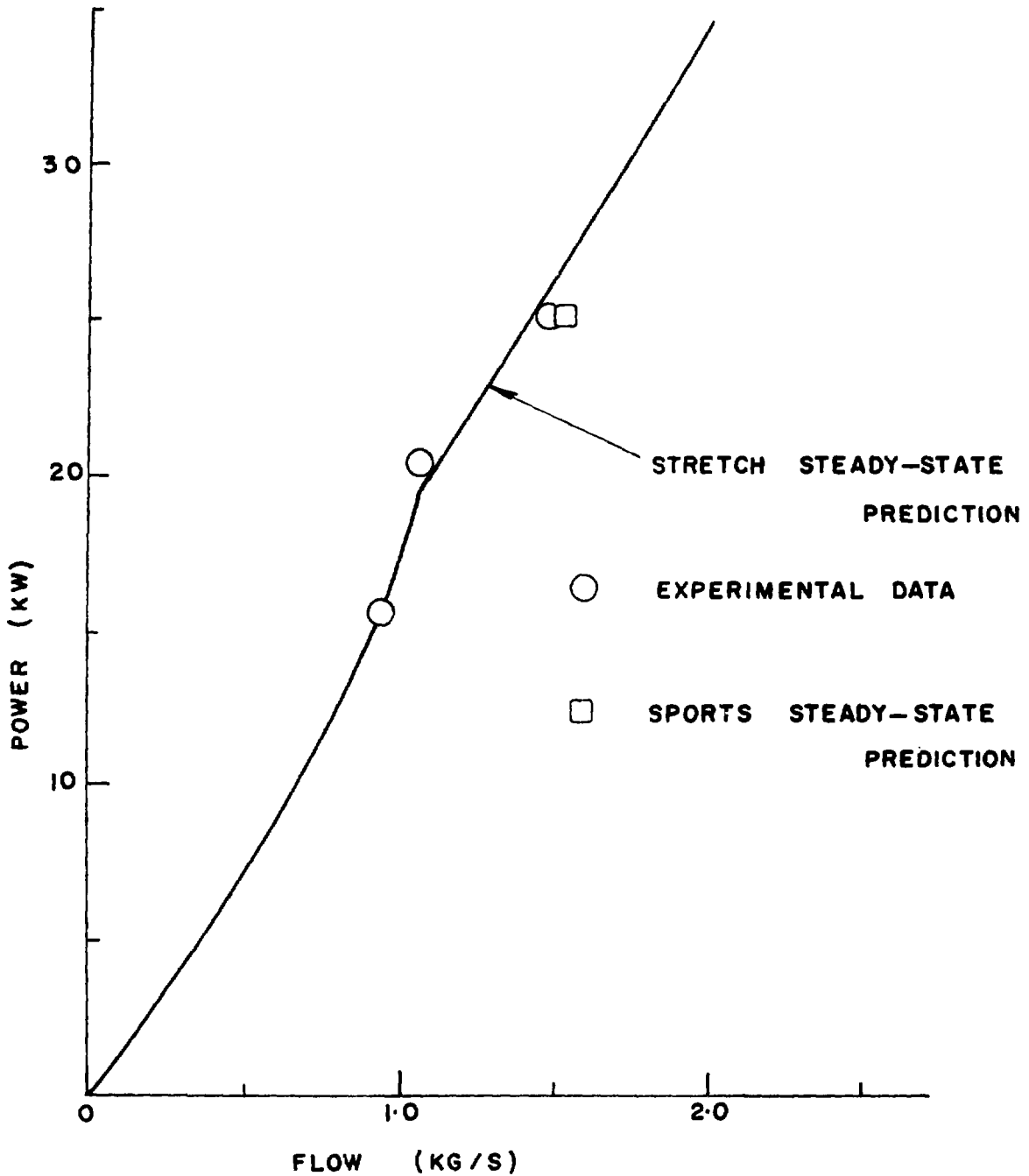
FIGURE 13

TWO-PHASE STEADY STATE FLOW CHARACTERISTICS
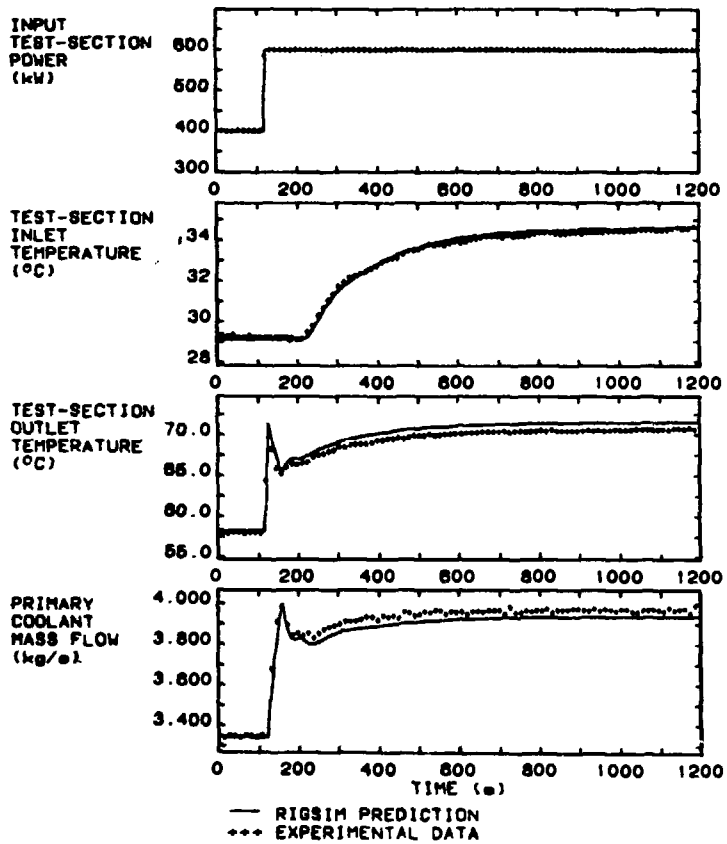
Left panel (Figure 14a):

INPUT TEST-SECTION POWER (kW) — 300, 400, 500, 600

TEST-SECTION INLET TEMPERATURE (°C) — 28, 30, 32, 34

TEST-SECTION OUTLET TEMPERATURE (°C) — 55.0, 60.0, 65.0, 70.0

PRIMARY COOLANT MASS FLOW (kg/s) — 3.400, 3.600, 3.800, 4.000

TIME (s) — 0, 200, 400, 600, 800, 1000, 1200

— RIGSIM PREDICTION
••• EXPERIMENTAL DATA

FIGURE 14a TRANSIENTS IN RESPONSE TO A STEP INCREASE IN TEST-SECTION POWER

Right panel (Figure 14b):

INPUT SECONDARY COOLANT-INLET TEMP. (°C) — 15.0, 20.0, 25.0, 30.0

HEAT EXCHANGER PRIMARY OUTLET TEMP. (°C) — 25.0, 30.0, 35.0, 40.0

TEST-SECTION INLET TEMP. (°C) — 25.0, 30.0, 35.0, 40.0

HEAT EXCHANGER SECONDARY OUTLET TEMP. (°C) — 40.0, 45.0, 50.0, 55.0

TIME (s) — 0, 200, 400, 600, 800, 1000, 1200

— RIGSIM PREDICTION
••• EXPERIMENTAL DATA

FROM [6]
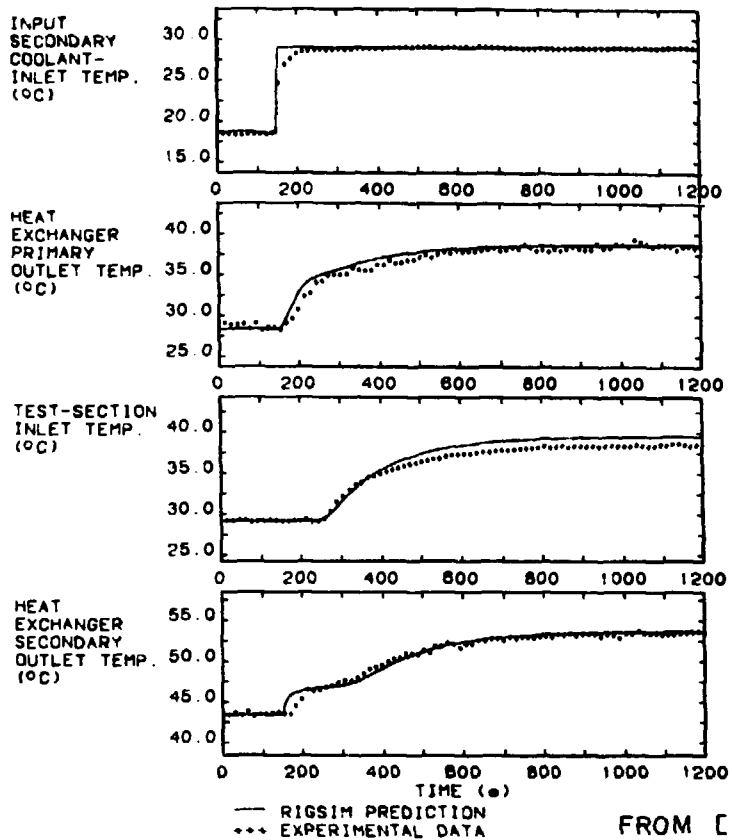
FIGURE 14b TRANSIENTS IN RESPONSE TO A STEP INCREASE IN SECONDARY COOLANT-INLET TEMPERATURE
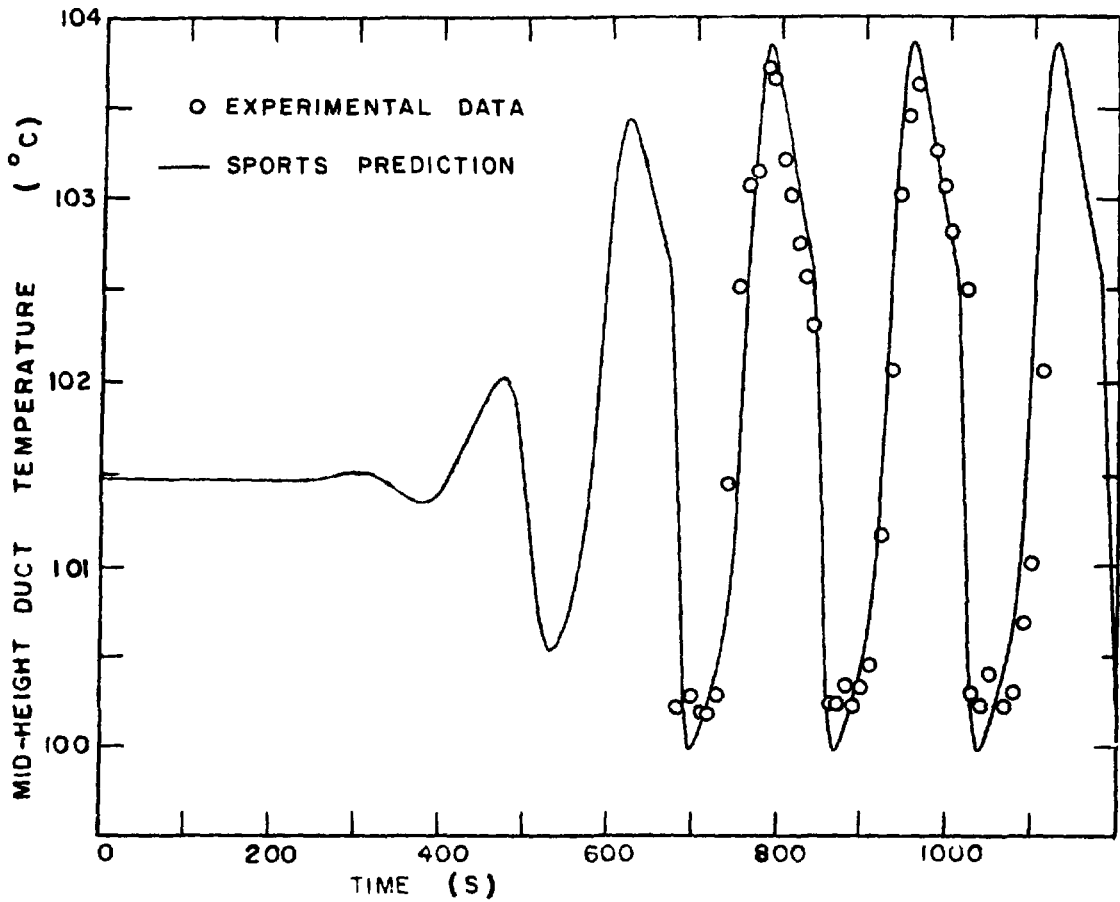
FIGURE 15

SPORTS PREDICTION OF FLASHING

COMPARED WITH EXPERIMENTAL DATA

INFORMATION FROM [7]