DEC 19 1967

ACNP-67533

MASTER

INITIAL TESTING
OF THE
LA CROSSE BOILING WATER REACTOR

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1. INTRODUCTION

This topical report covers the low-power nuclear startup program, conducted from July 3 to August 25, 1967, of the La Crosse Boiling Water Reactor (LACBWR) designed and constructed by Allis-Chalmers. This reactor is being operated under Provisional Operating Authorization No. DPRA-5, issued July 3, 1967, which authorizes Allis-Chalmers to use and operate the reactor at powers up to 165 Mwt. Initial operation is limited to 1 Mwt until the completion of certain systems which are required for power levels above 1 Mwt.

The LACBWR is a nuclear power plant of nominal 50 Mw electrical output, which utilizes a forced-circulation, direct-cycle boiling-water reactor as its heat source. The plant is on the east bank of the Mississippi River in Vernon County, Wisconsin, approximately 1 mile south of the village of Genoa, Wisconsin, and approximately 19 miles south of the city of La Crosse, Wisconsin.

The reactor and its auxiliary systems are within a steel containment building. The turbine-generator and associated equipment, the control room for both turbine and reactor controls, and plant shops and offices are in a conventional building adjacent to the containment building. Waste-handling facilities (including facilities for processing liquid wastes and for packaging, decontaminating, and temporarily storing solid wastes) are in a separate building.

The reactor core has an equivalent diameter of 5.09 ft and an active height of 6.92 ft. It contains a total of 72 fuel assemblies, with a nominal uranium content of 8600 kg. Each assembly is enclosed within a removable section of shroud. The channels between adjacent shrouds guide the control rods. The startup core uses both zirconium shrouds and stainless-steel shrouds. The use of stainless-steel shrouds initially reduces the amount of reactivity to be controlled and improves the power distribution within the core.

Each fuel assembly consists of a 10 x 10 array of fuel rods 0.396 in. OD by 83 in. long (active length), on a 0.565-in. pitch. The rods consist of 0.020-in.-thick stainless-steel tubing, filled with 0.350-in.-OD uranium dioxide fuel pellets. The uranium is enriched to 3.63 percent in uranium-235.

The core includes 29 cruciform control rods with B_4C pellets as absorber. The pellets are contained in Inconel tubes, which are packed in cruciform-shaped sheaths of stainless steel; each sheath is 0.3125 in. thick and has a 9.75-in. cruciform span and an active length of 83 in.

The control-rod extensions penetrate the bottom head of the reactor vessel. Normally the rods are positioned by electro-mechanical drive mechanisms that withdraw them downward out of the core to increase reactivity and insert them upward into the core to decrease reactivity. The rods are scrambled upward by hydraulic motors.

1.1 DESIGN DATA

The following are summaries of general design data:

1.1.1 Core

Active diameter	61 in.
Active height	83 in.
Active volume	3980 liters
Total uranium content	8600 kg
Initial U-235 enrichment	3.63%
Water/uranium dioxide volume ratio (unrodded)	2.63
Water reflector thickness	17.0 in.

1.1.2 Fuel Assemblies

No. of assemblies in core	72
No. of fuel rods per assembly	100
Arrangement of rods in assembly	10 x 10
Outside rod diameter	0.396 in.
Rod pitch	0.565 in.
Cladding material	Type-348H stainless steel
Cladding thickness	0.020 in.
UO ₂ density	10.41 g/cm ³
UO ₂ diameter	0.350 in.
Active fuel length	83 in.
End plug material	Type-348H stainless steel

1.1.3 Nuclear Data

Average thermal flux, initial	1.5×10^{13} n/cm ² -sec
Core burnup:	
initial core	12,700 Mwd/ST
equilibrium core	15,000 Mwd/ST
Multiplication factor of cold, clean core (k_{eff})	1.198
Multiplication factor of hot, clean core (k_{eff})	1.167
Shutdown multiplication factor of cold, clean core with all rods in (k_{eff})	0.930

1.1.4 Heat Transfer and Fluid Flow

Average power density in core	41.4 kw/liter
Max. power density in core	164.8 kw/liter
Recirculation flowrate	10,750,000 lb/hr (30,000 gpm)

Heat flux:	
average	109,100 Btu/hr-ft ²
maximum at 100% power	434,700 Btu/hr-ft ²
calculated burnout	810,000 Btu/hr-ft ²
burnout safety factor at 100% power	1.86
burnout safety factor at 120% power	1.55
UO ₂ temperature:	
average at 100% power	930 F
maximum at 100% power	3850 F
maximum at 120% power	4750 F

1.1.5 Control Rods

Total number	29
Pitch (triangular)	9.009 in.
Shape	cruciform
Total blade span	9.750 in.
Thickness	0.313 in.
Length of poison section	83 in.
Absorbing material (80 tubes per cruciform)	B ₄ C

1.1.6 Control-Rod Drive Mechanism

Number of drives	29
Type of drives	electric motor shim, hydraulic motor scram, roller-screw
Location of drives	beneath reactor vessel
Direction of rod withdrawal from core	downward
Direction of scram	upward
Velocity-normal insertion and withdrawal	20 in./min
Scram time to 90% of rod travel	2.0 sec
Scram time to full insertion	2.5 sec

1.1.7 Reactor Vessel

Materials of construction:	
vessel	A302B carbon steel
cladding	Type-304L stainless steel
Diameter, inside	8.25 ft
Thickness of cylindrical portion (including cladding)	4 in.
Thickness of cladding	0.188 in.
Height, inside	37 ft
Design pressure	1400 psig
Design temperature	650 F
Vessel closure type	bolted, double closure seal with between-seal leakoff
Inner thermal shield:	
material	Type-304 stainless steel
thickness	1.25 in.
Weight of contained water and steam during operation	20.92 tons

2. SUMMARY

The first few days of July were spent completing the preoperational test program required prior to fuel loading. An indoctrination meeting attended by all site personnel was held during which the evacuation procedures, containment building entry and exit requirements, and the low power program were discussed. Site and containment building evacuation drills were satisfactorily performed.

Loading of fuel into the LACBWR core was started on July 10, 1967; criticality was achieved, with a 10-element core, at 7:35 pm c.s.t. on July 11, 1967. Loading was resumed on July 13th, and a 72-element core was achieved on July 24, 1967. The full core k_{eff} is 1.20 ± 0.005 . The shutdown margin is 3.5 percent Δk , which is much greater than the required 0.5 percent Δk .

The antimony rod, activated at Babcock and Wilcox's test reactor, was received on August 9, 1967 and was placed in the reactor on August 21st. Due to the low count rates obtained on one of the source range channels, provisions have been made to place a second Sb-Be source in the reactor prior to power escalation.

The program was carried out with very few equipment malfunctions or difficulties. Those that took place were promptly corrected and resulted in no significant loss of testing time.

Gas leakage from some of the control rod drive accumulators was the only equipment malfunction of any significance which occurred during the program. In seven out of eight cases, the leakage was slow enough so that it did not interfere with operation of the reactor. In the eighth case (rod drive number 4) it was decided to put the drive in its fully-inserted position until it could be overhauled. All the drive accumulators will be disassembled and inspected during the coming final construction period.

The lower spacer grid of fuel element number 24 was damaged during the fuel loading. The element was left out of the core and, starting with element number 25, binoculars were used to better observe and guide the entry of the spacer bands in the shroud cans.

A criteria has been developed for the loading of fuel elements in the core without damaging the spacer grids. A cable tensiometer with a range of 40-600 pounds has been purchased from the Pacific Scientific Company. This device can easily detect a change of 20 pounds in the weight of our elements. During August, the tensiometer was successfully tested, using a shroud plug. The tensiometer was also tested again during the performance of Test 108.

3. TESTING

3.1 TEST 101 - FUEL LOADING AND INITIAL CRITICALITY

3.1.1. Preloading Core Description

The shroud can configuration utilized for the initial fuel loading, the instrument locations, and the source location are shown in Fig. 101.1. The 5-curie PuBe source was located 24 in. from the top of the fuel. Channels A, B, and 6 were placed in such a way that their centers were 24 in. from the top of the core. Channels 1 and 2 were placed such that their centers were 41 in. below the top of the core. This instrumentation was installed inside aluminum tubes ~3.5 in. OD, and were positioned in the center of their respective lattice positions.

3.1.2. Fuel Loading

The fuel loading steps were those described in the Low Power Test Program and summarized below.

<u>Step</u>	<u>No. of Elements Loaded in this Step</u>	<u>Total No. of Elements in Core</u>
1	4	4
2	2	6
3	1	7
4	1	8
5	1	9
6	1	10

The location of the first nine elements in the core lattice is shown in Fig. 101.1. The objective of the particular placement utilized was to achieve criticality with stainless steel shrouded elements only, in order to facilitate the interpretation of the experimental data. The tenth element was loaded in position D-6 after moving chamber A to C-1 and renormalizing the count rates.

The pre-startup prediction for criticality was 8 to 9 elements (8 elements were predicted to have a k_{eff} of 1.002), and criticality was actually achieved on the 10th element. The 10-element core had a measured $k_{eff} = 1.01$.

At each loading step, a forecast of the number of elements for criticality was made, along with a forecast of the shutdown margin for the next loading step. These forecasts were based on the results of the reciprocal count rate versus mass plots being kept during initial loading.

A number of reciprocal count rate plots were generated, and points were added at each loading step. These included:

- (a) CR_0/CR for all rods out vs. number of elements
- (b) CR_0/CR for all rods in vs. number of elements
- (c) CR_0/CR for most reactive rod out vs. number of elements
- (d) CR_0/CR vs. all rod bank position at each loading step.

A typical reciprocal count rate plot vs. the number of fuel elements is shown in Fig. 101.2. The count rate of the four-element core was used as the base count rate (CR_0) and the number of elements were then plotted against the ratio of the count rate of the four-element core to the count rate (CR_0) for each particular loading step. The data obtained from Channels 1 and 2 (Fig. 101.1) for extrapolation to the critical number of elements, although data from Channel A was also plotted for information.

A reciprocal count rate plot versus rod bank position was made on approaching criticality as shown in Fig. 101.3. A similar plot was generated at each loading step so that the control rod positions for criticality could be forecast prior to criticality. As shown by Fig. 101.3, criticality was achieved on the 10th element with the rod bank at ≈ 56 in.*

During the reduction and interpretation of the low power program data, it was brought out that, on the average, there is a 1.67-in. difference between the indicated and the actual rod positions, i.e., actual rod position = dial position - 1.67 in.

Table 101-1 gives (in inches) the position above the fuel of the poison section of each rod when the dial indication is zero.

*All rod positions given in this report are dial reading unless otherwise indicated.

TABLE 101-1

POISON DISTANCE ABOVE TOP OF FUEL (ROD FULLY INSERTED)

<u>Rod No.</u>	<u>In.</u>
1	1.647
2	1.779
3	1.592
4	1.620
5	1.665
6	1.565
7	1.808
8	1.605
9	1.638
10	1.627
11	1.685
12	1.680
13	1.643
14	1.685
15	1.750
16	1.590
17	1.594
18	1.668
19	1.627
20	1.672
21	1.651
22	1.532
23	1.793
24	1.710
25	1.720
26	1.815
27	1.672
28	1.605
29	1.792
avg. distance = 1.67 in.

LACBWR CORE CONFIGURATION AND INITIAL LOADING STEPS

FIG. 101.1

RECIPROCAL MULTIPLICATION VS. NUMBER OF FUEL ELEMENTS

INITIAL APPROACH TO CRITICALITY - 10-ELEMENT CORE

FIG. 101.3

3.2 TEST 102 - FUEL LOADING TO FULL CORE

3.2.1 Fuel Loading Description

After criticality had been achieved and the reactivity of the 10-element core established, the loading of fuel elements continued. The loading steps followed were the ones described in Test 102 of the Low Power Nuclear Startup Program, except that two additional loading steps were added to the original proposed sequence. A step from 22 elements to 30 elements was added to reduce the number of elements and hence reactivity added in that step, and a step from 68 to 72 elements was added in order to more gradually approach the minimum shutdown with a single-rod-out condition. These additional steps were discussed with the Safety Committee, and after approval changes were made to the test procedure. The loading steps are summarized in Table 102-1.

TABLE 102-1

FUEL LOADING STEPS

<u>Step No.</u>	<u>No. of Elements Loaded in this Step</u>	<u>No. of Elements in Core</u>
6	1	10
7	1	11
8	2	12
9	2	14
10	4	16
11	6	22
12	8	30
13	6	36
14	12	48
15	20	68
16	4	72

The sequence used for the location of the elements in the core matrix is shown in Fig. 102.1. Also shown in Fig. 102.1 are the source and detector locations utilized during the loading.

3.2.2 Reactivity Determination

The reactivity predictions for the 10-element core were low by ~ 2.5 percent Δk . By the time the 12- and 14-element cores were established, a tentative reason for the difference was advanced: an underestimate of the leakage and/or an underestimate of the flux in the control rod water channels, which would result in the absorption in the stainless steel cans being higher than calculated. If either or both of

these effects existed, it was reasoned that the reactivity gained per loading step would be higher than that which had been predicted when loading into zirconium cans in the fuel matrix later in the loading program, i.e., after 12 elements. This was the case, as is indicated in Table 102-2, and it appears that most of the small core reactivity difference was regained by the time that the core was fully loaded.

Based on the two methods of following the core loading reactivity, it was estimated that the final full core k_{eff} was between 1.19 and 1.21. More information on this parameter is given in Test 105 - Boric Acid Measurements.

After each loading step, the reactivity added in the step, the total reactivity in the core, and the shutdown margin (see Test 104) were determined. Two methods were used to extrapolate from the observed parameters (the rod bank position and the ρ /in. of the bank) to the parameters of interest (reactivity loaded per step and total reactivity in the core). These methods were:

- (a) on-site reduction of the data, and performance of hand calculations; and
- (b) computer calculations normalized to the measured parameter.

3.2.2.1 On-Site Reduction. This method was utilized to obtain quick estimates and projections for the purpose of continuing the test program in a safe and efficient manner. To obtain an estimate of the reactivity loaded during loading step N, the average of the ρ /in. of the 29-rod bank at step N-1 and step N was multiplied by the difference in the 29-rod bank critical positions from step N-1 to loading step N. The results of these calculations were also periodically checked by constructing a differential rod worth curve from the measured ρ /in. and then integrating under this curve.

Beginning with the 16-element core, it was no longer possible to measure the differential rod worth of the 29-rod bank by a positive period. This was a result of the higher differential worth experienced as the rod bank moved into the core, coupled with the positioning accuracy of the control rods. To obtain a desirable positive period, it was necessary to withdraw the total rod bank only a few tenths of an inch. At the time it was felt that uncertainty in the actual rod position could be a significant fraction of the bank movement, and the uncertainty in the differential worth measurement would be large. In order to avoid this situation, the 29-rod bank was calibrated by measuring a partial rod bank and by determining the ratio of the partial bank worth to total bank worth. From the 16-element core through the 36-element core, the partial bank worth was measured by the period method; and the ratio of the partial bank worth to total bank worth was based on estimates using geometric considerations. From the 48-element to the 72-element core, the following method was used:

- (a) 29-rod bank critical obtained
- (b) a partial bank of rods (between 4 and 7 rods) withdrawn ~ 0.75 in., and the period measured

(c) a partial bank (the same as in b) inserted ~ 0.75 in. from the 29-rod critical positions, and counts taken

(d) 29-rod bank inserted ~ 0.75 in. from critical, and counts taken

(e) the differential rod worth of the 29-rod bank equals the differential rod worth of the partial bank times the ratio of the subcritical count rates.

Table 102-2 shows the loading steps and reactivities obtained by the method just described.

TABLE 102-2

<u>LOADING STEPS AND REACTIVITIES ADDED</u>			
<u>Loading Step</u>	<u>No. of Elements in Core</u>	<u>% Reactivity Increase from Previous Step</u>	<u>Total Reactivity Added in %</u>
6	10	0.9	0.9
7	11	1.1	2.0
8	12	1.8	3.6
9	14	2.2	5.8
10	16	1.9	7.7
11	22	3.2	10.9
12	30	3.3	14.2
13	36	1.2	15.4
14	48	1.4	16.8
15	68	2.0	18.8
16	72	0.2	19.0

Using the definition

$$k_{\text{eff}} = e^{\sum \Delta \rho}$$

the method utilized on site gives a 72-element unrodded $k_{\text{eff}} = 1.209$.

3.2.2.2 Normalized Computer Calculations. Prior to startup, a three-dimensional calculational model of the core had been set up for FLARE computations on the computer. After each loading step, the measured parameters of bank position and differential rod worth were put into the computer code. The calculational input was adjusted until the computer model results duplicated the experimental data. With this normalization completed, the computer model was extended to determine other core parameters.

TABLE 102-3

RATIO OF PARTIAL BANK WORTH TO TOTAL BANK WORTH

<u>No. of Elements</u>	<u>29-Rod Bank</u>		<u>28-Rod Bank***</u>	
	<u>Computer Ratio</u>	<u>Site Ratio</u>	<u>Computer Ratio</u>	<u>Site Ratio</u>
16	0.42	0.415**	0.25	0.25
22		0.42**		
30	0.27	0.26**		
36	0.24	0.29**		
48	0.28	0.24*	0.22	0.16*
68		0.12*		0.13*
72	0.18	0.103*	0.16	0.12*

*Done using subcritical count ratio for partial bank to full bank worth.

**Done by geometrical ratio for partial bank to full bank worth.

***One rod all out.

Table 102-3 compares the computer-determined ratio of partial bank worth to the total bank worth. The table also compares partial bank worth to the worths determined from on-site geometrical consideration and by the subcritical count method.

The core k_{eff} as determined after the normalization is given in Table 102-4 for some of the steps.

TABLE 102-4

NORMALIZED COMPUTER KEFF VS. ELEMENTS

<u>No. of Elements</u>	<u>k_{eff}</u>
10	1.009
14	1.058
16	1.074
30	1.131*
48	1.170
72	1.191

*This step not fully normalized.

3.3 TEST 103 - CONTROL ROD CALIBRATION

3.3.1 Bank Criticals

During the testing program with the 72-element core, the reactor was taken critical on the 29-rod bank seven different times with the following results for the bank average:

TABLE 103-1

CRITICAL POSITIONS OF THE 29-ROD BANK

<u>Average, In.</u>	<u>Average - 11.511, In.</u>
11.567	0.056
11.496	-0.015
11.455	-0.056
11.502	-0.009
11.519	0.008
11.507	-0.004
<u>11.534</u>	<u>0.023</u>
Mean = 11.511 in.	Standard Deviation = 0.032 in.

Of the 0.032 in. deviation (which is about 8¢ in reactivity) it was determined that 0.020 in. is due to the fact that several people read the same positions on the indicator to the nearest 1/16 in. It is concluded that the 29-rod bank accuracy is adequate for detection of any 0.6 percent reactivity discrepancies as required by the Technical Specifications.

3.3.2 Symmetry

In the process of individual rod calibrations, 28-rod bank criticals were obtained with the one rod in the ring being measured while fully inserted in order to verify core symmetry.

The average Δ is a measure of the relative worths of a rod in different rings of rod groups in the 9.8411-in.-high core (11.511 in. - 1.67 in.); as expected, the interior rods are stronger over this distance due to the higher radial flux towards the center of the reactor core.

Within each of the seven rod ring groups above the maximum deviation of Δ from the average Δ in the ring is a possible measure of a small lack of symmetry: 0.006 in., 0.039 in., 0.002 in., 0.021 in., 0.011 in., and 0.014 in. respectively for the groups in Table 103-2.

TABLE 103-2

ROD SYMMETRY DATA

<u>Symmetric Rods</u>	<u>$\Delta = 28\text{-Rod Bank} - 11.511, \text{ in.}$</u>	<u>Average Δ, in.</u>	<u>Max. Δ Dev., in.</u>
2, 3, 4, 5	0.418, 0.427, 0.422, 0.418	0.421	0.006
7, 9, 11, 13	0.275, 0.279, 0.230, 0.293	0.269	0.039
6, 8, 10, 12	0.239, 0.241, 0.243, 0.243	0.241	0.002
15, 16, 18, 19, 21, 22, 24, 25	$\left\{ \begin{array}{l} 0.159, 0.163, 0.179, 0.170 \\ 0.190, 0.192, 0.150, 0.163 \end{array} \right\}$	0.171	0.021
14, 17, 20, 23	0.072, 0.085, 0.063, 0.076	0.074	0.011
26, 27, 28, 29	0.054, 0.045, 0.072, 0.063	0.058	0.014

3.3.3 Single Rod Calibrations

3.3.3.1 Calibration Curve Shape Measurements. The period method of obtaining calibrated points on individual rods was supplemented by another method based on a continuous change in count rate as a rod is driven in from critical. This other method was employed for two reasons:

(a) subcritical count rate transients are much simpler and faster than positive period measurements; and

(b) since the subcritical method is essentially continuous, it affords a better opportunity of detecting possible local anomalies in the shape of the rod calibration curve.

Figure 103.1 shows the channel 2 count rate as the center rod was steadily driven in from critical at 1/3 in./sec. Between 83 in. and 53 in. the count rate is unchanged, indicating no measurable reactivity worth. The rate of decrease of the count rate is a maximum when the rod position is at 11 in. Therefore, it is deduced that this point is the location of the maximum reactivity worth per inch.

The rod number 1 curve in Fig. 103.2 is the slope of Fig. 103.1 normalized to one period measurement of rod worth: 12¢/in. at 11 in. after correcting for delayed neutron effects. (The slope of Fig. 103.1 is not zero at 0 in. because of delayed neutrons, whereas the rod worth is in fact essentially zero there.) It can be seen that the three other period measurements' points are within a few percent of the curve, verifying the equivalence of both methods.

3.3.3.2 Period Measurements. In order to obtain more than one measurement by the positive period method on a rod, rods 6, 8, 10 and 12 in the bank were repositioned, thus giving different critical positions for the rod being calibrated. For each of the seven types of rods, the rod was positioned critical near its peak worth point and then supercritical (by typically 20¢), thus giving a near-peak worth measurement. In addition, as indicated in Figs. 103.2 and 103.3, period calibrations were also done at selected points elsewhere on the curves. The shape of the curves shown over regions away from period measurements is defined by the subcritical transient technique.

The area under the calibration curves may be used as a total worth figure for each rod calibrated. Another indication of the total worth is the distance moved by the remaining 28 rods between two criticals: one with the calibrated rod at 83 in. and one with it fully inserted. These results are as follows:

TABLE 103-3

INTEGRATED WORTHS OF INDIVIDUAL RODS

<u>Rod No.</u>	<u>Integrated Worth, ¢</u>	<u>28-Rod Bank Difference, In.</u>
1	176	
4	202	0.930
10	154	0.949
11	289	1.020
20	71	0.199
21	256	0.728
28	58	0.164

It can be seen that, as one might expect, the period calibration results are approximately proportional to the bank differences (remembering the inherent inaccuracies in small bank differences due to variances in readability).

3.3.4 Bank Calibrations

3.3.4.1 Methods. Calibrations of the 28- and 29-rod bank are available by a number of methods described in other sections of this report. These methods are:

(a) Partial bank method, in which a period method calibration on a few rods is ratioed up to a full bank result, based either on how the partial bank and full bank affect subcritical count rates (see Test 102), or how the computer results show partial banks and full banks differ (see below).

(b) Rod drop method, using the count rate transient following a 2-sec insertion (see below).

(c) Calibrated rod trade method, in which a rod, calibrated by the period method, is traded for the remaining bank at criticality. (See Table 103-3.)

Another method, based on calibration against the amount of boric acid added, was used. (See Test 105.) The reason for having a variety of methods is the fact that readability errors prevent an accurate bank calibration by the period method when the bank worth exceeds about 50¢/in.; therefore, it is necessary to compare a number of less direct methods.

3.3.4.2 Partial Bank Method. To calibrate the 28-rod bank (number 20 out) by the partial bank (number 1, 5, 12 and 23) period method, the relative worths of the partial bank and full bank were determined by comparing computed differential worths (see Table 102-3) as well as by using subcritical counts (see Test 102):

$$\text{28-rod bank worth} = \text{measured partial bank worth} \times \frac{\text{computed 28-rod worth}}{\text{computed partial bank worth}}$$

or

$$\text{28-rod bank worth} = \text{measured partial bank worth} \times \left\{ \begin{array}{l} \text{count rate ratio of partial} \\ \text{bank insertion to 28-rod} \\ \text{bank insertion} \end{array} \right.$$

The results are as follows:

28-rod bank worth at 11.89 in. = 265¢/in. (using computed worths)

28-rod bank worth at 11.89 in. = 337¢/in. (using count rates)

Since the FLARE code results on the computer were able to match the ¢/in. worth of the partial bank within a few percent, one has confidence in the computer's ability to give the correct ratio of partial bank and full bank worths. It is believed that there may be a systematic error in the use of sub-count rates to compare partial banks and full banks.

The 29-rod bank was also calibrated by the partial bank method:

29-rod bank worth at 11.61 in. = 266¢/in. (using computed worths)

29-rod bank worth at 11.61 in. = 355¢/in. (using subcritical count rates)

As might be expected, the 266¢/in. value is rather close to values obtained for 28-rod banks.

3.3.4.3 All-Rod Insert Method. A variation on the well-known rod drop method of calibrating control rods was used as one of the methods of obtaining bank calibrations. At the completion of a criticality test, it is usual to shut down by an all-rod insert.

By first having just 2 sec of rod insertion ($2 \text{ sec} \times 1/3 \text{ in./sec}$ equals $2/3 \text{ in.}$ insertion), taking count rate data for 1 min, and then having the remainder of the insertion, one can calibrate the bank between critical and 0.666 in. below critical.

As applied here, the ratio of the count rate at critical before insertion to the integrated count rate in the 10 to 30-sec time interval after insertion is used as the experimental measure of rod worth. This quantity is converted to cents reactivity using theoretical solutions of the neutron kinetic equations (H. Etherington, ed., Nuclear Engineering Handbook, First Edition, 1958, McGraw-Hill Book Co., Inc., pp. 6-115). Result for this reactivity, decided by the distance moved by the rods, is as follows:

28-rod bank (with rod number 20 out): 191¢/in. at 11.08 in.

Unlike large reactivity rod drop tests in which geometrical changes in the flux distribution can disturb the results, the small perturbation rod drop test here should give reasonable answers, though this result is somewhat below the partial bank method result.

3.3.4.4 Calibrated Rod Trade Method. The 28-rod bank with either rod number 4 or rod number 11 out has also been calibrated by using the quotient of the interpreted worth of rod number 4 or rod number 11 (as measured by the period method above in Table 103-3) bank and the distance moved (0.930 in. and 1.020 in.) by the bank to maintain criticality when either rod number 4 or rod number 11 is moved its entire 83 in. The results were:

28-rod bank (number 4 out) worth at 11.4 in. = 217¢/in.

28-rod bank (number 11 out) worth at 11.2 in. = 283¢/in.

They tend to confirm the partial bank (using computer worths) method for the 28-rod bank with rod number 20 out.

Finally, in the process of calibrating rods number 1 and 11, the critical positions of the 6-, 8-, 10- and 12-rod bank was compared with those of the calibrated rod, thus giving a calibration of this bank over a limited range. This gives the following results:

4-rod bank worth = 40.6¢/in. at 12.25 in.

4-rod bank worth = 35.6¢/in. at 14.0 in.

CHANNEL 2 COUNT RATE DURING INSERTION OF ROD NO. 1

FIG. 103.1

DIFFERENTIAL WORTH CURVES FOR RODS NOS. 1, 4, AND 11

DIFFERENTIAL WORTH CURVES FOR RODS
NOS. 10, 20, 21, AND 28

FIG. 103.3

3.4 TEST 104 - SHUTDOWN MARGIN

3.4.1 Method and Results of Shutdown Margin Determination During Loading

At each loading step, a determination of the minimum shutdown margin was made, and a projection of the minimum shutdown margin present at the next loading step also made. A number of different methods were employed to make these determinations; some determinations were made immediately at the site, and others were carried out using an off-site computer (as described in Test 102).

The off-site determination of shutdown margin utilized a three-dimensional calculational model (FLARE) normalized to the experiment as described in Test 102. The measured bank position with the most reactive rod withdrawn, and the differential worth of this one-rod-out partial bank (see Tests 102 and 103) were used in the computer model until, by altering constants, a reasonable reproduction of the experimental measurements was obtained. With the computer model normalized to the experiment, the calculation was extended to the 28 rods fully inserted in the most-reactive-rod-out condition. In this manner, the computer in effect generates the differential rod worth curve of the one-rod-out bank and is made to pass through one or more experimental points. The assumption is made that the shape of the generated curve from the critical position to fully inserted is valid when normalized in this manner.

The on-site determination of a minimum shutdown had one prime objective: to demonstrate, conservatively, at each loading step that the shutdown margin with the most reactive rod removed was greater than 0.5 percent Δk , i.e., that the reactor was subcritical in this condition by more than 0.5 percent Δk , and that fuel loading could safely continue.

This determination for the early loading steps was made by assuming that the shutdown margin decreased by the amount of reactivity added to the unrodded core. The initial base point for this approach was the shutdown margin of the 10-element core as determined by the normalized computer method. As the computer results for other cores were generated, the base point was readjusted. Since the shutdown margins being projected were ten times or more greater than the minimum allowable, this approach was quite satisfactory for the early loading steps.

During later loading steps, when the control rods were further inserted into the core, the minimum shutdown margin was determined by the second method described in Test 104 of the Low Power Nuclear Startup Program Report. This approach has been labeled the "triangulation" method. It assumes that the differential rod worth determined at the fuel loading step of interest is at the peak of the differential rod worth curve. By linearly extrapolating this point to zero, a conservative (or minimum) differential rod worth curve is constructed from the critical rod bank position to the fully inserted position. The shutdown margin is taken as the integral from the critical position to the fully inserted position. As a further conservative precaution, the point at which the differential worth is assumed to be zero was taken as +4.0 in., since this is about where the theoretical curve indicates a near-zero value. An example of the "triangulation" method is given in Fig. 104.1, where the 36-element core data are used for purposes of illustration.

SHUTDOWN MARGIN FOR 36-ELEMENT CORE

FIG. 104.1

Again, these on-site determinations were only used for establishing minimum shutdown margins in order to safely continue fuel loading. The normalized computer method was utilized to determine the final exact shutdown margin. As long as the on-site methods gave shutdown margins (SDM) on the order of \$8 to \$10, it was not necessary to wait for the computer results to continue loading. Had the SDM approached 0.5 percent Δk , it was planned to halt loading until the computer results were available. Table 104-1 gives the computer SDM determinations after the initial normalization. For these determinations, it was assumed the most reactive rod was that determined by pre-startup calculations.

TABLE 104-1

COMPUTER SDM DETERMINATIONS

<u>No. of Elements</u>	<u>SDM, Δk</u>
10	0.149
14	0.122
16	0.120
30	0.094
48	0.09
68	0.09
72	0.06

3.4.2 28-Rod Bank Criticals

Subsequent to the loading of the core, further measurements were made and data obtained to facilitate this task of determining the minimum shutdown margin of the fully loaded core.

The critical position of the 28-rod bank with the 29th rod fully withdrawn was measured for each differently located rod. This data is given in Table 104-2.

TABLE 104-2

28-ROD BANK CRITICAL POSITIONS

29-Rod Bank Crit. Pos. = 9.84 in.*

<u>Rod Withdrawn</u>	<u>28-Rod Bank Critical Position*</u>
#1	9.388 in.
#4 (2, 3, 5)	9.286 in.
#10 (6, 8, 12)	9.135 in.
#11 (7, 9, 13)	9.051 in.
#20 (14, 17, 23)	9.705 in.
#21 (15, 16, 22, 18, 19, 24, 25)	9.303 in.
#28 (26, 27, 29)	9.749 in.

*Actual rod position. For dial reading add 1.67 in.

The rod that was actually measured is shown without parentheses at the left in the "Rod Withdrawn" column of Table 104-2; the rods that are symmetrically located are shown in parentheses in this column.

It should be noted that the data in Table 104-2 show which control rod is the strongest when withdrawn from the 29-rod bank critical position; but further reduction and analysis were required to determine which rod is the strongest when withdrawn from the 29-rods-fully-inserted condition.

From the data in Table 104-2, the configuration with the least shutdown margin was determined by normalizing the calculational model to the measurements given in the table.

3.4.3 Small Bank Criticals

At the conclusion of Test 107, another set of measurements were made as an addendum to Test 104. These measurements had the objective of obtaining data which would aid in determining the shutdown margin. The critical position of a 4-rod bank on the periphery of the core was measured with all other rods fully inserted. The 4-rod bank consisted of rods number 10, 19, 20 and 21. A period measurement on the 4-rod bank was not practical because of the relatively high worth of this bank; instead, a period measurement was made by withdrawing rod number 20 from the bank.

In addition to the 4-rod bank measurements, 3-rod bank measurements were also performed. The critical position of rods number 10, 19 and 21 were measured with rod number 20 fully withdrawn and with all other rods fully inserted. As in the case of the 4-rod bank, a period measurement was made by withdrawing a single rod (rod number 10) from the bank. Table 104-3 summarizes the results of the 3- and 4-rod measurements.

TABLE 104-3

3- AND 4-ROD BANK CRITICALS

Description	Rod No. and Position*				Rod Used for Period	¢/in. of Rod
	10	19	20	21		
4-rod critical	17.1 in.	17.61	16.94 in.	17.97 in.	#20	21.3 ¢/in.
3-rod critical	15.19 in.	15.57 in.	out	14.94 in.	#10	35 ¢/in.

*Actual rod position. For dial reading, add 1.67 in.

3.4.4 Final Normalization of Computer Model to Experimental Data

A wide spectrum of input constants was examined, and many varied combinations were tried. A set was found that reproduced most of the critical configurations (Table 104-4). A final set of FLARE calculations was then performed to determine the minimum shutdown margin with the most reactive rod removed from the core. Calculations were first made, using the normalized computer model, for the various 28-rod bank positions. These results are shown in Table 104-4.

TABLE 104-4

COMPARISON OF CALCULATED AND EXPERIMENTAL EIGENVALUE

<u>Critical Rod Configuration</u> (In. Withdrawn)	<u>Calculated Eigenvalue</u>
29-rod bank at 11.51 in.	1.000
25 rods inserted rod #20 withdrawn rods #10, 19, 21 at 16.93 in.	1.000
25 rods inserted rods #10, 19, 20, 21 at 18.84 in.	0.9890
28 rods at 10.97 in. rod #21 out	0.9984
28 rods at 11.52 in. rod #28 out	1.0013
28 rods at 10.96 in. rod #4 out	0.9977
28 rods at 10.81 in. rod #10 out	0.9992
28 rods at 10.72 rod #11 out	0.9954
28 rods at 11.06 in. rod #1 out	0.9980
28 rods at 11.38 in. rod #20 out	1.0033

As may be seen from the table, the calculated and experimental eigenvalues agree to within ± 0.5 percent for all cases except the 4-rod bank critical. The 1 percent Δk difference in the 4-rod bank critical position was not peculiar to the final set of normalization parameters used; rather, it was common to all combinations that were tried.

The calculated shutdown margins were then obtained, using the same normalized computer model as was used to reproduce the critical configurations. These values are shown in Table 104-5.

TABLE 104-5

CALCULATED SHUTDOWN MARGIN (SDM)

<u>Control Rod Withdrawn</u> (28 Rods Inserted)	<u>Calculated K_{eff}</u>	<u>Calculated SDM</u> (% Δk)
#1 out	0.9404	5.755
#4 out	0.9468	5.083
#10 out	0.9727	2.645
#11 out	0.9532	4.224
#20 out	0.9828	2.049
#21 out	0.9658	3.261
#28 out	0.9643	3.702

From Table 104-5, rod number 20 is the most reactive rod; rods number 10 and 21 respectively are next in order of reactivity.

As might be expected, calculated shutdown for the rods away from the core periphery (rods number 1, 4 and 11) was fairly insensitive to the parameters used for normalization. The variation in the shutdown margin for these rods was only ~ 0.7 percent Δk for the extremes in normalization parameters. Conversely, the rods near the periphery were very sensitive to the normalization parameters, and they exhibited variations of greater than a factor of two for the extremes.

For the 3-rod bank with rod number 20 withdrawn, a normalization was sought which would give a $\zeta/\text{in.}$ closer to the measured value. The greatest change in the calculated $\zeta/\text{in.}$ that could be obtained while still having the calculated critical positions agree with the measured, was ~ 10 percent, which was still smaller than the measured values. Every set of parameters that reproduced the 3-, 28-, and 29-rod bank critical positions, yielded the same calculated shutdown with rod number 20 withdrawn, i.e., ~ 2 percent Δk .

All of these cases yielded about the same actual shutdown margin when defined as:

$$\Delta k) \text{ actual shutdown} = \Delta k) \text{ calculated shutdown} \left[\frac{(\zeta/\text{in.}) \text{ meas.}}{(\zeta/\text{in.}) \text{ calc.}} \right]$$

The actual shutdown margins as defined were from 3.2 percent Δk to 3.5 percent Δk .

In order to obtain a separate estimate of the shutdown, another approach was employed. A set of normalization parameters were selected with the intent of getting closer to the measured $\zeta/\text{in.}$ regardless of the value of the calculated k_{eff} for the 3-rod bank critical. This approach yielded a much higher $\zeta/\text{in.}$ than normalizing to the critical position (~ 35 percent higher). However, the calculated k_{eff} for the critical positions was 0.975. These calculated results can be used to infer the shutdown margin in a manner similar to that just described.

$$\Delta k) \text{ actual shutdown} = \left[k_{\text{eff}}^{\text{crit.}} - k_{\text{eff}}^{\text{inserted}} \right]_{\text{calc.}} \left[\frac{(\phi/\text{in.})_{\text{meas.}}}{(\phi/\text{in.})_{\text{calc.}}} \right]$$

where

$k_{\text{eff}}^{\text{crit.}} = \text{calc. } k_{\text{eff}}$ at the critical rod position

$k_{\text{eff}}^{\text{inserted}} = \text{calc. } k_{\text{eff}}$ with one rod out, 28 rods in

The shutdown margin using this normalization approach was 3.8 percent Δk .

Based on both approaches of normalizing calculations to experiment, i.e., normalizing to the critical position and normalizing to the $\phi/\text{in.}$, a shutdown margin of 3.5 percent Δk has been determined as being a reasonable value.

3.5 TEST 105 - CORE EXCESS REACTIVITY

3.5.1 Boric Acid Addition and Removal

In order to determine the excess reactivity in the 72-element core with all rods out, and to obtain a bank calibration in the process, boric acid was added to the reactor water in five steps. Another reason for boration is the desire to irradiate foils in a control rod configuration somewhat representative of that likely to be encountered during power operation. Finally, the reactivity worth of boric acid offers another means of checking the theoretical calculation model used, since agreement between calculations and experiment here would mean the relative neutron absorptions in water versus fuel are being computed accurately.

Table 105-1 shows the steps used in acid addition. The first step consisted of weighing out only 10.2 lb of boric acid, dissolving it in a special tank, and draining the tank into the reactor during decay heat pump operation. This step was used to prove out the workability of this procedure. The system water volume was kept constant during the entire test by using reactor water pumped to the dissolving tank and returning it to the reactor. It was found that the 885-gpm capacity decay heat pump is more than adequate for mixing. Samples from the top and bottom of the reactor water, taken as soon as 30 min after the end of the addition, showed that an equilibrium concentration to within titration accuracy (~ 2 ppm) had been reached. Visual indication of the increase in boric acid concentration was also immediately evident from the conductivity recorder; it increased $1 \mu\text{mho/cm}$ for every 840 ppm added.

TABLE 105-1

BORIC ACID CONCENTRATION DATA

<u>Step No.</u>	<u>Total Pounds of H_3BO_3 in Reactor Water</u>	<u>Titration Result, ppm</u>	<u>Inferred System Volume, gal.</u>
1	10.2	152	8040
2	70.3	1015	8310
3	148.3	2101	8470
4	254.3	3572	8540
5	394.3	5489	8610
6	--	78	--

A discrepancy in the test appeared when the water volumes of Table 105-1, obtained from the mass and ppm, were higher than the measured volumes. Before the test, 6996 gal were obtained from a primary system water fill, corrected for volume not used in this test; after the test, 6750 gal were measured leaving the system during a water dump. Also, the above table shows a gradual increase in the effective borated volume over a period of several days, as if spreading into undrainable "dead" areas, or as if stagnant volumes of water were occurring. The volume discrepancy was resolved when sampling of the refilled

primary system, after dumping 6750 gal of primary water, indicated significant concentrations of boric acid. This is further evidence that volume in the order of 1500 gal of "non-drainable" water exists (water whose location in the system was identified).

A later check of the piping flow sheets showed that the drain points used to drain the system prior to filling from the overhead storage tank did not completely drain all systems. A major portion of the bypass purification system was not drained because of its elevation relative to the drain point. The check valve in the decay heat system precluded complete draining of this system including the decay heat cooler. Other dead legs were also found to occur on lateral off the primary forced circulation system. The volume of these dead areas accounts for a major portion of the excess water volume discovered following boration of the system.

The sixth step in Table 105-1 is the boric acid concentration after several dumps and refills of the primary system. An additional check on this titrated result appears in Table 105-2. The critical position of the 29-rod bank at 78 ppm is essentially halfway between the 0 and 152 ppm values, indicating that at this stage the removal of boric acid was almost completed; but the presence of the still-remaining boric acid was accurately known.

3.5.2 Criticality Tests

After each acid addition, the reactor was taken to critical in order to obtain critical positions of the control rod banks and to carry out calibrations of these banks by various methods. The critical position data is as follows:

TABLE 105-2

BANK CRITICAL POSITIONS

<u>Step No.</u>	<u>ppm of H₃BO₃</u>	<u>29-rod bank, in.</u>	<u>24-rod bank, in.</u>	<u>22-rod bank, in.</u>
0	0	11.511	13.406	14.332
1	152	11.792	13.810	14.773
2	1015	13.373	16.089	17.545
3	2101	15.903	20.725	23.321
4	3572	21.732	32.920	42.298
5	5489	47.355	--	--
6	78	11.648	--	--

The 24-rod bank used excludes rods number 1, 6, 8, 10 and 12, which remain fully inserted; in the 22-rod bank, rods number 1, 7, 11, 18, 19, 24 and 25 remain fully inserted.

Although the furthest withdrawn position of the 29-rod bank is seen to be only a little beyond the core midplane, this position in fact represents a point quite nearly completely withdrawn in terms of reactivity. The data correlation in Fig. 105.1 shows that a fully withdrawn state requires approximately an additional 450 ppm beyond the maximum 5489 ppm added.

CORRELATION OF BORIC ACID CONCENTRATION AND
WITHDRAWN CRITICAL POSITION OF THE 29-ROD BANK

FIG. 105.1

3.5.3 Bank Calibrations

Prior to adding boric acid, calibration of the 28- and 29-rod banks was obtained by the following methods (see Test 103):

(a) Partial bank method, ratioing a period calibration of a few rods up to a full bank result by either subcritical count rates or by computer worths;

(b) Rod-drop method, using a 2-sec insertion;

(c) Calibrated rod trade method, using the critical position of the bank and a period method calibrated rod.

During the boric acid test, these methods were continued and were applied at various positions achieved by the banks.

The addition of known amounts of boric acid offered a fourth and presumably quite accurate method of obtaining a bank calibration: one obtains the entire shape of the calibration curve by noting distances moved by the bank per ppm of boric acid added. Then, at one point on this curve, an accurate absolute bank calibration (achieved by using the period method on the entire bank) is used to normalize the calibration curve to cents per in.

Figure 105.1 was used to correlate the bank positions at withdrawn positions H with the ppm of boric acid present. This almost straight line was a preferred method of keeping data during the test because it allowed one to make accurate predictions of expected bank positions at successive boric acid additions. The theoretical basis for the straightness of the slope in Fig. 105.1 is based on the criticality relation:

$$1 + \frac{M^2 \pi^2}{(H + \lambda)^2} + M^2 B_r^2 = \frac{k_{\infty o}}{1 + \alpha c}$$

$$c = \frac{k_{\infty o}}{\alpha (1 + M^2 B_r^2)} \times \frac{1}{1 + \frac{M^2 \pi^2}{(H + \lambda)^2 (1 + M^2 B_r^2)}} - \frac{1}{\alpha}$$

where c is the ppm of boric acid, α is a constant, $k_{\infty o}$ is k_{∞} without any boric acid, M^2 is the migration area, B_r^2 is the radial buckling, and λ is an effective total axial reflector savings of the unrodded height H core. Evidently, c is expected to be linear with respect to

$$\left[1 + \frac{49.2}{(H + 2)^2} \right]^{-1},$$

where H is in inches, $\lambda = 2$ in. (being somewhat arbitrarily chosen), and where a value of $M^2 = 33 \text{ cm}^2$ was used.

The 29-rod bank was calibrated by the period method only at 48.83 in., with 9.6 ¢/in. resulting. (At other bank positions, the motion of the bank would be too small to read accurately and also to give an acceptable positive period.) At this position, the slope of Fig. 105.1 is 25.3 ppm/in. Hence

$$\text{boric acid worth at 48.83 in. and 5489 ppm} = \frac{9.6 \text{ ¢/in.}}{25.3 \text{ ppm/in.}} = 0.38 \text{ ¢/ppm.}$$

In order to obtain the entire calibration curve of Fig. 105.2, the slope of Fig. 105.1 at various points in ppm/in. is multiplied by the boric acid worth at that concentration; namely

$$\frac{0.45}{1 + 0.18 \frac{c}{5489}} \text{ ¢/ppm}$$

(since at $c = 0$ ppm, the boric acid is 18 percent stronger than at 5489 ppm, due to less total absorption at $c = 0$ ppm). The curve is therefore normalized to the period calibration of the bank at 5489 and 48.83 in.

Also shown (see Fig. 105.2) are calibration points obtained by other methods. Except for the X's, which are from the partial bank method using subcritical counts (and known to have perhaps systematic errors), the agreement between the curve and other methods is good.

Prior to boric acid addition, and during the test, there were calibrations of 24-rod and 28-rod banks. Most of the values in Table 105-3 have been reported in Test 103 but they are now worth comparing to the 29-rod calibration shown in Fig. 105.2

TABLE 105-3

BANK CALIBRATIONS BEFORE AND DURING THE BORIC ACID TEST

Rod configuration	Calibration for this configuration			Value of 29-rod bank calibration from boric acid
	By rod trade	By partial bank using counts	using computer	
24-rod bank (rod #1, 6, 8, 10 & 12 inserted)		227 ¢/in. at 13.71 in.		200 ¢/in. at 13.71 in.
28-rod bank (rod #20 out)				257 ¢/in. at 11.89 in.
28-rod bank (rod #20 out)		337 ¢/in. at 11.61 in.	265 ¢/in. at 11.61 in.	
28-rod bank (rod #4 out)	217 ¢/in. at 11.4 in.			
28-rod bank (rod #11 out)	283 ¢/in. at 11.2 in.

29-ROD BANK CALIBRATION

Computer results for the 28- and 29-rod bank worths at the same location indicate that these should be equal to within a few percent. Recognizing this, one sees that the rod trade method above lends support to the validity of the boric acid method of rod calibration. The partial bank (based on subcritical counts) and rods insert methods again give results above and below, respectively, the results from the boric acid calibration of the 29-rod bank; while the partial bank method, using computer factors, gives a result in good agreement with results obtained by using the boric acid method.

3.5.4 Core Excess Reactivity

The core excess reactivity has been determined by two semi-independent methods: analysis of the boric acid test results, and the use of extrapolated test results together with PDQ cell calculations of k_{∞} vs ppm. The two methods are in good agreement; accordingly, it is concluded that the core (cold, unrodded) k_{eff} is 1.20 ± 0.005 .

Boric acid test results are shown in Fig. 105.1 and in Table 105-2. In order to extrapolate to the fully withdrawn ppm values for each control rod bank, the following analysis was performed. For the 29-rod bank, a radial buckling of 0.009, an M^2 of 33 cm² and a clean k_{∞} of 1.245 were assumed. (The parameters chosen were the pre-startup calculated values, but the results are relatively insensitive to the values of these parameters.) The upper reflector savings required to match the 0 ppm 29-rod bank critical position was calculated as 5.4 cm from

$$k_{eff} = k_{\infty} (1 + B^2 M^2)^{-1} = 1$$

where

$$B^2 = B_r^2 + B_z^2$$

$$B_z^2 = \left(\frac{\pi}{Z + RS} \right)^2$$

Z = bank position (cm withdrawn)

For each succeeding 29-rod critical bank position, values of $k_{eff} - 1$ were calculated from the measured Z and the above constants. A curve of $k_{eff} - 1$ vs ppm was drawn for the five measured points and extrapolated to the $k_{eff} - 1$ value corresponding to the full core height of 82.92 in. (See Fig. 105.3.) Thus, it was determined that 5940 ppm would have been required for an all-rods-out just-critical core. The corresponding k_{eff} value is 1.202. The advantage of using the $k_{eff} - 1$ vs ppm curve is that it is approximately linear, as would be expected on theoretical grounds. The bank position vs ppm curves has too much curvature to allow unique extrapolation.

A similar procedure was used for the 24- and 22-rod banks (Figs. 105.4 and 105.5) except that different radial bucklings were used, i.e., those required for criticality at the

measured 0 ppm bank positions, and the corresponding axial bucklings. The difference between these radial bucklings and the actual geometric bucklings is a measure of the 5- and 7-rod worths. The corresponding reactivities were added to the extrapolated $k_{eff} - 1$ values for 82.92 in. to get the all-rods-out reactivities. The extrapolated results were $k_{eff} = 1.196$, (4860 ppm) and $k_{eff} = 1.196$, (4320 ppm), respectively, for the 24- and 22-rod banks. The average of the inferred k_{eff} values for the three different rod banks provides the core excess reactivity, which is $k_{eff} = 1.198$.

Previously calculated k_{∞} vs ppm results may also be used to obtain core reactivity. Figure 105.6 shows PDQ cell results for Zr and s.s. shrouded elements. The slopes of these curves are -0.0360 and -0.0300 Δk per 1000 ppm, respectively. Using reactivity weighting factors of 0.40 and 0.60 for s.s. and Zr shrouded elements (these factors being determined from cold, clean unrodded FLARE power distributions), the average actual "core" slope is -0.0324 Δk per 1000 ppm. In addition to reducing k_{∞} , the boric acid increases leakage because of increased neutron absorption in the reflector. It is estimated that 6000 ppm will reduce the reflector savings to about one-half of its unborated value. This is equivalent to approximately -0.0025 Δk or -0.0004 Δk per 1000 ppm. Hence, the total boric acid reactivity effect is -0.0328 Δk per 1000 ppm. Using the 5940 ppm all-rods-out value inferred above, the core excess reactivity is $k_{eff} = 1.195$, which is in excellent agreement with the values quoted above.

INFERRED EXCESS REACTIVITY VS. BORIC ACID
CONCENTRATION - 29-ROD BANK

FIG. 105.3

INFERRED EXCESS REACTIVITY VS. BORIC ACID
CONCENTRATION - 24-ROD BANK

FIG. 105.4

INFERRED EXCESS REACTIVITY VS. BORIC ACID
CONCENTRATION - 22-ROD BANK

FIG. 105.5

k_8 VS. BORIC ACID CONCENTRATION
(PDQ CELL CALCULATIONS)

FIG. 105.6

3.6 TEST 107 - NEUTRON FLUX DISTRIBUTION

3.6.1 Irradiation of Foils

Prior to the principal foil irradiation run, a test run with foils in fuel element H-6 was performed to verify that a suitable count rate could be obtained from a selected irradiation of 84×10^{-5} amp sec and 39×10^{-5} amp-sec, according to readings on channels B and number 6 ion chambers respectively (located in-core). The selected irradiation was based on calculations using neutron flux sensitivity of ion chambers in combination with the time-integrated flux required to adequately activate gold. It was concluded, after counting the test foils, that it was preferable to increase the exposure time by a factor of six.

The final and principal foil run irradiated 100 foils placed in 7 elements as follows:

- 13 in F-6 to obtain an axial distribution
- 13 in H-8 to obtain an axial distribution
- 11 in F-6 to obtain an interpin distribution
- 10 in H-8 to obtain an interpin distribution
- 10 in F-6 to obtain an intra-element distribution
- 10 in H-8 to obtain an intra-element distribution
- 17 distributed among F-7, F-8, F-9, G-6 and K-6 to obtain a radial distribution
- 16 distributed among elements F-6, F-7, F-8 and H-8 to obtain fluxes on the edges and corners
- 100 foils total

Of these, nine foils were within cadmium covers in order to measure the epithermal flux at selected locations. The seven elements utilized for the foil measurements are shown in Fig. 107.1 and are marked with the letter "F".

The irradiation run consisted of bringing the reactor critical (42.298 in. bank) when 3523 ppm of boric acid was present, and then proceeding to 42.863 in. on the 22-rod bank (rods number 1, 7, 11, 18, 19, 24 and 25 fully inserted) for an 81-sec period. At 0.333×10^{-5} amps on the channel number 6 ion chamber, the reactor power rise was stopped and held constant for 600 sec. This gave an integrated current of $681 \times 0.333 \times 10^{-5} = 227 \times 10^{-5}$ amp sec.

After shutdown, the elements with foils were pulled up; the foils were taken to the counting room and were counted in a scintillation counter for the next couple of days. The count rate for each foil was corrected for decay back to the time of shutdown and was also converted to total disintegrations (of all γ energies) per min, using the counter efficiency. This figure, divided by the foil weight (typically 0.05 gm), is the dpm/gm plotted in the results below. (See Figs. 107.2 through 107.7.)

3.6.2 Axial Flux Distribution

An axial flux distribution was obtained in the center element F-6, and an average element H-8 was obtained from foils in a plastic holder in the element. The results for both bare and cadmium-covered foils are shown in Fig. 107.2. As expected, the flux is cosine-shaped in the region between the bank tip and fuel top, and it exponentially decreases in the rodded region below the bank tip.

A number of cold core physics characteristics are readily obtainable from Fig. 107.2. By fitting $\cos \left[B_z (Z - Z_0) \right]$ to the flux shape in the region above the tank, one finds $B_z = \pi / (H + \lambda)$ to be 0.0688 in.^{-1} , where $H = 40.63 \text{ in.}$ and $\lambda = 5 \text{ in.}$ as the (reasonably expected) effective top-plus-bottom reflector savings on the 40.63-in.-high fuel zone above the bank. By fitting $e^{-\gamma(Z - Z_1)}$ to the rodged region, one obtains its axial buckling $\gamma^2 = 0.00388 \text{ cm}^{-2}$.

3.6.3 Radial Flux Distribution

Foils were inserted at three axial locations in elements along the north-south and the east-west directions in order to obtain radial flux distributions. The results, shown in Fig. 107.3, reflect chiefly the effects of the seven inserted control rods superimposed on the usual Bessel function distribution. One can see the depressing effect of the center rod in the east-west direction, as it is three elements away from the other inserted rod number 19; this is less evident in the north-south direction, for the center rod is only two elements away from the inserted rod number 11. The highest point in Fig. 107.3 is essentially at the peak core flux location in the G-6 element, both axially and radially.

3.6.4 Interpin Flux Distribution

The detailed behavior of the flux between individual fuel pins was measured on the center fuel element F-6 in its corner next to the inserted center rod, and at an average element H-8 away from control rods. Tapes with foils spaced at 0.25-in. increments were located through the element at an axial height of 69.34 in. as shown in Fig. 107.4. It can be seen that alternate adjacent pairs of foils are on fuel surfaces and in water passages.

Figure 107.5 shows the flux depression effect of the center rod on foils number 12 and 73 near the rod: the outer row of pins has about four-fifths the flux of the row adjacent to it. In the case of element H-8, one can see a more representative detail of the flux away from control rods. Here, superimposed on the local coolant channel to fuel surface variations, is the overall gradient of the radial flux in the core, foil number 145 being approximately 2 in. nearer the center of the core than foil number 68.

3.6.5 Intra-element Flux Distribution

The flux shapes across elements in a horizontal plane were measured in the center element F-6 and in an average element H-8. Foils were located midway between fuel pins and along the elements' perpendicular centerlines, as indicated in Fig. 107.4.

The results in Fig. 107.6 for F-6 show the effect of the center rod depressing the flux at this element's east and south edges. Also the east-west flux gradient exceeds the north-south gradient because, in the case of the latter, the rod number 11 inserted nearby tends to allow less flux rise away from the center rod. In the case of H-8, the results tend to show more the overall radial flux pattern in the core.

3.6.6 Edge and Corner Fluxes

As shown in Fig. 107.4, four-edge foils were located on outer fuel pins to obtain the flux gradient across the entire element. The results of such foils on two fuel elements are as follows:

$$\text{At 76.34 in. up on element F-6: } \frac{\text{north edge}}{\text{south edge}} = \frac{6.35 \times 10^6 \text{ dpm/gm}}{3.01 \times 10^6 \text{ dpm/gm}} = 2.11$$

$$\text{At 69.34 in. up on element H-8: } \frac{\text{east edge}}{\text{west edge}} = \frac{10.02 \times 10^6 \text{ dpm/gm}}{7.04 \times 10^6 \text{ dpm/gm}} = 1.42$$

Since there are no inserted control rods near H-8, its east-west ratio is essentially the same as the overall lateral traverse result shown in Fig. 107.6. For the F-6 element, however, the above data, when compared to Fig. 107.6, show the flux-depression effect of the inserted center rod on the edge row of fuel.

Corner foils, as indicated in Fig. 107.4, were used to give further information about flux gradients across elements, as well as to give flux levels at the in-core ion chamber thimbles number 7 and 9 located at the east and north tip of rod number 10 respectively. Figure 107.7 gives the results. The effect of inserted control rod number 11 on the east sides of elements F-7 and F-8 is readily seen. The diagonal radial gradient across element H-8 is approximately $\sqrt{2}$ times the east-west gradient given above.

3.6.7 Comparison of Measured and Calculated Flux Distributions

An analysis of the flux distributions across the element and at the corners of the element has been completed. Curves showing a comparison between calculated and measured fluxes along the element centerlines have been generated for element F-6. The calculated curves are obtained from PDQ cell calculations. Element F-6 was chosen for the comparison because its location in the core most closely resembles the physical constraints placed on the cell calculation, i.e., zero current boundary conditions. As is shown in Figs. 107.8 and 107.9, the agreement between calculations and measurement is good in the east-west traverse, while the north-south plane shows some variation between calculations and measurement near the edge of the fuel. This variation is expected, since the zero current boundary conditions in the PDQ cell calculations do not represent physical reality, i.e., there is a gradient across the element. This difference between calculated PDQ cell flux distributions and measured flux distributions is increased for the other elements largely because the gradient of the flux across the element becomes more pronounced as it is farther from the center of the core.

For element H-8, there is sufficient foil data to compare calculated and measured local peaking factors. The foil data for H-8 is shown in Fig. 107.10. The foils along the element centerlines are averaged to get an "average" element value of 7.142×10^6 dpm/gm. Using the cadmium ratio of 0.48 and a conversion factor to thermal flux of 1.45×10^8 n/cm²/sec = 3.7×10^6 dpm/gm, the "average" thermal flux in H-8 is 1.455×10^8 n/cm²/sec. The measured peak thermal flux is:

$$\frac{12.2}{7.142} \times 1.455 \times 10^8 \text{ n/cm}^2/\text{sec} = 2.485 \times 10^8 \text{ n/cm}^2/\text{sec}$$

Analysis of a PDQ cell calculation indicates that the "average" thermal flux defined in this manner (i.e., from the foils on the element centerline) would be approximately 7.5 percent lower than the true cell average value. Assuming that the measured centerline average flux is also low by this same amount, the measured peak thermal flux is:

$$\frac{12.2}{7.14} \times 1.45 \times 10^8 \times 0.925 = 2.3 \times 10^8 \text{ n/cm}^2/\text{sec}$$

The average calculated power in element H-8 from the FLARE calculation is 2.50 at 70 in. from the bottom of the active fuel. This height corresponds closely to that at which the measurements were made. The average thermal flux in the fuel is $0.5 \times 10^8 \text{ n/cm}^2/\text{sec}$, hence the average thermal flux in the fuel region at the location of the measurements is:

$$0.5 \times 10^8 \text{ n/cm}^2/\text{sec} \times 2.5 \times 1.166 = 1.458 \times 10^8 \text{ n/cm}^2/\text{sec}$$

where the factor of 1.166 is the average thermal flux in the fuel element normalized to the average value in the UO_2 .

From PDQ cell calculations the peak-to-average flux (i.e., the value of the flux at the edge of the pin normalized to the average in the fuel region) is 1.65. This yields a calculated value of the peak thermal flux of:

$$1.458 \times 10^8 \text{ n/cm}^2/\text{sec} \times 1.65 = 2.406 \times 10^8 \text{ n/cm}^2/\text{sec}$$

This is approximately 5 percent higher than the measured value of $2.3 \times 10^8 \text{ n/cm}^2/\text{sec}$ reported above, and it is well within the uncertainties of the calculations and measurements. It is seen that the method utilized for predicting the power peaks in the core yields reasonable results.

During the analysis of the latest set of calculations, the question arose as to whether the ratios of average foil measurements in two individual fuel elements was the same as the ratio of the center foil measurements in the same elements. Previously, foil data had been plotted normalized to a single point in the corresponding FLARE calculations. From this, radial as well as axial distributions could be inferred from the one set of measurements.

An analysis of the foil measurements in elements F-7, F-8 and H-8 was performed to determine whether or not the ratio of center foil measurements was the same as that of "average" foil measurements. Since foils were placed only on the corner pins and on the element center in F-7 and F-8, only those corresponding foils are considered in element H-8. The average flux and the flux in the center foil for each element are shown in Table 107-1.

TABLE 107-1

MEASURED CENTERLINE AND "AVERAGE" FLUX*

Element	H-8	F-8	F-7
C Flux	7.21	8.99	9.68
"Avg" Flux	7.62	9.02	9.10

*Disintegrations/min/gm $\times 10^{-6}$

This data shows that the ratio of center foil flux in element H-8, when compared to that in F-7, is ~12 percent different than the ratio of the average of the five foils in H-8 to F-7. Also, the ratio of center foils in F-8 to F-7 is 0.929, while the ratio of "average" foil measurements is 0.991 or ~7 percent difference. Therefore, care must be exercised when comparing the measured and calculated fluxes because the calculated values are all based on the element average while most of the measurements were made at the center of the element. It was concluded that the only basis for a comparison of the measured axial flux distributions with calculated power distributions would be on an individual element basis. Figures 107.11 and 107.12 show the measured and calculated axial distributions in seven fuel elements. Each of the measured distributions is normalized to the FLARE result at the point 65 in. from the bottom of the active fuel. The results show good agreement between calculated and measured axial flux distributions.

Relative radial flux distributions are obtained in the following manner: elements F-7, F-8 and H-8 are considered for comparison. Since elements F-7 and F-8 contain five foils (four corner foils and a centrally located foil), only those corresponding foils are considered in H-8. These foils are averaged, and an average flux in F-7 and F-8 is obtained relative to H-8. Next the foils across the center of elements F-6 and H-8 are averaged, and the average flux in F-6 is obtained relative to element H-8. The measured and calculated fluxes relative to element H-8 are shown in Table 107-2.

TABLE 107-2

MEASURED AND CALCULATED FLUX RELATIVE TO ELEMENT H-8

Relative Flux	H-8	F-6	F-7	F-8
Measured	1.0	1.12	1.19	1.18
FLARE	1.0	1.15	1.17	1.13

If it is assumed that the ratio of the actual average flux in the fuel elements is the same as the ratio of fluxes obtained by averaging the foils as just described, a comparison with calculations can be made. This is not immediately apparent, since it was shown earlier that the ratio of foils at the center of the elements is not the same as the ratio of the average flux in the elements. To test this assumption, the PDQ full-core calculation was examined and revealed that the ratio of averages based on the corresponding five points in two elements, such as H-8 and F-7, agrees to within 4 percent of the ratios of the true average elemental powers. Therefore, the comparison was made, and the results of Table 107-2 show the calculated and measured values to be within the uncertainty of the basic assumption.

3.6.8 Power Level Determination

The reactor power level during the foil irradiation has been determined from the combination of foil data and a three-dimensional calculation.

A three-dimensional calculation using the FLARE program was done during the boric acid test for the conditions at the time of foil irradiation, i.e., 3572 ppm boron, 7 rods inserted (see Fig. 107.1) and 22 rods at the critical position. Element F-6 was arbitrarily taken as the basis for normalization of the calculations and the foil data. As shown in Fig. 107.8, the flux shape from the FLARE calculation was normalized to the foil data. For expediency, a single point was used as the normalization, and the resulting curves were compared. The agreement was satisfactory, and the calculated ratios of fluxes were used to determine power.

$$\Phi_{th} = \frac{(DPS/cm^3/sec) \text{ sub.cd.}}{\sum_a A_u I_i}$$

$$\lambda = 3 \times 10^{-6} \text{ sec}$$

$$\sum_a A_u = 4.0 \text{ cm}^{-1}$$

$$I_i = 680 \text{ sec (irradiation time)}$$

$$DPS/cm^3/sec = DPG/MIN. \cdot \frac{\rho}{60} = \frac{DPG}{MIN.} \times 0.322$$

The flux in the water in element F-6 at 60 in. from the bottom of the core is:

$$\Phi_{th}^{H_2O} \left(\begin{array}{l} \text{AT 60 in.} \\ \text{IN F-6} \end{array} \right) = \frac{(6.78 \times 10^6 - 3.1 \times 10^6) 0.322}{4.0 \times 680 \times 3 \times 10^{-6}} = 1.45 \times 10^8 \text{ n/cm}^2/\text{sec}$$

The flux in fuel is:

$$\Phi_{th}^{FUEL} \left(\begin{array}{l} \text{AT 60 in.} \\ \text{IN F-6} \end{array} \right) = \frac{1.45 \times 10^8}{1.22} = 1.2 \times 10^8 \text{ n/cm}^2/\text{sec}$$

Using calculated value of $\bar{\Phi}$ (60 in. IN F-6) / $\bar{\Phi}$ core:

$$\bar{\Phi}_{\text{FUEL}} = \frac{1.2 \times 10^8 \text{ n/cm}^2/\text{sec}}{2.4} = 0.5 \times 10^8 \text{ n/cm}^2/\text{sec}$$

Finally:

$$\text{Power} = \frac{\bar{\Phi} \sum_f V_{\text{FUEL}}}{3.1 \times 10^{10}}$$

where:

$$\sum_f = 9.2$$

$$V_{\text{FUEL}} = 0.34 \times 10^5$$

$$\text{POWER} = 500 \text{ watts}$$

CONTROL ROD CONFIGURATION FOR INITIAL POWER OPERATION FIG. 107.1

AXIAL FLUX DISTRIBUTION THROUGH TWO FUEL ELEMENTS

FIG. 107.2

RADIAL FLUX DISTRIBUTION AT VARIOUS HEIGHTS ABOVE THE
BOTTOM OF THE FUEL

FIG. 107.3

265

ACNP-67533

272

PR-AP-2

192

FOIL LOCATIONS TO OBTAIN INTER-PIN AND INTRA-ELEMENT
FLUX DISTRIBUTION

FIG. 107.4

DETAILED FLUX MAP THROUGH ELEMENTS F-6 AND F-8

FIG. 107.5

LATERAL FLUX TRAVERSES THROUGH THE CENTERLINES OF ELEMENTS F-6 AND H-8

FIG. 107.6

VALUES OF CORNER FUEL ACTIVITIES FOR THREE FUEL ELEMENTS

FIG. 107.7

CALCULATED AND MEASURED LOCAL FLUXES ELEMENT F-6: NORTH-SOUTH TRAVERSE

CALCULATED AND MEASURED LOCAL FLUXES ELEMENT F-6: EAST - WEST TRAVERSE

FIG. 107.9

Foils at 76.34 in. normalized to those at 69.34 in. by ratio of flux at the center of the element.

$$\frac{7.21}{5.62} = 1.27$$

* in DPM/gm x 10⁻⁶

FOIL DATA*, ELEMENT H-8

FIG. 107.10

CALCULATED AND MEASURED AXIAL FLUX DISTRIBUTION

FIG. 107.11

CALCULATED AND MEASURED AXIAL FLUX DISTRIBUTION

FIG. 107.12

3.7 TEST 108 - INSTALLATION OF Sb-Be NEUTRON SOURCE

The antimony rod was received from Babcock and Wilcox's test reactor on August 9, 1967, and was lowered in the spent fuel storage well on August 19. The following Monday (August 21), the source-handling crew practiced Sb rod handling and placement into the Be target using an unirradiated rod. Once a satisfactory technique was developed, the irradiated rod was removed from the shipping cask and was loaded into the beryllium target. The neutron source was then transferred to the reactor core and placed in core position D-1 following the flooding of the reactor upper cavity and spent fuel storage well to approximately 700 ft el. At this time, the strength of the antimony rod was approximately 1900 cures. (See Fig. 108.)

The all-rods-fully-inserted count rate on nuclear channels 1 and 2 was 1.2 cps and 11.0 cps respectively with the detectors inserted 12 ft 5 in. in their tubes (tangent to the reactor vessel). Since it was desirable to increase the count rate in channel 1 by at least a factor of two, different methods of accomplishing this were tried, as explained below. By fully inserting the channel 1 detector in its lower tube, the count rate was increased by a factor of 1.5 to approximately 1.8 cps. The source was then removed from core position D-1 to G-1, and the channel 2 detector was fully inserted. This gave a better detector-core-source geometry, and the final count rates obtained were: channel 1: 7.4 cps, and channel 2: 2.8 cps. Figure 108.2 shows the effect of moving the detectors in the lower tubes for the two source positions.

Next, an attempt to go critical was made, but it had to be discontinued when no significant change in the subcritical count rate had been observed after the 29-rod bank was within 1/2 in. of the critical position established by a previous critical for the same configuration. A vertical view of the core-detector geometry indicated that the startup detectors, by being in the lower tubes, were shielded from the active core by approximately 26 in. of rodded core. However, if the detectors were moved to the upper tubes, the amount of rodded core shielding was reduced to approximately 10 in.

Therefore, the detectors for channels 1 and 2 were placed in the upper tubes, and those for channels 5 and 6 were placed in the lower tubes. The rods-fully-inserted count rate of both detectors decreased following the move to the upper tubes. Table 108-1 gives the results of all the attempts made to increase the count rate, especially that of the channel 2 detector (source in position G-1).

TABLE 108-1
COUNT RATES FOR VARIOUS DETECTOR CONFIGURATIONS

<u>Location of detector</u>	<u>Location of paraffin</u>	<u>Ch. 1</u>	<u>Ch. 2</u>
Fully inserted - lower tube	none	7.4 cps	2.76 cps
Fully inserted - upper tube	none	--	0.67
Fully inserted - upper tube	back side to act as reflector	10.5	0.75
Tangent to vessel - upper tube	back side to act as reflector	--	0.60
Fully inserted - upper tubes	channel 1 - back side only		
	channel 2 - all 3 sides to act		
	as reflector and moderator	10.5	1.43

Since the move to the upper tubes had increased the source-detector distance, the source was raised to see if the count rate could be increased. Figure 108.3 shows the effect the source height has on detector count rate. As can be seen, the effect was mainly one of count rate reduction. But, in an effort to ensure that the nuclear channels would respond properly during the next approach to critical, the source was left 18 in. above its fully seated position. In this condition, with both source and detectors much closer to the active core, a second approach to criticality was made with satisfactory results. Figure 108.4 shows the amount of channel overlap obtained between channels 1, 2, 3, 4, 5 and 6.

With the problem of the nuclear instrumentation's response to reactivity changes solved, the only problem left was that of ensuring a minimum count rate of 1 cps with the reactor shut down before rods can be withdrawn. Two solutions, both of which were followed, are:

- (a) the addition of a second source, and
- (b) the use of more sensitive detectors; the present ones have a sensitivity of 12 counts/nv.

ANTIMONY ROD CALIBRATION PERFORMED AT TEST REACTOR

FIG. 108.1

EFFECT OF COUNTER POSITION IN LOWER TUBE

FIG. 108.2

EFFECT OF SOURCE HEIGHT ON DETECTOR IN UPPER TUBES

FIG. 108.3

DETECTOR INTER-CALIBRATION CURVE

FIG. 108.4

4. OPERATIONS

4.1 REACTOR PERFORMANCE

The reactor was operated for approximately 200 hr over a 44-day period at power levels less than 1.65 kw. There were 30 scrams during this period. However, none of the scrams were caused by a real neutron flux increase beyond the limits set for the safety instrumentation.

A summary of the scrams follows:

(a) Seven of the scrams occurred from noise in the in-core instrumentation during loading when one control rod was half cocked with the remaining 28 rods fully inserted. The cables for the in-core instrumentation in the reactor vessel were extra long, temporary runs which were susceptible to local vibration.

(b) Five of the scrams were scram tests, manually conducted on individual rods for scram timing.

(c) Two of the scrams were manual scrams conducted as tests of rapid negative reactivity insertions from critical (rod drop tests).

(d) Two of the scrams were caused by noise on in-core instrumentation when the reactor was subcritical.

(e) Seven of the scrams were caused by noise on the in-core instrumentation when the reactor was critical but at least a decade below the scram settings.

(f) Six of the scrams occurred when the reactor was critical or slightly supercritical and the operator either turned the range switch on the wide-range channel in the wrong direction (so that the reading went in the upscale direction) or else neglected to uprange the instrument when increasing power level.

(g) One scram of the type described in (f) above also occurred when the reactor was subcritical.

4.2 EQUIPMENT PERFORMANCE

In general, equipment performance was good. The nuclear instrumentation for neutron level detection was modified for fuel loading operations in order to have in-core instrumentation. This necessitated extra long runs of temporary cabling which were exposed and which resulted in more noise problems than would be the case with normal installation.

The primary purification system performed very well, maintaining the conductivity considerably below 1 micromho per cm. The beds also performed the final cleanup of the boric acid from the low power test program. The beds were expended in this final cleanup and were replaced.

The following malfunctions occurred during this reporting period:

4.2.1 Control Rod Drive Number 26

Prior to any fuel loading, an oil leak was discovered on control rod drive number 26. The Pydraul oil was leaking by the low pressure seal between the hydraulic motor and the transmission assembly. This resulted in the lubricating oil being displaced out of the transmission assembly overflow. The leakage was slow, and the only immediate problem was the oil mess on the floor below the mechanism. This drive was not needed until after 36 elements were loaded in the core. Therefore, the control rod was kept fully inserted, and the drive was disabled in this position until it could be removed and repaired. The hydraulic lines and electrical cables were disconnected. This drive was removed; the seal was replaced; and on July 17, 1967 the drive was reinstalled and scram tested.

4.2.2 Control Rod Drive Control Switch

Late on the third shift of July 25, 1967, the control rod drive control switch on the bench board became "sticky" in operation. It would not spring-return to the OFF position. It could be turned off readily, and all its contacts performed satisfactorily.

On the following morning, the switch was inspected and a nut on one end plate was found loose. The nut was tightened and the shaft lubricated; the switch then performed satisfactorily. Later on the 26th of July, the switch started to stick again in the same manner. The switch was removed and taken apart. It was found that the end plates were misaligned and were binding on the shaft. The end plates were aligned, and the switch was placed back in service. It has worked satisfactorily since that date. In addition, a spare switch has been ordered.

4.2.3 Control Rod Drive Number 11

On July 26, 1967, control rod drive number 11 would not withdraw from approximately 50 in. The drive was inserted a couple of inches, and it then withdrew satisfactorily.

Based on this occurrence, it is our opinion that the orifice in the one-way restrictor which bypasses the hydraulic motor (and thus allows the motor to pump oil when being driven by the electric motor) was plugged with some foreign material. When the drive is driven in the insert direction, the hydraulic motor pumps in the opposite direction and would tend to push the obstruction from the orifice. This particular type of malfunction does not affect the drives' insertion or scram capability. Test scrams were performed on rod number 11 after this occurred, and they were satisfactory.

4.2.4 Nuclear Instrumentation Channel N-6

On July 11, 1967, Channel N-6, Picoammeter, failed to indicate properly.. This was evident by the fact that, on the 60×10^{-4} percent scale, the indication on this channel abruptly leveled off, even though the reactor was on a long supercritical period.

The reactor was made subcritical shortly thereafter, and after a slight delay the indication on this channel decreased as expected.. The reactor was shut down, and during the following morning this channel was checked out and found to be malfunctioning.. Repairs were made, and the channel was put back in service.

During this malfunction, Channel B, which is also a Picoammeter with in-core detector, was providing scram backup protection.

4.2.5 Nuclear Instrumentation Channel N-1

While taking the last sets of subcritical counts on July 24, 1967, it was discovered that Channel N-1 was giving excessive noise counts, resulting in a poor statistical count rate.. The following morning, prior to reactor operation, the current pulse amplifier was replaced with a spare unit, and the noise counts in Channel N-1 returned to normal.. A shorted capacitor was replaced in the malfunctioning current pulse amplifier, and the amplifier's performance was then satisfactory.

5. RADIOLOGICAL PHYSICS AND CHEMISTRY

5.1 RADIOLOGICAL PHYSICS

During the low power test program, the following maximum dose rates to personnel were recorded:

- (a) handling of irradiated fuel (Test 107) 1 rem/hr
- (b) handling of antimony source 25 mrem/hr

The fuel elements loaded in the reactor had a maximum dose rate of 3 mr/hr at contact. Alpha smears were taken of fuel elements, and all results were <250 dpm/ft².

The second core fuel elements received at the site for storage in the new fuel storage racks gave a maximum dose rate of 4 mr/hr at contact. Alpha smears taken of all fuel elements were <250 dpm/ft².

Radiation surveys were conducted in the reactor containment building during low power testing. A maximum reading of 1 rem/hr was observed over the core at the 701-ft level with the reactor critical and at a power level of ~ 1.5 kw.

5.1.1 In-Plant Health Physics Monitoring Data

The highest containment air average particulate beta, 30 min after sampling, was 2.4×10^{-9} μ Ci/cc.

The highest containment air average particulate beta, 6 hr after sampling, was 6.2×10^{-12} μ Ci/cc.

5.1.2 Environmental Monitoring

5.1.2.1 Environmental Surface Water Activity Measurements for May, 1967.

Sample Identification	Gross β - γ , Suspended μ Ci/ml	Gross β - γ , Dissolved μ Ci/ml	Gross α , Suspended μ Ci/ml	Gross α , Dissolved μ Ci/ml
Stoddard	$9.3 \times 10^{-9} \pm 3.5 \times 10^{-9}$	$16.8 \times 10^{-9} \pm 3.9 \times 10^{-9}$	$<1 \times 10^{-9}$	$3.1 \times 10^{-9} \pm 0.9 \times 10^{-9}$
Dam #8	$7.8 \times 10^{-9} \pm 3.5 \times 10^{-9}$	$17.9 \times 10^{-9} \pm 4.0 \times 10^{-9}$	$<1 \times 10^{-9}$	$1.4 \times 10^{-9} \pm 0.7 \times 10^{-9}$
Boat Launch	$<5 \times 10^{-9}$	$6.9 \times 10^{-9} \pm 3.8 \times 10^{-9}$	$<1 \times 10^{-9}$	$1.3 \times 10^{-9} \pm 0.6 \times 10^{-9}$
Victory	$<5 \times 10^{-9}$	$16.5 \times 10^{-9} \pm 3.8 \times 10^{-9}$	$<1 \times 10^{-9}$	$<1 \times 10^{-9}$
La Crosse	$<5 \times 10^{-9}$	$16.3 \times 10^{-9} \pm 3.8 \times 10^{-9}$	$<1 \times 10^{-9}$	$3.3 \times 10^{-9} \pm 0.9 \times 10^{-9}$

5.1.2.2 Environmental Surface Water Activity Measurements for June, 1967.

Sample Identification	Gross β - γ Suspended $\mu\text{Ci/ml}$	Gross β - γ Dissolved $\mu\text{Ci/ml}$	Gross α Suspended $\mu\text{Ci/ml}$	Gross α Dissolved $\mu\text{Ci/ml}$
Stoddard	$<5 \times 10^{-9}$	$<5 \times 10^{-9}$	$<1 \times 10^{-9}$	$<1 \times 10^{-9}$
Dam #8	$<5 \times 10^{-9}$	$5.1 \pm 1.7 \times 10^{-9}$	$<1 \times 10^{-9}$	$<1 \times 10^{-9}$
Boat Launch	$11.0 \pm 2.3 \times 10^{-9}$	$11.3 \pm 1.8 \times 10^{-9}$	$<1 \times 10^{-9}$	$<1 \times 10^{-9}$
Victory	$<5 \times 10^{-9}$	$11.6 \pm 1.8 \times 10^{-9}$	$<1 \times 10^{-9}$	$<1 \times 10^{-9}$
La Crosse	$<5 \times 10^{-9}$	$<5 \times 10^{-9}$	$<1 \times 10^{-9}$	$<1 \times 10^{-9}$

5.1.2.3 Environmental Air Particulate Activity Measurements for May, 1967.

Sample Identification	Min. Radioactivity Conc. ($\mu\text{Ci/cc}$)	Avg. Radioactivity Conc. ($\mu\text{Ci/cc}$)	Max. Radioactivity Conc. ($\mu\text{Ci/cc}$)
La Crosse	$0.05 \times 10^{-12} \pm 0.01 \times 10^{-12}$	$0.08 \times 10^{-12} \pm 0.01 \times 10^{-12}$	$0.13 \times 10^{-12} \pm 0.01 \times 10^{-12}$
LACBWR	$0.07 \times 10^{-12} \pm 0.01 \times 10^{-12}$	$0.09 \times 10^{-12} \pm 0.01 \times 10^{-12}$	$0.11 \times 10^{-12} \pm 0.01 \times 10^{-12}$

5.1.2.4 Environmental Air Particulate Activity Measurements for June, 1967.

Sample Identification	Min. Radioactivity Conc. ($\mu\text{Ci/cc}$)	Avg. Radioactivity Conc. ($\mu\text{Ci/cc}$)	Max. Radioactivity Conc. ($\mu\text{Ci/cc}$)
La Crosse	$0.03 \pm 0.01 \times 10^{-12}$	$0.058 \pm 0.01 \times 10^{-12}$	$0.13 \pm 0.01 \times 10^{-12}$
LACBWR	$0.03 \pm 0.01 \times 10^{-12}$	$0.058 \pm 0.01 \times 10^{-12}$	$0.09 \pm 0.01 \times 10^{-12}$

5.1.3 Waste Disposal

A total of 61,987 gal of liquid waste with a radioactivity concentration of $<1.0 \times 10^{-7}$ $\mu\text{Ci/ml}$ was discharged into the Mississippi River during the program.

A total of 5400 gal of liquid waste with a radioactivity concentration of 1.455×10^{-7} $\mu\text{Ci/ml}$ was discharged into the Mississippi River during August, 1967. This discharge was diluted with low-pressure service water during release.

5.2 WATER CHEMISTRY

Results of systems analyses are reviewed daily. All results were within acceptable limits during this period.

Reactor Purification Influent Sample	Maximum	Minimum
pH	8.4	5.45
Conductivity (μ mhos/cm)	2.7	0.43
Chloride (ppm)	0.05	<0.02
Dissolved oxygen (ppm)	8.0	4.8
Gross (μ Ci/ml)	$<1.77 \times 10^{-8}$	$<9.7 \times 10^{-10}$
Radiochemical iodine (μ Ci/ml)	$<8.6 \times 10^{-8}$	$<3.8 \times 10^{-10}$

Note: Above analysis reflects boric acid additions made during the low power testing program.

6. MAINTENANCE

The principal maintenance performed on reactor systems during the program was that associated with the malfunctions previously described in Sec. 4.2, Equipment Performance. In addition, regular preventive maintenance was performed throughout the plant in accordance with manufacturers' instructions.

In order to avoid excessive cycling of the high pressure service water pump for the purpose of only supplying service water to the turbine condenser cooling water monitor eductor pump, a modification was made to the system. It is now possible to operate the eductor pump through the use of low pressure service water.

A rod scram time test, performed on July 31, 1967, showed the slowest rod to have a measured time of 2.242 sec and the fastest rod to have a time of 1.880 sec.

Technical Specifications tests were performed satisfactorily on the following equipment during the program:

- (a) area radiation monitors
- (b) liquid radiation monitors
- (c) gas radiation monitors
- (d) particulate radiation monitors
- (e) diesel generator
- (f) reactor water level
- (g) full scram circuits hot short test
- (h) nuclear instrumentation
- (i) control rod drive full range exercise
- (j) boron injection system and high pressure core spray controls and valves
- (k) emergency generator auto transfer
- (l) noninterruptible bus transfer switch
- (m) hydraulic scram solenoid valves
- (n) reactor building pressure switch
- (o) reactor building MCC 1A undervoltage scram
- (p) 2400-v busses 1A and 1B undervoltage scram
- (q) CRD accumulator oil level scram and gas pressure scram
- (r) reactor building isolation system