Atomic Energy of Canada Limited

EXPERIMENTAL BUCKLINGS OF "SIMULATED BOILING" H_2O-COOLED 28-ELEMENT NATURAL UO_2 FUEL IN HEAVY WATER MODERATOR

by

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Chalk River, Ontario

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ABSTRACT

Bucklings of "simulated boiling" H_2O-cooled 28-element natural UO_2 fuel assemblies in heavy water moderator were determined using the few rod substitution technique in the ZED-2 critical facility. Boiling coolant was simulated by bubbling air through H_2O in a maximum of seven assemblies in the central test zone of a 55 assembly core. Coolant density was varied by adjustment of air flow rate and addition of minute quantities of a surface tension reducing agent. The purpose of the experiment was to obtain bucklings for coolant density values between the limits of 1.0 and 0 $g.cm^{-3}$. Measurements were carried out at six different coolant densities in this range.

The critical heights of the substituted cores were analyzed by the source-sink theory code MICRETE3, following the procedure developed at the Chalk River Nuclear Laboratories. Within experimental errors results indicate a linear variation of buckling with coolant density between the end point values.

An interesting outcome of this experimental program was the technique of three region substitution. It was found that use of a two region pseudo-reference core, wherein the central test region contained buffer fuel assemblies of known spectral properties which more closely matched those of the test fuel, had advantages. In this case progressive substitution resulted in three region cores and resultant test fuel bucklings were more consistent.

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Laplaciens expérimentaux d'ensembles combustibles en "ébullition simulée"

par M. Srinivasan et K.J. Serdula

Résumé

Les laplaciens d'ensembles combustibles en "ébullition simulée" comportant 28 éléments de bioxyde d'uranium naturel, refroidis par eau légère et modérés par eau lourde, ont été déterminés par la technique de substitution de quelques barres dans l'assemblage critique du ZED-2. Le caloporteur en ébullition a été simulé en faisant barboter de l'air dans l'eau légère de pas plus de sept ensembles dans la zone centrale d'essai d'un coeur comportant en tout 55 ensembles. On a fait varier la densité du caloporteur en ajustant le débit de l'air et en ajoutant en très petites quantités un agent réducteur de la tension de surface. Le but de cette expérience était d'obtenir les laplaciens correspondants à des valeurs de densité du caloporteur variant entre 1.0 et 0 $g.cm^{-3}$. Les mesures ont été effectuées pour six densités de caloporteur comprises dans cet intervalle.

Les hauteurs critiques des coeurs substitués ont été analysées par le code MICRETE 3 suivant la technique développée à Chalk River. Dans la limite des erreurs expérimentales, les résultats donnent une variation linéaire du laplacien en fonction de la densité du caloporteur dans l'intervalle indiqué ci-dessus.

Un résultat intéressant de ce programme expérimental a été la technique de la substitution des trois régions. Il s'est avéré avantageux d'utiliser un coeur de pseudo-référence de deux régions où la région centrale d'essai contenait des ensembles combustibles tampons ayant des propriétés spectrales connues correspondant mieux à celles du combustible d'essai. Dans ce cas, une substitution progressive a permis d'avoir des coeurs de trois régions et les laplaciens obtenus pour le combustible d'essai étaient plus constants.

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1. INTRODUCTION

The properties of boiling H_2O-cooled natural UO_2 fuelled heavy-water moderated systems are of direct interest to Atomic Energy of Canada Limited in its CANDU-BLW program, in particular to the Gentilly Power Reactor[1]. The core design for this power reactor was based on physics data available on air, H_2O and organic cooled lattices covering the range of properties of boiling H_2O-cooled lattices. Zero energy experiments with "simulated boiling" H_2O-cooled natural UO_2 lattices were scheduled in the ZED-2 critical facility[2] at Chalk River, to investigate neutron dynamics in cores with fluctuating coolant voids. During this program buckling and other lattice parameters were determined at intermediate coolant densities to check performance of codes. This report describes the determination of buckling, by the substitution technique, of simulated boiling H_2O-cooled 28-element natural UO_2 fuel on a 30 cm triangular lattice pitch, for coolant densities in the range between 1.0 and 0.4 $g.cm^{-3}$.

The lattice was similar in fuel area and fuel/coolant/moderator ratio to that of the D_2O moderated boiling H_2O-cooled Gentilly Power Reactor lattice. Experimental bucklings for the H_2O and air-cooled cases used in the analysis were taken from earlier Chalk River measurements in ZED-2.

2. FUEL AND LATTICE ARRANGEMENT

The 28-element UO_2 fuel assemblies used in the simulated boiling experiments were essentially the same as those reported in AECL-2606[3] and AECL-2691[4]. Fig. 1 shows a cross-sectional view of these assemblies. Details of the fuel assembly

cross-sectional areas are summarized in Table I. Density of the UO_2 is 10.45 $g.cm^{-3}$. Twenty-four pellets are sheathed in Zircaloy-2 tubes to form a full length element. Elements are arranged in a four, eight, sixteen array on circles of diameters 2.326 cm, 5.304 cm and 8.412 cm respectively. Although each element contains 47.73 cm of fuel, zircaloy end caps and nuts plus Al end plates and boss increase the overall cluster length to 49.67 cm. The reference fuel assemblies consisted of five of these fuel bundles stacked in a type 50S Al 'calandria tube', plus a sixth fuel bundle of the 7 element UO_2 type[5] above them, to ensure that the top extrapolation length was independent of moderator level in the tank. Throughout the experimental program, the D_2O moderator level in the reactor was well below the top of the fifth bundle.

The basic reference core used in the experiments was 55 fuel assemblies on a 30 cm pitch triangular lattice. All assemblies were of the 28-element type described above except one in an outer position which was an 18-element UO_2 fuel assembly[6]. The fuel area and reactivity worth of this was almost the same as the 28-element assembly. Fig. 2 is a schematic view of the core configuration.

Test fuel assemblies were of the 28-element type except a maximum amount of material was removed from the Al bundle end plates to permit free flow of air bubbles between the aligned bundles. Also the bottom part of these assemblies was altered to facilitate introduction of air bubbles through a bank of hypodermic needles. The maximum number of air-water mixture cooled test assemblies substituted during the experiments was seven; the other 48 assemblies were air-cooled.

3. SIMULATION OF BOILING H_2O COOLANT

3.1 Method of Simulation

Boiling H_2O coolant was simulated by air-water mixtures obtained by bubbling air through the bottom of the test fuel assemblies containing a static column of H_2O. The seven central test fuel assemblies each contained an array of hypodermic needles at the bottom, mounted on a plenum and connected to a check valve fed by air inlet lines. Fig. 3 shows a schematic diagram of the air bubbling equipment. Air was obtained from a standard process instrument supply line at 100 psi. To remove oil or other contaminants a filter was included in the air supply line. The effect of possible pressure fluctuations in the air supply caused by changing demands at other sites was reduced by incorporating two pressure regulating valves (PRV's), one between the supply and the surge tank and the second on the individual supply lines before the needle valve used for flow adjustment. This reduced the air pressure in two steps from 100 to 40 and 40 to 20 psi. In addition to dampening supply pressure fluctuations it also helped reduce the safety-hazards problem. The air flowed from the second PRV to a needle valve, through a flow indicator into the check valve of the fuel assembly. After passing through the plenum, the hypodermic needles and H_2O, the air was exhausted outside the reactor through a line at the top of each fuel assembly. Since dry air was used, a water spray in the surge tank raised the humidity of the air and reduced H_2O losses from the test assemblies.

It was found necessary to add minute quantities of n-propanol, a surface tension reducing agent, to the H_2O coolant to prevent slugging and maintain laminar flow, especially, for the lower density values.

3.2 Monitoring and Control of Coolant Density

Early experiments showed that direct inference of the density of the air-water mixture through measurement of air flow rate was not possible. This was considered to be due to effects of water temperature, impurities and dissolved gases on bubble formation. An indirect method of derivation of the density, from the position of the air to air-water interface was worked out. This is based on the volumetric expansion of the water column on introduction of air bubbles.

The interface in each assembly was detected by means of an axial array of six thermistors positioned 1.5 cm apart in the vicinity of the interface. The operating principle is simply that heat transfer rate is higher in air-water mixtures than in air, when a current of sufficient magnitude to induce self heating is passed through the thermistor. The thermistor outputs were displayed on a fast multichannel pen recorder. From the traces it was easy to determine the position of the interface. In practice the thermistor positions were pre-calculated and fixed and the air flow rate adjusted to position the interface at the third thermistor or between the third and fourth thermistor. Fig. 4a shows in schematic form a thermistor circuit.

Each of the seven test fuel assemblies had its own thermistor probe of six thermistors; therefore there were 42 thermistors to monitor. Since there was only one 8-channel recorder available the assembly probes were monitored sequentially. Fig. 4b is a schematic of the probes monitoring system. Adjustment of air flow rate during reactor operation to maintain the required interface levels was done manually whenever necessary.

Variation of coolant density was mainly done by adjustment of air flow rate. For lower density values, n-propanol was added to the H_2O to a maximum of 0.1% by volume to maintain smooth flow of air. This is not expected to alter the neutron parameters of the coolant since n-propanol has an effective chemical composition of the form $(CH_2)_n$.

3.3 Calculation of Average Coolant Density

If a volume v_0 of H_2O expands and occupies a volume v_1 on the introduction of air bubbles then the average density ρ of the air-water mixture is given by

$$\rho = \frac{v_0}{v_1}$$

If the cross-sectional area of the coolant channel is the same throughout its height then ρ could be obtained simply from the heights of the water column before bubbling and the air-water mixture column after bubbling.

$$\rho = \frac{h_0}{h_1}$$

In the fuel assembly case the volume available for the coolant was the sub-channels between the UO_2 pencils of the 28-element assembly. On introduction of air, the air-water mixture expanded above the top of the fifth fuel bundle. This resulted in the upper part of the air-water mixture occupying the full cross-sectional area of the pressure tube. This increased area was accounted for in the calculation of the average coolant density. The level of the interface was always above the moderator level during the measurements.

In these experiments the test fuel assemblies were filled with H_2O to a known level and the air flow rate increased until the interface reached a pre-determined level as detected by the thermistor probes, giving the required coolant density. In practice the static H_2O level was set, for operational safety, at the top of the fifth fuel bundle (or the fourth fuel bundle for lower density values). This practice facilitated measurements. It was possible to obtain different coolant densities by merely withdrawing the thermistor probes to pre-calculated levels and increasing the air flow rate correspondingly, without changing the quantity of H_2O in the assemblies.

The expression used in the calculation of the average coolant density was

$$\rho = \frac{V_{H_2O}}{A_2 h - 5\, V_{fuel}}$$

where

ρ = average coolant density

V_{H_2O} = measured volume of H_2O in fuel assembly

A_2 = pressure tube cross-sectional area

h = effective height of air-water mixture in assembly

V_{fuel} = volume of one fuel bundle

5 = number of bundles covered by air-water mixture.

3.4 Density Calibration

The above method of calculation of coolant density gives only the average density value over the height of the air-water mixture column. However the local coolant density

varies with height. In such a case the effective density value within the active reactor core region of the fuel assembly would be different from the overall average.

To gain an insight into the nature of the vertical variation of coolant density that may exist, some out of pile measurements were carried out on a mock-up of the fuel assembly. It was found that an important source of error was the sudden increase in cross-sectional area available for the air-water mixture as it emerged from the top of the fifth fuel bundle into the pressure tube region above where the area for coolant flow increased by a factor of 2.5 compared to the coolant area in the bundle. The coolant density in this region was higher than the coolant density inside the fuel bundles. Consequently the effective coolant density within the active reactor region was lower than the overall average value. The variation of coolant density with height within the active reactor region was found to give rise to relatively minor errors only.

A second set of critical height measurements was therefore carried out with the pre-bubbling H_2O level in the fuel assemblies adjusted so that on bubbling the air-water mixture interface level was at the top of the fifth fuel bundle. This was done with air flowing in each of the central seven assemblies and for each of the six average coolant density values used in the original substitution runs. A plot of critical height of the substituted core against the average coolant density in the seven test fuel assemblies resulted in a smooth curve (Fig. 5). Also plotted on the same figure is data obtained with the original method of bubbling plotted against overall average coolant density. Based on these

calibration measurements a table of the effective coolant densities within the reactor core, valid for each of the density substitutions was prepared (Table II). Quoted error bars of $\pm .015$ g.cm^{-3} include estimates due to the possible contribution of vertical coolant density variation, axial cosine flux distribution and neutron importance variation effects.

4. SUBSTITUTION MEASUREMENTS

4.1 Two Region Substitution

In the substitution technique of buckling determination[7], fuel assemblies of known properties including buckling are substituted with one or more test assemblies whose buckling is to be derived. Critical heights of the two region cores are measured as a function of number of test assemblies substituted. The critical height data is analysed by an appropriate reactivity code to give the buckling or other property of the test fuel assemblies. Some input data on the test fuel, discussed in Section 5.2, are required for the analysis.

In the substitution experiments reported the maximum number of test assemblies substituted was seven, with three intermediate steps of 1, 3 and 5 substitutions. Fig. 6-(a) depicts the selected progressive substitution stages; only the central seven positions involving test fuel are shown as the surrounding 48 lattice sites contained reference fuel. There are alternative possiblilities for the 3 and 5 assembly substitutions, but measurements were not carried out for those.

The reference fuel used in these experiments was air-cooled 28-element UO_2 assemblies while the test fuel was the

same but with "simulated boiling" coolant. Substitution involved introduction of air to produce air-water mixtures in the fuel assemblies. In practice the reactor was shut down after a reference core critical height measurement and a known amount of H_2O added into the test fuel assemblies at the selected sites. Air flow was then adjusted to position the interface at the pre-calculated level and the reactor pumped up to critical. On account of the long procedure it was more practical to repeat the critical height measurements for a fixed number of test assemblies substituted, but with various coolant densities, rather than for a given coolant density value with varying number of test assemblies. The latter procedure is preferable to minimize errors due to reactivity drifts, non-reproducibility of coolant density, etc.

4.2 Three Region Substitution

A second series of substitution critical height measurements was carried out as follows. A two region core was used as the reference where the central seven assemblies were filled to a known height with H_2O. The critical height of this pseudo-reference core was measured. Air flow to the central assembly was slowly adjusted to bring the interface to a level corresponding to the selected density, and the critical height of this substituted core noted. For the next measurement, air flow was maintained at the same rate in the central three assemblies. The procedure was repeated for the 5 and 7 assembly substitution at the same coolant density. Finally air flow was gradually stopped and the

critical height of the pseudo-reference core noted again. It was thus possible to carry out all the substitutions for a given density in the same day. At the end of the series when measurements had been completed for all substitutions at all density values, the critical height of the one region all air-cooled reference core was recorded.

This method of substitution resulted in three region cores. (Fig. 6b). Although the reference cores were different in the two and three region series, progressive substitution led to the same seven assembly substituted test core. The critical heights of the all air-cooled 55 assembly core and the pseudo-reference core with seven H_2O-cooled central assemblies, represented the upper and lower bounds for critical heights of the two and three region series.

4.3 Extended Substitution Series

In this series the test assemblies were H_2O-cooled, reference assemblies being air-cooled as before. Details of the substitutions are illustrated in Fig. 7. The first half of the series was identical to a conventional two region progressive substitution method for 1, 3, 5 and 7 substituted assemblies. The second half of the series may be called an inverse substitution experiment since substituted (test) assemblies were air-cooled and 'reference' assemblies were H_2O-cooled in the central test region. This half may also be looked upon as being one possible way of carrying out the 6, 4, 2 substitution stages of the conventional progressive substitution scheme or alternately as a special case of the

three region method where the coolant density of the test assemblies is 0 $g.cm^{-3}$. An important point is that the first and second halves of the series form a complementary pair with a 'mirror image' (test and reference sites interchanged) in the second half for every substituted core in the first half (Fig. 7).

The aim of this series was to provide data for the purpose of checking alternate schemes of data analysis. In particular it was hoped that information on fuel interaction effects could be extracted.

4.4 Summary of Corrected Critical Moderator Level Results

The simulated boiling substitution experiments extended over a period of several months commencing August 1968. The D_2O moderator temperature was in the range 21.5 ± 0.5°C. However moderator purity steadily deteriorated from 99.68 atom % to 99.63 atom % D_2O during the period of the experimental program. The resultant increments in the critical heights of the all air-cooled and seven H_2O-cooled plus 48 air-cooled cores were approximately 1.5 cm and 2.5 cm, corresponding to a ΔB^2 of ~.03 m^{-2}. All critical heights were normalized to the September 10th, 1968 moderator purity of 99.66 atom % D_2O. No correction was applied for the small temperature variations. The seven assembly substituted core critical heights served as an additional normalization check between the two and three region heights.

Table III summarizes the corrected critical height results of the simulated boiling substitution program. The same data is plotted in Fig. 8, 9 and 10. Although the two

and three region series data are independent, they are plotted in the same figures for comparison. The two region data in Fig. 8 show a small curvature as a function of the number of substituted assemblies, while the three region points lie on a straight line. The three region points for coolant density of 0 $g.cm^{-3}$ were obtained in the second half of the extended substitution experiment. The shape of the three region curve for density 0 $g.cm^{-3}$ is quite similar to the two region curve for density 1.0 $g.cm^{-3}$. This feature is also visible in Fig. 10 where all the points fall close to a smooth curve. The implications of this are discussed in Section 6.3 .

Fig. 9 contains plots of critical moderator level vs. effective coolant density for various numbers of substituted assemblies. The continuous lines are nominal fits. Scatter of the points around the smooth curves is largely due to uncertainty in the effective coolant density values rather than in measured critical moderator levels.

5. METHOD OF ANALYSIS

5.1 Code MICRETE3

MICRETE3 is a two group line source-sink model heterogeneous theory code developed at Chalk River by J.D. Stewart[8] [9]. It was originally written in APEX language for the G-20 computer at CRNL. A FORTRAN version for use on a CDC 6600 machine has since been made available. The essential features of this code may be summarized as follows:

(a) - Fuel rods are represented by line sources of fast neutrons and line sinks of thermal neutrons.

(b) - Slowing down and diffusion in the moderator is described by conventional diffusion theory, characterized by a slowing down area, L_s^2 and diffusion area L^2.

(c) - The reflector is assumed to have the same properties as the moderator.

(d) - The 'source' strength of the rod is characterized by a parameter, η, which is actually the product $\epsilon\eta$ of the conventional four factor notation.

(e) - Thermal neutron sink strength of the fuel element is given by a parameter α. The expression for α used in the analysis is related to the thermal utilization factor, f_{rod}, diffusion area, L^2 and geometrical properties b and d of the cell as follows:

$$\alpha = \left[\frac{2L^2}{f_{rod}b^2} - \ln\left(\frac{b}{d}\right) + \frac{3}{4}\right]^{-1}$$

where b ≡ equivalent cell radius

d ≡ lattice pitch

(f) - Resonance absorption in the cell is assumed to be proportional to (1 - P) times the fast flux at a specified

radial distance c from the rod axis. It enters the two group calculation as a uniform sink of thermal neutrons inside a radius c (usually set equal to the cell boundary radius) from the rod axis. The parameter P is the conventional resonance escape probability factor for the lattice and is an input quantity along with α and η.

(g) - The code requires the following additional input parameters:

D_F = cell-averaged fast diffusion coefficient

D = cell-averaged thermal diffusion coefficient

κ_F = cell-averaged inverse slowing down length

κ = moderator inverse diffusion length.

These definitions of 'moderator' diffusion properties and α were recommended by Okazaki and Craig[10] following extensive investigations.

(h) - MICRETE3 can take into account a radial reflector, either finite or infinite, but it assumes the reflector material is identical to that of the moderator in the core.

5.2 MICRETE3 Analysis of Substitution Experiments

Application of MICRETE3 to the analysis of substitution experiments was described by Okazaki and Craig[10]. The only change introduced in the present analysis was in the manner of adjustment of P of the reference fuel in order to obtain agreement between MICRETE3 buckling and the experimental reference flux-map buckling value. The procedure adopted in this report is shown schematically in Fig. 11.

The only experimental information directly entering the analysis was the extrapolated critical heights of the reference and substituted cores. All other input quantities were calculated using the lattice recipe code LATREP[11]. The first and last blocks in Fig. 11 represent large imaginary hypothetical cores mocked up for the purpose of deriving bucklings. Radial bucklings are obtained in a manner analogous to a flux mapping experiment; namely by fitting an appropriate Bessel function to the radial distribution of MICRETE3 calculated fluxes[10]. Axial buckling is taken as $(\pi/H)^2$: H being the extrapolated critical height of the mock-up. The total buckling is the sum of axial and radial bucklings.

In the first block of Fig. 11 the P of the reference fuel is adjusted until the experimental buckling value is obtained. In the second block, a mock-up of the reference core of the substitution experiments, this adjusted P_{ref} is used and the radius R of reflector outer boundary is altered until agreement is obtained between calculated and measured extrapolated critical heights of the reference core.

The third block mocks up the actual substituted two or three region core. Here P_{test} of the fuel assembly is adjusted to give the observed critical height. Although any one of the calculated input parameters could have been adjusted the choice of P was dictated by the fact that its calculation by LATREP and representation in MICRETE3 are the least accurate of all the input parameters. In the last step this adjusted P_{test} is used along with other LATREP calculated inputs in a large one region hypothetical test core mock-up to obtain the buckling of the test lattice.

MICRETE3 is basically a reactivity code. The outcome of the analytical procedure is a single number, namely an adjusted P_{test} value of the test fuel which satisfies the reactivity balance conditions of the substitution experiment. Drawbacks and errors of the LATREP calculated input parameters, approximations of geometry and physical representation of the reactor in MICRETE3, uncertainties in measured critical moderator levels, errors in derivation of axial extrapolation lengths,etc. are all reflected as a reactivity discrepancy between experiment and MICRETE3. This is compensated for in the analytical procedure by adjusting P and R. The adjusted value of P may therefore not be in agreement with the true resonance escape probability of the lattice. Sensitivity of MICRETE3 calculations to various input parameters has been discussed in detail in reference (10).

5.3 Code MICRETEX

The four block scheme of analysis of substitution critical heights to yield the buckling of the test lattice directly was incorporated in MICRETEX - a modified version of MICRETE3. It can be seen from Fig. 11 that the procedure is quite symmetric. The code accepts an input reference lattice buckling value and exits with the calculated test lattice buckling using the measured reference and substituted core critical heights. Note that for each stage of the progressive substitution procedure a separate B^2_{test} value would result.

5.4 Input Parameters Used in the Analysis

Apart from the critical moderator heights of the reference and substituted cores, the only other input parameters obtained experimentally were (i) axial (top plus bottom) extrapolation lengths and (ii) buckling of the air-cooled reference lattice. The former was obtained from an axial flux distribution measurement carried out during the substitution program. Appendix A contains details of this measurement and derivation of the top and bottom extrapolation lengths. The derived total extrapolation length of 23 cm for the seven H_2O-cooled assembly plus 48 air-cooled assembly core was, within experimental errors, the same as that obtained for the 55 air-cooled assembly core. Therefore 23 cms was added to all measured critical moderator levels to yield extrapolated critical heights.

Experimental bucklings of the ZED-2 28-element UO_2 fuel, for a moderator-to-fuel volume ratio of 14.7 corresponding to 30 cm triangular lattice pitch or 27.94 cm square lattice pitch, determined earlier at Chalk River for air and H_2O-cooled cases are summarized in Table IV. Based on these, buckling values of $4.04 \pm .03\ m^{-2}$ for the air-cooled lattice and $1.17 \pm .05\ m^{-2}$ for the H_2O-cooled lattice were derived for the temperature and purity conditions of the simulated boiling experiments of $21.5 \pm 0.5^{\circ}C$ and 99.66 ±0.01 atom % D_2O. The experimental buckling of the H_2O-cooled lattice was not used in the analysis. It served only as a check on the accuracy and validity of the measurements and method of analysis.

All calculated input parameters used in the analysis were obtained with the Chalk River lattice recipe code LATREP; these parameters are summarized in Table V.

6. RESULTS AND DISCUSSION

6.1 Two Region Results

Results discussed in this section were obtained using the experimental critical height data of Table III and the input parameters listed in Tables IV and V. Use of experimental data at 21.5°C and moderator purity of 99.66 atom %, and calculated parameters at 25°C and 99.72 atom % purity will not contribute any significant errors. Absolute accuracy of the various calculated parameters at any given temperature and purity could be in error. However relative values given by most codes are reliable. By use of the correct reference lattice buckling, valid for the experimental conditions of moderator temperature and purity, in normalization of P_{ref}, effects due to systematic errors in input data are annulled to a large extent. Besides in this case, temperature and purity differences between the input experimental data and calculated data had opposing effects, minimizing further errors due to input data.

Test lattice buckling results obtained from analysis of two region critical heights are summarized in Table VI . The major feature is the variation of B^2 with N, the number of substituted test assemblies. In general the buckling approaches the expected test lattice buckling value as N increases. This type of behaviour has been noted by others[12][13] also at Chalk River when the code MICRETE3 was applied to few

rod substitution measurements. It was shown by Okazaki[12] that a plot of B^2 against the inverse of N falls on a reasonably straight line (except for the one assembly substitution point). This line extrapolated to the flux-map value of B^2 of the test lattice at $1/N = 0$. Fig. 12 shows the results of Table VI plotted against 1/N. For the case of coolant density of 1.0 $g.cm^{-3}$, a line through the 3, 5 and 7 assembly substitution points extrapolates to a test lattice buckling of $1.23 \pm .10\ m^{-2}$. This compares well with the value of $1.17 \pm .05\ m^{-2}$ given as the experimental flux-map buckling value for the H_2O-cooled 28-element UO_2 lattice (Table IV). An accuracy of $0.1\ m^{-2}$ has been suggested[10] for bucklings derived by such an extrapolation technique. The difference is therefore well within the estimated errors. The close agreement gives confidence in the validity of the experimental technique, input parameters used, and method of analysis. Since the air-cooled reference assemblies and the H_2O-cooled test assemblies discussed above represent the extreme values of coolant density, it is reasonable to assume that the bucklings derived by an interpolation technique for intermediate coolant densities between 0 and 1.0 $g.cm^{-3}$ are also valid.

The anomalous pattern of the slopes and angles of the extrapolation lines could be attributed to the scatter of the two region experimental points, especially for the 3 and 5 assembly substitutions. Note that the .52 and .62 density points occurring on the lower side of the lines in Fig. 9, have caused the slopes of the extrapolation lines for these two density cases to depart in an opposite direction from the general trend in Fig. 12.

Resultant test lattice bucklings from the two region experiments for simulated boiling H_2O-cooled lattices are given in Table VI.

6.2 Three Region Results

For the three region measurements the reference lattice was the two region core with seven central H_2O-cooled assemblies. The method of analysis adopted was similar to that used for the two region data except for a slight modification. An additional step was introduced between blocks 2 and 3 of Fig.11. The measured extrapolation critical height of the pseudo-reference core was fed in along with the adjusted values of P_{ref} and $R_{reflector}$. P_{buffer} of the H_2O-cooled assemblies was then adjusted until the pseudo-reference core became critical at its observed height. These values of P_{ref}, P_{buffer} and $R_{reflector}$ were then fed into a three region substituted core calculation and P_{test} of the simulated boiling H_2O-cooled test fuel assemblies was adjusted to make the core critical at its extrapolated critical height. This P_{test} was used in the last calculation to obtain the test lattice buckling in a large one region hypothetical core mock-up.

The results of analysis of three region data are summarized in Table VII. Although test lattice bucklings did vary with N, the number of test assemblies substituted, the variation was much smaller than in the two region case. In Fig. 13 the results are shown graphically. Not only do the one assembly substitution results fall on the extrapolation lines, but the lines themselves are more horizontal when compared to the two region results. This is attributed

to the presence of the initial H_2O-cooled assemblies, giving rise to a spectrum which matches that of the simulated boiling H_2O-cooled lattices better than the spectrum produced by the air-cooled assemblies. The fact that results are obtainable, even with single assembly substitution, quite close to the expected value is noteworthy. The difference between the single assembly value and the extrapolated value is $\lesssim .1\ m^{-2}$ in all cases except for the point at a density of $0.41\ g.cm^{-3}$.

5.3 Inverse Substitution Analysis

The critical heights of substituted cores with H_2O-cooled assemblies belonging to the conventional two region series and inverse substitution cores of the extended series fall on an almost smooth curve (Fig. 10). This suggests the flux gradient in the central region approaches zero as the amount of H_2O in the central test region is increased, resulting in approximately equal statistical weights for any fuel assembly in the test region.

Results of analysis of extended substitution data by the two region method are summarized in Table VIII and Fig. 14. While there may be a small difference in the patterns formed by the inverse substitution points and the two region series (1, 3, 5, 7) points, the difference is insufficient to warrant an alternate scheme of analysis to extract information on fuel interaction effects.

7. SUMMARY AND CONCLUSIONS

Table IX and Fig. 15 summarize test lattice buckling results of both two region and three region experiments. The

agreement between the results obtained by the two different techniques is satisfactory. Also plotted in Fig. 15 are the bucklings for coolant density values of 0 $g.cm^{-3}$ and 1.0 $g.cm^{-3}$, measured previously in ZED-2 (Table IV). The bucklings for intermediate coolant densities obtained in the simulated boiling experiments fall within experimental errors on the straight line joining the points at the extreme densities. The experiments indicate a linear interpolation between the end point results may be adequate.

The three region substitution method yields more consistent bucklings and presents a useful technique when spectral mis-match problems are suspected between reference and test fuel.

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We express our sincere thanks to Miss V. Kaminski for typing the manuscript.

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TABLE I

CROSS-SECTIONAL AREAS FOR 28-ELEMENT UO_2 FUEL AND REFERENCE LATTICE CELL

(30 cm Triangular Lattice Pitch)

All figures are in units of cm^2

Total cell	779.43
Moderator outside Calandria tube	651.99+
Difference =	127.436

Areas for mid-cluster plane (Reference and Test Assemblies):

UO_2 Fuel	44.405+
Total air (including calandria annulus area of 30.580 cm^2)	31.207
Coolant inside pressure tube	30.672
Al Pressure tube	9.754
Zircaloy-2 sheathing	5.897
Al Calandria tube (505)	5.501
Total =	127.436

Areas for Homogenized-Bundle End-Region*:

	Reference	Test
Coolant	43.329	49.933
Al	38.474	30.000
Air	35.411	35.411
Zr-2	10.222	12.092
Total =	127.436	127.436

+V_{mod}/V_{fuel} = 14.7

*Volumes can be obtained by multiplying by gap between UO_2 which is 1.94 cm for reference and 1.64 cm for test assemblies.

TABLE II

COOLANT DENSITY CALIBRATION RESULTS

Average Coolant Density ($g.cm^{-3}$)	Effective Coolant Density in Reactor Core ($g.cm^{-3}$)
1.0	1.00
0.9	0.90 ± .015
0.8	0.77 ± .015
0.7	0.62 ± .015
0.6	0.52 ± .015
0.5	0.41 ± .015
0	0

TABLE III

SUMMARY OF CORRECTED CRITICAL MODERATOR LEVELS

(Moderator Purity : 99.66 atom %; Temperature 21.5 ± .5°C)

	TWO REGION					THREE REGION			
N →	0	1	3	5	7*	5	3	1	0
ρ DENSITY ($g.cm^{-3}$) ↓	Critical Moderator Heights (cm)								
1.0	199.8	204.1	213.1	223.2	234.4	234.4	234.4	234.4	234.4
0.9	199.8	203.7	212.2	221.3	231.2	232.1	233.0	233.9	234.4
0.77	199.8	203.3	211.0	218.8	226.7	228.8	231.1	233.3	234.4
0.62	199.8	202.5	208.5	214.6	221.5	225.1	228.8	232.6	234.4
0.52	199.8	202.2	207.0	212.2	218.2	222.7	227.4	232.1	234.4
0.41	199.8	202.2	206.5	210.6	214.3	220.1	225.8	231.6	234.4
0.0	199.8	199.8	199.8	199.8	199.8	208.3	217.7	228.3	234.4

N = Number of substituted assemblies

* The seven assembly substituted values are common to both series

TABLE IV

PREVIOUS CHALK RIVER EXPERIMENTAL BUCKLING

RESULTS ON 28-ELEMENT UO_2 FUEL FOR V_{Mod}/V_{Fuel} RATIO OF 14.7

COOLANT	LATTICE PITCH cm	EXPERIMENTAL TECHNIQUE	B^2 (m^{-2})	MODERATOR CONDITIONS		REFERENCE NO.	BUCKLING VALUE USED IN ANALYSIS OF THIS REPORT (21.5°C, 99.66 ATOM % D_2O)
				TEMPERATURE	PURITY ATOM % D_2O		
AIR	30 △	One region, flux map	4.059±.012	25°C	99.72	(3)	
	27.94 □	One region flux map	3.977±.042	25°C	99.70	(4)	4.04±.03 m^{-2}
	27.94 □	Multi region, flux map	3.920±.043	25°C	99.70	(4)	
	27.94 □	One region, flux map	4.00± .04	21.3°C	99.72	(14)	
H_2O	27.94 □	Multi region, flux map	1.166±.018	25°C	99.70	(4)	
	27.94 □	Two region substitution	1.12 ±.10	25°C	99.65	(13)	1.17±.05 m^{-2}
	27.94 □	Two region substitution	1.22 ±.1	25°C	99.63	(12)	

TABLE V

LATREP CALCULATED INPUT PARAMETERS

USED IN THE ANALYSIS

Effective Coolant Density ($g.cm^{-3}$)	0 (REF)	0.41	0.52	0.62	0.77	0.9	1.0
Average* Coolant Density ($g.cm^{-3}$)	0	0.5	0.6	0.7	0.8	0.9	1.0
D_F^+ (cm)	1.2189	1.1849	1.1782	1.1716	1.1651	1.1586	1.1522
D (cm)	.9830	.8947	.8886	.8836	.8796	.8762	.8734
κ_F (cm^{-1})	.0842	.08982	.09094	.09205	.09315	.09423	.09534
κ (cm^{-1})	.01022	.01022	.01022	.01022	.01022	.01022	.01022
α	.3024	.3120	.3134	.3148	.3164	.3180	.3197
η	1.2965	1.2226	1.2092	1.1961	1.1833	1.1716	1.1586

* The calculated data pertain to average coolant density rather than effective coolant density values.

+ Definitions of Parameters are given in Section 5.1.

TABLE VI

TWO REGION BUCKLING RESULTS

(Units of m^{-2})

N \ $\rho(g.cm^{-3})$	0.41	0.52	0.62	0.77	0.9	1.0
1	2.9624	2.8789	2.7142	2.2246	1.9849	1.7395
3	2.8747	2.7991	2.5011	1.9732	1.7245	1.5415
5	2.9251	2.7526	2.4740	1.9589	1.6554	1.4247
7	2.9908	2.6770	2.4133	1.9775	1.6008	1.3365
Extrapolated	3.04	2.63	2.39	1.94	1.52	1.23

TABLE VII

THREE REGION BUCKLING RESULTS

(Units of m^{-2})

N \ $\rho(g.cm^{-3})$	0.41	0.52	0.62	0.77	0.9	1.0
1	2.8857	2.6304	2.3663	1.9735	1.6314	-
3	2.9337	2.6544	2.4068	1.9735	1.6059	-
5	2.9553	2.6752	2.4133	1.9859	1.6019	-
7	2.9908	2.6770	2.4133	1.9775	1.6008	-
Extrapolated	2.991	2.683	2.408	1.980	1.598	-

TABLE VIII

EXTENDED SUBSTITUTION SERIES RESULTS

N	B^2_{TEST} (m^{-2})
1	1.7395
2*	1.5661
3	1.5415
4*	1.5010
5	1.4247
6*	1.4162
7	1.3365
Extrapolated	1.20 ± .01+

*Inverse Substitution Cores

+The value obtained from extrapolation of 1, 3, 5, 7 points only was (1.23 ± .01)

TABLE IX

SUMMARY OF TEST LATTICE BUCKLING RESULTS

28-ELEMENT UO_2 FUEL, 30 cm TRIANGULAR LATTICE PITCH

(Moderator Purity : 99.66 atom %; Temperature = 21.5°C)

COOLANT DENSITY ($g.cm^{-3}$)	BUCKLING (m^{-2})	
	2-Region	3-Region
0.0 (Air)	4.04*	
.41	3.04	2.99
.52	2.63	2.68
.62	2.39	2.41
.77	1.94	1.98
.90	1.52	1.60
1.00 (H_2O)	1.23	-

*Experimental value used for input in analysis

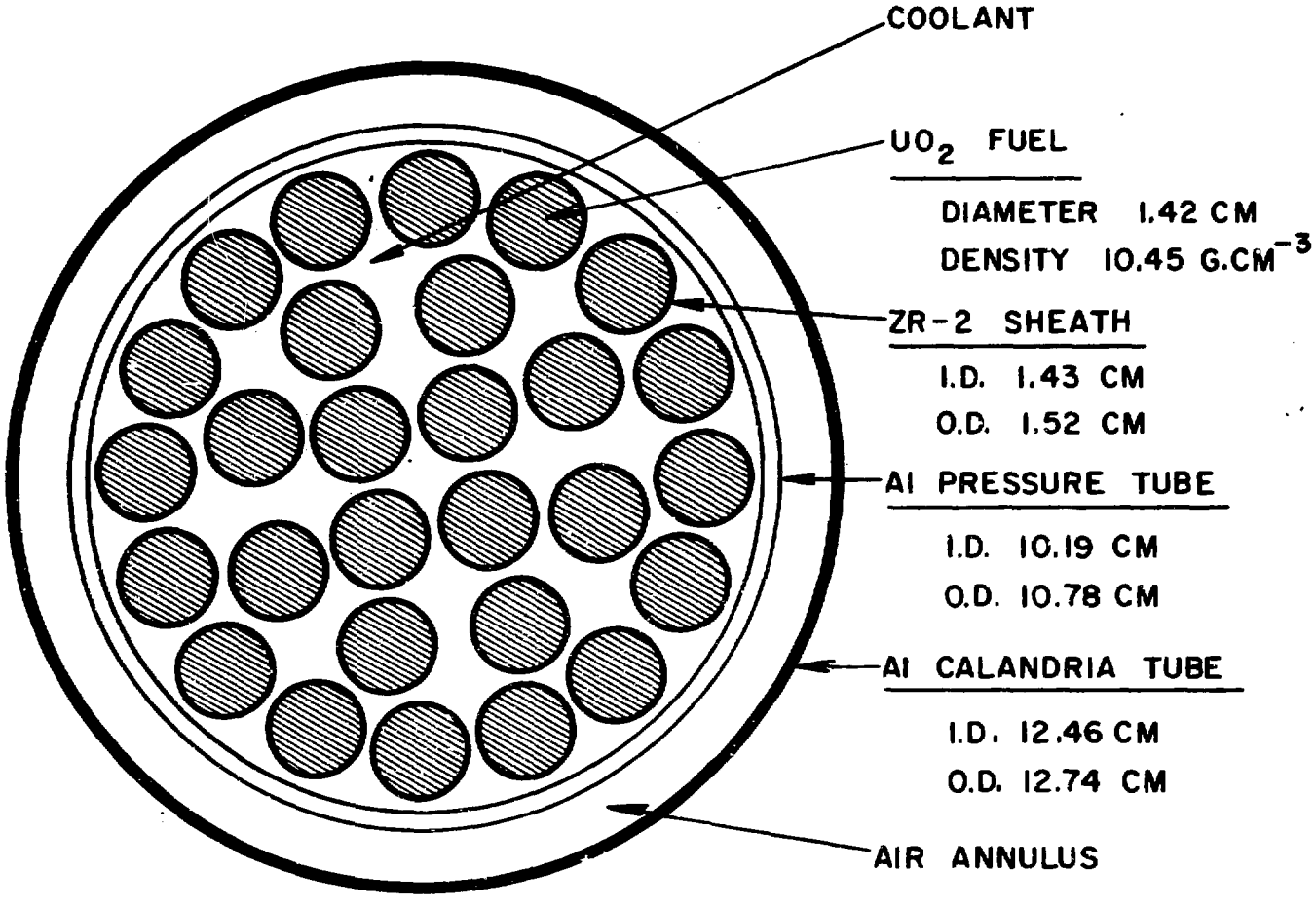


FIG 1 28-ELEMENT NATURAL UO_2 FUEL ASSEMBLY: CROSS-SECTION

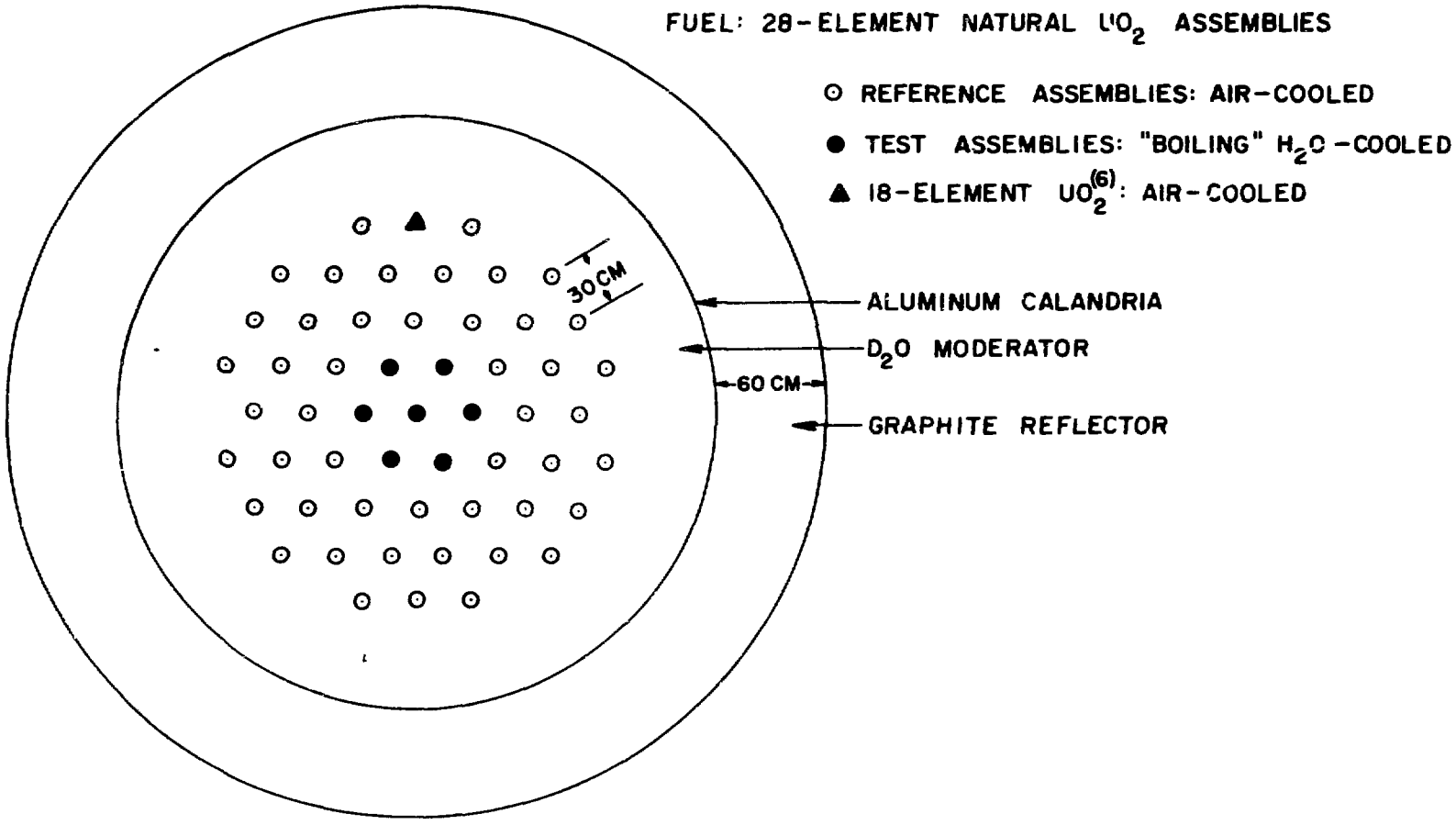


FIG 2 SUBSTITUTION EXPERIMENTS: CORE CONFIGURATION

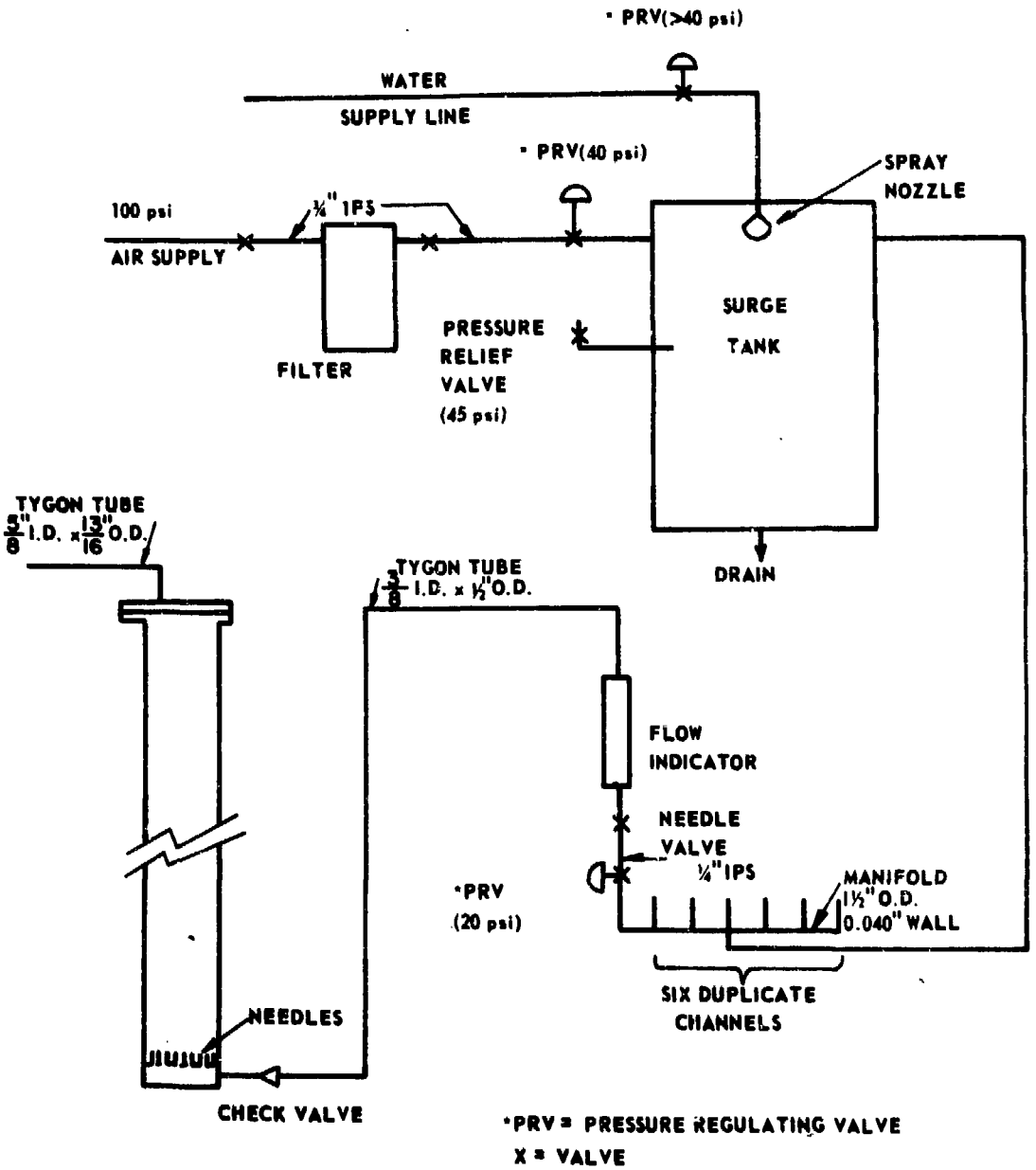


FIG 3 SCHEMATIC DIAGRAM OF AIR BUBBLING EQUIPMENT

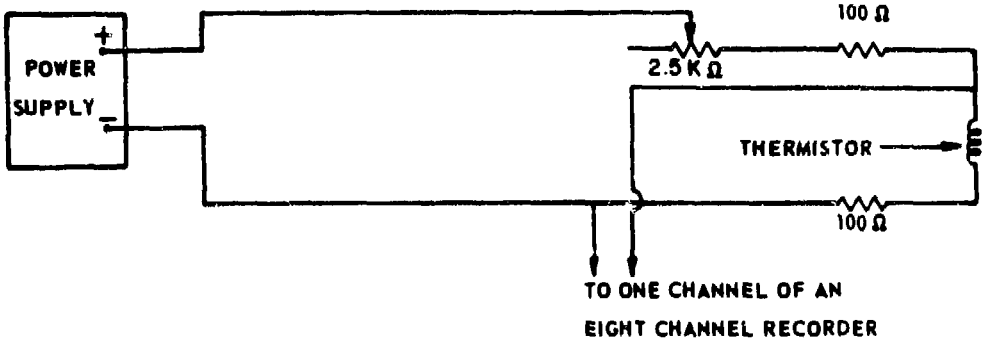


FIG 4a. SCHEMATIC OF A THERMISTOR CIRCUIT. SIX IDENTICAL THERMISTORS MAKE UP ONE PROBE WITH COMMON POWER SUPPLY BUT OUTPUTS GO TO INDIVIDUAL RECORDING CHANNELS.

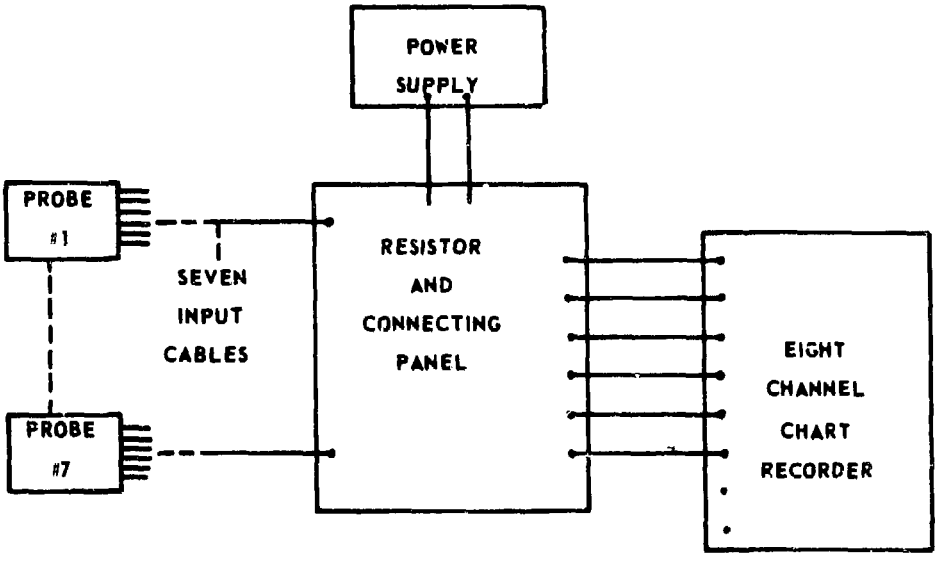


FIG 4b. SCHEMATIC OF MONITORING SYSTEM FOR THERMISTOR PROBES

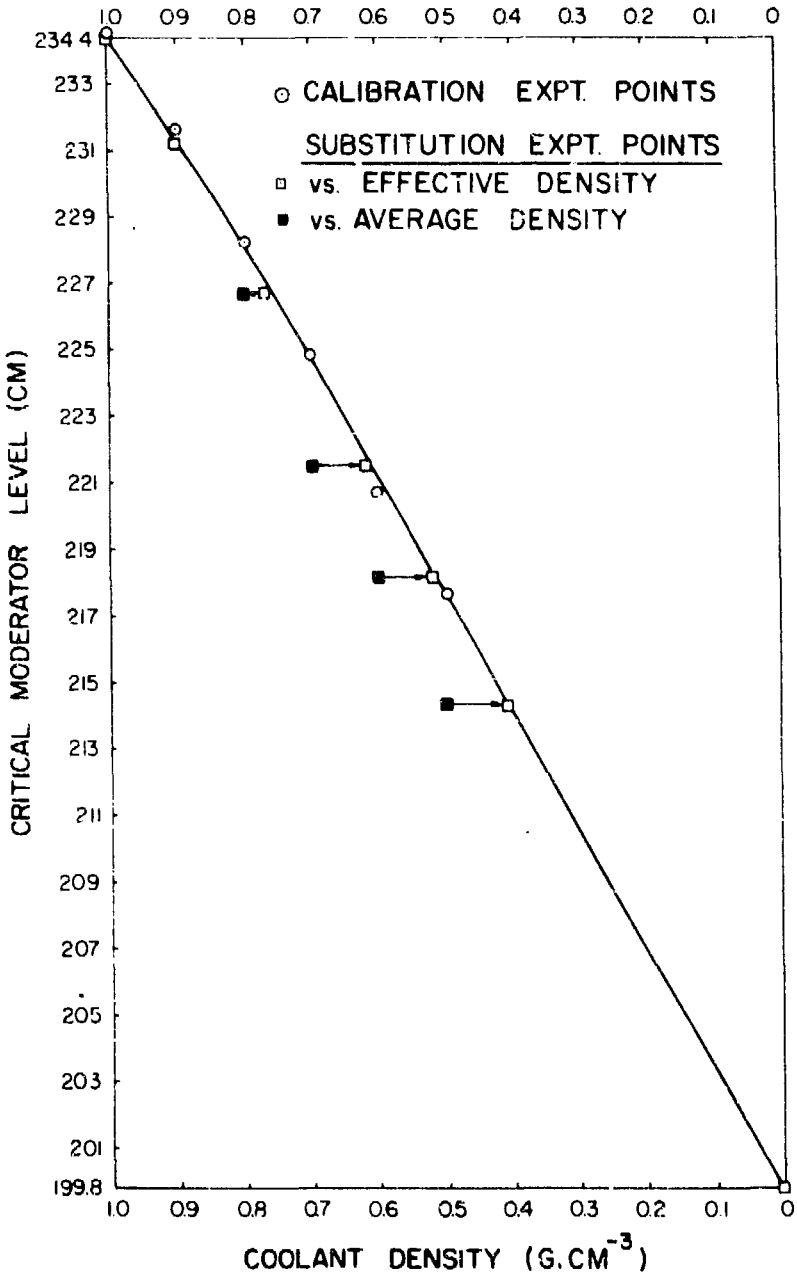


FIG. 5 COOLANT DENSITY CALIBRATION CURVE

FIG 6a TWO REGION SUBSTITUTION

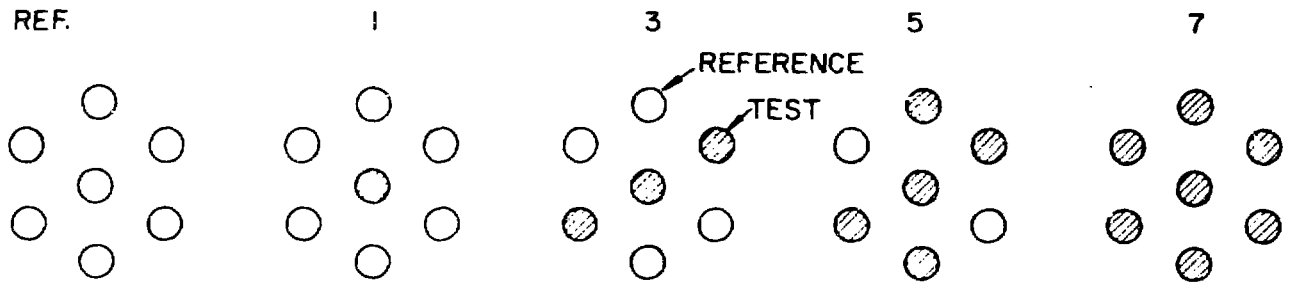


FIG 6b THREE REGION SUBSTITUTION

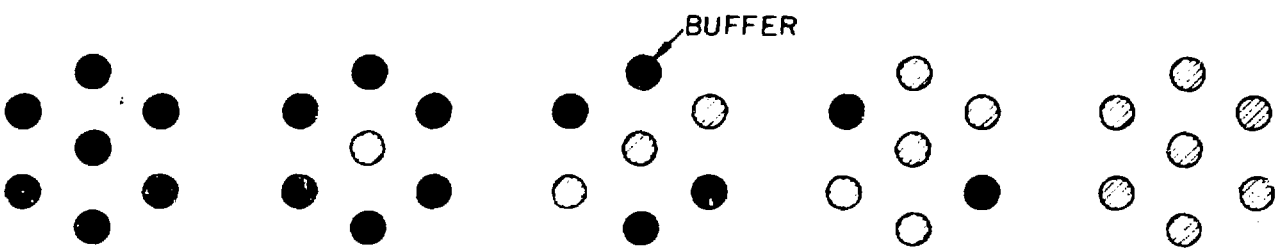
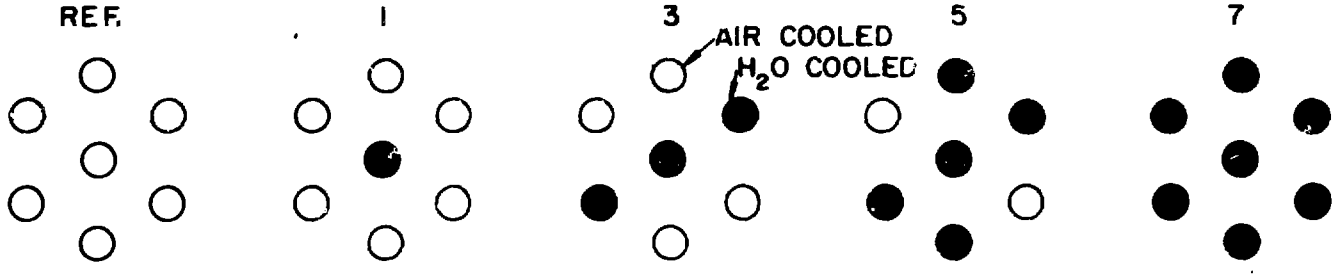


FIG 6 STAGES IN PROGRESSIVE SUBSTITUTION MEASUREMENTS

CONVENTIONAL SUBSTITUTION:



INVERSE SUBSTITUTION:

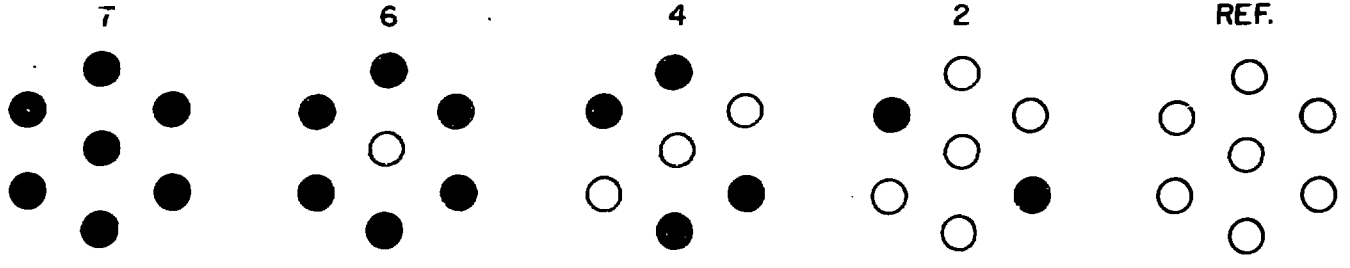


FIG 7 STAGES IN EXTENDED SUBSTITUTION SERIES

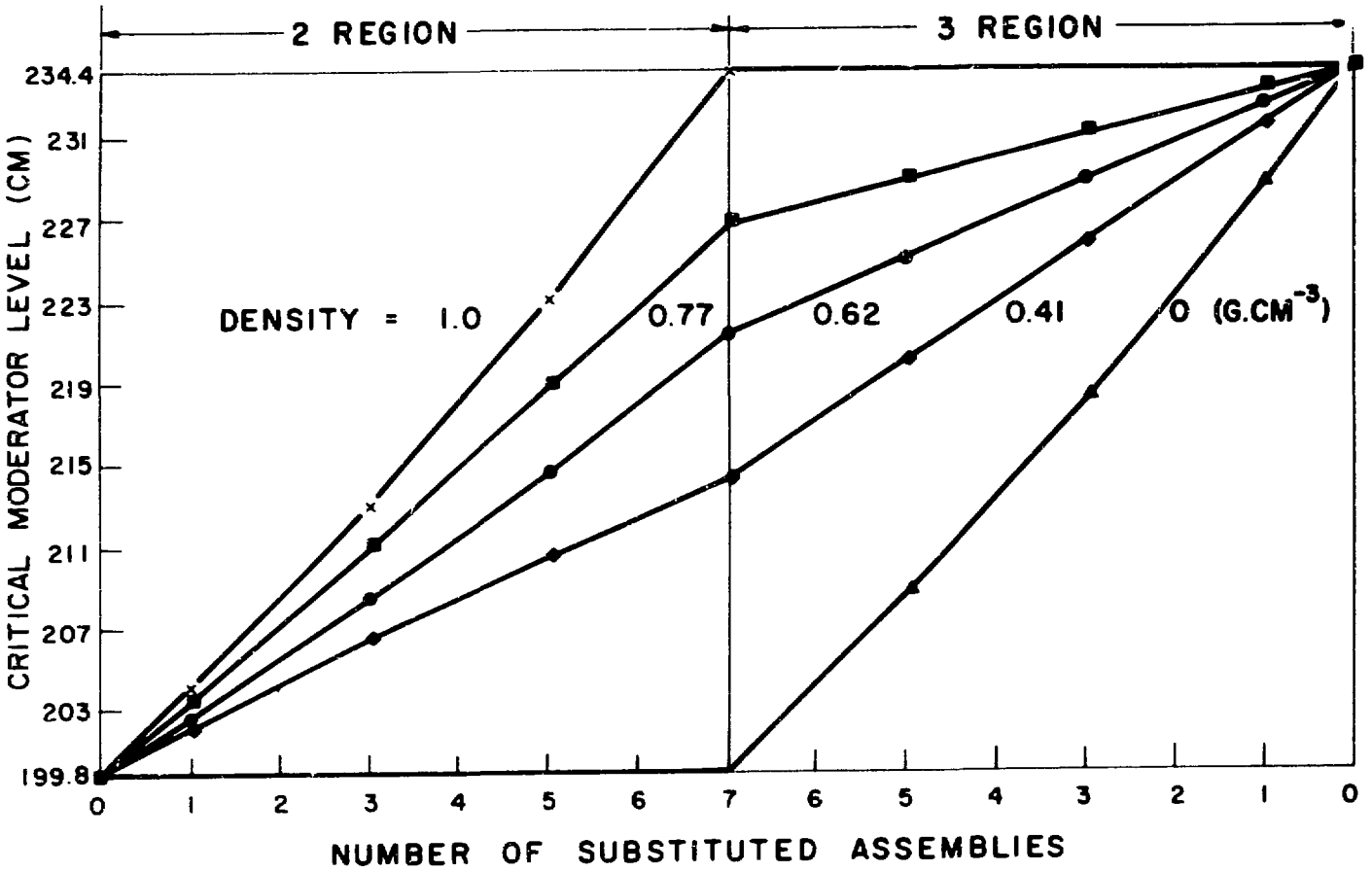


FIG 8 CORRECTED CRITICAL MODERATOR LEVELS – TWO AND THREE REGION SAMPLE DATA (99.66 ATOM % D_2O PURITY, 21.5 °C)

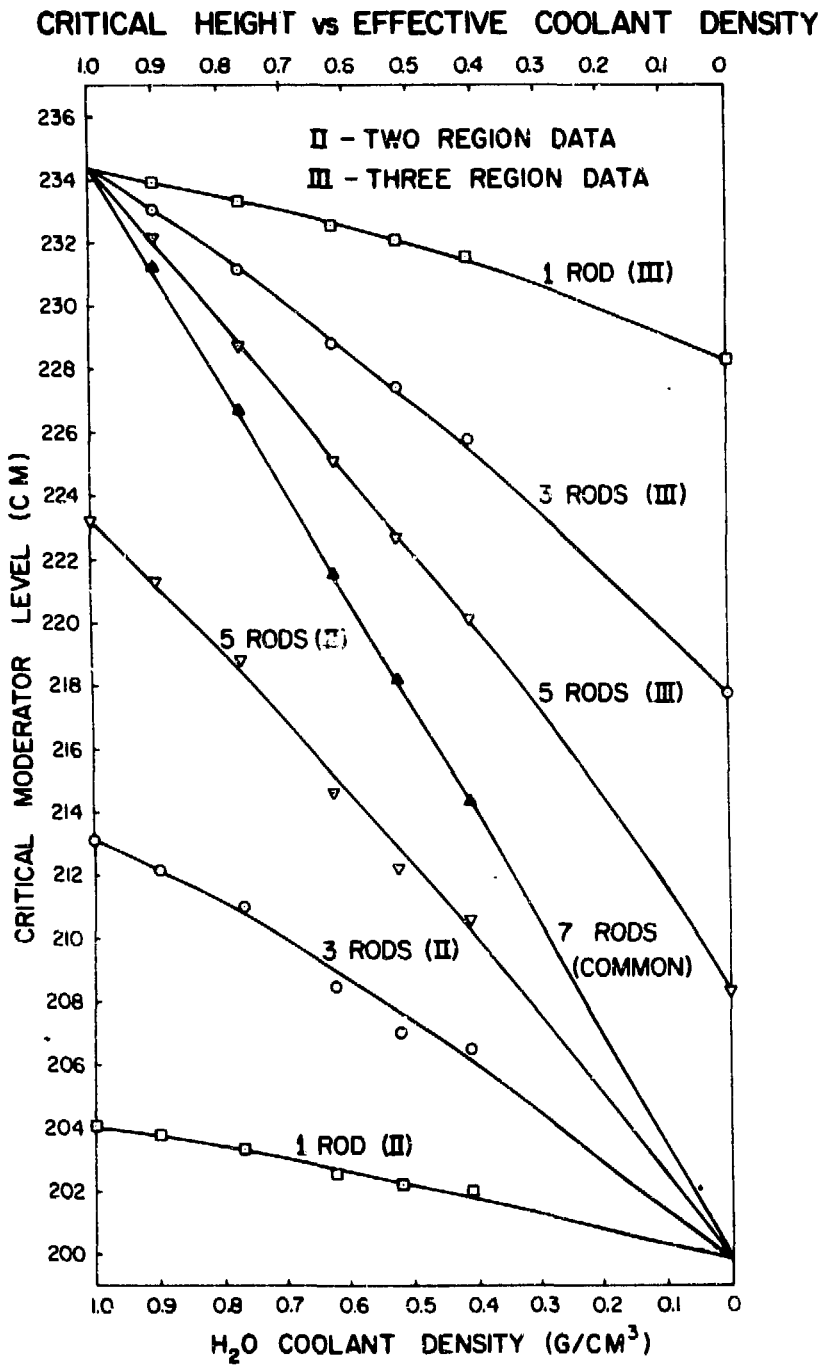


FIG 9 CRITICAL MODERATOR LEVEL vs. EFFECTIVE COOLANT DENSITY IN TEST ASSEMBLIES

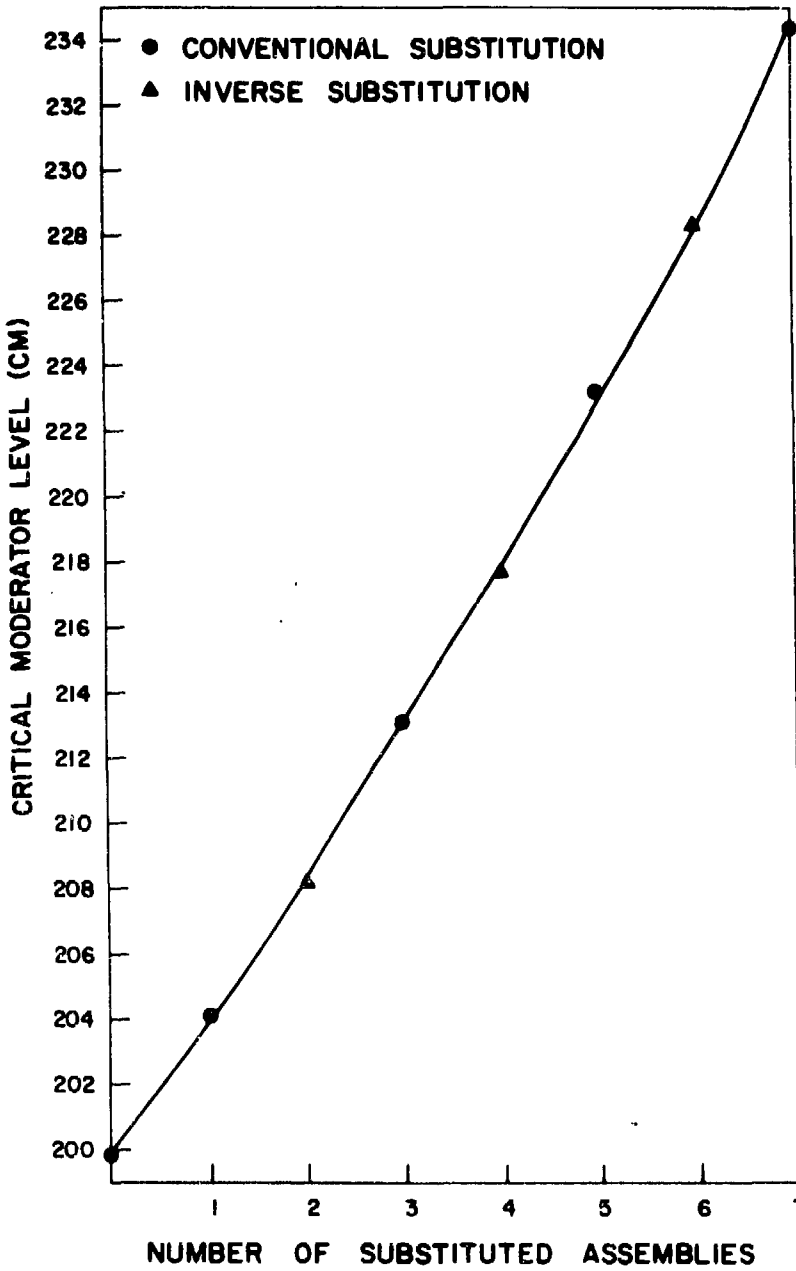


FIG 10 EXTENDED SUBSTITUTION SERIES: CRITICAL MODERATOR LEVEL DATA

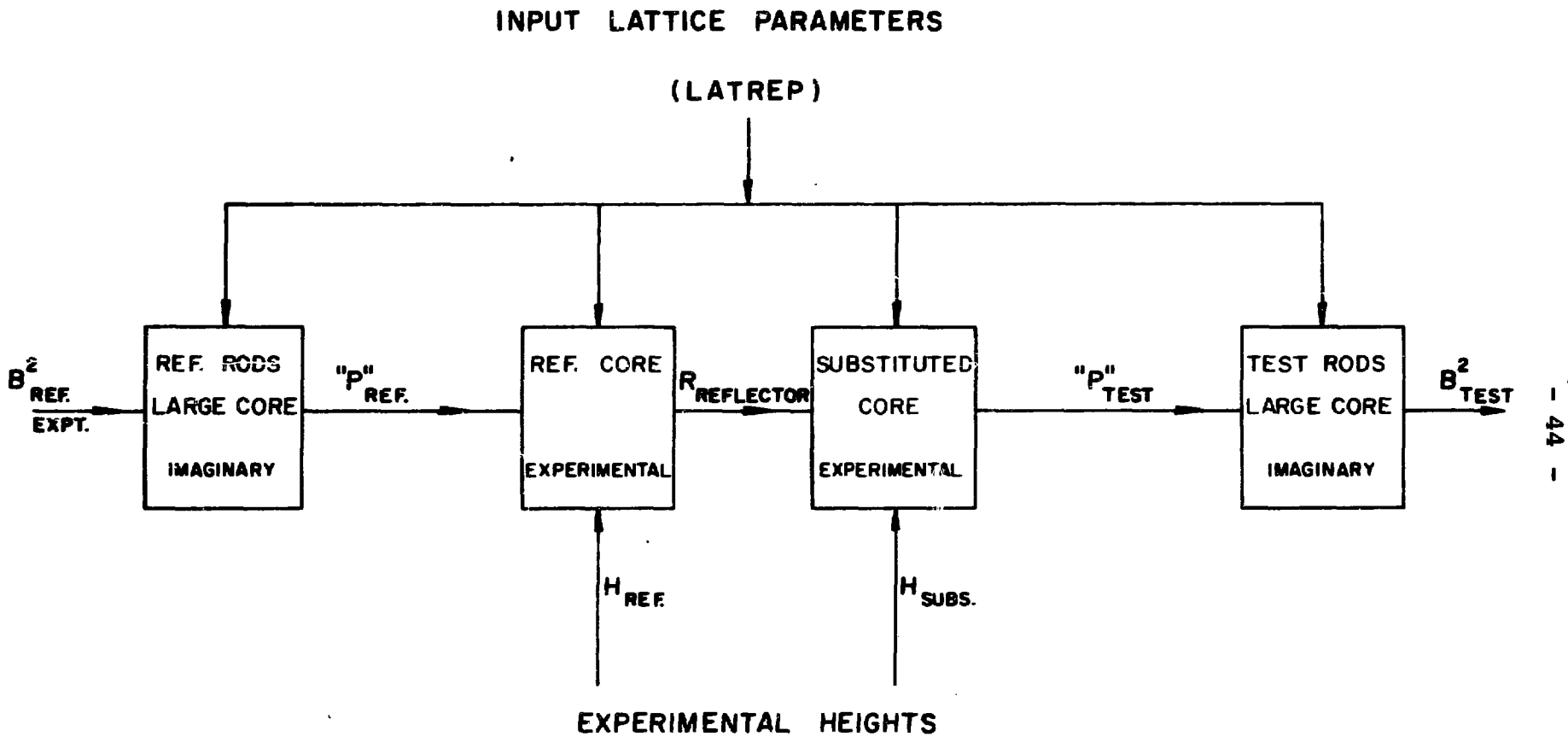


FIG 11 MICRETE 3 ANALYSIS OF SUBSTITUTION EXPERIMENTS

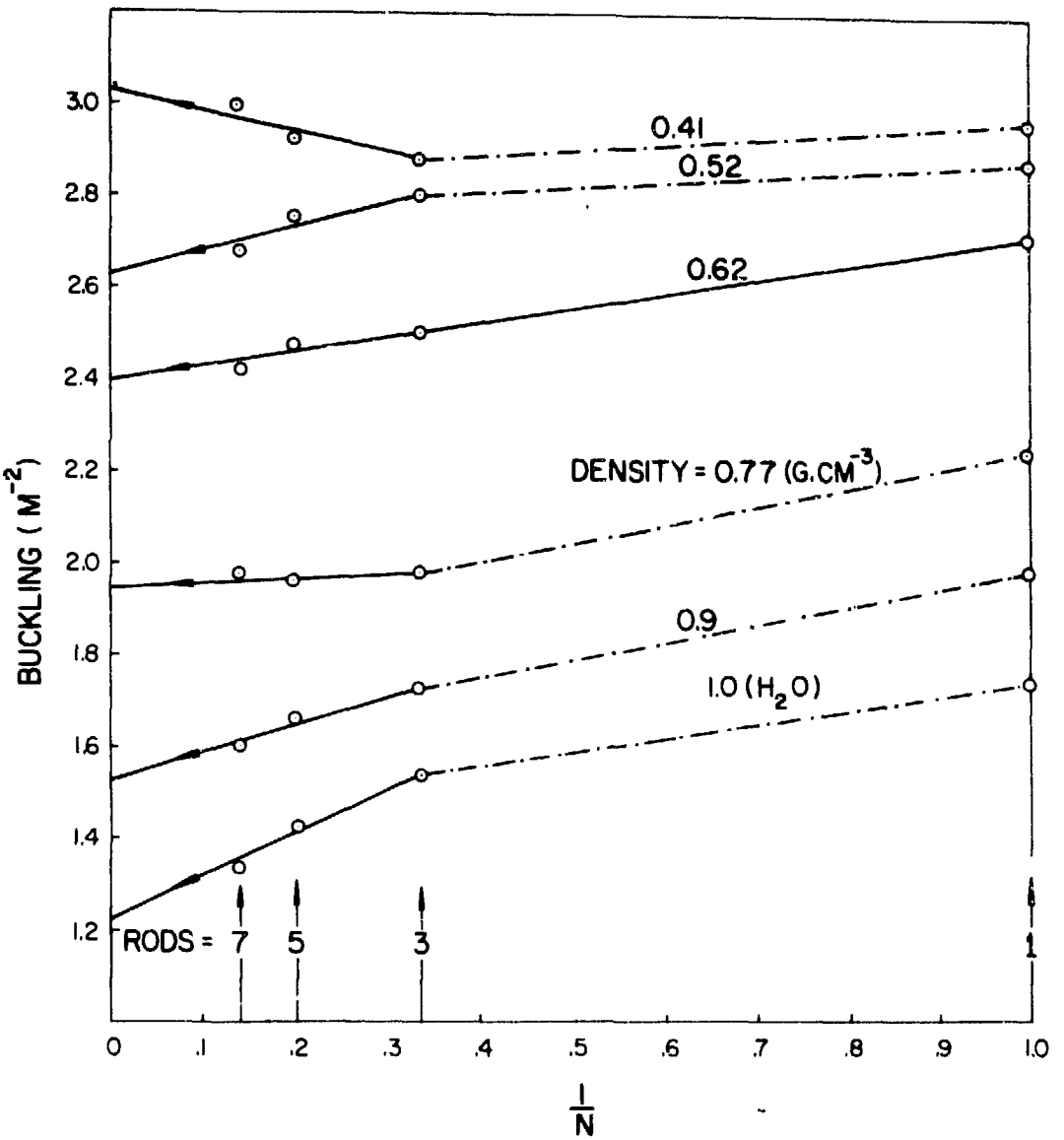


FIG 12 TWO REGION BUCKLING RESULTS

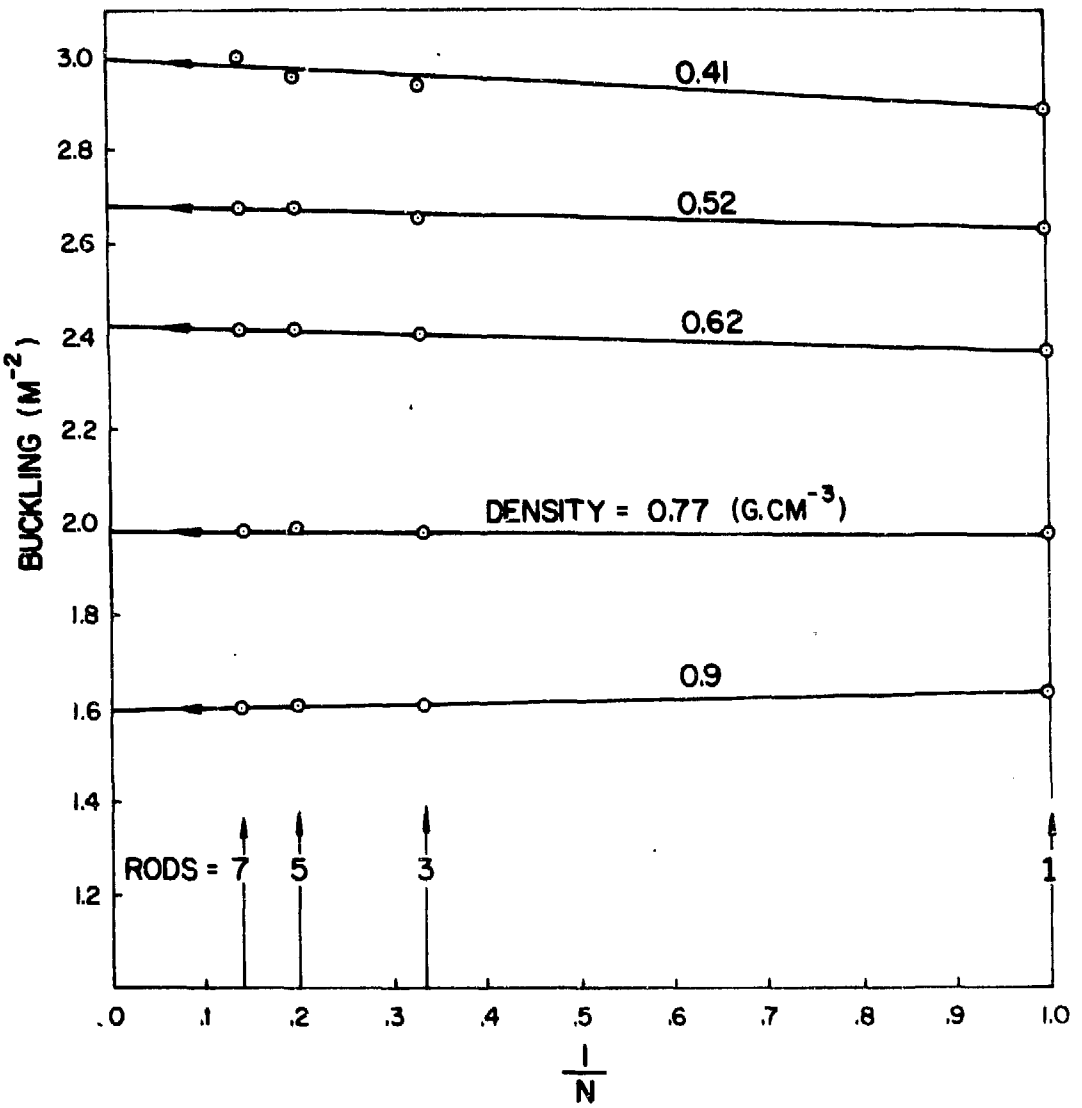


FIG 13 THREE REGION BUCKLING RESULTS

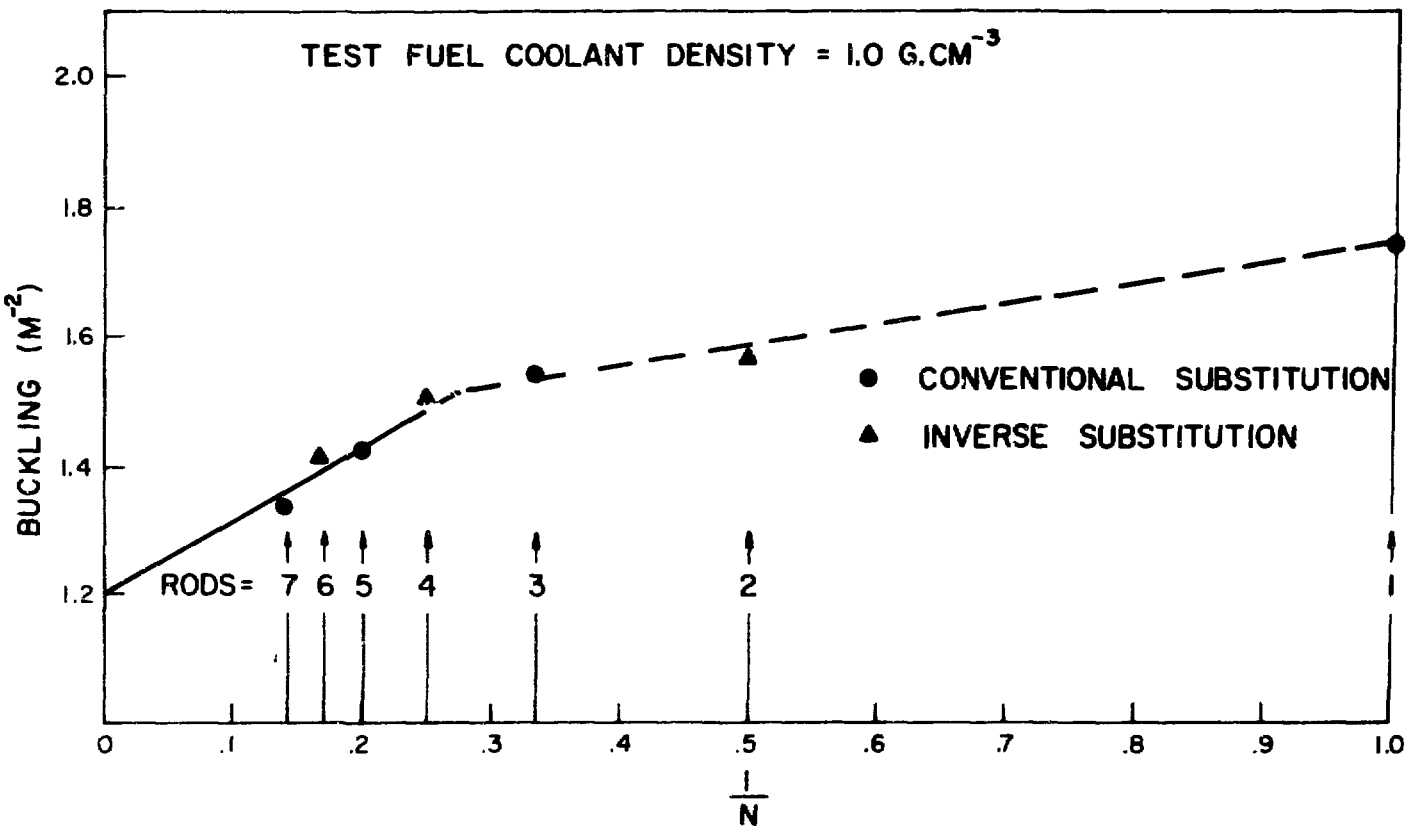


FIG 14 EXTENDED SUBSTITUTION SERIES: BUCKLING RESULTS

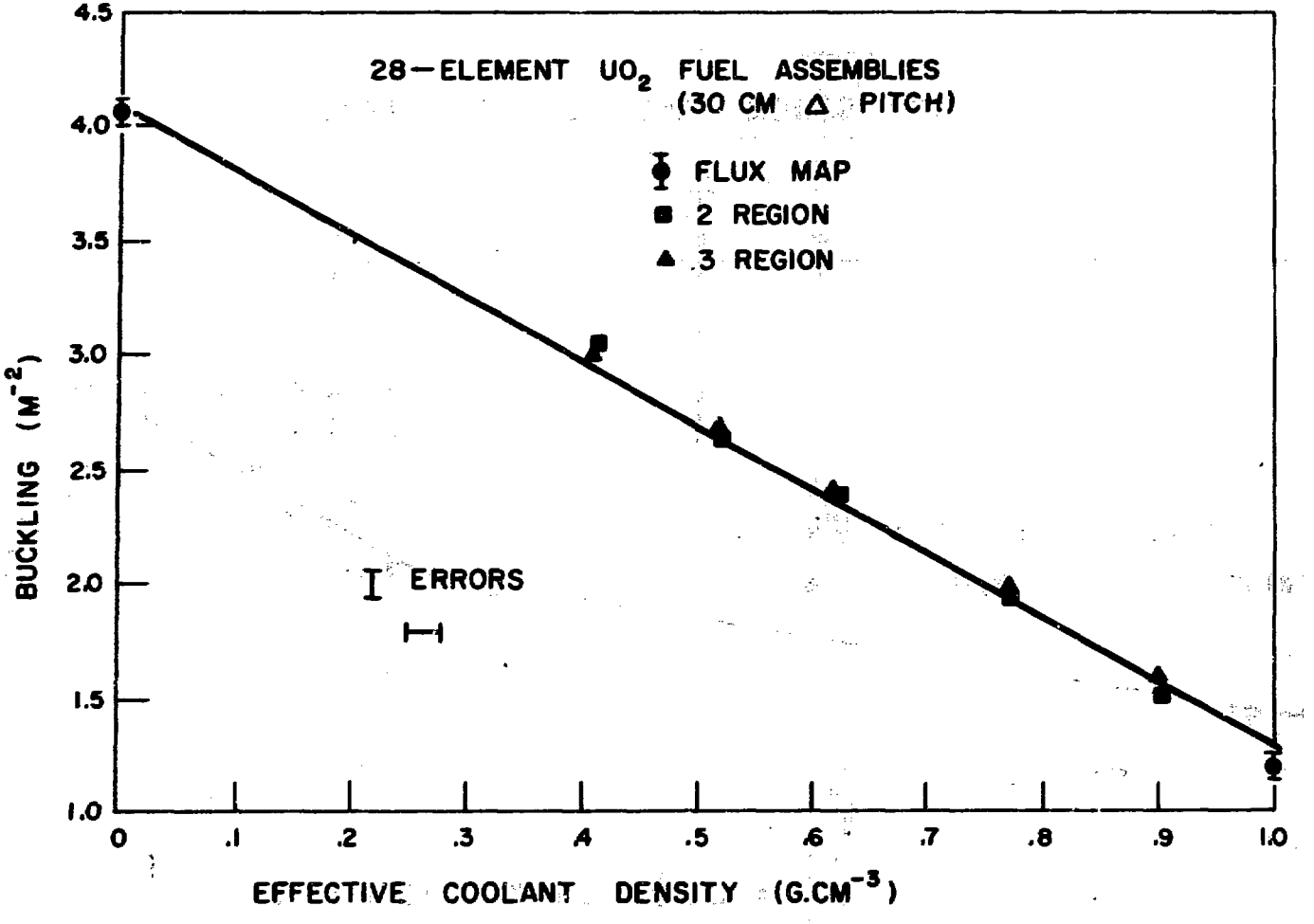


FIG 15 TEST LATTICE BUCKLING vs EFFECTIVE COOLANT DENSITY
(99.66 ATOM % D_2O, 21.5°C)

APPENDIX A

AXIAL FLUX DISTRIBUTION MEASUREMENTS

Axial distribution of thermal neutron density was measured along cell boundary positions at three different radial distances from the core axis. This was repeated both for the all 55 assembly air-cooled reference core and the two region pseudo-reference core with the central seven assemblies H_2O-cooled. The purpose was to check if the total axial reflector savings was a function of critical moderator level and altered in any way by the presence of H_2O in the central test zone.

Strings of Mn foil detectors spaced 10 cm apart spanning the core height were irradiated at radial positions 15, 75 and 150 cm from the core centre. At each distance two sets of foils were used, one on either side of the centre in a diametrical plane. The relative activity of the irradiated foils was measured using an electroscope. A data reduction program processed the data to yield numbers proportional to the neutron flux, after applying corrections for background, foil weights, activity decay etc.

A function of the following form was least square fitted to the activity data:

$$A(Z) = A_o \cos \alpha (Z - Z_o)$$

where Z = axial height of foils from an arbitrary reference level

$A(Z)$ = relative activities

A_o, α and Z_o are the fitted parameters.

Fits were repeated with a reduced number of points in the central region. The fitted α's tended to reach an equilibrium value as the outer points were dropped. Table A-I summarizes the results, the α's quoted being equilibrium values. Results indicate

(1) Within experimental errors, presence of H_2O-cooled fuel assemblies in the central test zone does not affect the total axial reflector savings.

(2) A value of 23 ± 1 cm may be taken as the effective total axial reflector savings to be added to the critical moderator levels of all the substituted cores, to give extrapolated critical heights.

(3) The total axial reflector savings in the outer D_2O reflector region at a radius of 150 cm appears to be a few centimeters higher. But since it remains the same on substitution, the analysis and the results are not affected.

TABLE A-I

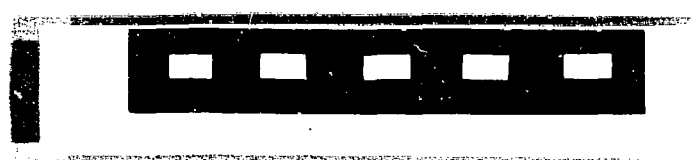
RESULTS OF AXIAL FLUX DISTRIBUTION MEASUREMENTS

Radial Distance cm	East/West Position -	α cm^{-1}	$\Delta\alpha$ cm^{-1}	$H_e = \frac{\pi}{\alpha}$ cm	h_c cm	$\delta_h = (H_e - h_c)$ cm	Mean of E and W cm	Error on Mean cm
I. ALL AIR-COOLED REFERENCE CORE (10th Oct., 1968):								
15	W	1.405	.005	223.6	201.6	22.0	22.2	±1
15	E	1.402	.005	224.1	201.6	22.5		
75	W	1.398	.005	224.7	201.6	23.1	23.5	±1
75	E	1.393	.005	225.5	201.6	23.9		
150	W	1.400	.010	224.4	201.6	22.8	27.5	±2
150	E	1.343	.050	233.9	201.6	32		
II. SEVEN CENTRAL ASSEMBLIES H_2O-COOLED (11th Oct., 1968):								
15	W	1.236	.005	254.2	235.9	18.3	23.2	±1
15	E	1.190	.008	264.0	235.9	28.1		
75	W	1.219	.005	257.7	235.9	21.8	22.2	±1
75	E	1.216	.005	258.4	235.9	22.5		
150	W	1.180	.010	266.2	235.9	30.3	27.9	±2
150	E	1.202	.005	261.4	235.9	25.5		

H_e = Extrapolated critical height

h_c = Critical moderator level

δ_h = Total axial reflector savings less 15 cm.



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