APPROACH TO CRITICAL IN CROCUS

1. INTRODUCTION

In order to understand the process of achieving a critical configuration with a reactor, it is important to define basic terms such as the start-up neutron source, the subcritical multiplication, the effective multiplication factor and the reactivity.

In a nuclear reactor, apart from the neutron production via fission reactions, neutrons are also produced as a result of certain other nuclear reactions and these are usually referred to as source neutrons. These neutrons provide useful information during shutdown and startup sequences, using neutron sensitive monitors placed in and around the reactor. Since changes in the neutron population reflect on the state of the reactor, it is important to verify that there is sufficient neutron flux (also when the reactor is shutdown), which can be detected by an appropriate choice of neutron detectors. The source neutrons needed for the purpose can be intrinsic or external. A brief introduction is given below about the nature of such sources.

1.1 Intrinsic neutron source

Several heavy nuclides present in the reactor undergo spontaneous fission, leading to the release of neutrons. A list of isotopes along with their specific neutron yields is given in **Table 1**.

Table 1 Spontaneous fission neutron sources.

Nuclide	Half-Life of Spontaneous Fission Source [years]	Decay Constant [s^{-1}]	Neutron Production [$s^{-1}g^{-1}$]
^{232}Th	1.0E+18	2.196E-26	1.20E-4
^{233}U	1.0E+15	2.196E-23	1.25E-1
^{235}U	1.8E+17	1.220E-25	7.57E-4
^{238}U	8.0E+15	2.745E-24	1.86E-2
^{239}Pu	5.5E+15	3.994E-24	3.21E-2
^{240}Pu	1.2E+11	1.830E-19	1.47E+3
^{252}Cf	66.0	3.328E-10	3.02E+12

Also, some of the heavier isotopes of uranium and plutonium emit alpha particles and, when boron is present in the fuel, these can interact with ^{11}B (80.1% of natural boron) to produce neutrons from the following reaction:

$$ {}^{11}_{5}B + {}^{4}_{2}He \Rightarrow {}^{14}_{7}N + {}^{1}_{0}n \qquad (1) $$

Neutrons may also be produced by the following gamma interaction with deuterium.

$$ {}^{2}_{1}H + \gamma \Rightarrow {}^{1}_{1}H + {}^{1}_{0}n \qquad (2) $$

There are abundant high energy photons that are produced from the fission products, and their intensity is governed by the operating power level and the duration of the reactor operation. The atom % of deuterium in the water ranges from 0.015% for a light water reactor to 99.8% for a heavy water reactor. Since the neutron emission is essentially due to the production of gamma rays, the source strength for such a source depends on the operational and shutdown history of the reactor.

1.2 External neutron source

Since the neutron source strength due to intrinsic sources is quite small in low-power reactors, it becomes essential to use an external neutron source to bring the reactor to critical in a reasonable period of time. All experimental reactors have provision for introduction of an external neutron source into the reactor, with the sole objective of being able to quantify the shutdown levels properly with the help of neutron sensitive monitors. The external neutron source is usually withdrawn back into its containment outside the reactor, once criticality has been achieved.

Among the standard external sources, ^{252}Cf (spontaneous fission source) is an important one. It emits about 3.0E+12 s^{-1} per gram of the nuclide. However, it is quite expensive and also has certain other drawbacks, viz. a relatively short half-life and the emission of high energy gamma radiation.

Several commercial neutron sources are based on the (α,n) reaction with beryllium as given below:

$$^{9}_{4}Be + {}^{4}_{2}He \rightarrow {}^{12}_{6}C + {}^{1}_{0}n \qquad (3)$$

These sources consist of metallic beryllium uniformly mixed with alpha emitter such as ^{210}Po, ^{226}Ra, ^{238}Pu and ^{241}Am. Another commonly available neutron source is based on the (γ,n) reaction given below:

$$^{2}_{1}H + \gamma \Rightarrow {}^{1}_{1}H + {}^{1}_{0}n \qquad (4)$$

The gamma radiation is usually obtained from a radioisotope such as ^{124}Sb (which can be produced in a reactor from neutron capture in ^{123}Sb). Unlike sources based on (α,n) reaction, photon-based sources do not need to be uniformly mixed. In fact, some designs employ a gamma source contained in a capsule, which is surrounded by a sleeve of beryllium.

1.3 Subcritical multiplication

When a neutron source is placed in a block of moderating material such as graphite, there will be a steady-state neutron flux (or density) distribution, which will depend upon the strength of the neutron source, the absorption of neutrons in the graphite and the neutron leakage. When a fissile material is added to the moderator, there will be an increase in the neutron flux at any given point in the assembly. This will depend upon the kind and amount of fissile material added, as also the geometrical arrangement used for the materials.

The multiplication *M*, of a subcritical fissile assembly of the above type, is defined as the ratio between the thermal neutron flux due to both the primary source and fission, and that due to the primary source alone. If a neutron detector placed in the assembly is assumed to measure, in the presence of a neutron source, a count rate (cps) proportional to the neutron flux, then the neutron multiplication *M* of the system is given by:

$$M = cps\ with\ fuel\ /\ cps\ without\ fuel = I_{wf}/I_{nf} \qquad (5)$$

where I_{wf} and I_{nf} denote neutron flux (or detector response) with and without the fissile material, respectively.

1.4 Effective multiplication factor

In reactor experiments, the spatial and energy distributions of greatest significance are those characteristics of the critical reactor, in which the effective multiplication factor, k or k_{eff} (defined as the ratio of the number of neutrons in one generation to that in the preceding generation), is unity.

In the case of a subcritical assembly with a neutron source, the distributions are more complex. However, in any case, the subcritical multiplication M can be related to the assembly's k value, which is less than 1. Thus, S source neutrons result in kS, k^2S, k^3S... neutrons in the 1st, 2nd, 3rd… generations, respectively. Hence, the M value for the assembly is given by:

$$M = \frac{S(1 + k + k^2 + k^3 + k^4 \ldots\ldots)}{S} = \frac{1}{1-k} \qquad (6)$$

There are certain practical difficulties to obtain an accurate value of k from a multiplication experiment, e.g. difficulties in distributing the source in the normal mode and in measuring the neutron count rates throughout the assembly, particularly when the measurements are required for a series of subcritical states.

Instead, one adopts an empirical approach, the count rate, $C_{wf}(k)$ of a neutron detector placed in or near the assembly being taken as a relative measure of the neutron population and being determined as a function of the corresponding k value. The multiplication M, as observed by such a detector, is then given by:

$$M_{obs} = C_{wf}(k)/C_{nf} \qquad (7)$$

where C_{wf} and C_{nf} correspond to the count rates with and without the fissile material, respectively.

1.5 Reactivity

For completeness, one needs to recall that the concept of a critical configuration can also be defined in terms of reactivity (ρ), defined by:

$$\rho = \frac{\delta k}{k_{eff}} = \frac{k_{eff} - 1}{k_{eff}} \qquad (8)$$

Clearly, the reactivity is zero for a critical reactor, negative for a subcritical assembly and positive for a supercritical reactor. It is common to express reactivity in pcm units (1 pcm=10^{-5}). For example, the maximum possible positive reactivity in CROCUS is +200 pcm, i.e. ρ = 0.002.

1.6 Approach-to-critical experiment

As indicated earlier, there are certain difficulties in relating M_{obs} to the true subcritical multiplication M. These arise from the sensitivity of the spatial and energy distributions of the neutron flux to k_{eff}, source location and assembly geometry. This means that the observed multiplication will depend on the position of the detector, its energy response and details of the experimental conditions. It is known, however, that for approximately k>0.95 and depending on the system, the neutron spectrum remains nearly constant and the flux increases almost proportionately throughout the reactor as k is increased.

Despite its drawbacks as a method to measure k, the multiplication experiment is performed routinely in the context of the safe loading, or "approach to critical", of any given multiplying assembly being brought to critical for the first time. Usually, an approach to critical is achieved by loading fuel material in small steps. It is, however, possible to demonstrate the basic principles by increasing the k value of a subcritical reactor in relation to a change in the moderator level or the withdrawal of a control rod. In an approach-to-critical experiment, the basic relations employed are those of Equations (6 & 7):

$$M_{obs} = \frac{C_{wf}(k)}{C_{nf}} = \frac{1}{1-k} \tag{9}$$

From the definition of k, it is evident that, for $k < 1$, the number of fissions, and hence neutron flux decreases from one generation to the next, the chain reaction being subcritical. In an approach-to-critical experiment, i.e. for achieving $k = 1$, the value of k needs to be changed slowly from a low value, while tracking the response of the detector which is proportional to the factor M_{obs} of Equation (9). By plotting $1/M_{obs}$, or $1/C_{wf}(k)$ (C_{nf} being constant), against the amount of fuel loaded and extrapolating the value to achieve the loading corresponding to $1/M_{obs} = 1/C_{wf}(k) = 0$, one gets the critical configuration. Clearly, as mentioned, one may be employing a different means of varying k, such as change in moderator level or control rod position. The basic procedural steps, however, are similar

An approach-to-critical experiment needs to be done extremely carefully, ensuring that, under any circumstances, criticality is not reached inadvertently. Such a situation could possibly arise because the slope of the plots obtained during an approach-to-critical experiment can have different shapes depending upon the choice of the location of the detector. As illustration, representative results are discussed for an Argonaut multiplication assembly, a cutaway view of which is shown in Fig. 1.

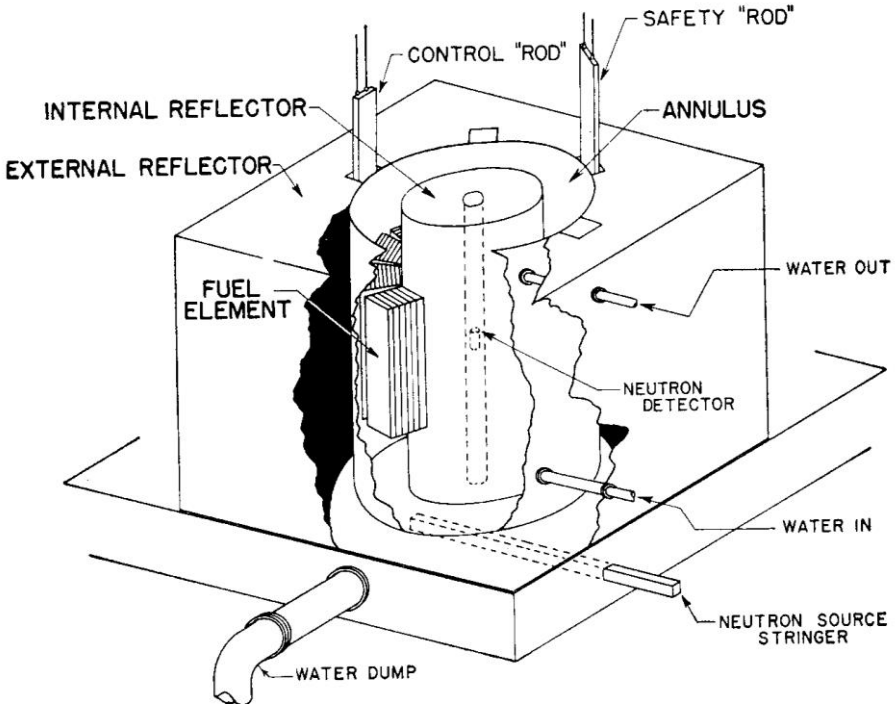


Fig. 1: Schematic view of the Argonaut multiplication assembly.

Fig. **2** illustrates the results of a representative approach-to-critical experiment, carried out using BF_3 detectors, the inverse count rates having being plotted for three different BF_3 detector positions. While an ideal curve would be a linear one, it can be seen that plot No. 1 & 3 are particularly dangerous for extrapolating to the critical mass unless one has carried out measurements with small loading steps till at least about 1.5 kg of ^{235}U (in this particular example). Plot No. 2, which reflects the neutron flux at a location closer to the fissile material, in comparison to the other locations, gives a significantly better description for extrapolation from a loading onwards of about 1.0 kg ^{235}U.

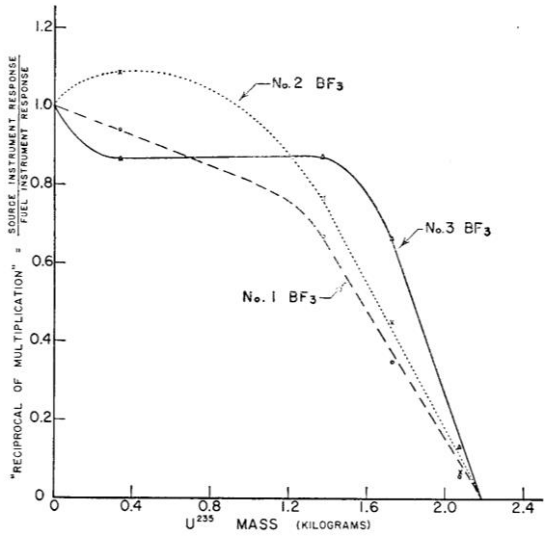


Fig. 2: An approach-to-critical experiment with the Argonaut multiplication assembly using BF3 detectors; 1) detector located at top of the central hole; 2) detector outside the reflector on the fuel side; 3) detector about 120 cm above the central hole.

A typical, well-behaved inverse counts plot is shown in Fig. 3, the prediction of the critical loading here being seen to be relatively reliable, although small-step measurements, even in this case, remain essential till quite near to the end. An often used prescription for calculating the next loading step during an approach to critical is to implement no more than about one-third of the increment predicted (to zero inverse counts) at any given stage [1].

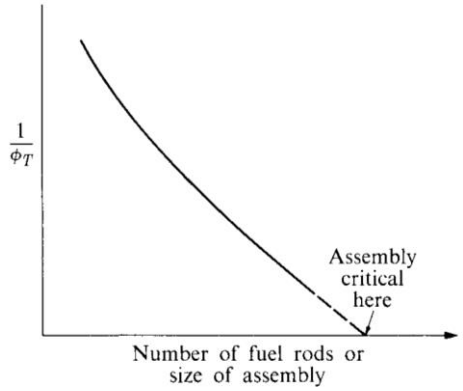


Fig. 3: A "well-behaved" inverse counts plot during an approach-to-critical experiment.

When a nuclear reactor is built, one needs more than just the critical mass of fissile material. For a power reactor, the excess reactivity is required to compensate for burnup effects and ensure a certain cycle length of steady-state operation for the loaded fuel. Even in the case of a zero-power reactor such as CROCUS, some slight excess reactivity is needed in order to be able to increase the neutron flux, i.e., establish different low power values. In all case, the excess reactivity has to be compensated, and this is usually done by introducing absorbing material in the form of control or regulating rods.

In the CROCUS facility, there are two types of approach-to-critical experiments which are possible. The increase of the moderator (water) level, without any use of the control rods, is clearly one possible means of varying the k_{eff} value. The other possibility is to have, at the start of the experiment, at least one of the control rods fully inserted and the moderator level at a specific height. The k_{eff} variation of the reactor can then be carried out in terms of small step withdrawals of the inserted control rod, in order to extrapolate to zero inverse-counts for the critical control rod position.

In the present context, the approach to critical for CROCUS reactor will be carried out by varying the water level.

2. PURPOSE

The purposes of this experiment are:

- Assessment of the suitability of each detector and its location for realistic prediction of the critical height.
- Prediction of the steps of the approach to critical using the inverse counts plot
- Determination of the critical water level with uncertainties.

3. SAFETY MEASURES

In order to work under the standard regulations for control of radiation exposure the safety measures during the experiment are:

- Personal dosimeter for each participant of the experiment (teachers, students and reactor operators) provided by the radiation safety group.
- Drinking and eating is not allowed in the control area.
- Each intended change in reactivity should be carefully evaluated and communicated to the reactor operator.
- The change in water level is to be carried out only by the reactor operator.
- Following a water level change, sufficient time (about 10 minutes) should be allowed for the flux to stabilise, before carrying out the data acquisition.
- The inverse-counts-vs.-water-level plot should be acquired “online”, in order to properly assess the next increment in water level.

4. INSTRUMENTS AND MATERIALS

This experiment measures the axial distribution of thermal neutron flux and the Cd ratio, therefore is done in the core of a nuclear reactor. It needs the following:

- The CROCUS reactor in subcritical state (operator).
- Spillway system for a precise setting and knowledge of the water level.
- Fission (2) and ionisation (2) chambers in the periphery of the CROCUS reactor core, with associated electronics.

5. EXPERIMENTAL PROCEDURES

- Switch on the power supplies of all the associated electronic units and verify the specified settings. Switch on the PC, if needed, for data processing or for using the MCS acquisition system connected to the fission chamber.
- Make sure that the CROCUS reactor has an initial water level such that the configuration is subcritical (k_{eff} ~0.96); this is done by the reactor operator.
- Make sure the startup Pu(α,n)Be neutron source is under the core before starting the measurements.
- Switch on the data acquisition for the fission chambers for a definite period of time. For a given measurement point, the data acquisition is initiated after a waiting time of ~10 min following the change in water level (to ensure stabilisation of the delayed neutron precursors). It is necessary to have sufficient counting statistics for each measurement point (assuming a Poisson distribution). A compromise between accuracy and measurement duration has to be found. Several series of measurements can also be used. Read the ionization current directly on the instrument.
- The water level increment in height is 10 mm in the first step. Increase the water level in steps of suitable height increase and repeat the above measurements.
- Finally, verify that the reactor is critical assuming the predicted water level. Consider the need of the source at that instant.

6. MAIN PARAMETERS MEASURED

In order to obtain the inverse-counts plot for each detector, it is necessary to collect the following data:

- Mark the water level position indicated by the spillway system.
- Collect the fission and ionisation chambers responses at each water level increase step.

7. TYPICAL REAL DATA COLLECTED

Examples of the typical real data collected:

Table 2: Experimental results for fission chambers for approach to critical with water level.

	Average Count Rate (counts/s)		
Water Level	**Fission Chamber 1**	**Fission Chamber 2**	**Fission Chambers Combined**
910 mm	20.97±0.70	20.72±0.69	20.84±0.59
918 mm	25.21±0.22	24.29±0.22	24.75±0.15
926 mm	31.07±0.39	30.92±0.39	30.99±0.23
934 mm	42.36±0.46	41.32±0.45	41.84±0.26
942 mm	66.74±0.58	65.05±0.57	65.90±0.33

Table 3: Experimental results for ionisation chambers for approach to critical with water level.

	Measured Current (Na)	
Water Level	**Ionisation Chamber 1**	**Ionisation Chamber 2**
900 mm	0.18±0.01	0.18±0.01
910 mm	0.22±0.01	0.22±0.01
930 mm	0.38±0.02	0.40±0.02
940 mm	0.62±0.02	0.64±0.02

8. DATA ANALYSIS, ASSUMPTIONS AND EQUATIONS

- The subcritical multiplication M can be related to the assembly's k value, which is less than 1. Thus, S source neutrons result in kS, k^2S, k^3S... neutrons in the 1st, 2nd, 3rd… generations, respectively. Hence, the M value for the assembly is given by:

$$M = \frac{S(1+k+k^2+k^3+k^4\ldots\ldots)}{S} = \frac{1}{1-k}$$

- One adopts an empirical approach, the count rate, $C_{wf}(k)$ of a neutron detector placed in or near the assembly being taken as a relative measure of the neutron population and being determined as a function of the corresponding k value. The multiplication M, as observed by such a detector, is then given by:

$$M_{obs} = \frac{C_{wf}(k)}{C_{nf}} = \frac{1}{1-k}$$

where C_{wf} and C_{nf} correspond to the count rates with and without the fissile material, respectively.

- It is also important to consider other parameters contributing to error apart from the counting statistics. More often, the data obtained need to be processed by additional steps such as addition, subtraction, multiplication and other functional manipulations, in order to arrive at the specific parameter of interest. The numerical values arising from these steps will be distributed in a way that is dependent on both the original distribution and the types of operations carried out.
- The standard deviation on a quantity u(x, y, z, …) can be derived from the standard deviation of each independent variable denoted by σ_x, σ_y, σ_z, etc., using the standard error propagation formula given by:

$$\sigma_u^2 = \left(\frac{\partial u}{\partial x}\right)^2 \sigma_x^2 + \left(\frac{\partial u}{\partial y}\right)^2 \sigma_y^2 + \left(\frac{\partial u}{\partial z}\right)^2 \sigma_z^2 + \ldots..$$

9. PRE-KNOWLEDGE REQUIRED FROM STUDENTS

The students should be familiar with following contents:

- Nuclear measurements: different types of neutron detectors, basic experimental set up.
- Reactor physics: neutron flux in a reactor, spatial and energetic distribution, control rod perturbations, reactivity, critical state.
- Radiation protection: dose rates, limits and all aspects regarding radiological protection, necessary to perform the experience under high security standards.

10.RESULTS

The number of counts during each acquisition interval at each water level height is shown in Table 2. In reality these values do not represent the true flux in these volumes because the detectors are not 100-percent efficient. This is not too much of a concern because the count rates will later be normalized. The factors used to alter the count rates will cancel out in the normalization process and therefore the calculation is unnecessary to perform.

Reported with the counts is the standard deviation involved in their measurement, calculated by Equation (5) where $\sigma_{\bar{x}}$, is the standard deviation in the average count rate, $\bar{x}$ is the average number of counts per acquisition time, N is the number of measurement intervals, and A is the acquisition time for each measurement. Each measurement interval was converted into a rate by dividing by the acquisition time. This equation only works for averages of measurements with identical acquisition times, as was the case in this experiment.

The *1/M* curve was created by normalizing the count rates at each augmented water level to base count rate measured when the water level was at 910 mm, creating a data plot with fitted linear curves that start at y = 1 and are extrapolated to y = 0. Equation (7) shows the normalization formula for calculating *M*, and Table 4 shows the values obtained for *M* for each detector and the average of all of the detectors combined.

Table 4 *M*, multiplication factors for each water level.

Water Level	Fission Chamber 1	Fission Chamber 2	Fission Chambers Combined
910 mm	1.00	1.00	1.00
918 mm	1.20	1.16	1.19
926 mm	1.48	1.47	1.49
934 mm	2.02	1.97	2.00
942 mm	3.18	3.10	3.16

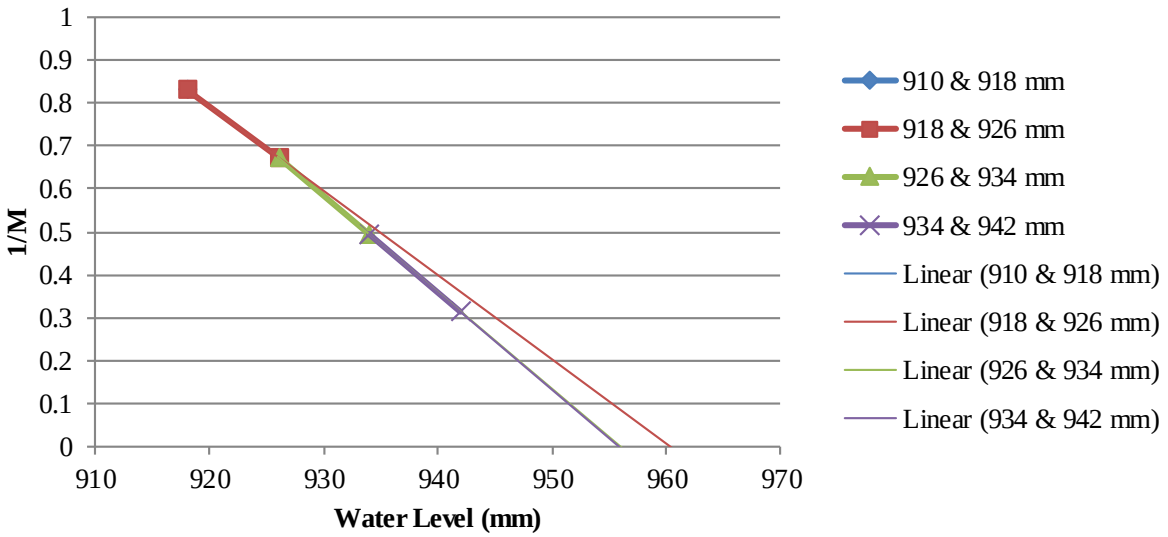


Fig. 4:Averaged and normalized *1/M* curve with extrapolated lines using averaged fission chambers measurements.

The extrapolated lines in Fig. 4 predict where the critical water level is from the values of the current and previous water levels. Only two values, the current and previous, were used for a number of reasons when creating the extrapolated *1/M* curve. The first and most important reason is that this resembles the practice performed in reactor operations. In a fuel loading procedure or reactor startup from hot, clean shutdown, it is essential that a super-critical condition is not reached. Accidental

supercriticality would be a safety and regulatory disaster. Therefore the criticality of the reactor is increased by small increments, after which the new ultimate critical condition is predicted. If the increment is kept sufficiently small, the operators can be assured that the reactor will always be within their control and subcritical.

Table 5 contains the predicted critical positions of each line, calculated by solving for x in each linear equation when y is set equal to zero.

Table 5 Linear equations and predicted critical positions for the average and normalized 1/C curve.

Water Levels Used in Extrapolation (mm)	Linear Equation	Predicted Water Level at Critical (mm)
910 & 918	y = -0.021x+20.123	958.4
918 & 926	y = -0.0196x+18.844	961.0
926 & 934	y = -0.0225x+21.486	955.4
934 & 942	y = -0.0226x+21.614	956.2

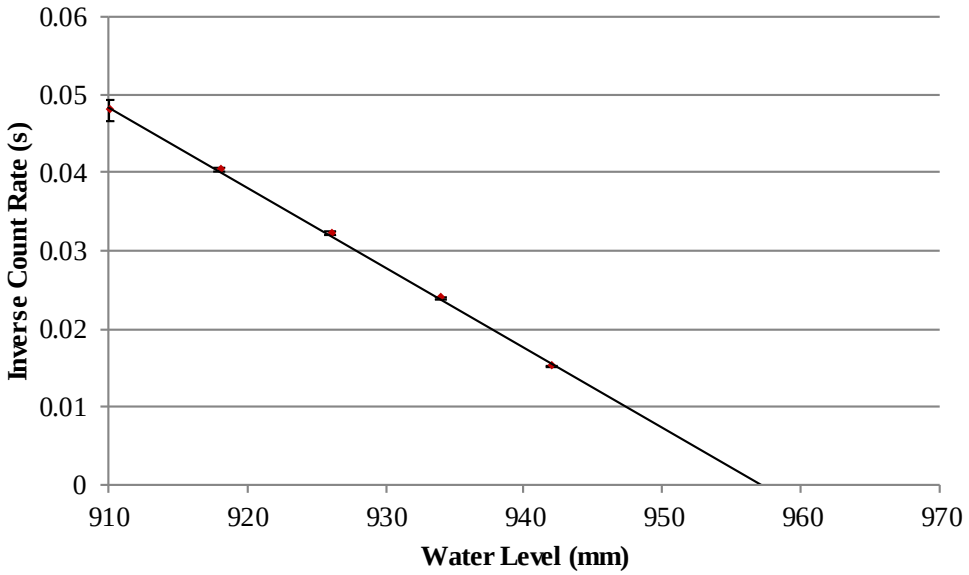


Fig. 5: 1/C curve using all measurements in extrapolation.

11.CONCLUSIONS

Multiple methods were investigated in an attempt to plot the most reliable *1/M* curve possible. Normalizing the count rates and plotting either all data points or the two most recent data points only gave comparably equivalent results to the inverse count rate plot. Considering this, in the case of CROCUS, each of these methods can be used as a valid approach.

The first few extrapolations deviated from the reference critical water level of (955.3±0.1) mm, confirmed by operator, by less than 24 pcm (~6.0 mm of water level). While approaching to criticality, the extrapolated critical position prediction began to fluctuate around (955.8±0.4) mm, within the statistical error of the experimental method. The obtained *1/M* curve confirmed the importance of taking multiple measurements, evaluating statistical errors and doing multiple predictions. The second extrapolation over-predicted the critical position of the water level to a value of 961.0 mm. If only this prediction was used and water was increased to this level without any further consideration, the reactor would have been in a supercritical state, therefore in an unstable regime.

12.REFERENCES

[1] A. E. Profio, Experimental Reactor Physics, John Wiley & sons Inc., 1976.