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ELK RIVER REACTOR OPERATIONS ANALYSIS PROGRAM

Annual Progress Report

July 1, 1965 - June 30, 1966

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U. S. Atomic Energy Commission

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by

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1. INTRODUCTION AND SUMMARY

During the period between July 1, 1965 and June 30, 1966 the reactor heat generated was 14,117.1 Mwd; and the net electrical power generated was 129,408 Mw-hr. Overall plant behavior was excellent except for some minor difficulties caused by leaking tubes in the evaporators.

Operations Analysis efforts during this period have been directed mainly toward establishing a core loading pattern which best meets the objectives of the Elk River Reactor, making recommendations concerning proposed changes to the technical specifications, analyzing the operation of the plant energy systems and the thimble cooling system, and studying other problems of particular interest to the Chicago Operations Office of the U. S. Atomic Energy Commission.

The measured position of the center regulating control rod as a function of core exposure has been compared with the previously reported curve of predicted rod position. Agreement between the predicted and actual lifetime is extremely good, thus adding to our confidence in the analytical methods used for predicting core nuclear characteristics. Remaining reactivity at the time of shutdown on April 15, 1966 was negligible at full power.

Progress is reported on the fuel cycle studies which established a fuel loading pattern for Core II. Since the choice of a loading pattern for the first batch of Core II fuel affects not only the second burnup interval but also the entire fuel cycle, patterns which met all of the other nuclear requirements were examined further to determine the type of equilibrium cycle which might result. At the time of actual core loading, changes in the recommended pattern were required due to a decision to replace fuel elements which had bowed fuel rods. The nuclear characteristics of the final loading pattern are included in this report.

Some changes were required in the wording of the technical specifications describing the use of Core II fuel elements and B_4C in-tube-type control rods. The proposed changes, the reasons for the changes, and the safety considerations relevant to these changes were studied during the report period. Some of the important aspects of these changes and their effect on reactor operation are discussed. Perhaps the most important change, from an operating standpoint, is one which limits the reactivity which may be held down by one rod. The new limitation is more severe than that imposed for the first core.

Plant energy transfer system data taken in March 1966 are compared with data taken in the previous three years of operation and with the design parameters. The heat balances are not as close this year as they were in 1965, nonetheless they are considered to be within station instrument accuracy. An analysis was made of the effect on power level of sealing off leaking evaporator tubes. To date the effect has not been detectable since only a small number of tubes have been affected. The limiting number of tube failures was determined to be 322 U-tubes per evaporator for full power operation.

Corrosion specimens inserted into the water boxes of the evaporators in June 1964 were removed for examination during the report period. The results of this examination show evidence that conditions exist in the evaporator water boxes which are conducive to stress corrosion cracking.

Analyses of the need for the addition of more capacity for collecting primary system condensate and of modifications to improve operational control of the control rod thimble cooling system are also reported. The former study concluded that the economic incentive for providing additional capacity for collecting primary system condensate was marginal. The latter study outlined modifications to the thimble cooling circuit to achieve the maximum amount of operational control.

Because of the phase-out of nuclear operations by Allis-Chalmers, this is the last annual report to be issued by the Atomic Energy Division on the Elk River Reactor Operations Analysis Program. The report covers the activities of the ERR-OAP project through the end of the contract period on August 31, 1966.

Abstracts of reports from all sources which were issued during this report period are included as Appendix A of this report.

2. REACTOR OPERATING SUMMARY

A tabular summary of ERR operation for this report period is given in Table 2-1.

The plant operated continuously from September 1, 1965 until April 15, 1966 except for three unscheduled outages totaling one-half day in the seven and one-half month period. On April 15, the plant was shut down as scheduled for partial refueling, inspections, modifications, and preventive maintenance. The plant was also shut down during the month of August 1965 in order to plug three defective tubes in evaporator #1. The plant availability during the twelve-month period was approximately 65 percent.

During the scheduled shutdown, all control rods were inspected. Horizontal cracks were observed in the absorber section of the regulating rod which is located in the center core position. Evidence of cracking of the 2 percent boron-stainless steel absorber metal near the rivets at the transition joint was noted on rods 2 and 5. All three of the above-mentioned rods were replaced with new B_4C in-tube-type control rods. Some cracking was found in the rod lifting pin gussets of 11 control rods. These gussets were modified by the addition of stainless-steel reinforcing plates riveted to the original gussets.

During the reloading operation, all Core I elements scheduled for continued use were inspected prior to their placement in the core. This inspection revealed some bowing of individual fuel rods in the fuel elements. The majority of those elements with distorted fuel rods had only a very slight bow in the individual pins. The ERR safety committee decided to remove all elements exhibiting any distortion until a criterion for acceptance was formulated. The removal of these elements necessitated the utilization of Core I spare elements and more highly exposed Core I elements which did not have bowed fuel rods as replacements.

Other work completed during the shutdown included the vessel inspection, primary piping inspection, plugging of evaporator tube leaks, modifications to the four rod scram circuits, and superheater conversion.

TABLE 2-1

ERR OPERATING HISTORY,* 1965-1966

month	reactor heat generated, Mwd	average power, Mwt	maximum power, Mwt	net electrical generation, kwhr	reactor power level factor, %
July 1965	1,402.7	45.12	58.2	12,812,114	86.7
Aug. 1965	0	0	--	---	--
Sept. 1965	1,692.2	56.3	58.2	15,743,276	96.9
Oct. 1965	1,732.9	55.9	58.2	15,942,908	96
Nov. 1965	1,683.6	56.1	58.2	15,574,267	96.4
Dec. 1965	1,740.6	56.1	58.2	15,917,383	96.4
Jan. 1966	1,767.8	57.0	58.2	16,157,750	97.9
Feb. 1966	1,562.9	55.8	58.2	14,208,361	95.9
Mar. 1966	1,744.0	56.3	58.2	15,846,272	96.7
Apr. 1966	790.4	26.35	58.2	7,205,809	45.3
May 1966	0	0	--	---	--
June 1966	0	0	--	---	--

* From ERR Monthly Operating Reports

3. CORE NUCLEAR CHARACTERISTICS

3.1 REACTIVITY HISTORY (TASK 102)

The objective of this task has been to determine the amount of excess reactivity (and, hence, the operating time) left in the core as the reactor operated, and to compare the predicted and measured characteristics as a function of time.

3.1.1 Core I Measurements

The observed position of the center regulating rod vs. core operation has been compared with the previously reported curve (see Ref. 1) of the predicted rod position vs. exposure. The predicted rod position curve was calculated assuming continuous operation at 58.2 Mwt with equilibrium xenon, samarium, and protactinium. Since the reactor has been operated primarily in the load-following mode, equilibrium concentrations of these isotopes are not usually present, and the observed rod positions were corrected to equilibrium conditions. In addition, the points (taken at nominal full power conditions) are corrected for variations in power by normalizing to a primary flowrate of 250,000 lb/hr.

Near the end of core life, the rod position and power history were followed more closely to get a better feel for the rate of burnup during this period. During the last month of operation, the reactor was operated primarily in the load-following mode with the actual operating power level (and the observed rod position) varying from hour to hour and the average daily power generation varying from day to day. Graphs of the average reactor power level and the observed rod position were kept on a daily basis using data supplied by RCPA in the ERR daily operating reports.

The average daily power generation is shown as a histogram in Fig. 3.1 for the last month of operation (March 15 to April 15, 1966). Variations within a 24-hr period are not shown. The maximum change in power level during any one day occurred on April 12 when the nominal power level ranged from 33.9 to 58.2 Mwt. The reactor reached the nominal full power level of 58.2 Mwt at least once every day during the month except on April 13, when the highest power achieved was 55.8 Mwt during the power escalation tests.

Also shown in Fig. 3.1 is the observed position of the regulating rod at a particular time during the day when the nominal power level was 58.2 Mwt. For the reason noted above, the observed position for April 13 is at 55.8 Mwt.

A comparison of the corrected measurements with the curve of the predicted rod position is shown in Fig. 3.2. The highest regulating rod position reported during the month in the daily operating reports was 57.9 in. from the bottom of the core on April 15. Examination of the reactor console data sheets for April 15 showed that the rod was withdrawn as far as 58.6 in. (both values uncorrected). The final corrected data point as plotted in Fig. 3.2 is 59 in. at approximately 7040 Mwd/MT. The corresponding predicted value was 60 in. at 7000 Mwd/MT.

The agreement between the predicted lifetime and the actual lifetime is extremely good and thus lends confidence to the present analytical methods being used for predicting core characteristics.

For a better analysis of the reactivity of the end of life core, a series of rod configurations at power were requested and the tests were performed by RCPA. The rod configurations and reactor conditions are given in Table 3-1.

TABLE 3-1

END OF CORE LIFE ROD CONFIGURATION TEST

(April 14, 1966, 58.2 Mwt, 940 psi)

time	rod positions: 12-rod bank	rod positions: reg. rod
2130	60 in. (out)	54.0 in.
2135	59.0 in.	58.82 in.
2200	60 in.	53.7 in.

Calculations were performed using these rod positions and assuming that the core had a burnup of 7000 Mwd/MT. The core constants (i.e., isotopic distribution, etc.) used in the analysis were those predicted when the rod position curve was generated in 1964. The approach was to place the rods, analytically, at the measured positions and calculate the corresponding k_{eff}. The results of the calculation are shown in Table 3-2.

TABLE 3-2

COMPARISON OF MEASURED AND CALCULATED k_{eff} AT END OF CORE LIFE

rod positions: 12-rod bank	rod positions: reg. rod	k_{eff}: calculated	k_{eff}: measured
60 in.	54.0 in.	0.99942	1.0000
59.0 in.	58.82 in.	0.99937	1.0000
60 in.	60 in.	1.00037	-------

The calculation gives the same k_{eff} (0.9994) for the two critical rod configurations, the difference between this and the all rods out condition is approximately 0.1% Δk, which is a good estimate of the reactivity left in the core at the time the measurements were made (April 14, 1966). Thus, the reactivity controlled by the reg. rod at this time was approximately 0.1% Δk. On the following day, the regulating rod had moved from 54 in. to 58.6 in., presumably due to xenon buildup, and hence, the actual reactivity left at the time of shutdown was negligible.

3.1.2 Core II Measurements

In June 1966, new fuel was inserted into the reactor core to replenish the reactivity lost due to burnup. The core was loaded to two different configurations, with critical measurements being taken after each loading. The basis for the loading patterns was the topical report on fuel cycle studies for the second core (see App. A-2.19). The first loading pattern chosen for testing by the reactor operator was a placement of fuel similar to one of the patterns investigated in ACNP-66522 (see App. A-2.19 and Fig. 3.3). The actual configuration used is shown in Fig. 3.4. The second and final loading pattern (see Fig. 3.5) resulted from a decision to replace fuel elements which showed evidence of bowing in individual fuel rods.

The calculations done prior to startup based on the configuration shown in Fig. 3.3 are shown in Table 3-3 and are compared with the measured characteristics of the pattern shown in Fig. 3.4.

TABLE 3-3

COMPARISON OF PREDICTED AND MEASURED CRITICAL CHARACTERISTICS

	13-rod bank	12-rod bank rod 5 fully out
calculated k_{eff} (Fig. 3.3)	0.9998 at 12.4 in.	0.9904 at 10.4 in.
calculated critical position (Fig. 3.3)	12.5 in.	11.5 in.
measured critical position (Fig. 3.4)	12.84 in.	12.3 in.
calculated ¢/in. (Fig. 3.3)	162¢/in. at 12.2 in.	150¢/in. at 10.03 in.
measured ¢/in. (Fig. 3.4)	167¢/in. at 12.8 in.	164¢/in. at 12.6 in.

The differences between the two patterns are that six of the Core I exposed elements were replaced and the arrangement of the Core II feed elements on the periphery was modified somewhat.

Subsequent to the critical measurements, calculations were done for the same rod positions as measured. The control rods were placed, analytically, at the measured critical positions and the k_{eff} and ¢/in. values were calculated. These results are shown in Table 3-4. The calculated reactivities are about 0.3 percent high in all cases. The ¢/in. values are also in good agreement except for the case with rod 5 out, which is

~10 percent underestimated. An examination of the experimental data shows a reproducibility variation of approximately 5 percent. The reason for the larger variation with rod 5 out may be due in part to the dummy fuel element placement, since the calculational model had been normalized to the dummy locations which were used in Core I, and that are also used in the final loading pattern of Core II.

TABLE 3-4

CALCULATED AND MEASURED RESULTS FOR LOADING PATTERN IN FIG. 3.4

rod configuration	k_{eff} calculated	k_{eff} measured	differential bank worth calculated	differential bank worth measured
13-rod bank at 12.84 in.	1.003	1.000	167.8¢/in. at 12.5 in.	167¢/in. at 13.0 in.
reg. rod out, 12-rod bank at 11.32 in.	1.003	1.000	136.6¢/in. at 12.5 in.	140¢/in. at 11.5 in.
rod 5 out, 12-rod bank at 12.36 in.	1.003	1.000	149.4¢/in. at 12.5 in.	164¢/in. at 12.6 in.

The final revised loading pattern, which is to be used as the operating core, used the normal dummy element location. Extensive changes to the pattern recommended under Task 201 (see App. A-2.19) were required because of the ERR Safety Committee decision to defer the use of 25 Core I exposed elements which showed a slight bowing of individual fuel pins. This decision made it necessary to use the 16 available Core I unexposed spare regular elements plus nine additional Core I exposed elements which originally had been scheduled for discharge. The recommended location of the 48 Core II feed elements and of the 20 Core I spiked elements (which had no bowed fuel pins) was retained. The remaining fuel elements were rearranged to approach as nearly as possible a symmetrical power distribution and to retain the burnup characteristics of the original recommended pattern. The loading pattern that resulted is shown in Fig. 3.5.

Some measured and calculated values for the final loading pattern are given in Table 3-5. The calculated k_{eff} values have an average variation from the measured values of 0.47% Δk.

TABLE 3-5

MEASURED AND CALCULATED RESULTS FOR FINAL LOADING PATTERN IN FIG. 3.5

rod configuration	k_{eff} calculated	k_{eff} measured	differential bank worth calculated	differential bank worth measured
13-rod bank at 12.86 in.	1.0043	1.000	162.2¢/in. at 12.50 in.	164¢/in. at ~13.0 in.
reg. rod out, 12-rod bank at 11.39 in.	1.0039	1.000	136.0/in. at 12.50 in.	138¢/in. at ~11.5 in.
rod 5 out, 12-rod bank at 12.16 in.	1.0058	1.000	127.0¢/in. at 12.50 in.	not measured

3.2 CONTROL ROD WORTH (TASK 101)

3.2.1 B4C In-Tube-Type Control Rods

A control rod inspection in May 1966 revealed the presence of horizontal cracks in the absorber section of the regulating rod. Cracking of the 2 percent boron-stainless-steel absorber was also noted in rods 2 and 5 near the rivets in the transition joint between the follower and absorber sections. These three rods were removed from the reactor core and new B_4C in-tube-type rods were installed in the regulating (R) position, and in positions 3 and 9. The remaining positions were filled with the acceptable boron-stainless-steel rods. Figure 3.4 shows the locations (shaded for emphasis) of the B_4C in-tube-type rods.

The presence of the two types of control rods in the core, as shown in the figure, can lead to different reactivity effects on the fuel elements depending on whether the fuel inside the shroud box is exposed to either boron-carbide or boron-stainless-steel rods alone, or to both a boron-carbide and a boron-stainless-steel rod. These reactivity effects are accounted for in the FLARE model by adjustment of the input parameters.

3.2.2 One Rod Out Shutdown Margin

The calculations of shutdown margin performed under Task 201 for the recommended fuel loading pattern indicated that the least shutdown margin should occur with rod 5 withdrawn from the core. Table 3-6 gives the critical bank positions and differential bank worth (¢/in.) values measured for the final loading pattern.

TABLE 3-6

CRITICAL BANK POSITION FOR FINAL LOADING PATTERN

description	critical bank position	measured, ¢/in.
13-rod bank	12.86 in.	164¢/in.
12-rod bank, rod 5 out	12.16 in.	not measured
12-rod bank, reg. rod out	11.39 in.	138¢/in.

Based on the fact that the 12-rod bank is lower with the reg. rod withdrawn than with any other rod withdrawn, it was concluded that the minimum shutdown margin actually occurs with the reg. rod withdrawn. Preliminary investigations of the data indicate that this generalization may not be valid.

An analysis was later made for the final fuel loading pattern. The control rods were set at the observed critical positions and calculations were made for three conditions:

(1) 13 rods in a bank
(2) 12 rods in a bank, rod 5 withdrawn
(3) 12 rods in a bank, reg. rod withdrawn

Also, the differential bank worth (¢/in.) values were calculated for bank positions as near to the measured positions as the calculational model would allow. The calculated and measured values are given in Table 3-5. As stated previously, the calculated k_{eff}'s are an average of 0.47% Δk high; and, the ¢/in. values are accurate to within 10 percent.

Calculations of ¢/in. were made as a function of bank position for both the reg. rod out case and the rod 5 out case; and, differential rod worth curves were generated. These are shown in Fig. 3.6. The curves are markedly different in that the peak bank worth occurs at significantly different positions. Based on the calculational results in Table 3-5 and in Fig. 3.6, it is possible for the core to have the smallest shutdown margin with rod 5 withdrawn, even though the 12-rod bank critical position for this condition is higher than that of the condition with the reg. rod withdrawn.

As a first attempt to relate the calculated results of Fig. 3.6 to the actual core, the curves could be normalized at the critical positions to the measured ¢/in. values. Unfortunately, there is no measured differential bank worth value for the rod 5 out condition; but it might be assumed that the calculated and measured values differ by no more than the difference shown in Table 3-3 (about 10 percent). Making this assumption and using the measured value for the reg. rod out case would still lead to the conclusion that the least shutdown margin occurs with rod 5 out.

This approach (i.e., assigning to the curves the measured value at the critical positions) assumes that the shape of the curves are independent of the ¢/in. value. Actually, the shapes of the curves are dependent upon the nuclear constants used to generate the ¢/in., and the final normalization must be done by varying the nuclear parameters.

Not only which rod is the most reactive but the absolute value of the shutdown margin are consequences of the foregoing discussion and, hence, decisions concerning reload reactivity and fuel placement are affected.

3.2.3 Excess Reactivity Held Down by One Rod

The establishment of a new technical specification limits the amount of reactivity which may be controlled by one rod to 0.030 Δk under operating conditions. (See Sec. 4.3.4.) To meet this specification, an initial rod positioning program different from that used for Core I had to be utilized.

In the Elk River reactor core, the center regulating rod is capable of holding down more reactivity in the hot operating core than any other single rod when it is fully inserted into an otherwise unrodded core. The necessity for a new rod program is seen by an examination of the calculated reactivity worth of a fully-inserted center rod given in Table 3-7. A rod worth capability of approximately 20 percent higher than the 0.030 Δk value is predicted.

TABLE 3-7

REACTIVITY WORTH OF FULLY-INSERTED CENTER ROD AT HOT OPERATING CONDITIONS

loading pattern	type of rod	worth of reg. rod fully inserted, Δk
Core I	boron-stainless steel	0.033
Core II (recommended pattern)	boron-stainless steel	0.032
Core II (recommended pattern)	B_4C in-tube	0.037
Core II (final loading)	B_4C in-tube	0.036

The rod program that was used in Core I consisted of keeping the center regulating rod fully inserted and withdrawing the 12-rod bank to meet power escalation and xenon buildup requirements. Then, with the 12-rod bank fully withdrawn, the center regulating rod was partially withdrawn to maintain criticality at the desired power level. In order to comply with the new technical specifications, the center regulating rod is withdrawn first for a short distance so that the reactivity controlled by the reg. rod is less than 0.030 Δk under operating conditions. The 12-rod bank is then withdrawn to meet power escalation and xenon buildup requirements.

In the recommended loading pattern, a B_4C in-tube-type regulating rod must be at least 15 in. from the bottom of the core when the 12-rod bank is fully withdrawn to ensure compliance with technical specifications; and in the final loading pattern, the same type of control rod must be at least 16 in. from the bottom of the core when the 12-rod bank is fully withdrawn.

3.3 POWER DISTRIBUTIONS (TASK 103)

3.3.1 Effect of Core II Rod Program

The operating rod configuration of the center regulating rod described in Sec. 3.2.3 leads to a substantially different critical position for the 12-rod bank from that observed in Core I, and results in higher power peaking factors for the same total power outputs. This change in the operating rod positions required that new limit curves be established for the minimum 12-rod bank height as a function of power to ensure remaining within the maximum heat flux of 313,000 Btu/hr-ft^2 as required by the technical specifications. The overall peak-to-average power corresponding to this heat flux limitation is given vs. power level in Table 3-8, and includes the local peaking factor.

TABLE 3-8

MAXIMUM POWER PEAKING FACTOR VS. POWER

power, Mwt	peaking factor
20	9.96
30	6.68
40	5.02
50	4.02
58.2	3.45

Based on the maximum allowable power peaking factors, limit curves were generated for the minimum 12-rod bank height, assuming that the regulating rod was positioned at 16 in. above the bottom of the core. Limit curves were generated both for zero xenon and for the equilibrium xenon level as a function of power. These curves are shown in Fig. 3.7. Figure 3.8 in addition to giving the limit curves, also gives (as dashed lines) the predicted 12-rod bank critical positions for zero and equilibrium xenon.

These estimated bank positions assumed that approximately 0.005 Δk is held by protactinium and samarium.

3.3.2 Power and Void Distributions

The calculated power and void distributions for both the ERR Core I and Core II loadings were requested by the Chicago Operations Office, USAEC, for the Core II fuel vendor during the report period, but prior to the time of actually loading Core II. The information presented below is both for Core I after approximately 4000 Mwd/MT exposure and for a tentative Core II loading pattern using 48 Core II feed elements and 100 Core I exposed elements.

The power and void distributions were determined from three-dimensional power-void calculations at full power equilibrium xenon conditions with the regulating rod controlling the core. The power and void distributions as given are the average in the axial (Z) plane. Figures 3.9 and 3.10 show the average horizontal void distributions for Core I at 4000 Mwd/MT and for a tentative Core II pattern, respectively. Figures 3.11 and 3.12 show the relative horizontal power distributions for Core I and Core II. Figure 3.13 shows the relative locations of the element types in the Core II pattern. Table 3-9 gives the average axial void distributions, and Table 3-10 gives the relative axial power distributions for both Core I and Core II.

TABLE 3-9

AVERAGE AXIAL VOID DISTRIBUTION*

core average	Core I at 4000 Mwd/MT	Core II uniform pattern
axial segment	30.0 percent	29.7 percent
12	42.6	42.6
11	42.1	42.1
10	41.3	41.1
9	40.1	39.8
8	38.5	38.0
7	36.4	35.7
6	33.5	32.6
5	29.6	28.8
4	24.5	23.8
3	18.0	17.8
2	10.5	10.6
1	3.4	3.5

*all values given refer to the void percent within the shrouds

TABLE 3-10

RELATIVE AXIAL POWER DISTRIBUTION

axial segment	Core I at 4000 Mwd/MT	Core II uniform pattern
12	0.175	0.214
11	0.332	0.401
10	0.491	0.579

Table 3-10 - Relative Axial Power Distribution (cont'd)

axial segment	Core I at 4000 Mwd/MT	Core II uniform pattern
9	0.657	0.750
8	0.836	0.915
7	1.032	1.075
6	1.256	1.228
5	1.473	1.365
4	1.616	1.473
3	1.639	1.517
2	1.468	1.440
1	1.024	1.042

3.3.3 Core I Rod Programs Using an Off-Center Regulating Rod

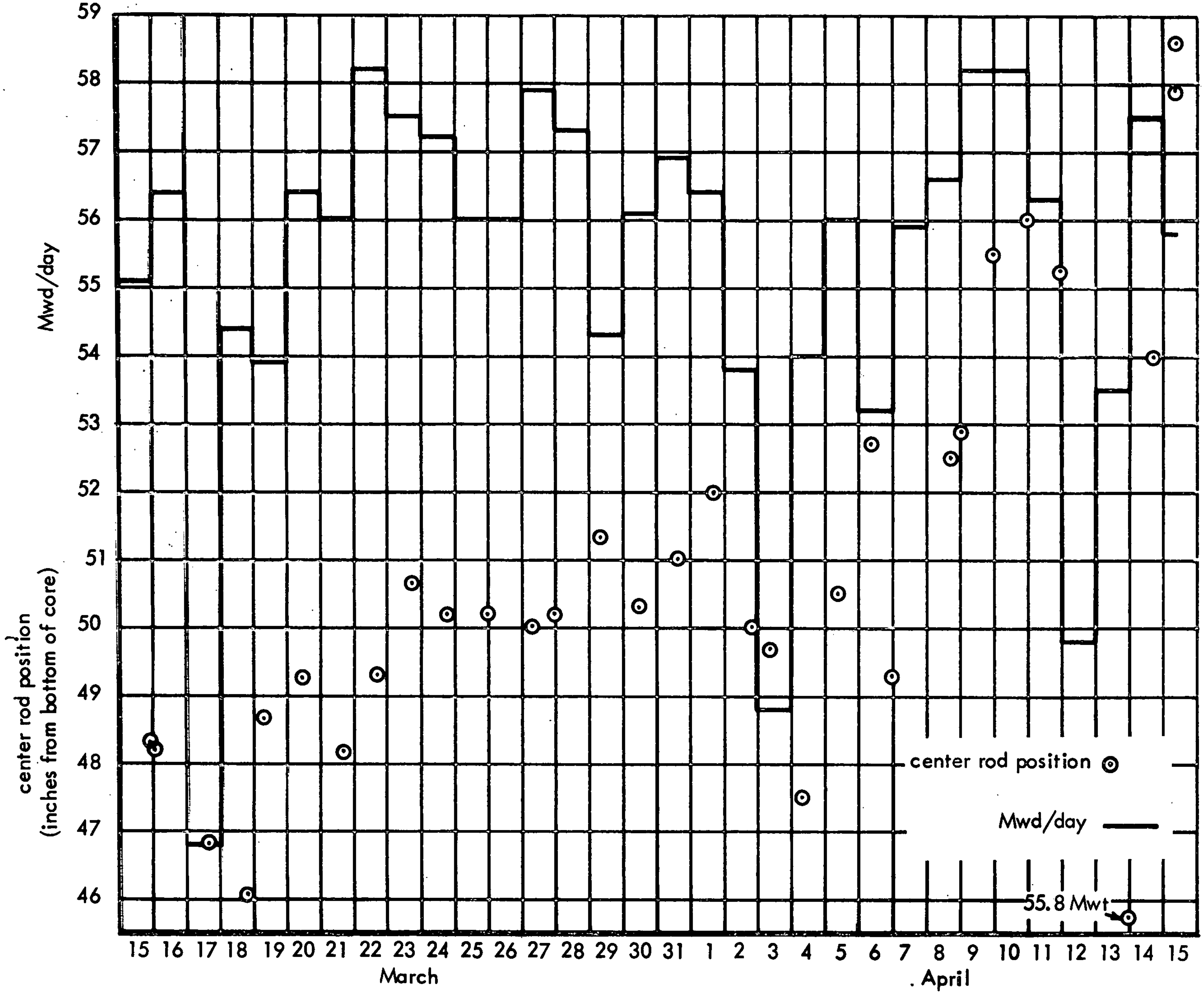
Due to the previous technical specification inspection requirements for the most highly exposed of the boron-stainless-steel control rods, there had been some interest in rod programs which would divide the exposure more evenly between the rods. An investigation had been conducted for Core I at approximately 2000 Mwd/MT. This corresponds to a critical position of the center rod at approximately 20 in. The calculations assumed 58.2 Mwt operation with equilibrium xenon. Alternate critical rod positions were found (in which the center regulating rod was not used) and the power distributions were calculated. These power peaking factors are shown in Table 3-11 compared to center rod operation. As can be seen from the table, all patterns investigated gave a total peak-to-average power at least 15 percent greater than the center rod pattern. Several rod patterns gave a lower average axial peaking but no rod pattern came within 15 percent of the average radial peaking.

TABLE 3-11

OPERATING POWER DISTRIBUTIONS FOR A UNIFORM LOADING PATTERN

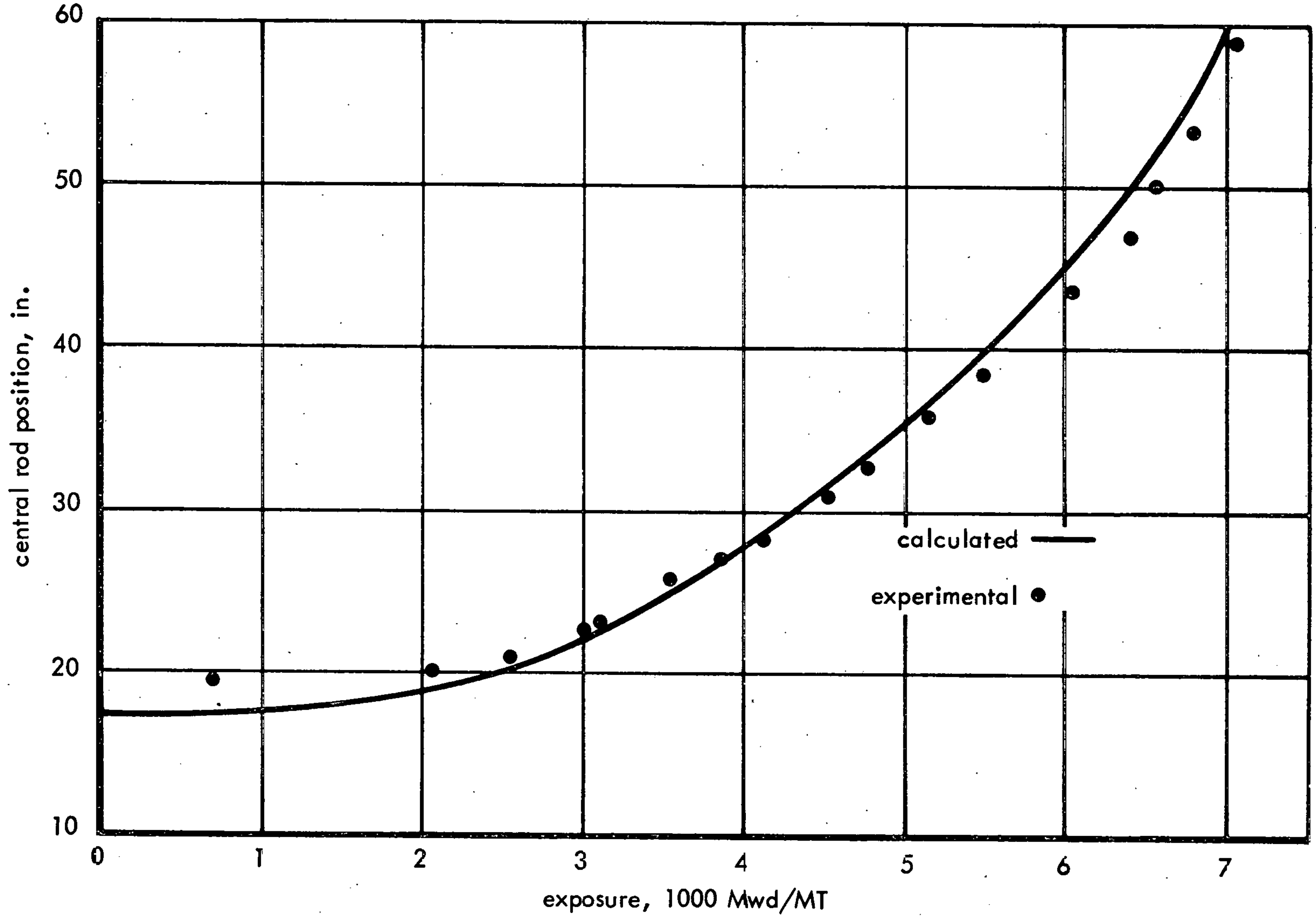
rod configuration	peak-to-average power* total	average axial	average radial
1. ℄ rod at 20 in.	2.71	1.81	1.44
2. rod 3 in, bank at 55 in.	3.30	1.64	2.00
3. rod 5 in, bank at 50 in.	3.17	1.63	1.95
4. 13-rod bank at 40 in.	3.43	2.00	1.70
5. rods 3, 9 at 25 in.	3.30	1.95	1.67
6. rods 4, 7 at 25 in.	3.22	1.83	1.81
7. rods 3, 9 at 30 in., bank at 50 in.	3.34	1.96	1.69
8. rods 3, 4, 8, 9 at 35 in.	3.38	1.98	1.68
9. rods 5, 6, 7, 11 at 30 in.	3.59	2.03	1.85
10. rods 3, 12 in	3.14	1.60	1.96
11. rods 2, 5, 12 in	3.36	1.60	2.10

*does not include local peaking factor



HISTOGRAM OF REACTOR THERMAL OUTPUT AND OBSERVED ROD POSITION

FIG. 3.1



REGULATING ROD POSITION VS. CORE EXPOSURE

FIG. 3.2

	1	2	3	4	5	6	7	8	9	10	11	12	13	14
A					F	R106	D	D	R78	F				
B			D	S22	R81	R50	R1	R98	R26	R108	S1	D		
C		D	F	F	S3	R101	R146	R56	R2	S16	F	F	D	
D		S15	F	S11	R84	F	R49	R24	F	R59	S13	F	S2	
E	F	R89	S12	R105	F	R5	F	F	R47	F	R62	S9	R82	F
F	R74	R79	R35	F	R54	F	R34	R80	F	R117	F	R102	R121	R137
G	D	R124	R51	R63	F	R83	F	F	R120	F	R10	R107	R140	D
H	D	R144	R118	R32	F	R123	F	F	R114	F	R64	R13	R85	D
I	R3	R88	R21	F	R31	F	R55	R75	F	R96	F	R41	R30	R91
J	F	R142	S20	R44	F	R71	F	F	R4	F	R52	S17	R136	F
K		S7	F	S19	R122	F	R39	R6	F	R69	S18	F	S6	
L		D	F	F	S14	R29	R129	R38	R135	S4	F	F	D	
M			D	S21	R141	R90	R95	R94	R104	R57	S8	D		
N					F	R128	D	D	R58	F				

RECOMMENDED CORE LAYOUT, ALTERNATE DUMMY LOCATION 1

FIG. 3.3

	1	2	3	4	5	6	7	8	9	10	11	12	13	14
A					R106	F	D	D	F	R78				
B			D	S22	R81	R50	R1	R98	R56	R108	S1	D		
C		D	F	F	S3	R101	R146	R26	R2	S16	F	F	D	
D		S15	F	S11	R84	F	R49	R24	F	R59	S13	F	S2	
E	R74	R89	S12	R105	F	R5	F	F	R47	F	R62	S9	R82	R137
F	F	R79	R46	F	R54	F	R34	R80	F	R117	F	R102	R121	F
G	D	R124	R51	R63	F	R83	F	F	R120	F	R10	R107	R140	D
H	D	R95	R118	R32	F	R123	F	F	R114	F	R64	R13	R94	D
I	F	R90	R21	F	R31	F	R55	R75	F	R96	F	R73	R104	F
J	R3	R142	S20	R44	F	R71	F	F	R4	F	R52	S17	R7	R91
K		S7	F	S19	R122	F	R39	R6	F	R15	S18	F	S6	
L		D	F	F	S14	R29	R88	R136	R135	S4	F	F	D	
M			D	S21	R141	R129	R144	R85	R30	R93	S8	D		
N					R128	F	D	D	F	R58				

CORE LAYOUT FOR INITIAL CRITICAL MEASUREMENTS

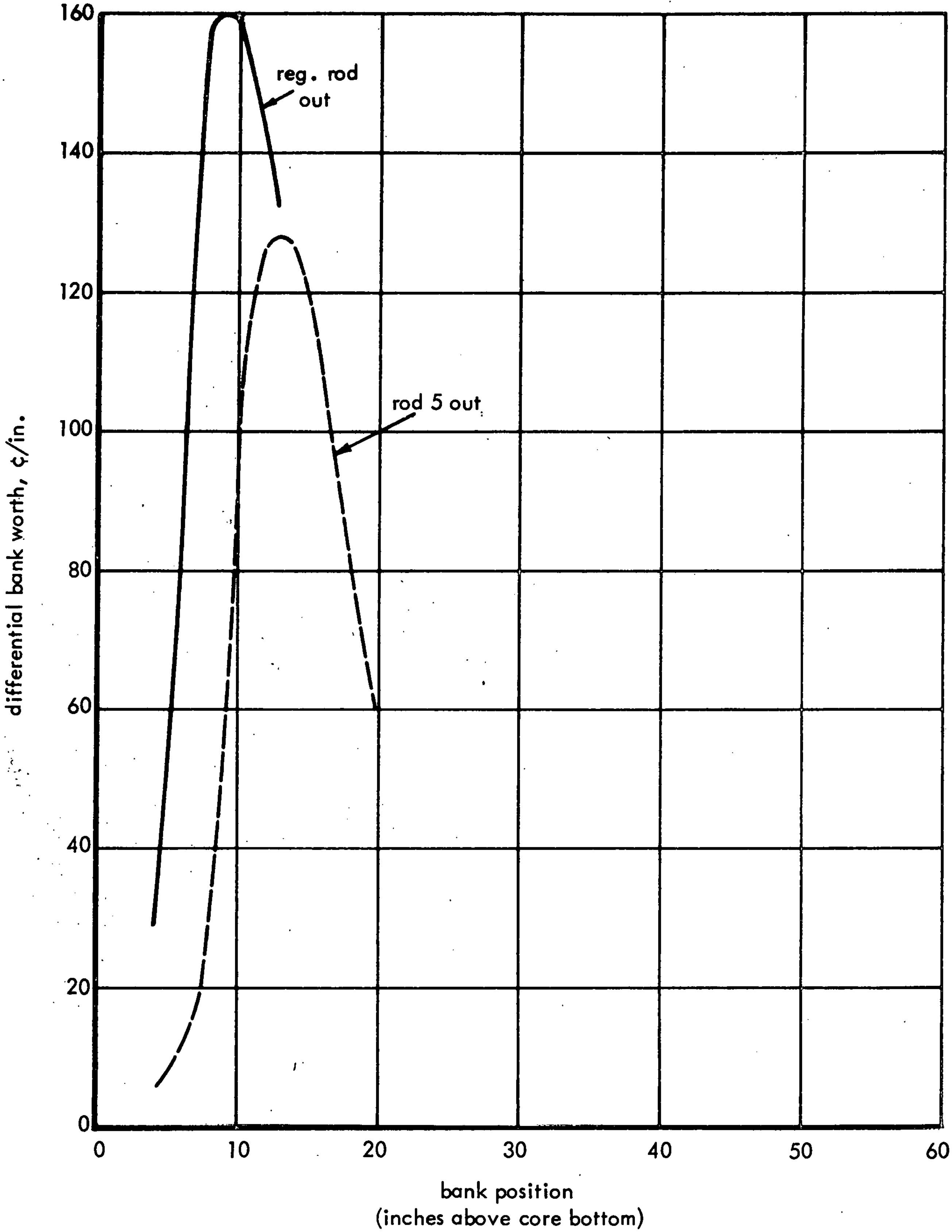
FIG. 3.4

	1	2	3	4	5	6	7	8	9	10	11	12	13	14
A					D	F84	R20	R97	F58	D				
B			D	S22	R127	R58	R54	R98	R50	R67	S1	D		
C		D	F102	F85	S3	R137	R135	R26	R121	S16	F104	F68	D	
D		S15	F90	S11	R85	F105	R22	R24	F86	R59	S13	F96	S2	
E	D	R115	S12	R105	F67	R113	F66	F77	R138	F76	R62	S9	R70	D
F	F91	R90	R91	F93	R100	F95	R34	R89	F94	R45	F65	R102	R31	F55
G	R60	R124	R51	R63	F74	R83	F75	F70	R120	F78	R10	R107	R96	R65
H	R103	R95	R118	R32	F83	R123	F82	F80	R114	F89	R2	R13	R94	R116
I	F88	R81	R106	F61	R133	F63	R55	R142	F72	R73	F64	R78	R104	F53
J	D	R139	S20	R44	F89	R15	F60	F81	R12	F92	R29	S17	R7	D
K		S7	F71	S19	R122	F62	R39	R6	F73	R84	S18	F87	S6	
L		D	F103	F56	S14	R74	R88	R136	R146	S4	F98	F69	D	
M			D	S21	R37	R129	R144	R5	R117	R93	S8	D		
N					D	F57	R11	R14	F54	D				

F - Core II feed element
R - Core I regular element
S - Core I spiked element
D - Dummy element

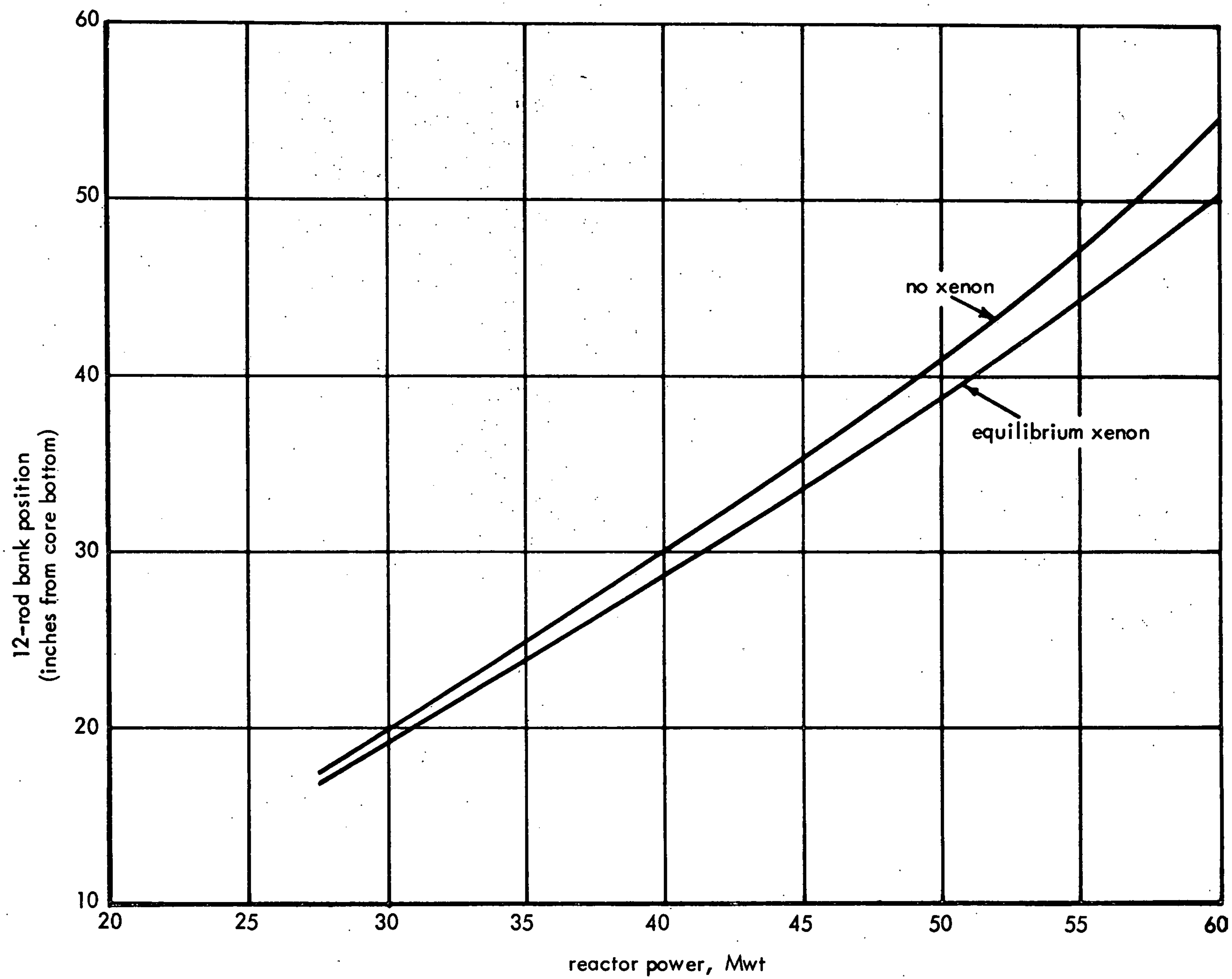
CORE LAYOUT FINAL LOADING PATTERN

FIG. 3.5



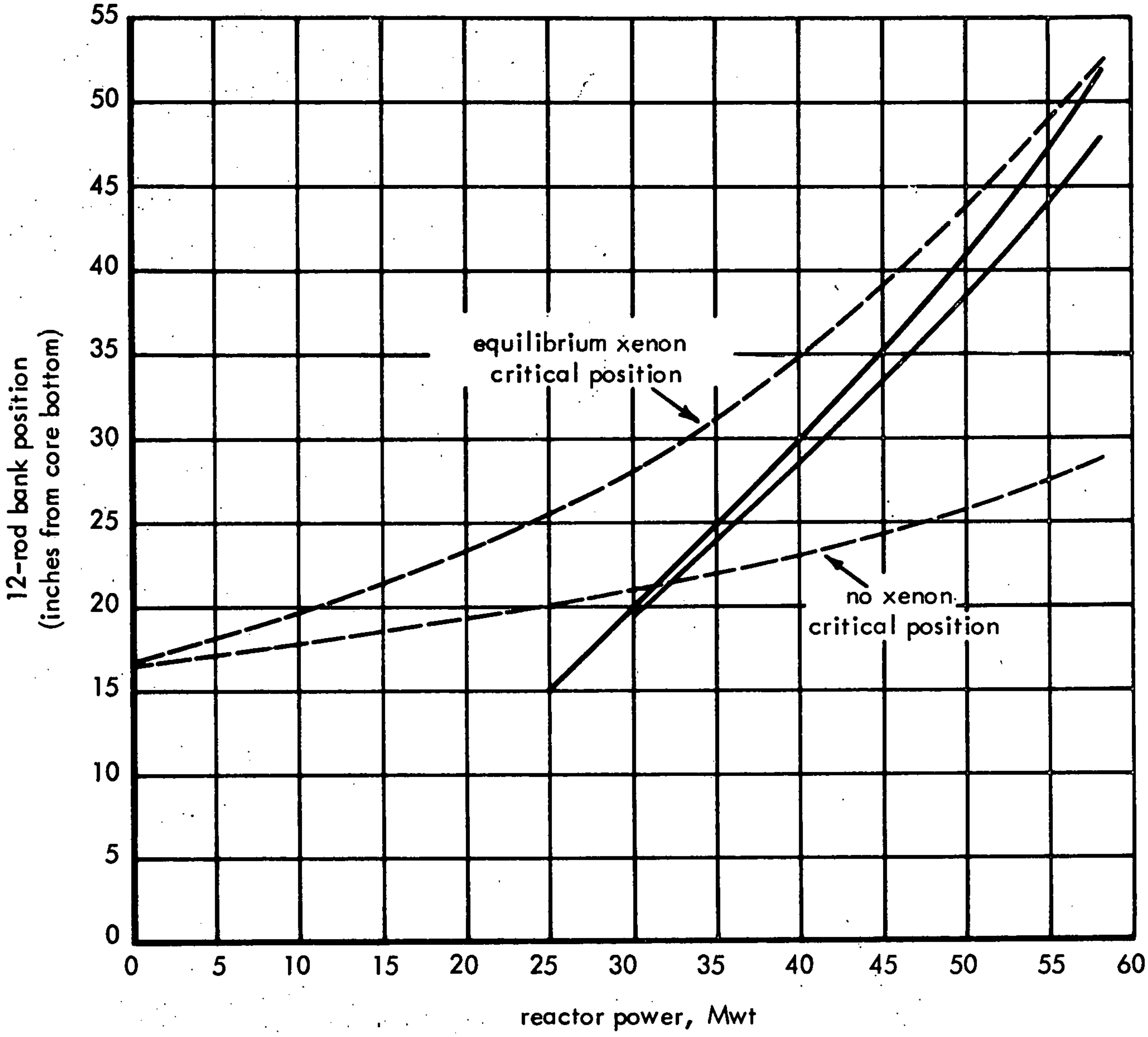
TWELVE-ROD BANK DIFFERENTIAL WORTH
VS. BANK POSITION FINAL CORE

FIG. 3.6



MINIMUM 12-ROD BANK POSITION VS. REACTOR POWER
(REG. ROD AT 16 IN.)

FIG. 3.7

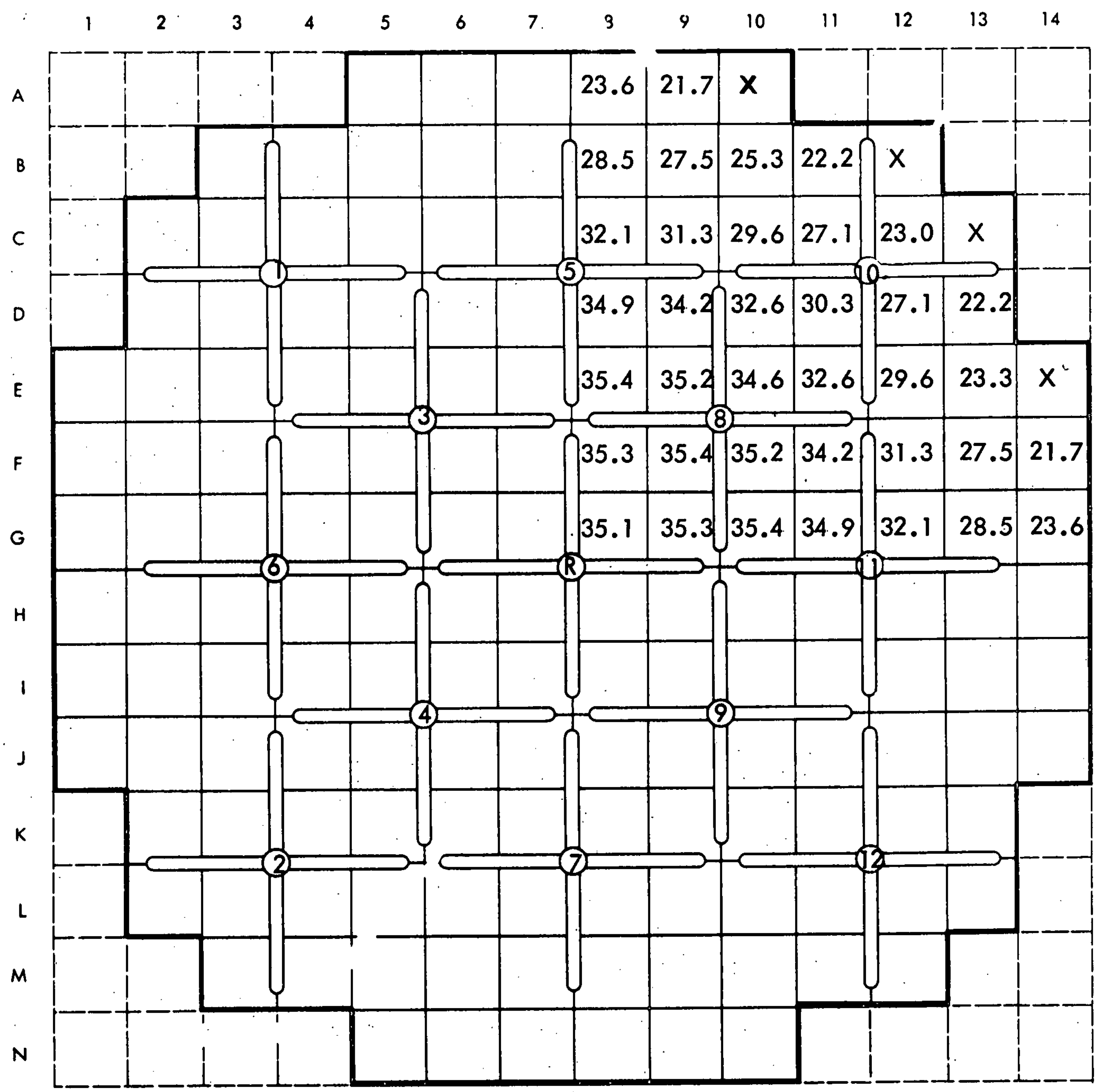


ESTIMATED 12-ROD BANK CRITICAL POSITION VS. REACTOR POWER
(REG. ROD AT 16 IN.)

FIG. 3.8

AVERAGE HORIZONTAL VOID DISTRIBUTION*

CORE I AT ~4000 MWD/MT



X denotes dummy elements

* values given refer to void percent within shrouds

CORE LAYOUT

FIG. 3.9

HORIZONTAL VOID DISTRIBUTION*

CORE II UNIFORM LOADING PATTERN

	1	2	3	4	5	6	7	8	9	10	11	12	13	14
A					X			23.5	22.1	X				
B			X					28.2	27.7	26.0	23.0	X		
C		X						31.4	31.1	30.3	28.3	24.5	X	
D								33.5	33.6	32.8	31.3	28.3	23.0	
E	X							34.1	34.3	34.2	32.8	30.3	26.0	X
F								32.0	33.5	34.3	33.6	31.1	27.7	22.1
G								30.2	32.0	34.1	33.5	31.4	28.2	23.5
H														
I														
J	X													X
K														
L		X											X	
M			X									X		
N					X					X				

1 2 3 4 5 6 7 8 9 10 11 12 R

X denotes dummy elements

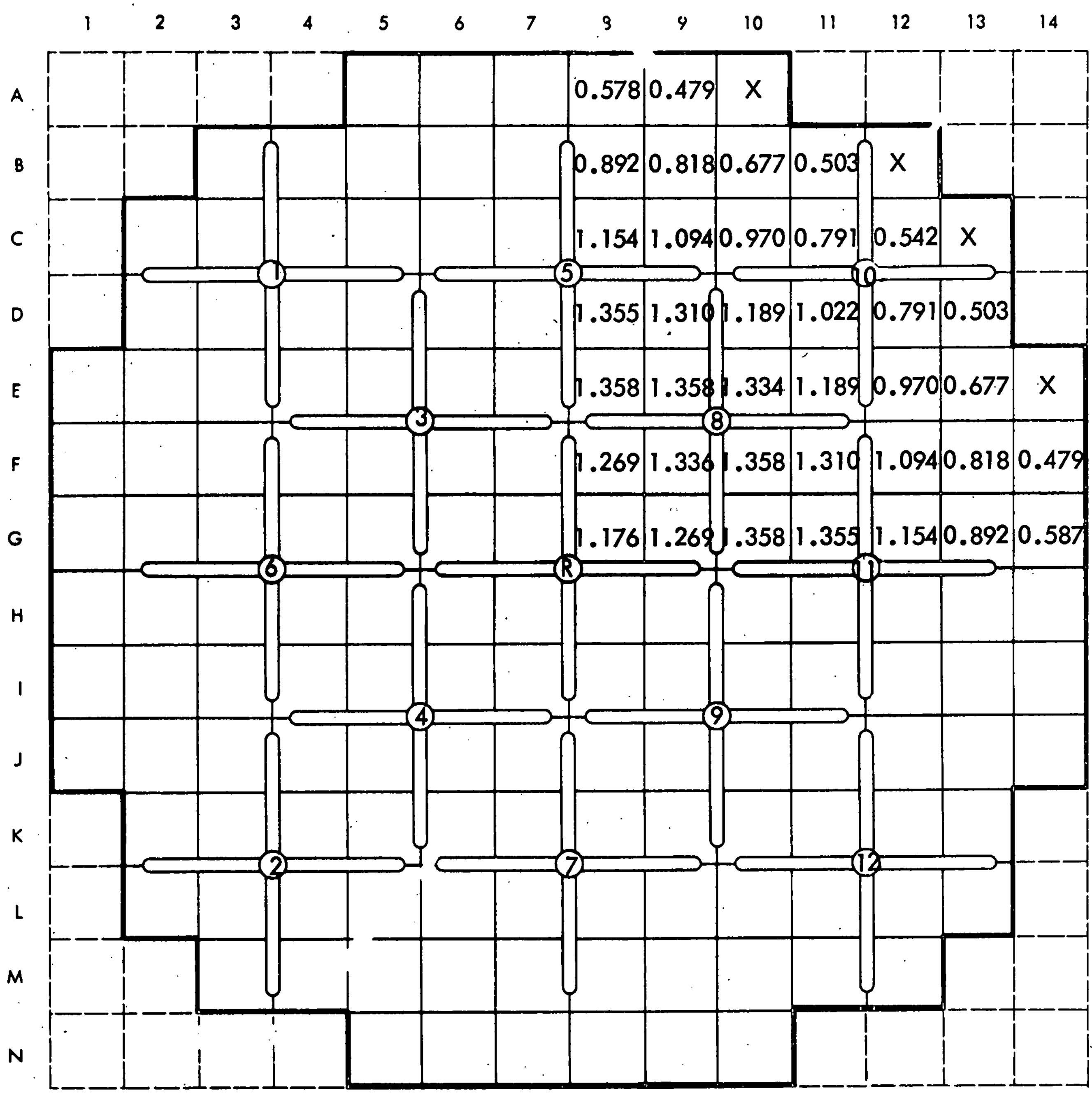
* values given refer to void percent within shrouds

CORE LAYOUT

FIG. 3.10

RELATIVE HORIZONTAL POWER DISTRIBUTION*

CORE I AT ~4000 MWD/MT



X denotes dummy elements

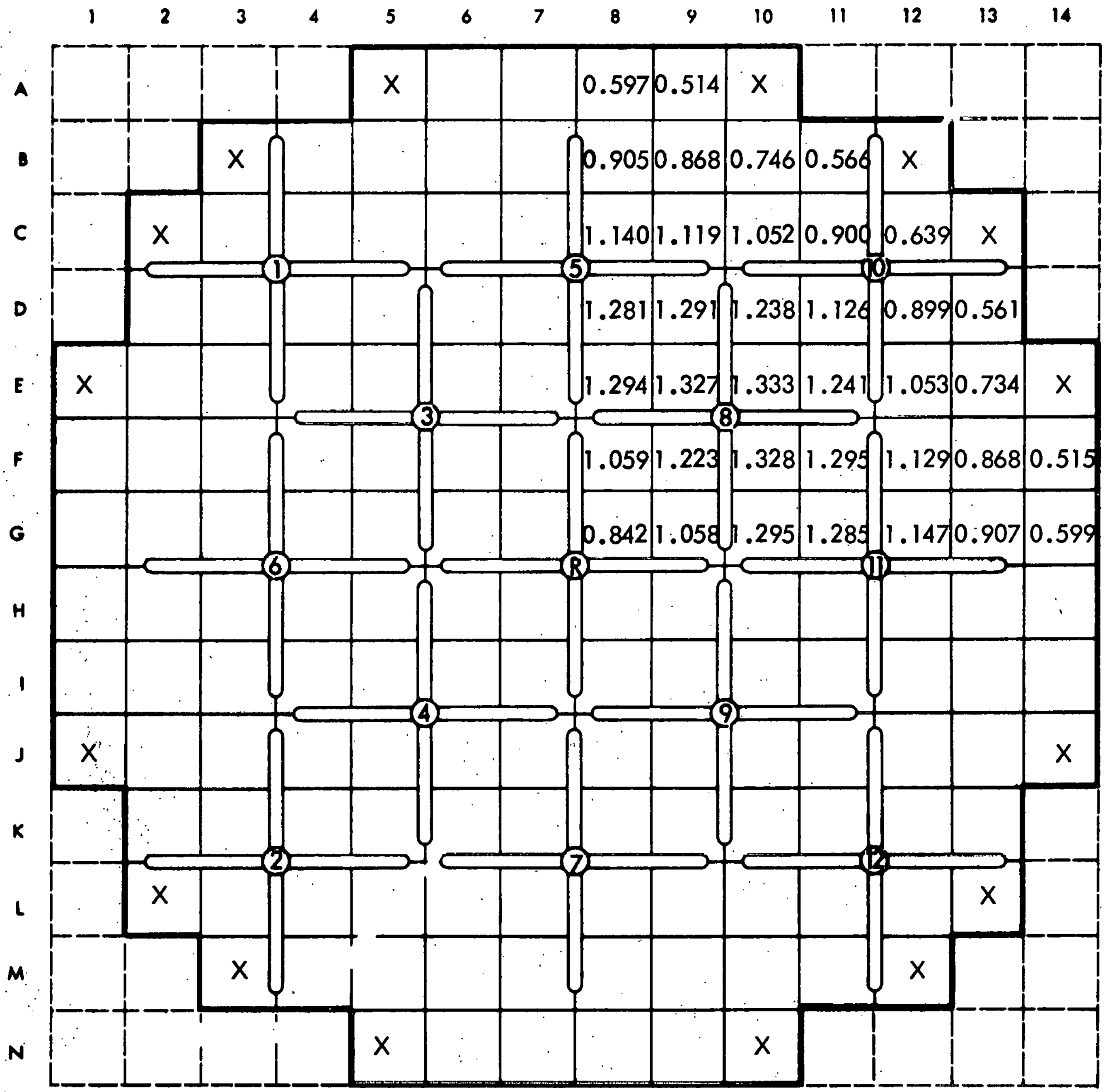
* does not include local peaking factor

CORE LAYOUT

FIG. 3.11

RELATIVE HORIZONTAL POWER DISTRIBUTION*

CORE II UNIFORM LOADING PATTERN



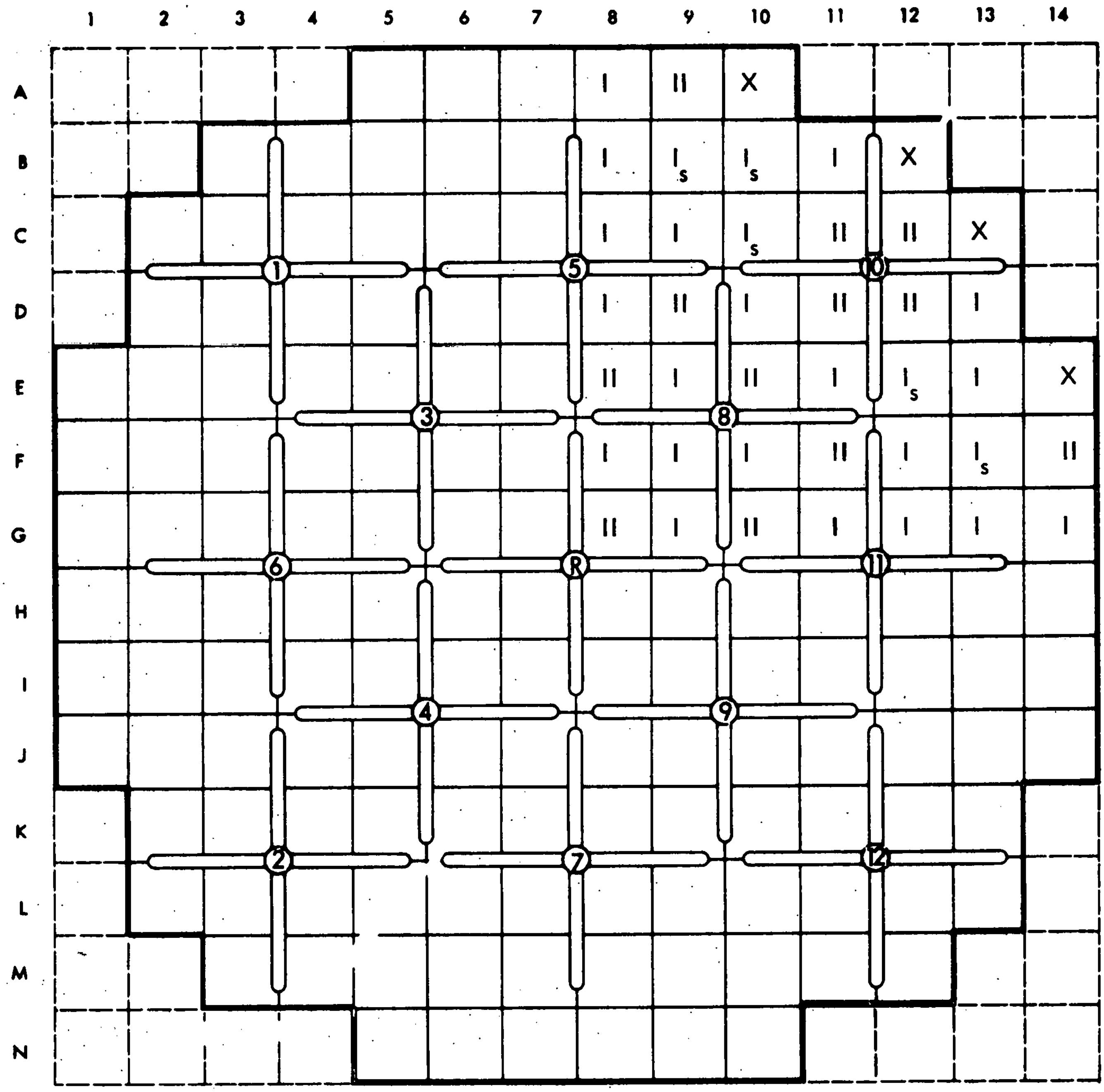
X denotes dummy elements

* does not include local peaking factor

CORE LAYOUT

FIG. 3.12

LOCATION OF CORE II FUEL ELEMENTS
PRELIMINARY LOADING PATTERN



I - Core I regular elements
I$_s$ - Core I spiked elements
II - Core II elements
X - Dummy elements

CORE LAYOUT **FIG. 3.13**

4. CORE MANAGEMENT

4.1 FUEL CYCLE STUDIES (TASK 201)

4.1.1 Major Objectives of the Fuel Cycle Study

The objective of Task 201 is to define the fuel management program which meets the thorium recycle objectives of the Elk River reactor. The primary effort, to date, has been concerned with the determination of a recommended loading pattern for the Core II fuel elements which, with a compatible rod program, would give desirable operating power distributions and fuel burnup characteristics, and would also satisfy the operating time requirements of the Rural Cooperative Power Association. A secondary objective of this effort was to develop, through calculations, the ability to specify (within three weeks of notification) a loading pattern which would satisfy any set of criteria set forth by the Chicago Operations Office, USAEC. It will be seen that this secondary objective achieved greater importance when the actual time for loading the core arrived.

Two specific goals for the fuel cycle were:

(1) to obtain an average burnup of approximately 14,500 Mwd/MT for the Core II fuel elements, and

(2) to obtain at least nine full power months of operation between refuelings.

The detailed results of the fuel cycle studies for the second core are contained in a topical report on the subject (see Ref. 2). The following sections review some of the more pertinent aspects of these results. For specific details the reader is referred to the topical report.

4.1.2 Constraints and Conditions Governing the Study

The study was governed by the following constraints and conditions:

(1) a minimum of 0.5% $\Delta\rho$ shutdown was required with the most reactive rod withdrawn from the core;

(2) the analysis of shutdown margins for the loading patterns was to be based on the use of boron-stainless-steel control rods;

(3) the Core II fuel contains thoria-urania fuel pellets loaded with U-235 to 4.4 wt. % of the total uranium plus thorium content;

(4) the reactivity insertion rates by control rod movement were to be within the Core I technical specification limitations;

(5) a desirable criterion was that the operating power distribution be such that the maximum heat flux would remain within the Core I technical specification limitation of 313,000 Btu/hr-ft^2; however, a pattern with other desirable characteristics would not be rejected solely for this reason;

(6) it was assumed that the initial Elk River core would operate until an average core exposure of 7000 Mwd/MT was achieved prior to discharge.

This last assumption was based upon lifetime calculations previously performed under Task 102, Reactivity History, and on the comparison of the predicted and observed rod position vs. core exposure. It was also recognized that upon inspection of the fuel prior to reloading, theconstraints and conditions might be expanded or changed. Thus, the study was directed toward a flexibility of being able to adjust the loading pattern according to the dictates at the actual time of loading.

4.1.3 Factors Governing the Choice of Loading Pattern

The major emphasis in the study was placed on a three-batch loading cycle of Core II elements (i.e., approximately one-third of the core is discharged at the end of each burnup period and replaced by fresh feed fuel). A separate phase of the study, however, concerned the effect of varying the size of the discharge group as well as incorporating the 20 Core I spares into the feed patterns. To investigate the three-batch loading cycle, it was assumed that 48 of the Core I regular elements (4.3 wt. % U-235) having the highest exposure would be discharged. The average exposure of these 48 elements was predicted (see Ref. 1) to be approximately 8900 Mwd/MT (assuming a core average exposure of 7000 Mwd/MT). The 100 remaining Core I fuel elements would at this time include 80 regular elements with an average exposure of 5350 Mwd/MT and 20 spiked elements (5.2 wt. % U-235) with an average exposure of 9100 Mwd/MT.

The first step in determining a tentative core arrangement was to select a loading pattern for the Core II feed fuel. This loading pattern could be of two basic types: either, (1) the dispersed (or uniform) type, in which the feed fuel is distributed more or less uniformly across the whole core; or, (2) the zoned type, in which the feed fuel is primarily grouped into one area, or zone, of the core. Examples of dispersed loading are shown in Figs. 4.1, 4.2, and 4.3, and are designated as patterns A, B, and C, respectively. Examples of zoned patterns are given in Figs. 4.4 and 4.5 (designated patterns D and E, respectively). Figure 4.4 shows a typical central loading pattern in which the feed fuel is grouped toward the center of the core; and, Fig. 4.5 shows a typical peripheral loading pattern in which the feed fuel is grouped around the core periphery. Starting with basic feed patterns similar to these, various patterns were assumed for the Core I exposed elements.

All tentative core arrangements were subjected to a determination of the room temperature excess reactivity and minimum stuck-rod shutdown margin. Patterns which exhibited acceptable room temperature characteristics were examined further on the basis of their hot, operating excess reactivity and power distribution. Several patterns which were still within the acceptable limits were then subjected to a detailed analysis of the burnup characteristics, in which the effects of rod programming were considered.

Since the choice of a loading pattern for the first batch of the Core II fuel can affect not only the second burnup interval but also the entire fuel cycle, patterns which were still attractive at this point were examined further to determine the type of equilibrium cycle which might result. The final recommendation, then, was the pattern considered best to satisfy the objectives and conditions delineated above for both the initial and subsequent partial reloads of the core.

Investigations involving the zoned patterns led to the conclusion that the advantages of this type of pattern were outweighed by the disadvantages. For example, the out-in pattern shown in Fig. 4.5 groups the feed fuel on the periphery of the core. This pattern had the advantage of a lower peak-to-average power and, possibly, a slight increase in the Core I average fuel exposure. The disadvantages included a low shutdown margin and a short operating time. For the three variations which were investigated, the shutdown margins were in the range of 0.7 to 0.3% $\Delta\rho$, and the operating time was approximately eight full power months. Since power peaking is not a real problem, the advantages offered by this pattern are of little value and the study was not pursued beyond the first two cycles.

The in-out pattern shown in Fig. 4.4 offers a longer cycle time for the first reload interval, and a slightly higher exposure for the Core II elements. However, the average exposure for the Core I elements would be slightly lower, the peak-to-average power would be higher (but not excessive), and the possible reactivity insertion rates from rod movement would be high. Since the longer operating time for the first cycle is not compatible with the projected shutdown schedule for plant maintenance, the advantages of this pattern were considered to be minimal.

Investigations of loading patterns involving the spare Core I elements and 1/4-core replacement instead of 1/3-core replacement also showed no particular advantages over the recommended pattern. However, the study of these patterns coupled with studies of the zoned patterns did provide the flexibility of being able to change the loading to satisfy new conditions and constraints that might be imposed at the time of refueling.

4.1.4 Characteristics of the Recommended Loading Pattern

The recommended loading pattern had 48 Core II elements distributed in a dispersed (uniform) array (see Fig. 4.6). The figure identifies the location of the new or feed fuel, designated by the letter F, and the specific recommended location of each remaining Core I fuel element. The Core I elements are designated by the letter R for regular elements and S for spiked elements. The spiked element serial numbers are also included.

This pattern would provide approximately 10.5 months of full power operation, has excellent power distribution characteristics, and has at least 2.5 percent shutdown margin for the cold, one rod out condition. Further, the projected characteristics of this type of loading offer the best assurance of meeting the objectives and conditions outlined earlier.

Pertinent information regarding this recommended pattern is summarized in Table 4-1.

TABLE 4-1

PREDICTED CHARACTERISTICS OF RECOMMENDED LOADING PATTERN

average exposure of removed Core I fuel	8900 Mwd/MT
average exposure of remaining Core I fuel	
1. spikes	9100 Mwd/MT
2. regulars	5350 Mwd/MT
operating peak-to-average power	
1. initial	2.51
2. maximum (4 months after refueling)	3.18
shutdown margin, most reactive rod withdrawn	2.7% $\Delta\rho$
operating time (full power months)	~10.5 months
projected Core II average fuel exposure	~15,000 Mwd/MT
projected Core I average fuel exposure	~11,800 Mwd/MT

One further result of the fuel cycle studies was that the shutdown margin could be increased significantly by a rearrangement of the dummy fuel element locations. This provided a backup in the event that measurements showed a smaller shutdown margin with the recommended pattern than was expected. This fuel pattern with the alternate dummy locations is shown in Fig. 3.3 while the regular dummy locations are shown in Fig. 4.6.

As a precaution, it was decided by the RCPA to load the reactor initially with the alternate dummy locations and, once the shutdown margin was established, to revert to the recommended pattern. The loading pattern actually used with the alternate dummy locations is shown in Fig. 3.4; and when compared with Fig. 3.3, differs slightly from the pattern obtained from the fuel cycle studies. The comparison of the predicted and measured characteristics was described in Sec. 3.1.2.

4.1.5 Final Loading Pattern for First Core II Reload

During the shutdown for refueling, the fuel elements were visually inspected and it was found that a number of the Core I regular elements had fuel pins that exhibited some degree of bowing. Of the 100 Core I elements scheduled to remain in the reactor, 26 of the regular (4.3 wt. % U-235) elements had one or more bowed pins (none of the spiked elements, 5.2 wt. % U-235, showed any signs of bowing). The bowing for the most part was minor and no pattern or common denominator such as high burnup or core location was apparent. A decision was made by the Elk River Safety Committee to postpone loading any element with pins that showed evidence of bowing.

Accordingly, a new fuel loading pattern had to be developed, the regulating rod worth redetermined for the new pattern, and the limit curves of Fig. 3.7 recalculated for the new pattern. As stated previously, one of the objectives of the fuel cycle studies was to be in a position to specify a new loading pattern within three weeks of notification. However, at the time of notification, which occurred during the Safety Committee meeting of June 23, 1966, the utility was ready to use power and it was expedient to develop and confirm a fuel pattern in as short a time as possible.

The 26 elements with bowed pins were replaced by 18 fresh Core I spare regular elements and 8 irradiated Core I regular elements which were originally to be discharged. A pattern, shown in Fig. 3.5, was obtained which left the feed Core II elements and the spiked elements unchanged, but required considerable reshuffling of the Core I regular elements. This final loading pattern was determined, the limiting regulation rod position recalculated, and the 12-rod bank limit curves (Fig. 3.7) with and without xenon reanalyzed ina period of four working days from the time of notification, which more than achieved the second objective of the fuel cycle studies.

4.1.6 Characteristics of Final Loading Pattern

The cold, zero power nuclear characteristics of the final loading pattern are given in Sec. 3.1.1. A summary of some of the more pertinent operating characteristics are given in Table 4-2.

TABLE 4-2

SUMMARY OF PERTINENT OPERATING CHARACTERISTICS

operating peak-to-average power	
1. initial	3.45
2. maximum	3.45
operating time (full power months)	~9.6
(Mwd/MT)	~4300

The projected fuel exposure for the Core I elements will depend upon the disposition of the elements with bowed tubes, and upon the future loading patterns which will now include Core I spares; the likelihood of additional bowed-pin elements will also play a part in the determination of the projected fuel exposure for the Core I elements.

The predicted regulating rod position vs. Mwd/MT for the final pattern is given in Fig. 4.7; and, as shown, the calculated irradiation time before the next refueling is 4300 Mwd/MT, or about 9.6 full power months. This predicted curve was generated assuming that equilibrium concentrations of protactinium and samarium were present.

4.2 FUEL ELEMENT EXPOSURE (TASK 203)

The purpose of this task is to provide the necessary information and recommend procedures for maintaining the Elk River reactor core fuel inventory as a function of operating time (megawatt days).

A report has been issued (see Ref. 3) which gives the method and necessary data for determining the inventory. This report has been compiled in a manner which enables the fuel inventory to be obtained and reported for three groups of elements; the Core I regular elements (originally 4.3 wt. % U-235), the Core I spiked elements (originally 5.2 wt. % U-235), and the Core II elements (originally 4.4 wt. % U-235). The time interval covered is from July 1966 to the second refueling which will occur approximately 9.6 full power months after refueling.

The calculational model used for determining the inventories is, in general, that used for the first fuel inventory report (Ref. 4, supplement 2) and discussed in Ref. 1.

One modification has been incorporated into the model since the initial isotopic inventory compilation was made. The point isotopic depletion calculation which serves as input to the three-dimensional analysis has been altered to incorporate more recent information. This modification does not alter k_∞ vs. time, but does perturb the isotopic concentrations slightly. These small changes have been included in the latest compilation.

The average burnup of each element in the core was determined for each 1000 Mwd/MT of core average burnup. The isotopic inventory of the total core was then obtained by summing the inventories of each individual element.

4.3 TECHNICAL SPECIFICATIONS FOR ERR CORE II (TASK 208)

The anticipated use of new (Core II) fuel elements and B_4C in-tube-type control rods necessitated changes in the technical specifications for the Elk River reactor. The majority of the changes required were descriptive in nature to account for differences in the physical makeup and materials of construction used in the Core II fuel and control rods. The remainder of the changes was related to the Reactivity Control Systems. The role of the ERR-OAP in these changes involved the initial preparation of recommendations to the Chicago Operations Office, USAEC, regarding the revised wording, the reasons for the changes, and the safety considerations relevant to the changes.

In the course of this work, technical specification limitations were recommended which were based on projected core characteristics. The recommended limitations do not necessarily conform to the characteristics of either the Core I or Core II loading patterns, but were based on typical loading patterns studied under Task 201, Fuel Cycle Studies. Some of the more pertinent information developed during the report period is reported below.

4.3.1 Reactivity Insertion Rates

To ensure that the proposed Core II loading patterns would meet the technical specifications for reactivity insertion rates, calculations were performed for the loading pattern with the highest excess reactivity. For these calculations, rods 5, 10, 11, and R were moved as a bank, and the maximum worth of rod 8 moving below the bank was calculated. When this rod configuration was measured in Core I, the maximum reactivity insertion rate was obtained. The results of these calculations are shown in Fig. 4.8, wherein the maximum reactivity worth of rod 8 is plotted as a function of the 4-rod bank position. From these and similar results, it was determined that the Core II reactivity insertion rates would be within the Core I technical specification limitations.

In response to a request initiated by the ERR Safety Committee, the reactivity insertion rate was also calculated for the hot, zero power core. This analysis was not made previously because the Core I startup data had indicated the hot, zero power case to be less severe than the cold case. Table 4-3 shows the results of this calculation compared to the room temperature condition for both boron-stainless-steel and B_4C control rods.

TABLE 4-3

MAXIMUM REACTIVITY WORTH OF CONTROL SYSTEM

rod configuration	type of rods	temperature	¢/in.
rods R, 5, 10, and 11	B-SS	20 C	45.0
banked: rod 8	B-SS	280 C	39.4
below the bank	B_4C	20 C	54.0
	B_4C	280 C	47.3

4.3.2 Cold Shutdown Margins

In the course of the investigation of changes required to allow the use of B_4C control rods, cold shutdown margins were calculated for the central and uniform loading patterns on the basis of B_4C rods. The results are given in Table 4-4 compared to the boron-stainless-steel rods.

TABLE 4-4

COLD k_{eff} WITH B_4C AND BORON-STAINLESS-STEEL RODS

rod type and configuration	loading pattern central	loading pattern uniform
no rods	1.153	1.123
boron-stainless-steel rods		
all rods	0.920	0.917
rod R out	0.974	0.950
rod 5 out	0.990	0.988
rod 10 out	0.957	0.970
B_4C rods		
all rods	0.901	0.903
rod R out	0.955	0.936
rod 5 out	0.984	0.982
rod 10 out	0.947	0.963
boron-stainless-steel with B_4C center rod		
all rods	0.918	
rod R out	0.974	

The use of all B_4C rods increases the total shutdown by 1.9% Δk and 1.4% Δk for the central and uniform loading patterns, respectively. The minimum stuck rod shutdown margin is increased by approximately 0.6% Δk for both patterns.

4.3.3 Worth of Most Reactive Rod Withdrawn

A calculation was made to determine the worth of the most reactive rod when withdrawn from the critical bank position. It was found that the complete withdrawal of rod 5 from the 13-rod bank critical position for the recommended loading pattern (Task 201) would add approximately 0.8% Δk to the core.

4.3.4 Peak-to-Average Powers

The peak-to-average powers were investigated for operating rod configurations differing from that used in Core I (12-rod bank out, reg. rod controlling). The calculated gross peak-to-average powers (excluding local peaking) are given in Table 4-5 for several rod configurations in the recommended pattern.

TABLE 4-5

PEAK-TO-AVERAGE POWER FOR VARIOUS ROD CONFIGURATIONS

rod configuration (inches from core bottom) reg. rod	12-rod bank	gross peak-to-average power
20	40	3.6
20	45	3.4
20	50	3.2
15	50	2.9
10	50	2.5

Based on the above analysis, it was recommended that the limitation on the maximum excess reactivity held down by the center regulating rod be no less than that of Core I (0.033 Δk). The actual limitation that was imposed was 0.030 Δk. The effect of this limitation on the rod programming and operating power distribution is discussed in Secs. 3.2.3 and 3.3.1.

The effect of the rotational orientation of the exposed Core I elements in the recommended loading pattern was also examined in response to an inquiry from the ERR Safety Committee. There are no physical constraints on the rotational orientation of fuel elements in the core; i.e., an element can be loaded with any one of its four corner pins oriented toward the center of the shroud box. It would be possible to maintain a record of the orientation by referring to the position of the engraved serial number on the corner of each element. The maintenance of this record, however, would require another accounting step and would complicate the loading procedure since, at present, there is no simple method to control the rotation of the element as it is lowered into the core.

The principal effect of the rotational orientation of an exposed fuel element would seem to be the effect on local peaking caused by the variation in exposure across the element. For the Core I elements in the proposed loading pattern, the ratio of the maximum to the minimum exposure across an element is less than 1.30, which corresponds to a 3-4 percent variation in the fission cross section. The effect on the local peaking of the orientation of an exposed fuel element would be a less than 5 percent variation.

4.3.5 Burnout Heat Flux Ratios

In conjunction with the thermal and hydraulic analysis performed under Task 201, the Core II burnout heat flux ratios were calculated for 100 percent and 125 percent of full power. For the evaluation of the burnout heat flux at 125 percent power, the extrapolated steam voids in the downcomer are 13 volume percent and the corresponding reduction in the thermal driving head is 2.28 ft. The 100 percent power values are 7 volume percent and 1.3 ft. Table 4-6 gives the maximum heat flux and the burnout heat flux ratio for both a zoned and a uniform Core II loading pattern, as compared to the Core I values.

TABLE 4-6

MAXIMUM HEAT FLUX AND BURNOUT RATIOS

	Core I	Core II uniform loading	Core II zone loading
maximum heat flux at 100 percent power (Btu/hr-ft^2)	238,100	217,300	224,600
burnout heat flux ratio*			
100 percent power	3.8	4.2	4.1
125 percent power	2.9	3.3	3.2

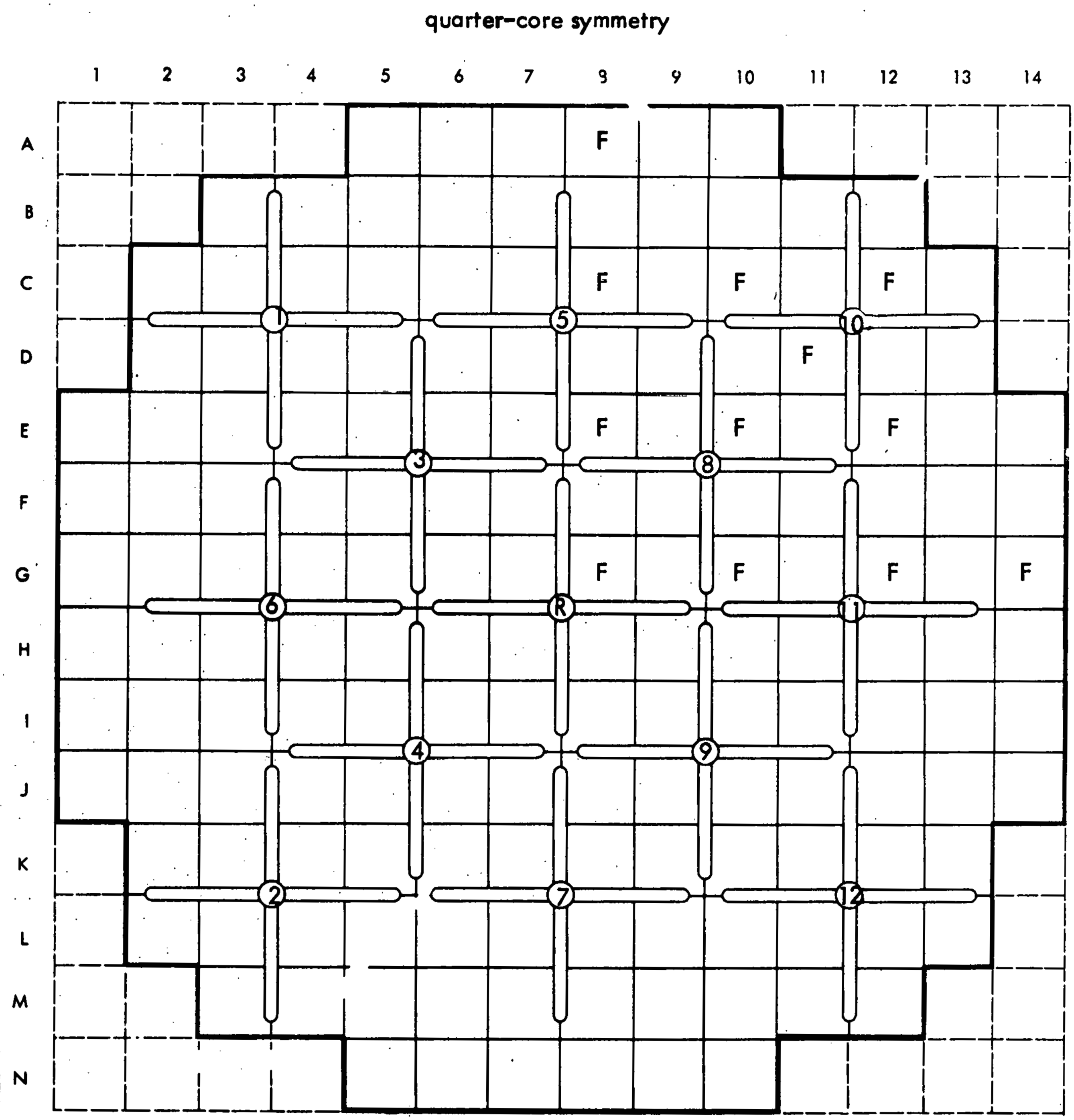
*based on APED-3892

The technical specifications for the ERR restrict the maximum heat flux to 313,000 Btu/hr-ft^2 at full power. From the values in Table 4-6, it is seen that the maximum heat flux for these loading patterns would have been well within this limit, if the rod program were the same as that used in Core I. The burnout heat flux ratios are limited by specifications to a factor of 3.2 based on the Griffith Correlation. Since the Griffith Correlation gives calculated burnout heat flux values nearly twice those reported in APED-3892, it is seen that the minimum burnout ratios are well within the specifications.

4.3.6 Criticality - Fuel Element Storage

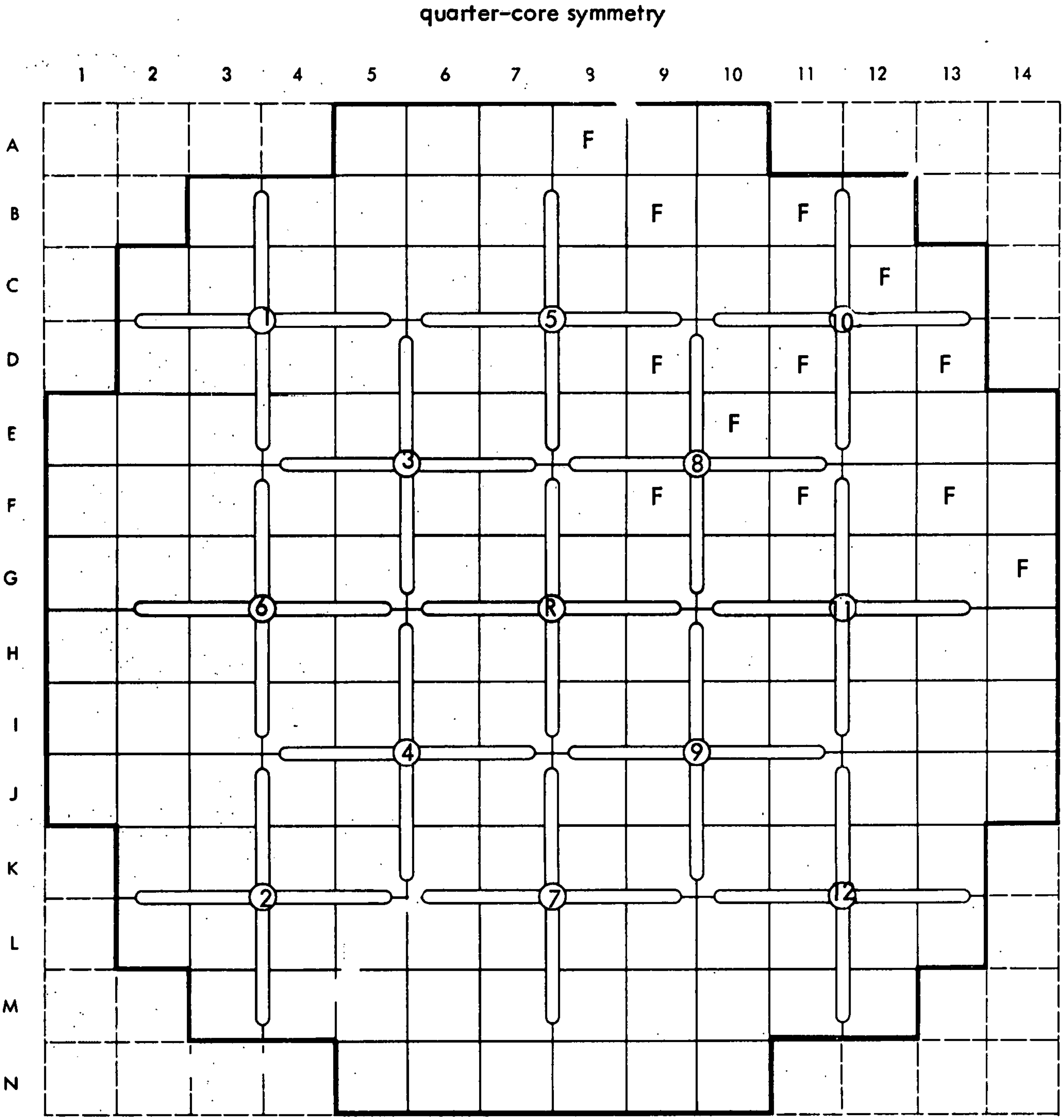
Drawings of the fuel element storage cabinets at the ERR were examined to determine if further calculations were needed to assure subcriticality with the Core II fuel elements. The most pessimistic assumption would be that the storage cabinets are flooded with water. A previous calculation, which assumed this condition, was performed for the fresh Core I spiked elements (5.2 wt. % U-235 with 600 ppm boron in the cladding). The calculation assumed an infinite row, one element wide, with a water reflector. The k_{eff} of this array was less than 0.72. In comparison, the reactivity of fresh Core II elements (4.4 wt. % U-235 with no boron in the clad) is within 0.1% $\Delta k/k$ of the Core I spiked elements. Therefore, it was concluded that the k_{eff} of Core II elements in flooded storage cabinets would also be less than 0.72.

A previous PDQ code calculation of the storage array in the fuel element storage well showed that the k_{eff} is less than 0.7 for the Core I spiked elements. In like manner, the k_{eff} for the Core II elements will be less than 0.7, since the reactivity of these elements is slightly lower than that of the Core I spiked elements.



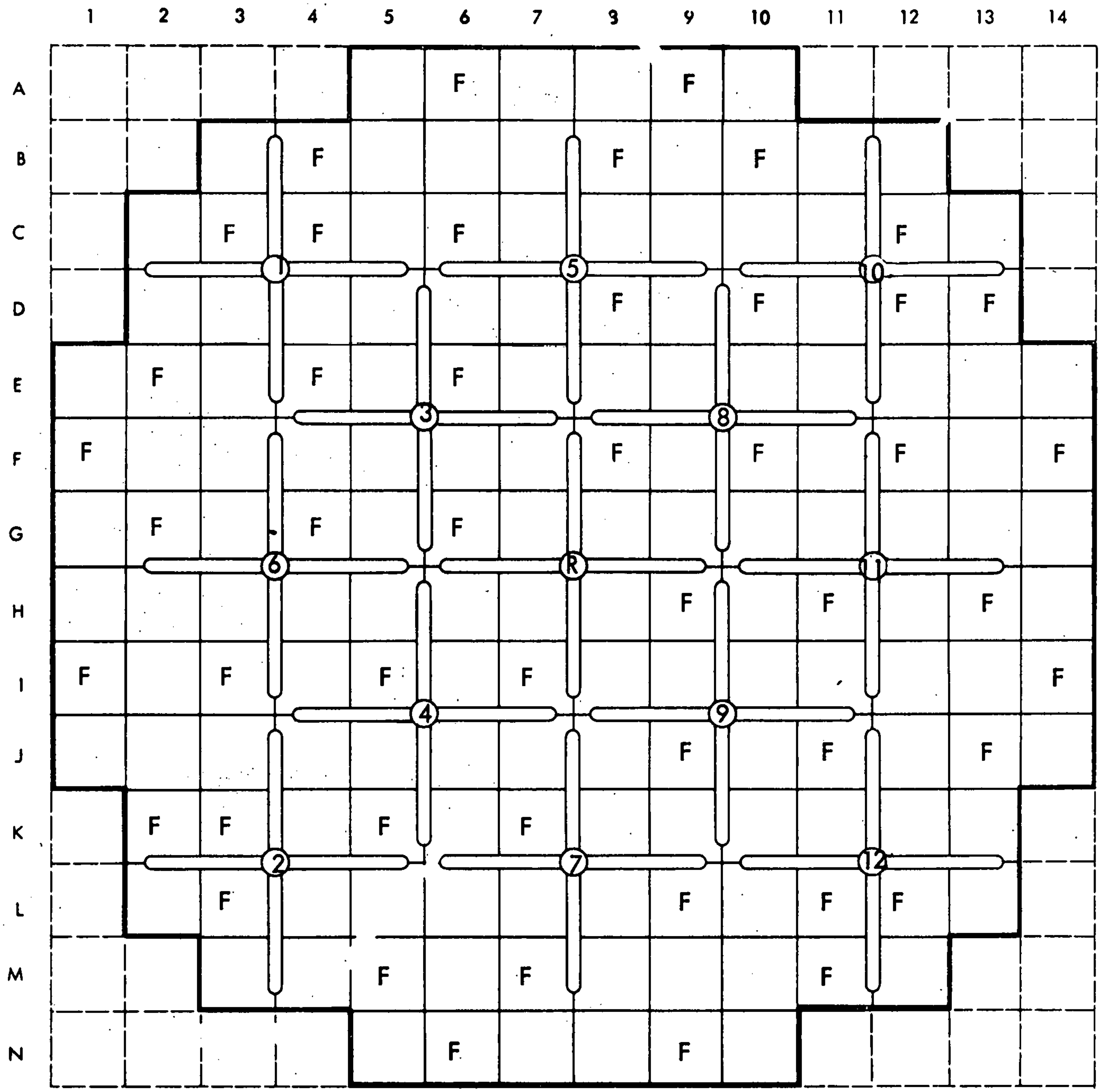
CORE LAYOUT, LOADING PATTERN A FOR THE 4.4 w/o U-235
CORE II FEED ELEMENTS

FIG. 4.1



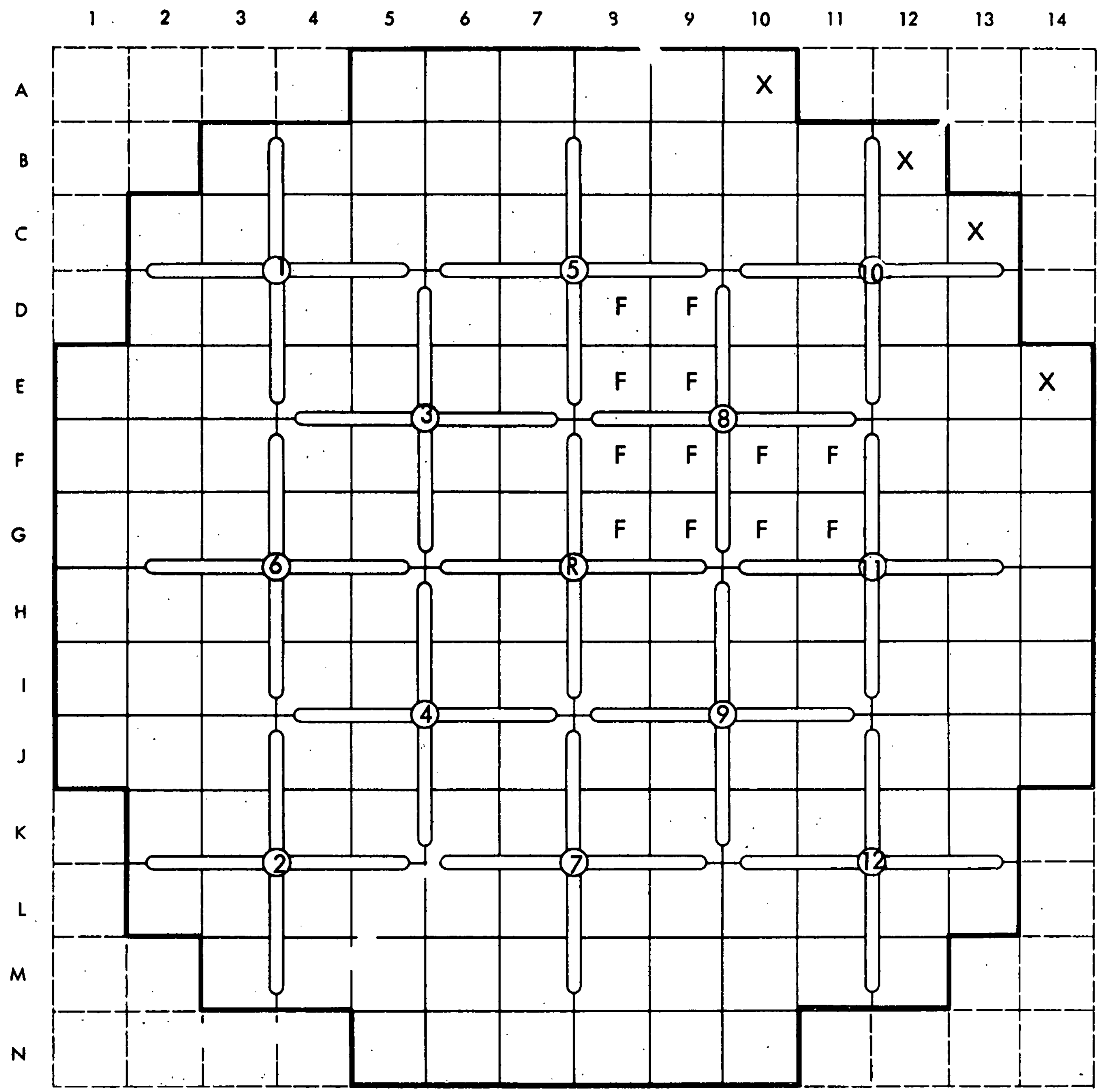
CORE LAYOUT, LOADING PATTERN B FOR THE 4.4 w/o U-235
CORE II FEED ELEMENTS

FIG. 4.2



CORE LAYOUT, LOADING PATTERN C FOR THE 4.4 w/o U-235
CORE II FEED ELEMENTS

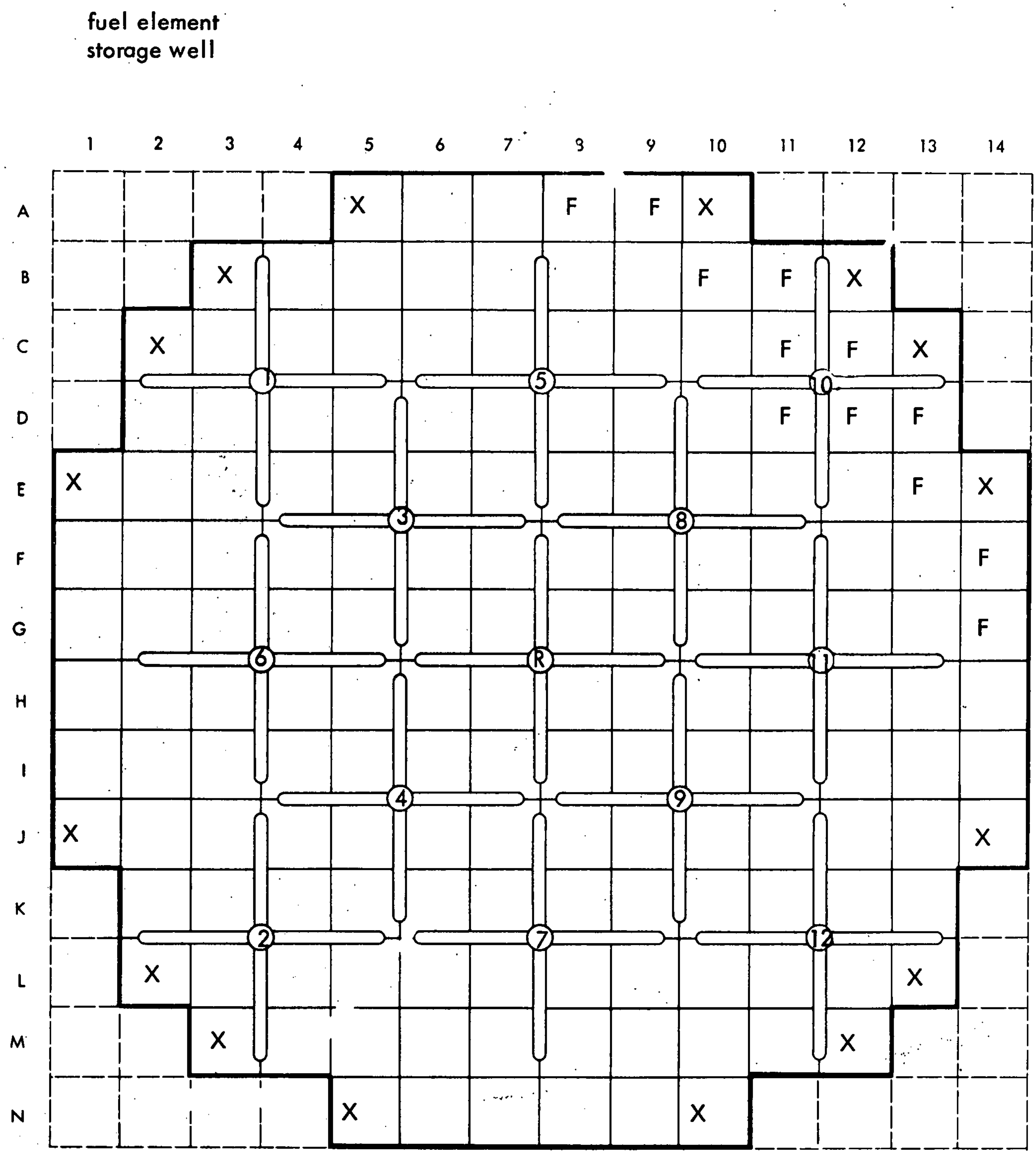
FIG. 4.3



X denotes dummy elements

CORE LAYOUT, LOADING PATTERN D FOR THE 4.4 w/o U-235
CORE II FEED ELEMENTS

FIG. 4.4



X denotes dummy elements

CORE LAYOUT, LOADING PATTERN E FOR THE 4.4 w/o U-235
CORE II FEED ELEMENTS

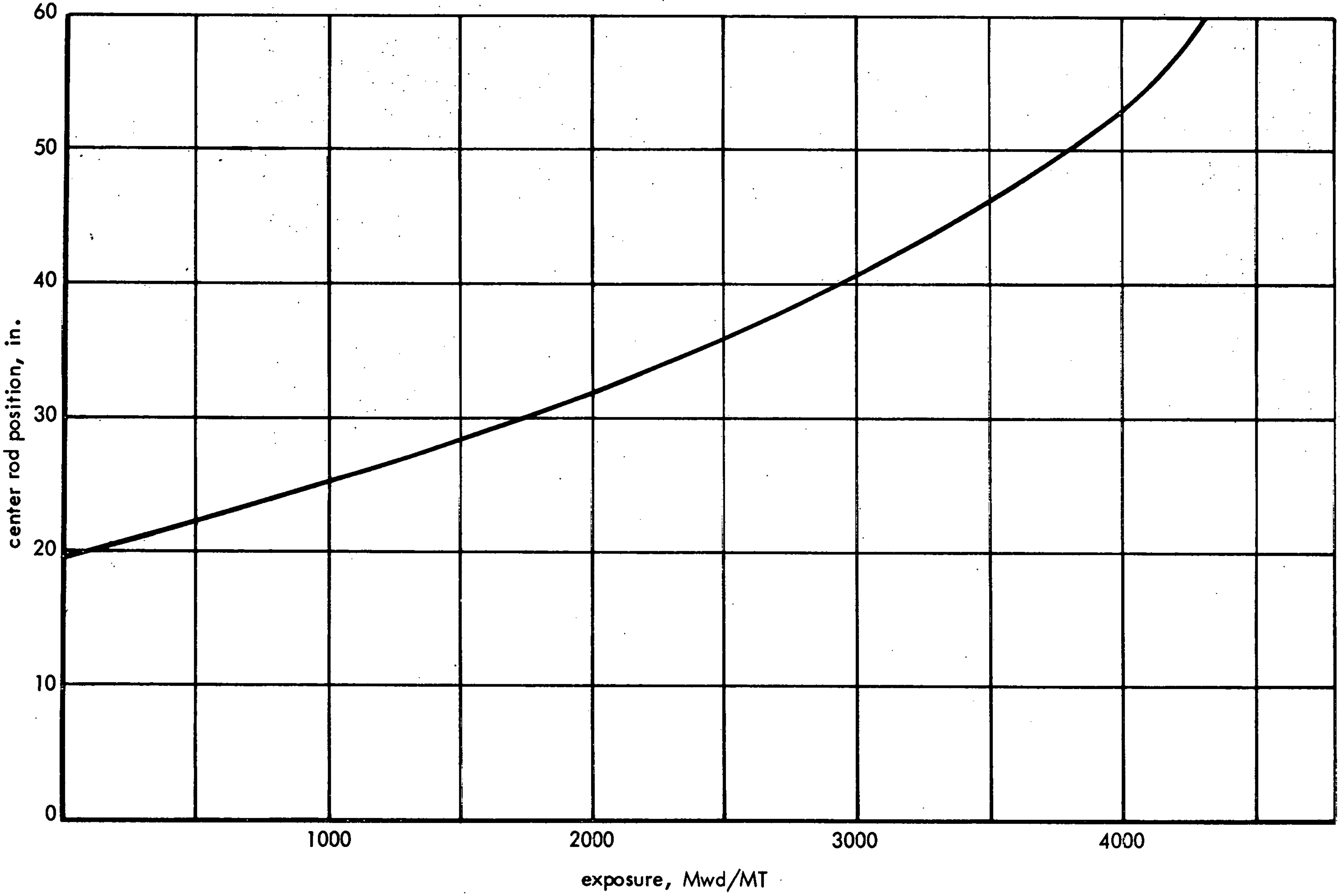
FIG. 4.5

fuel element
storage well

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	
1 A					D	F	R106	R78	F	D					1
2 B			D	S22	R81	R50	R1	R98	R26	R108	S1	D			2
3 C		D	F	F	S3	R101	R146	R56	R2	S16	F	F	D		3
4 D		S15	F	S11	R84	F	R49	R24	F	R59	S13	F	S2		4
5 E	D	R89	S12	R105	F	R5	F	F	R47	F	R62	S9	R82	D	5
6 F	F	R79	R35	F	R54	F	R34	R80	F	R117	F	R102	R121	F	6
7 G	R74	R124	R51	R63	F	R83	F	F	R120	F	R10	R107	R140	R137	7
8 H	R3	R144	R118	R32	F	R123	F	F	R114	F	R64	R13	R85	R91	8
9 I	F	R88	R21	F	R31	F	R55	R75	F	R96	F	R41	R30	F	9
10 J	D	R142	S20	R44	F	R71	F	F	R4	F	R52	S17	R136	D	10
11 K		S7	F	S19	R122	F	R39	R6	F	R69	S18	F	S6		11
12 L		D	F	F	S14	R29	R129	R38	R135	S4	F	F	D		12
13 M			D	S21	R141	R90	R95	R94	R104	R57	S8	D			13
14 N					D	F	R128	R58	F	D					14

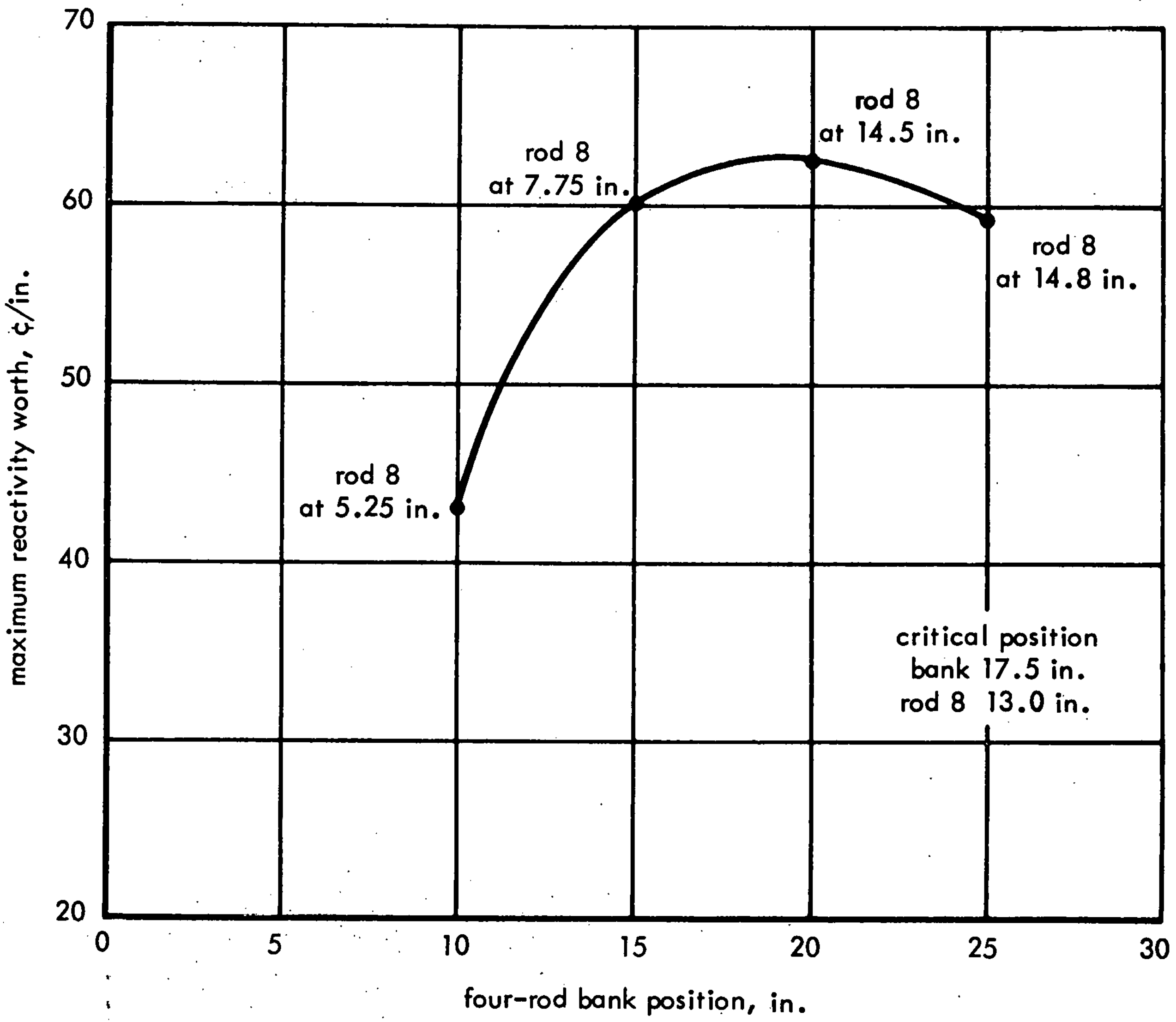
CORE LAYOUT, RECOMMENDED LOADING PATTERN

FIG. 4.6



REG. ROD POSITION VS: BURNUP

FIG. 4.7



MAXIMUM REACTIVITY WORTH OF ROD BELOW BANK
VS. BANK POSITION

FIG. 4.8

5. PLANT ENERGY TRANSFER SYSTEMS

5.1 SYSTEM PERFORMANCE

Operating data were obtained during the year for heat balance and equipment performance calculations. Table 5-1 compares the data of March 1966 with data taken during the month of March in the two previous years. The overall plant behavior continues to be excellent, and no areas of real concern have been observed. The overall plant average power is given in Table 5-2. Table 5-3 compares the equipment heat duties with the design values for the three-year period. Table 5-4 compares the overall heat transfer coefficients for the evaporators and subcoolers with those calculated for previous years. These comparisons show essentially no change in the equipment behavior with the possible exception that the heat load carried by evaporator #1 is lower than that for evaporator #2.

Leaking tubes were encountered in evaporator #1 in April 1964, in July 1965, and in April 1966. Three leaking tubes were also found in evaporator #2 in April 1966. The locations of the affected tubes are shown diagramatically in Fig. 5.1; the leaking tubes are identified as circles.

The superheater performance is compared with that of previous years in Table 5-5. The temperature distribution throughout the unit (see Fig. 5.2) continues essentially unchanged from the established norm or design figures. The absence of hot spots confirms the presence of clean heat transfer surfaces.

5.2 SYSTEM ANALYSIS

5.2.1 Deviations Between Primary and Secondary System Heat Loads

The heat balances are not as close this year as in 1965, but are considered to be within station instrument accuracy. The heat load carried by evaporator #1 is significantly lower than that for evaporator #2; a satisfactory explanation for this behavior has not been developed from the data on hand. It is noted that the primary steam pressure is 11 psi higher than in March 1965, and that the overall heat transfer coefficient in evaporator #1 is somewhat lower than in March 1965. The first thought is that this might be attributed to either a fouling of heat transfer surface or a buildup of noncondensible gases in the evaporator.

The fouling of heat transfer surface does not seem to be significant because the overall heat transfer coefficient is still high and, in fact, is higher than was calculated in March 1964. The reduction cannot be attributed to tubes which were blanked off because the attendant reduction in heat transfer surface is very small.

TABLE 5-1

PRIMARY AND SECONDARY SYSTEM DATA COMPARISON WITH PREVIOUS YEARS

Primary System	March 1964	March 1965	March 8, 1966	Design
Primary System				
reactor steam pressure, psia	937	940	951	935
reactor steam temperature, °F	536	537	539	534
pri. cond. flowrate, loop #1, lb/hr	121,162	123,722	118,135	129,000
pri. cond. flowrate, loop #2, lb/hr	120,840	124,536	129,128	129,000
pri. cond. temperature, exit evap. #1, °F	520	533	532	534
pri. cond. temperature, exit evap. #2, °F	521	528	525	534
pri. cond. temperature, exit subc. #1, °F	434	435	437	450
pri. cond. temperature, exit subc. #2, °F	431	432	432	450
Secondary System				
steam quality, %	99.7	99.8	99.73	99.75
steam pressure, evap. #1, psia	---	737	740	715
steam pressure, evap. #2, psia	---	740	740	715
steam pressure, inlet suphtr., psia	710	706	690	698
steam flowrate, evap. #1, lb/hr	114,800	112,200	108,404	112,500
steam flowrate, evap. #2, lb/hr	113,800	109,225	117,311	112,500
feedwater flow, evap. #1, lb/hr	111,300	110,334	106,821	---
feedwater flow, evap. #2, lb/hr	111,400	108,127	112,137	---
steam pressure, exit suphtr., psia	617	620	634	620
steam temperature, exit suphtr., °F	824	827	824	830
steam flowrate, exit suphtr., lb/hr	229,833	224,050	223,530	225,000
feedwater return temperature, °F	363	354	359	350
feedwater temp. exit subcooler #1, °F	473	471	480	464
feedwater temp. exit subcooler #2, °F	471	470	470	464
sec. system flowrate, average, lb/hr	229,211	221,312	223,940	225,000
coal to superheater, lb/hr	4654	4705	4688	4573
heating value of coal, Btu/lb	12,400	12,173	11,984	12,730

TABLE 5-2

OVERALL PLANT AVERAGE POWER

	March 1964	March 1965	March 1966	Design
reactor plant heat rate, Btu/hr	193.9×10^6	194×10^6	193.1×10^6	197×10^6
reactor power, Mwt	56.7	56.8	56.5	58.2
superheater heat rate, Btu/hr	49.75×10^6	48.5×10^6	49.4×10^6	50.64×10^6
superheater power, Mwt	14.6	14.3	14.5	14.8
total plant power, Mwt	71.3	71.1	71.0	73.0
total plant power, Mwe	23.8	23.0	23.4	22.0
efficiency, %	33.4	32.3	33.0	30.2
plant heat rate, Btu/kwhr	10,350	10,530	10.360	11,250

TABLE 5-3

EQUIPMENT HEAT DUTIES, Btu/hr, COMPARISON WITH PREVIOUS YEARS

	March 1964	March 1965	March 8, 1966	Design
evaporator #1[a]	82.6×10^6	82.5×10^6	78.8×10^6	
evaporator #1[b]	85.5×10^6	83.7×10^6	80.2×10^6	85×10^6
deviation from average, %	± 1.71	± 0.72	± 0.88	
evaporator #2[a]	82.0×10^6	83.9×10^6	87.2×10^6	
evaporator #2[b]	85.0×10^6	81.8×10^6	85.0×10^6	85×10^6
deviation from average, %	± 1.84	± 1.33	± 1.28	
subcooler #1[a]	12.5×10^6	14.2×10^6	13.1×10^6	
subcooler #1[b]	13.9×10^6	14.4×10^6	14.2×10^6	13.8×10^6
deviation from average, %	± 5.3	± 0.7	± 4.03	
subcooler #2[a]	12.8×10^6	13.85×10^6	14.0×10^6	
subcooler #2[b]	13.5×10^6	13.87×10^6	13.6×10^6	13.8×10^6
deviation from average, %	± 2.8	± 0.7	± 2.9	
Purification System				
regenerative cooler, Btu/hr	1.11×10^6	1.2×10^6	1.19×10^6	1.28×10^6
purification cooler, Btu/hr	0.8×10^6	0.7×10^6	0.74×10^6	1.45×10^6
Shield Cooling System				
shield cooler, Btu/hr	0.14×10^6	0.16×10^6	0.21×10^6	0.35×10^6

(a)/ Based on primary system data
(b)/ Based on secondary system data

TABLE 5-4

OVERALL HEAT TRANSFER COEFFICIENTS COMPARISON WITH PREVIOUS YEARS

	U - Btu/hr (ft^2) ($^{\circ}F$)			
	March 1964	March 1965	March 1966	Design
evaporator #1	542	624	580	556
evaporator #2	539	633	627	556
subcooler #1	612	543	575	445
subcooler #2	609	553	590	445

TABLE 5-5

SUPERHEATER PERFORMANCE

	March 1965	March 8, 1966	Design
steam inlet pressure, psig	692	677	684
steam inlet temperature, $^{\circ}F$	504	502	503
steam flowrate, lb/hr, average	221,312	223,940	225,000
steam quality, inlet, %	99.78	99.73	99.75
steam outlet pressure, psig	620	620	620
steam outlet temperature, $^{\circ}F$	827	826	830
oxygen in flue gases, %	5.5	5.9	6.0
coal fires, lb/hr	4705	4688	4573
coal heating value, Btu/lb	12,173	11,984	12,730
heat output, Btu/hr x 10^{-6}	48.5	49.4	50.6
heat input, Btu/hr x 10^{-6}	57.2	56.3	58.2
efficiency, %	84.8	87.7	87.0

The buildup of noncondensibles is a possibility which is refuted only by the fact that the primary condensate temperatures at the exits of evaporators #1 and #2 indicate more subcooling in evaporator #2 than in evaporator #1. If we postulate that there are more noncondensibles in evaporator #1, then one would expect more subcooling in evaporator #1 and this is not indicated by the data. A further postulation could be that the noncondensibles are carried under intermittently by the condensate. In this case, the temperature of the condensate leaving the evaporators would vary. That is, for at least part of the time the condensate leaving evaporator #1 would be at a lower temperature than that leaving evaporator #2. The data obtained were not complete enough to allow an analysis of this possibility. As a result, the recommendation was made that the temperatures of the condensate leaving the evaporators be recorded and monitored as an indicator of performance. One should also observe whether or not these temperatures change when the off-gas system is put into operation. A reduction in primary system pressure after venting is also a significant indication of noncondensibles. In any event, the performance of the evaporators should be followed to see if a pattern is developing which would warrant a more detailed examination.

5.2.2 Evaporator Tube Failures

The discovery of leaking tubes in the evaporators prompted an analysis of the restrictions imposed on power operation by sealing off the leaking tubes. To date, the effect of sealing off the tubes has not been detectable since a maximum of only about one percent of the tubes has been affected. The limiting number of tube failures was determined to be 322 U-tubes per evaporator for full power operation. Full power can be maintained by compensating for the loss in heat transfer surface by increasing the primary steam pressure. This increases the primary side temperature in the evaporators resulting in a greater temperature difference and, therefore, a greater driving force to transfer heat to the secondary side. The limit for increase of primary steam pressure is determined by the design pressure of the reactor vessel. For operation above the initial 58.2 Mwt power, the number of allowable tube outages decreases correspondingly. The relationships between primary pressure and U-tube requirements are given in Tables 5-6 and 5-7 and in Fig. 5.3.

The evaporator design was based on the use of demineralized water; and fouling was not considered to be a problem. However, since the evaporator shell-side water solids are concentrated by evaporation, fouling of tubes is a possibility. The effect of fouling was considered and calculations have shown that fouling of tubes would have about the same effect on the reduction of heat transfer as would the loss of tubes. The only difference would be the rate at which each deficiency occurs. A deficiency due to tube fouling can be corrected by appropriately cleaning the evaporator shell side. Up to the present, tube fouling has been negligible since there has been no apparent decrease in heat transfer coefficients.

2

TABLE 5-6

HEAT TRANSFER SURFACE - PRESSURE RELATIONSHIP
58.2 Mwt OPERATION

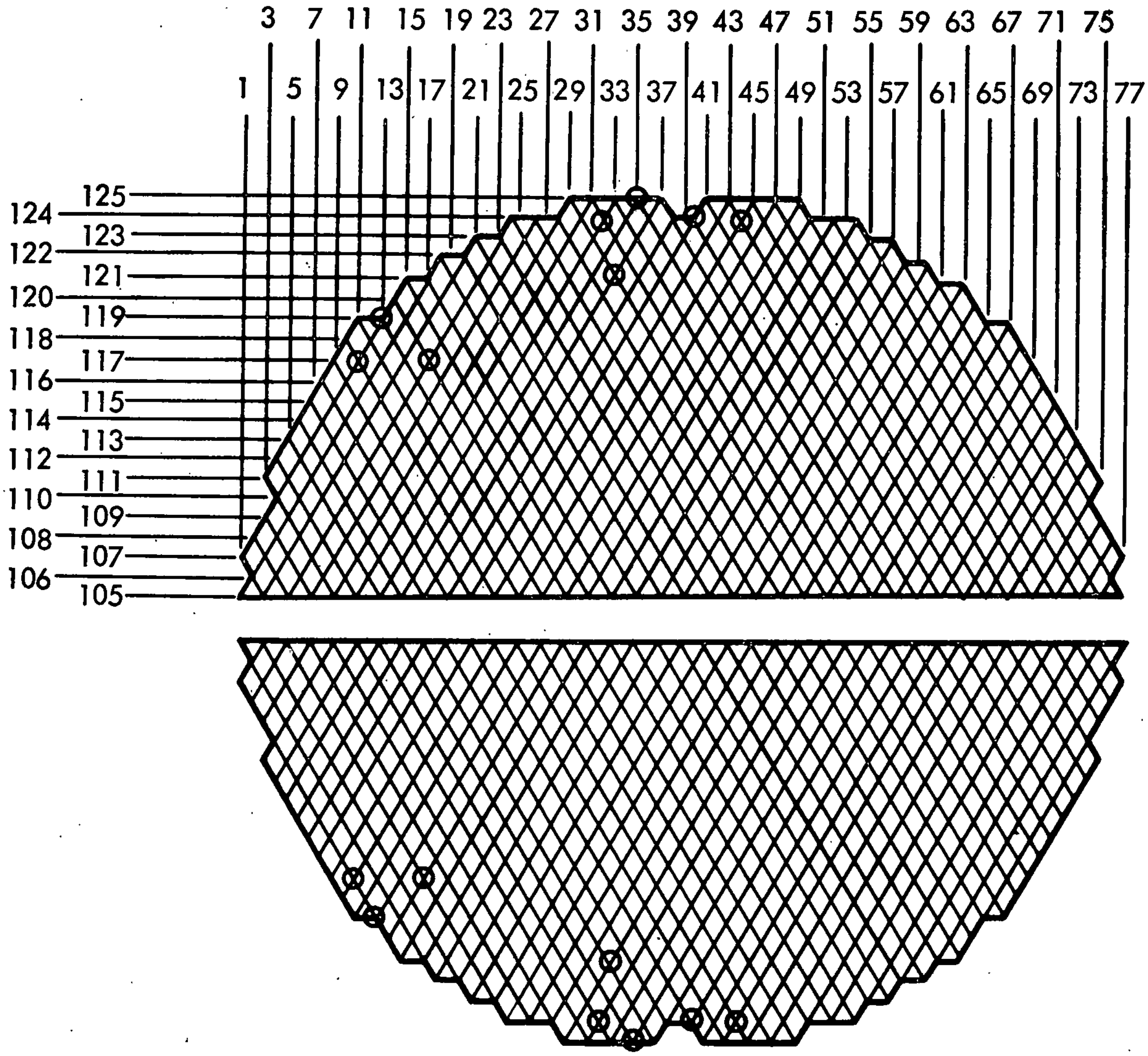
No. Tubes	Heat Transfer Surface, ft^2	Overall U	Δt	Primary Temp., °F	Primary Press., psig
642	4710	556	32.5	538	936
600	4402	556	34.8	540	948
500	3668	556	41.7	547	1008
400	2934	556	52.1	558	1102
300	2201	556	69.5	575	1263

TABLE 5-7

HEAT TRANSFER SURFACE - PRESSURE RELATIONSHIP
72.75 Mwt OPERATION

No. Tubes	Heat Transfer Surface, ft^2	Overall U	Δt	Primary Temp., °F	Primary Press., psig
642	4710	556	40.6	546	1029
600	4402	556	45.9	551	1045
500	3668	556	56.6	562	1140
400	2934	556	68.8	574	1257

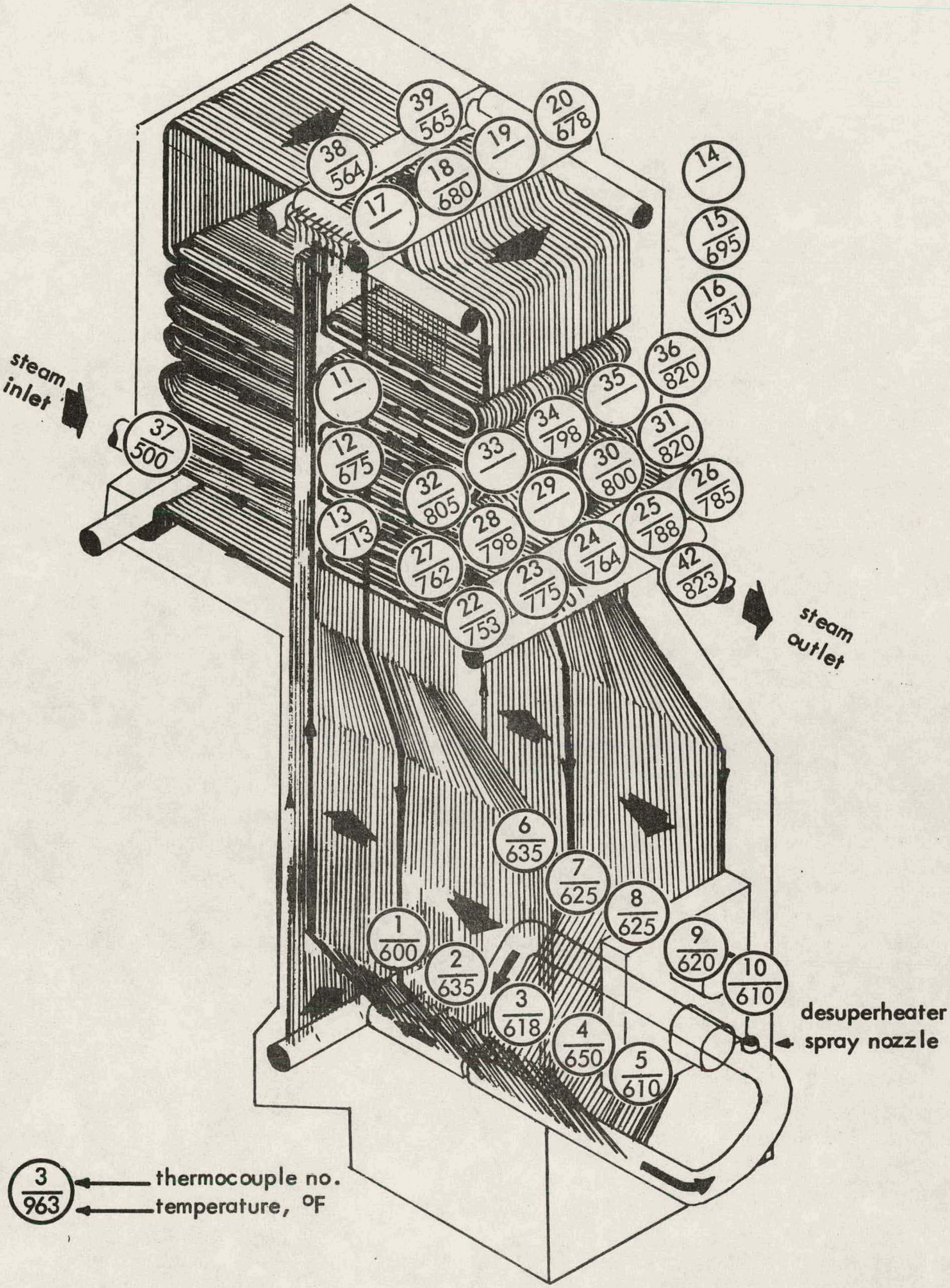
The cause of the tube failures has not been determined. Corrosion specimens inserted in the water boxes of the evaporators (see Task 615) have shown evidence of stress corrosion cracking in the stressed specimens. The sensitized specimens have shown evidence of intergranular attack. Although the tube failures may be related to these failure mechanisms, they may also be related to the environment on the secondary side of the tubes or to a statistical probability that a certain small percentage of deficient tubes will result from the fabrication of the equipment. This matter will undoubtedly be the subject of a continuing evaluation.



date plugged	evap	tube location	approximate leak location
4-19-64	#1	119, 13; 124, 32	top section, 60 in. from tube sheet
7-29-65	#1	117, 17; 121, 33 125, 35	mid-way on tube bend top section, 37 in. from tube sheet
5-25-66	#2	117, 11; 124, 40 124, 44	U-bend section of tube
6-9-66	#1	124, 40; 124, 44	U-bend portion of tube

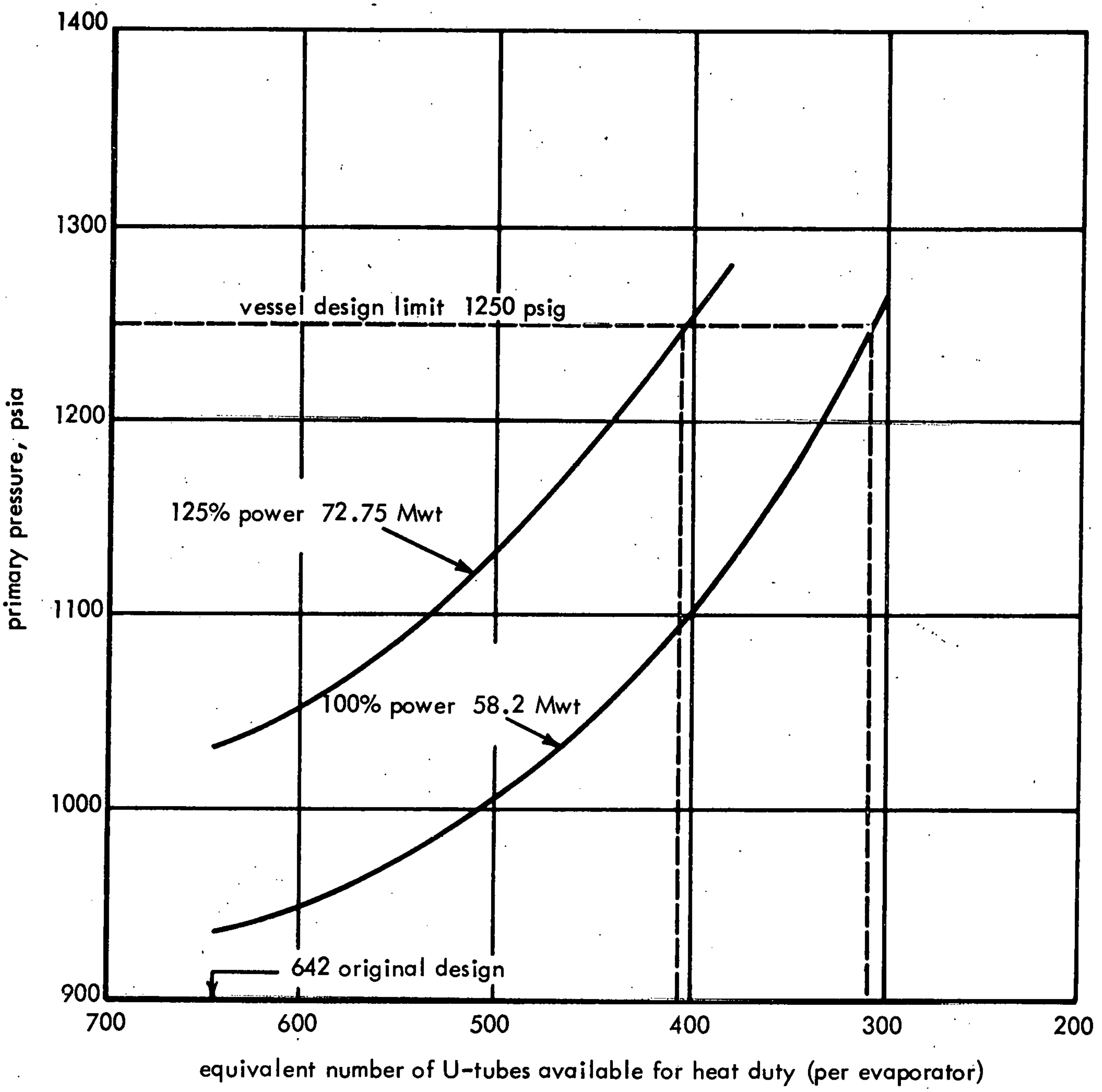
LOCATIONS OF EVAPORATOR TUBE LEAKS

FIG. 5.1



ERR SUPERHEATER

FIG. 5.2



REACTOR OPERATING PRESSURE REQUIREMENTS VS. TUBE OUTAGE

FIG. 5.3

6. PRIMARY AUXILIARY SYSTEMS

6.1 ERR PRIMARY WATER MAKEUP (TASK 401)

When the power of the reactor is raised from 70 percent to 100 percent of full power, water is dumped from the primary system to avoid having too high a water level in the reactor. In making such a power change, approximately 250 gal of water are removed from the reactor to the collecting tank. Since the collecting tank volume is only 50 gal, primary water must be discarded to waste via the 3000 gal retention tanks rather than being returned to the reactor when the power level is reduced. During the report period, a study was performed (see Ref. 5) to determine what savings, if any, would accrue to the Commission by the addition of more capacity for collecting primary system condensate at the Elk River Reactor Plant.

6.1.1 Mode of Operation

In the present mode of operation, water is dumped from the reactor when the power level is being increased from 16 Mwe. The rate of dumping is governed by the rate of power change. If the change is gradual, the dump is made 2 in. (50 gal) at a time. Water is drained from the purification system to the collecting tank manually by an operator. The control room operator discharges the collecting tank contents by activating a remotely operated valve from the control room. The general practice is to allow the level to build up from 25 in. to 29 in. before dumping starts. Dumps are made in the 29-in. to 22-in. range on the reactor level indicators. The level alarm points are set as shown in Table 6-1.

TABLE 6-1

REACTOR LEVEL ALARM POINTS

34 in.	- high level scram
29 in.	- high level alarm
25 in.	- normal operating level
22 in.	- low level alarm
12 in.	- low level scram

It was previously the practice to dump water when the reactor power level was at approximately 11 Mwe so that the reactor water level was just above the previous low level alarm point (i.e., about 20 in.). As noted in the July 1964 Monthly Operational Report, the water level increased to 29 in. at full power and no reactor water drainage was required during escalation. The study did not include an investigation of the various methods of manipulating the reactor water level; however, the mode of operation does influence the annual water usage and the annual cost of water dumped from the primary system.

6.1.2 Water Usage and Recovery Methods

The method of approach in this study was first to determine the average amount of water dumped (per month) to the retention tanks, which could be attributed to changes in power level of the reactor. Three different cases, or modes of operation, were considered:

Case 1 represents the lowest water usage situation wherein power swings can be made from 50 to 100 percent of full power without dumping or adding water. The estimated water usage is 2550 gal/month.

Case 2 represents a slightly higher water usage situation wherein the water is dumped as per the present practice, but estimates are based on spring and fall load swings with no abnormal number of shutdowns or power level changes. The estimated water usage is 3570 gal/month.

Case 3 represents the water discarded at the ERR for the period July 1964 through August 1965. This usage (7900 gal/month) was estimated by tabulating the total number of gallons of water released to the river as recorded in the monthly operating reports. An average of 65 gal/day was deducted for primary system leaks, and 26,740 gal were deducted from the total to represent water which was probably dumped from the reactor cavity and fuel element storage well following the control rod inspection.

The next step in the study was to determine the methods which could be used to recover and reuse the water dumped during power escalation. Two methods were considered:

Scheme 1 involves piping the water from the purification system discharge line to the overhead water storage tank via the fuel element storage well overhead line.

This method entails the installation of piping, valves, and a limiting orifice between the purification regenerative heat exchanger and the fuel element storage well discharge line. A schematic diagram is shown in Fig. 6.1. Water would be drained from the reactor system at the regenerative heat exchanger (following the demineralizers) and discharged to the overhead water storage tank through the fuel element storage well discharge line. Reuse of the water is effected by return into the system via the existing collecting tank and makeup pumps. Removal of coolant after demineralization provides the necessary water cleanup and should prevent radioactive contamination of the stored water as long as the reactor water quality is good. If fuel element leaks should develop, one might have to revert to the practice of dumping the water.

The estimated capital cost of this scheme is $3348.

Scheme 2 involves piping the water from the purification discharge line to a new auxiliary 300-gal storage tank. Reuse of the water would be effected via the present makeup system.

This second scheme duplicates much of the equipment of the first and adds a stainless-steel tank, additional valves, and controls. A schematic diagram is shown in Fig. 6.2. The system constitutes the same type of flow by pressure as Scheme 1, but discharges to an elevated 300-gal makeup tank. The stored water could be released to the collecting tank gradually, as makeup is required. Instrumentation, such as level indication, level alarm, and dump valve operation, would provide the necessary control from the control room panel.

The estimated capital cost of this scheme is $6544.

6.1.3 Recovery of Capital Equipment Costs

The final step in this study was to determine the number of years required to recover the capital equipment costs out of savings in water usage. Some of the important assumptions which were made in the cost study were: (1) high quality demineralized water costs $4.70 per 1000 gal; (2) analytical costs involved in dumping the retention tanks are $25 per 3000 gal; (3) equipment costs are not capitalized; and (4) no attempt is made to apply present worth techniques to the annual savings.

The number of years required to recover the capital equipment cost out of savings in water usage was determined by dividing the total job cost by the yearly savings. The results are shown in Table 6-2.

TABLE 6-2

RECOVERY OF CAPITAL EQUIPMENT COSTS

case	yearly savings, $	years to recover capital equipment costs	
		Scheme 1	Scheme 2
1	93.84	36	69
2	253.37	13	26
3	930.58	3.6	7

A cursory examination of the water dumping practices seemed to indicate that some savings could be made by a judicious manipulation of the reactor water level. In view of the rather marginal economic advantages of a further investment in the plant, it was recommended that an attempt be made to reduce the amount of water dumped by operational controls.

6.2 ERR THIMBLE COOLING SYSTEM (TASK 404)

6.2.1 System Performance

An increase in water flow to the control rod drive seals was observed during the report period. The increase was greatest in the No. 4 drive seal; and, in fact, this increase was large enough to warrant a study (see Ref. 6) to determine whether modifications should be made to the control rod thimble cooling system to improve the operational control of seal flows.

The control rod drive and water seal assemblies are shown in Fig. 6.3. The water seal is composed of two parts: (1) a single unit seal ring and diaphragm on the reactor side; and (2) a labyrinth or "pressure breakdown" seal containing 10 seal rings and diaphragms on the drive side. Seal water is injected between the two seal compartments at 25 to 30 psi above the reactor system pressure. Under design conditions, approximately one-half of the seal water was to have passed up into the reactor, cooling the thimble and rack; the other half to pass through the segmented "pressure letdown" seal for discharge at atmospheric conditions. Seal leakage of the water is limited when the labyrinth causes the fluid to lose velocity pressure as it is throttled through the radial width of the annular orifices in the seal. Theoretically, the seal rings float on a thin film of fluid and should never touch the shaft, thereby exposing no wear points in the seal. In the case of the No. 4 seal, it is possible that the clearance increased because of metal wear or corrosion.

The control rod seal flowrates for ten successive dates from 1962 into 1965 are shown in Table 6-3. In general, the flowrates have increased gradually with time and usage. The greatest increase appears to have occurred in the No. 4 seal following the March to July, 1964 shutdown. The flow data listed in Table 6-3 are too variant, even for the individual seals, to permit reaching a logical conclusion as to the cause of the increased leakage.

Three control rod drives (Nos. 3, 6, and 12) were completely dismantled and inspected during the November 1964 shutdown (see Ref. 7). The seal shafts showed circumferential lines on the chrome plating which were diagnosed as being associated with rubbing of the seal rings against the shaft. There was no evidence of corrosion. The chrome plating was removed from one of the shafts and the shaft was dye-penetrant inspected. The conclusion was that cracks were nonexistent and that surface score marks were not detrimental to the shaft. On the above basis, the excess seal leakage for the No. 4 seal may be attributed to seal-to-shaft wear.

6.2.2 System Analysis

Labyrinth seals limit leakages by closely controlling the annular clearances between the rotating shaft and the stationary housing. Manufacturing tolerances are specified so that these clearances are made as small as possible consistent with the operating requirements. As seal wear occurs, leakages increase gradually and the limit for leakage is governed either by tolerable limits of radioactivity released to the atmosphere or by plant operating efficiency. The flowmeter ranges were low for practical operating conditions, having been based on seal manufacturing tolerances rather than on a projected change in flow with operating time.

TABLE 6-3

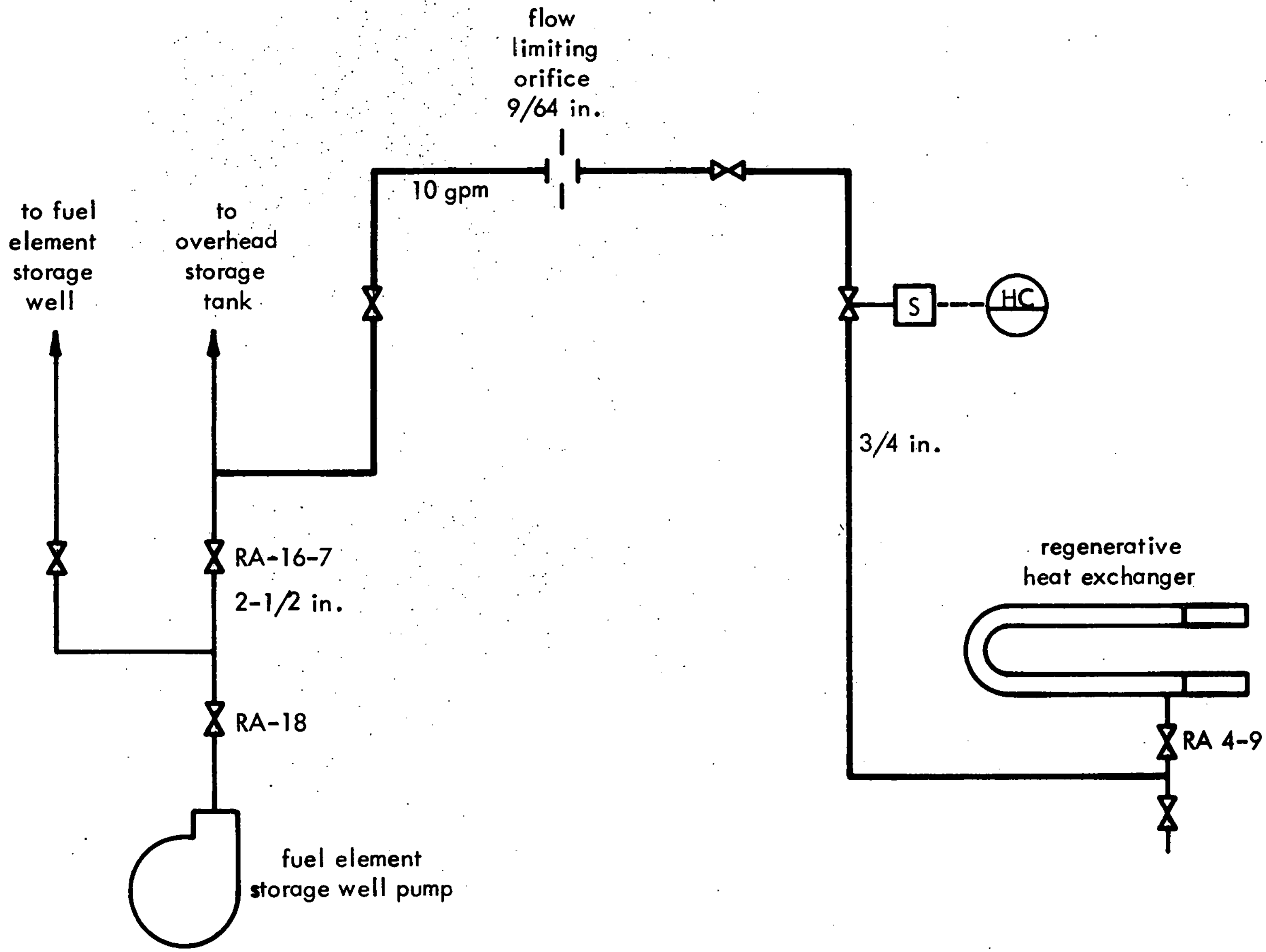
CONTROL ROD SEAL FLOWRATES, 1962 - 1965

	preop. test 1/25/62 rods seated	preop test 1/25/62 rods raised 7 in.	7/2/63 60% power	2/5/64 100% power	2/19/64 100% power	3/3/64 100% power	3/19/64 100% power	8/6/64 50% power	1/18/65 100% power	10/30/65 100% power
total flow L-14*	151	740	>700	465	>700	>700	>700	>700	>700	>700
seal flow*										
1	16	16	30	28	26	24	33	33	26	36
2	10	5	13	0	0	15	21	23	21	24
3	0	10	8	12	11	5	12	11	16	17
4	7	92	82	30	45	10	45	>110	>110	762
5	10	10	15	0	0	18	16	16	21	25
6	10	8	16	13	16	18	21	21	33	30
7	0	0	5	6	6	0	8	7	8	7
8	16	14	26	0	0	23	21	30	30	30
9	10	10	16	17	17	21	21	19	26	24
10	0	3	11	12	10	glass broken	glass broken	5	8	6
11	0	0	3	2	2	2	0	43	8	13
12	12	14	5	0	0	0	12	17	8	10
13	14	16	28	25	31	29	26	13	16	21

*flows in cc/min.

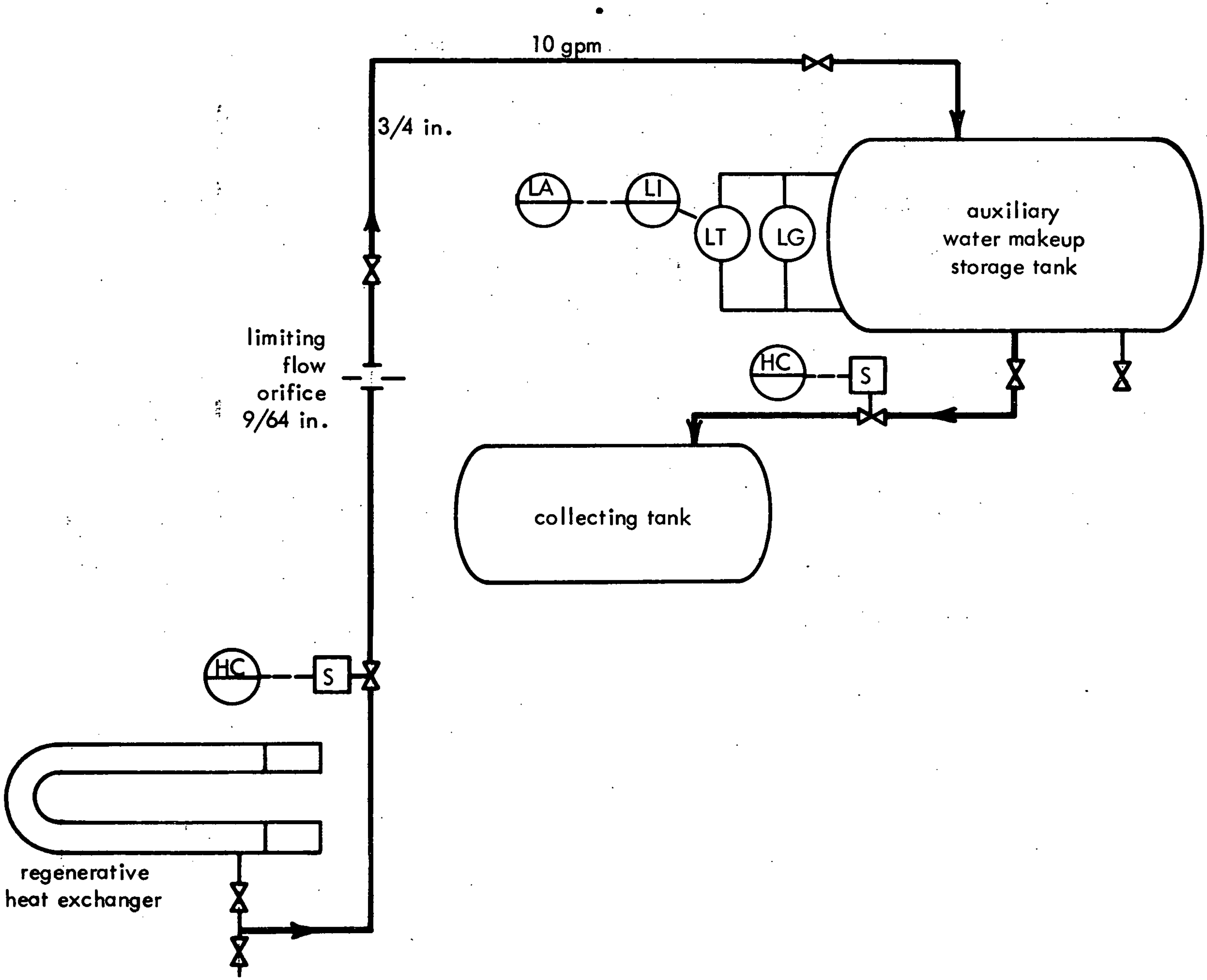
Concern was expressed about the possibility that excessive water flow in one of the thimbles would overcool the thimble and create a stressed area at the junction between the thimble and the bottom reactor head. Seal water flow into the reactor is probably greatest when outleakage from the high pressure seal is lowest. As the seals wear, the seal outleakage will increase while the reactor inleakage probably decreases. This deduction is based on the fact that the restriction to flow into the reactor is a single seal ring and diaphragm segment with a low pressure differential across the seal, and the restriction to flow to the collecting tank is a labyrinth seal constructed as a ten-segment seal ring diaphragm unit having a high pressure differential across the seal. The maximum allowable flow of water into the reactor via the individual thimbles was not determined as a part of this study. However, there is substantial evidence that flows on the order of 1000 cc/min do not present any stress problems not previously analyzed.

The study concluded that to achieve the maximum amount of operational control the thimble cooling circuit should be modified to conform to the diagram shown in Fig. 6.4. Elk River Reactor Topical Report, ACNP-65619 (Ref. 6) includes a detailed analysis of the flowmeter ranges, control valve requirements, preventive maintenance suggestions, and a cost estimate for the proposed modifications.



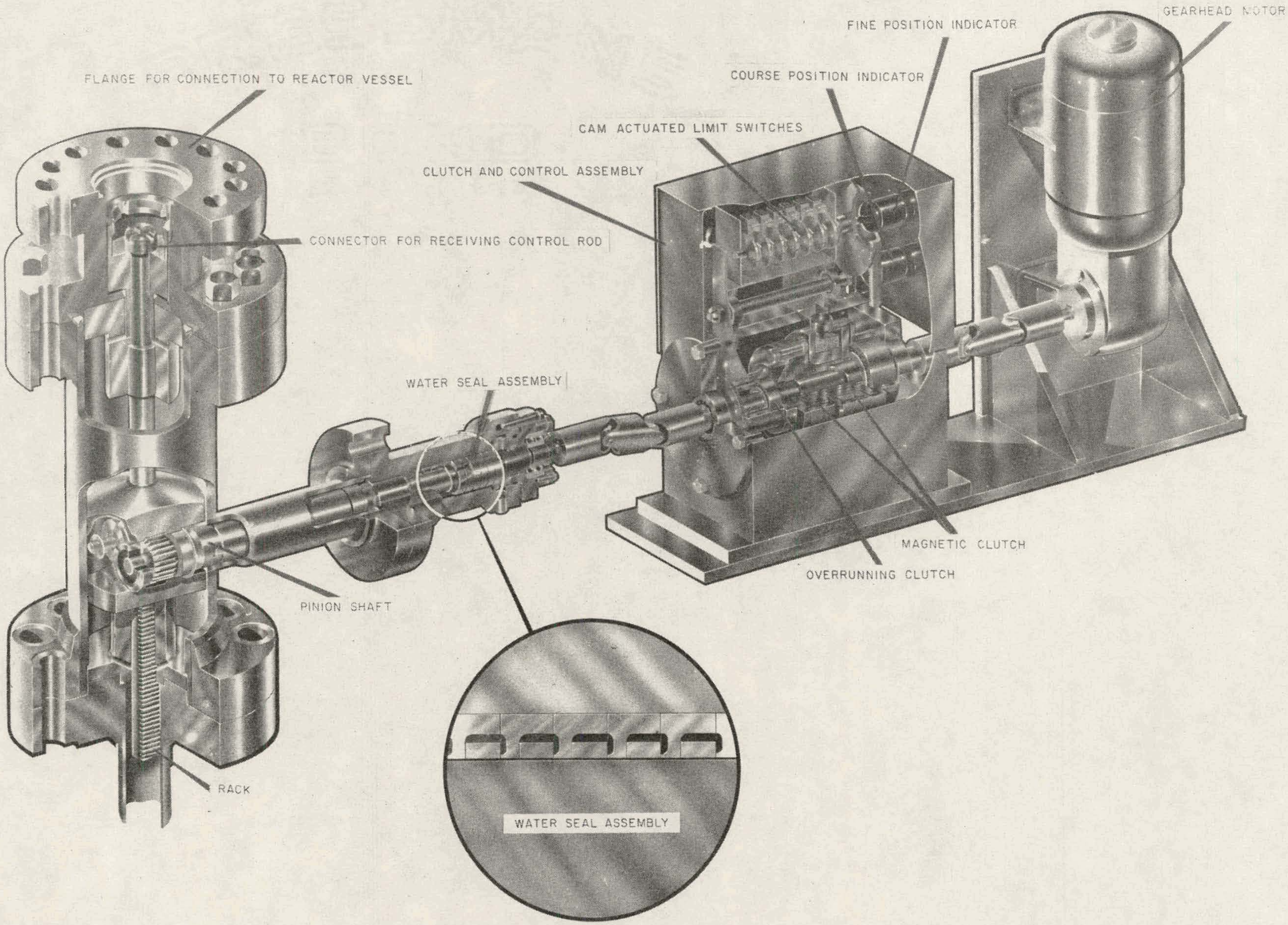
ERR RECOMMENDATION FOR EXCESS PRIMARY WATER DUMP
DURING POWER ESCALATION, SCHEME 1

FIG. 6.1



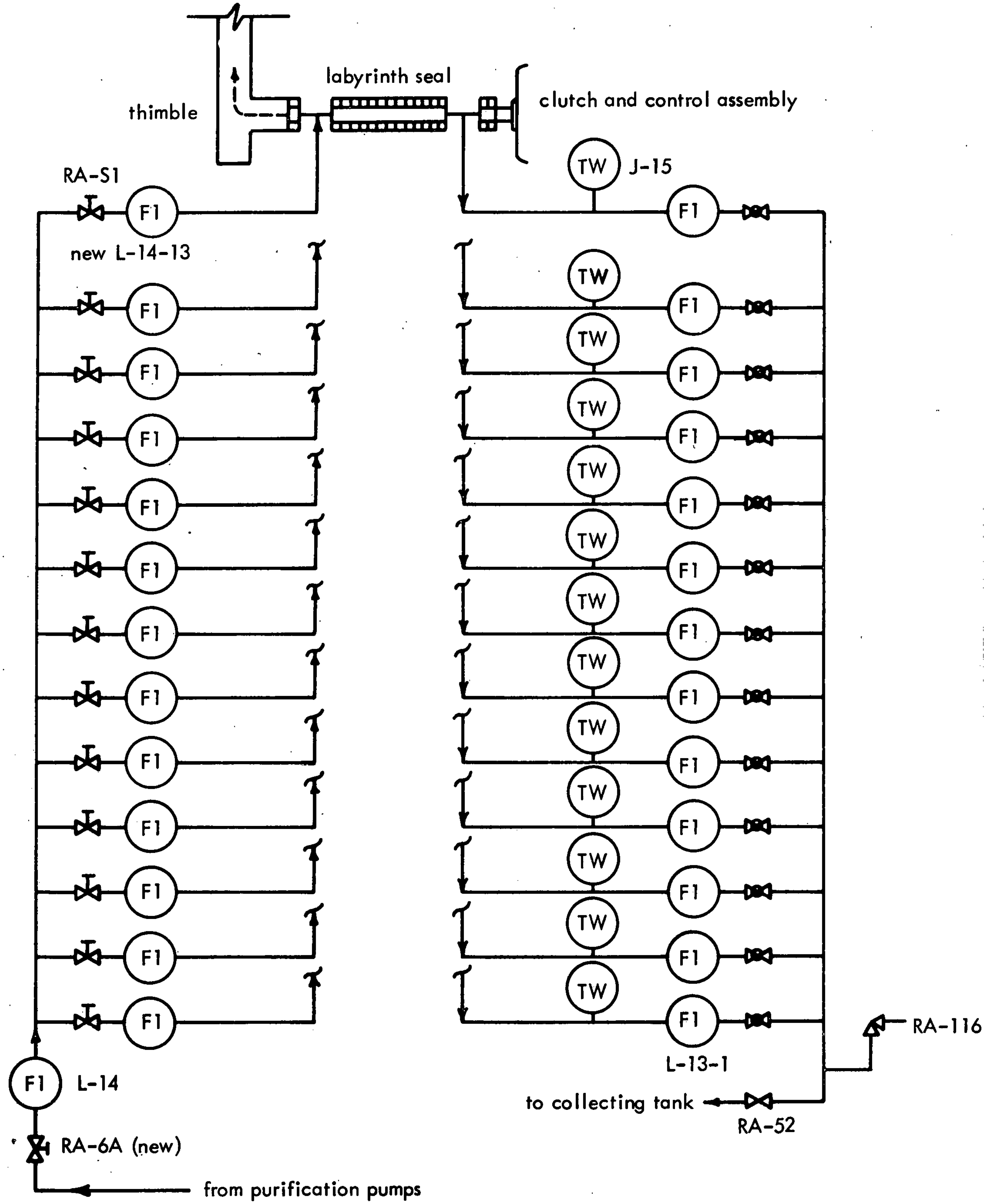
ERR RECOMMENDATION FOR EXCESS PRIMARY WATER DUMP
DURING POWER ESCALATION, SCHEME 2

FIG. 6.2



CONTROL ROD DRIVE, ELK RIVER REACTOR

FIG. 6.3



ERR THIMBLE COOLING SYSTEM (PROPOSED MODIFICATION)

FIG. 6.4

7. MISCELLANEOUS EVALUATIONS

7.1 CORROSION SAMPLES AND TESTS (TASK 615)

7.1.1 Background

In April 1964, the ERR-OAP project recommended that corrosion specimens be inserted into the evaporator water boxes of the Elk River reactor. This recommendation was made based on the results of the analysis of radiolytic gas in the primary steam during operation of the reactor at full power in the nonvented condition. The Chicago Operations Office, USAEC, concurred in the advisability of this task, and specimens were subsequently prepared and installed early in June 1964. The test as designed, is qualitative in nature and was intended to determine whether the conditions in the water boxes are conducive to abnormally high corrosion rates. In addition, stressed "U-bend" samples were inserted to determine whether conditions exist which are conducive to stress corrosion cracking. Examination of the specimens after exposure showed that while the corrosion rates are not abnormal there is evidence that conditions exist which are conducive to stress corrosion cracking.

7.1.2 Inspection History

The initial installation of specimens is described in detail in Ref. 8, which is the first report on the subject. The corrosion specimens were removed from the lower water box of evaporator #1 and visually examined in October 1964. The results of this examination are reported in Ref. 9. Since there was no evidence of deleterious effects, the test specimens were reinserted on the same day for further exposure. The next removal for inspection occurred on August 1, 1965. At that time, evaporator #1 was opened and the specimens were removed for examination. Two coupons of each type (as rolled, annealed, and sensitized) were taken from each fixture. The "U-bend" sample was removed from the lower water box fixture. The remaining specimens were reinstalled for further exposure. The results of a laboratory examination of the removed specimens are given in Ref. 10. These results gave the first indication that conditions might exist which are conducive to stress corrosion cracking. On April 26 and 27, 1966 all remaining specimens were removed for a laboratory examination. Reference 11 is a report which describes the results of the final examination of specimens originally installed. The fabrication history of replacement specimens is described in Appendix A of this report.

7.1.3 Summary of Results

After exposure, the individual coupons were weighed and then electrolytically descaled using an inhibited sulfuric acid procedure. The specimens were reweighed and the corrosion product film thickness and the amount of metal corrosion were determined. A correction was applied to the descaled specimen weight to account for metal removed during the descaling operation. The results of the corrosion tests are summarized in

Table 7-1. In this summary the corrosion rates in mils/year are calculated using the following equation:

$$\frac{(W_1 - W_3)\ (0.04724)}{(A)\ (M)\ (\rho)} = \text{corrosion rate, mils/year}$$

where W_1 = original weight of specimen, mg
W_3 = corrected descaled weight of specimen, mg
A = area of specimen, dm^2
M = months of exposure
ρ = density = 7.84 g/cc
0.04724 = conversion factor to mils/year

With the exception of the sensitized coupons, the corrosion was small. The sensitized specimens differed from the others in the extreme difficulty in removal of radioactivity by descaling. The reason for this difficulty was attributed to the intergranular attack observable in the metallographic sections shown in Fig. 7.1.

The "U-bend" samples were decontaminated (descaled) and examined metallographically for evidence of cracking. The surface of the "U-bend" specimen was ground down several thousandths of an inch at the highly stressed outer corner region. After polishing, several distinct cracks were visible under magnification. Etching revealed that the cracks were predominantly transgranular over most of their length, changing to intergranular near the apex. A typical crack is shown in Fig. 7.2. All of the stressed specimens exhibited similar cracks.

7.1.4 Environment

The corrosion coupons and the stressed specimens were exposed to near saturated steam conditions of 936 psig and 539 F in the water boxes of the evaporators. The amount of moisture present in the incoming steam to the evaporators is of the order of 1/2 percent. The water solids carryover in the steam does not exceed one part per billion. The outlet water boxes contain steam condensate with an atmosphere of steam and noncondensibles. Chloride concentrations in the reactor water have been maintained at nondetectable levels during the entire reactor operational period. The analytical technique used for the determination of chlorides is sensitive to about 35 parts per billion of chlorides. The pH of the water has been in the range of 6 to 8.

The steam leaving the reactor is sampled and analyzed routinely for oxygen and hydrogen (see Ref. 12). The results of these analyses are shown in Table 7-2. The oxygen and hydrogen content of the steam in the water box is not measured on a routine basis; however, data taken during recent recombiner tests (see Ref. 13) indicate that the concentrations are on the order of 1100 ppm oxygen and 130 ppm hydrogen. Some observers (Refs. 14 and 15) have postulated that higher oxygen concentrations (greater than 5000 ppm) may have occurred during the earlier days of reactor operation.

7.1.5 Conclusions

The main conclusion reached as a result of these qualitative tests was that the environment in the water boxes of the evaporators is conducive to stress corrosion cracking. As a result of this conclusion, an investigation of representative welds in the primary piping system was conducted prior to reactor startup. A total of 10 welds were selected for ultrasonic inspection. The inspection was completed on June 2, 1966 and no defects were detected. It is expected that routine ultrasound inspections will be performed at intervals during the future operation of the reactor.

TABLE 7-1

RESULTS OF CORROSION TESTS AFTER EXPOSURE

Evaporator #1 - 8 Months

sample	corrosion rate (mils/year) inlet water box	corrosion rate (mils/year) outlet water box
as rolled	0.006	0.006
annealed	0.014	0.007
sensitized	0.201	0.071

Evaporator #1 - 22 Months

sample	corrosion rate (mils/year) inlet water box	corrosion rate (mils/year) outlet water box
as rolled	0.013	0.003
annealed	0.004	0.004
sensitized	0.242	0.037

Evaporator #2 - 22 Months

sample	corrosion rate (mils/year) inlet water box	corrosion rate (mils/year) outlet water box
as rolled	0.002	----
annealed	0.003	0.004
sensitized	0.064	0.031

TABLE 7-2

RADIOLYTIC GAS CONCENTRATIONS IN REACTOR EFFLUENT STEAM

date		oxygen		hydrogen	
month	year	maximum	minimum	maximum	minimum
Dec.	1964	131	32	---	---
Jan.	1965	64.4	29.3	---	---
Feb.	1965	33.4	---	---	---
Mar.	1965	45.4	12.44	3.55	1.1
Apr.	1965	168	11.6	4.74	1.1
May	1965	24.8	16.4	2.0	1.5
June	1965	66.9	13.9	3.3	1.53
July	1965	17.56	15.9	1.71	1.4
Sept.	1965	29.34	29.34	1.96	1.96
Oct.	1965	27.6	11.2	4.2	1.4
Nov.	1965	4.14	4.11	2.22	1.29
Dec.	1965	5.86	2.03	2.18	1.15
Jan.	1966	8.05	2.02	1.69	1.4
Feb.	1966	7.5	6.03	1.52	1.4
Mar.	1966	6.05	1.63	1.57	0.76

7.2 CRITICALITY ANALYSIS OF THE ELK RIVER REACTOR SHIPPING CASK (TASK 617)

The objective of this task was to determine the weight percentage of cadmium which would have to be added to the aluminum used in fabricating the fuel basket of the ERR 28-element spent fuel shipping cask to ensure that the cask would meet the criticality requirements of 10 CFR 71.

7.2.1 Background

Knapp Mills, Inc. designed a 28-element fuel basket for the spent fuel shipping cask which is intended to be used for foreign and domestic shipments of spent fuel from the Elk River reactor. A drawing of the cask is shown in Fig. 7.3. The number of elements in the cask is large enough to present a criticality problem, if the usual conservative assumptions are made regarding the reactivity of the fuel to be shipped. In order to overcome this problem it was decided to add 1 percent by weight of cadmium to the aluminum used to fabricate the fuel basket. The details of the criticality analysis are included in Elk River Reactor Topical Report, ACNP-66551 (see Ref. 16). The results are summarized below.

7.2.2 Summary of Results

Criticality calculations were performed which considered the effect of various element types and of various amounts of cadmium added to the 28-element fuel basket on the multiplication factor of an infinite cask array. An infinite cask array was assumed in all calculations because the requirements of 10 CFR 71 state, in effect, that an unlimited number of packages (shipping casks) must be considered and water moderation must be assumed to exist wherever it would increase the reactivity of the package. It was determined that the addition of 1.0 wt. % cadmium to the aluminum used in fabricating the fuel basket is sufficient to reduce the multiplication factor to less than 0.80 with even the most conservative assumptions regarding the reactivity of the fuel. Figure 7.4 shows the effect of varying the amount of cadmium added to the aluminum.

The criticality calculations also evaluated the multiplication factor of an infinite array of fuel baskets (without the cask portion). In this case, the results showed that the addition of 1.0 wt. % cadmium is sufficient to reduce the multiplication factor to less than 0.85 with the same conservative assumptions regarding the reactivity of the fuel. Once again, Fig. 7.4 shows the effect of varying the amount of cadmium added to the aluminum.

It was concluded that 1.0 wt. % cadmium uniformly dispersed in the 28-element fuel basket is entirely sufficient to meet the criticality requirements of 10 CFR 71 for any combination of fuel elements now available and for any fuel presently envisioned for future use in the Elk River reactor core.

7.2.3 Method of Analysis

The Elk River reactor 28-element spent fuel shipping cask consists of the present ERR-PNPF outer cask, itself (see Ref. 17), including the biological shielding and its associated structure, and the inner fuel basket (see Ref. 18). The fuel basket is in the form of a cylindrical casting with 28 individually machined cavities for the fuel elements. This casting is composed of Type 6061 aluminum containing a uniform dispersion of cadmium.

The Elk River reactor fuel elements which will be available for future shipments are of the following types:

(1) 148 Core I regular elements: the unirradiated elements contain fuel pellets which consist of a mixture of thoria (ThO_2) and fully enriched urania (UO_2). The U-235 content is 4.3 wt. % (metal basis). The fuel cladding is Type 304 stainless steel containing 600 ppm natural boron.

(2) 22 Core I spiked elements: these are identical to the Core I regular elements except that the U-235 content is 5.2 wt. % (metal basis).

(3) 150 Core II elements: these are similar to the Core I elements except that the fuel cladding is Type 348 stainless steel containing no added boron; the pellets contain 4.4 wt. % U-235 (metal basis).

The following conservative assumptions were used in the calculations:

(1) All fuel elements were assumed to be unirradiated even though actual discharge exposures will be on the order of 8000 Mwd/MT.

(2) All fuel elements were assumed to have no boron in the cladding.

(3) In the final analysis, all fuel elements were assumed to contain 5.2 wt. % U-235 (metal basis) with no boron in the clad.

To meet the requirements of 10 CFR 71, all calculations were made for an infinite array of shipping casks; and water was assumed to be present in the fuel element cavities of the fuel basket and between the casks.

The calculational methods which were used in the criticality analysis of the shipping cask have been proved and tested against experiment in the models that have been used for the Elk River reactor. A short description of the nuclear and spatial models that were used is given below.

The criticality analysis was performed using three neutron energy groups. The energy ranges of these groups are given in Table 7-3.

TABLE 7-3

NEUTRON ENERGY GROUPS

group	energy range
1	10^7 ev to 5.53×10^3 ev
2	5.53×10^3 to 0.625 ev
3	0.625 ev to 0 ev

Slowing-down spectra (above 0.625 ev) were generated for the shipping cask materials by the GAM-1 code (see Ref. 19). The diffusion parameters for the two upper energy groups were then averaged over these spectra. The thermal spectra (below 0.625 ev) were calculated by the TEMPEST-II code (see Ref. 20) and the thermal diffusion parameters were averaged directly over these spectra except in the case of the fuel element regions. Since a fuel element assembly consisting of fuel, cladding and moderator materials is necessarily treated as a homogenized region in all two-dimensional diffusion calculations, the necessary parameters were developed in the following manner.

To account for the actual intracell flux distribution through the fuel, cladding and moderator, multi-energy thermal group parameters for the individual materials were used in one-dimensional multigroup transport calculations. The calculations yielded multigroup intracell flux distributions which, in combination with the multi-energy thermal parameters for each material, were used to derive space- and energy-averaged parameters for a homogenized fuel element region over a single thermal group.

The above calculations give the three-group diffusion parameters for the shipping cask materials which are necessary for the spatial calculations described below.

The spatial model that has been used in the criticality analysis of the 28-element shipping cask consists of a detailed two-dimensional mockup of the components of the cask using the PDQ-5 code (see Ref. 21). In this mockup, each fuel element, the aluminum fuel basket, the lead shield, etc., are separately represented as distinct regions. (Refer to Fig. 7.3.)

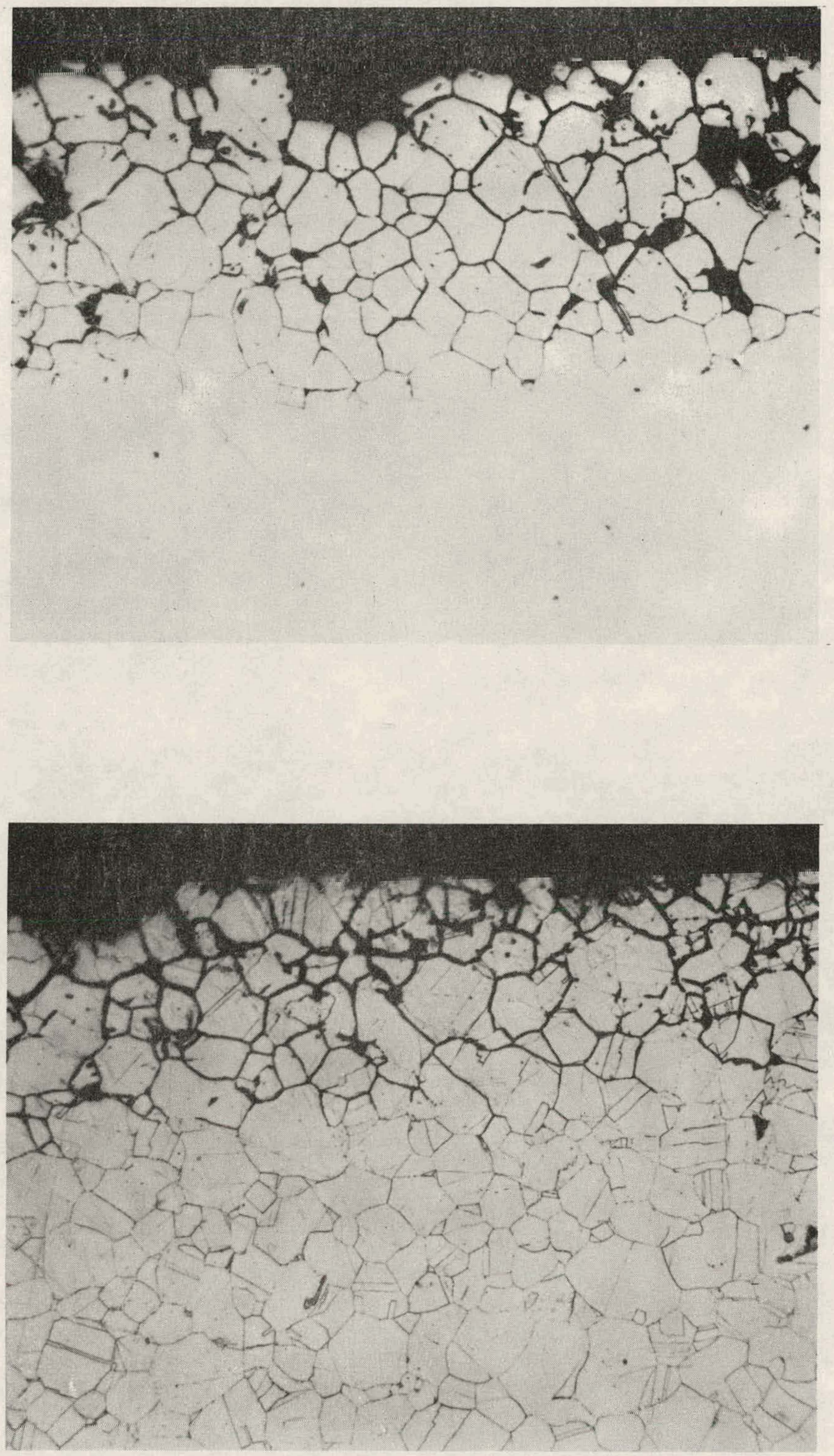
For all calculations which have involved symmetric groupings of similar fuel elements (e.g., 28 Core II elements), it has been possible to use a quarter-cask mockup. This mockup is represented as a 68 x 54 PDQ-5 mesh. For calculations involving mixtures of different fuel types, a half-cask mockup has been used; this mockup is represented by a 43 x 75 mesh.

As previously stated, an infinite array of shipping casks has been considered in order to comply with 10 CFR 71. It has been assumed in the calculations that the shipping casks are arranged in a rectangular array and are touching at the outer points of their respective shields.

The presence of cooling fins and other outer structures which would tend to increase the separation are neglected, since these could conceivably be damaged by some hypothetical accident. Water was assumed to be present in both the fuel element cavities and in the open spaces between the shipping casks.

Using the nuclear and spatial models outlined above, criticality calculations were performed as a function of the amount of cadmium in the fuel basket. The cadmium was assumed to be uniformly dispersed throughout the aluminum fuel basket. The effective multiplication factor for the cask array was calculated both with and without the assumption of axial leakage. In addition, the infinite multiplication factor was calculated for the fuel basket assembly alone.

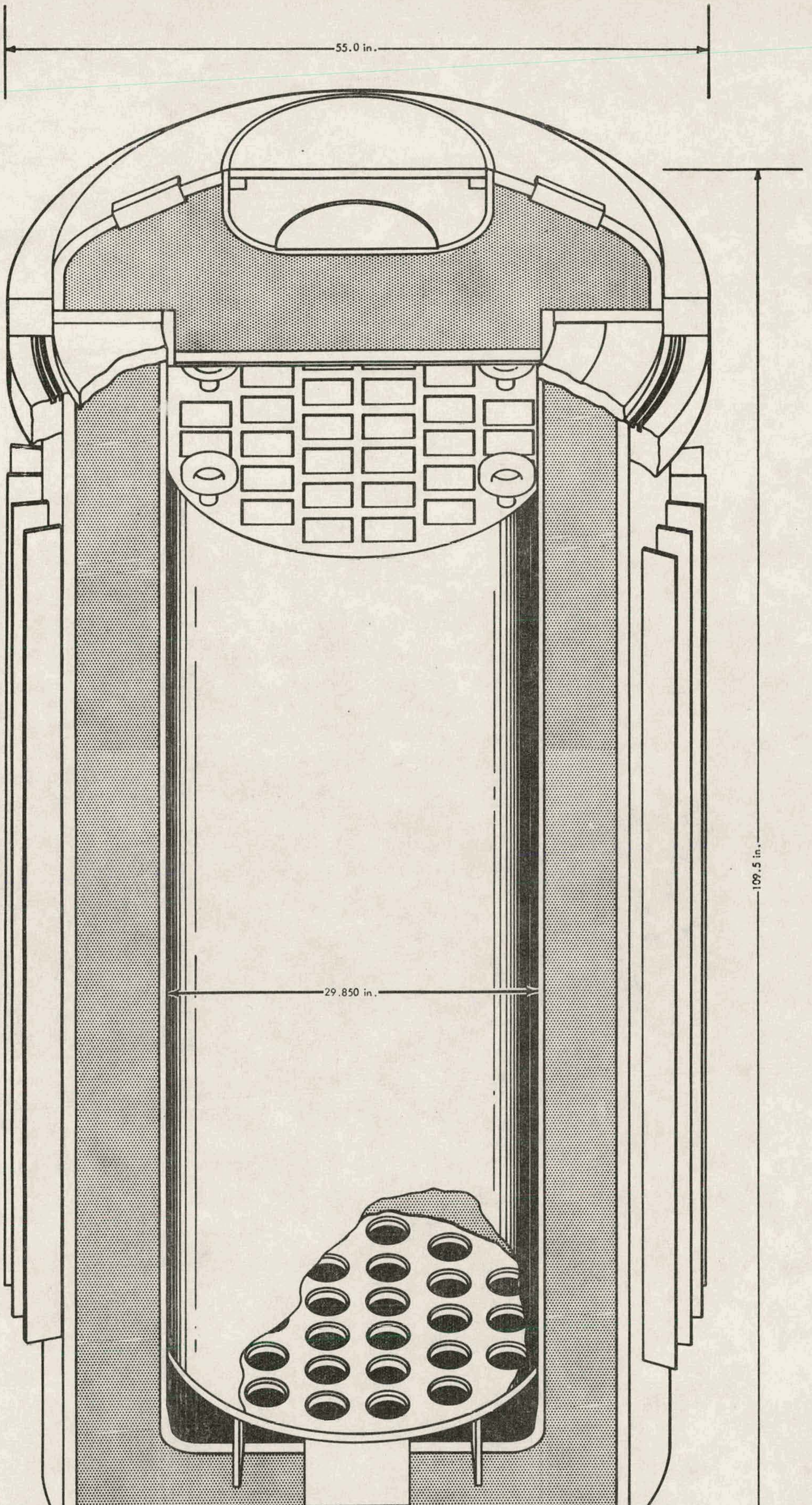
The multiplication factors were first determined for the zero cadmium condition with the following fuel loadings: 28 Core I spiked elements; 21 Core I regular, and 7 Core I spiked elements; 28 Core II elements; and 28 Core I regular elements. The multiplication factors for the 28 spiked element and Core II element cases were recalculated for the final analysis and those for the remaining two cases were obtained by utilizing the results of Ref. 22 and the more recent calculations. Further calculations were then made over a range of cadmium content from 0 to 1 wt. % using 28 Core I spiked, non-borated (5.2 wt. % U-235) elements in the fuel basket.



METALLOGRAPHIC SECTION OF SENSITIZED COUPON #65
300X UPPER AS POLISHED, LOWER ETCHED. INTERGRANULAR
ATTACK HAS PENETRATED TO A DEPTH OF ABOUT 6.5 MILS

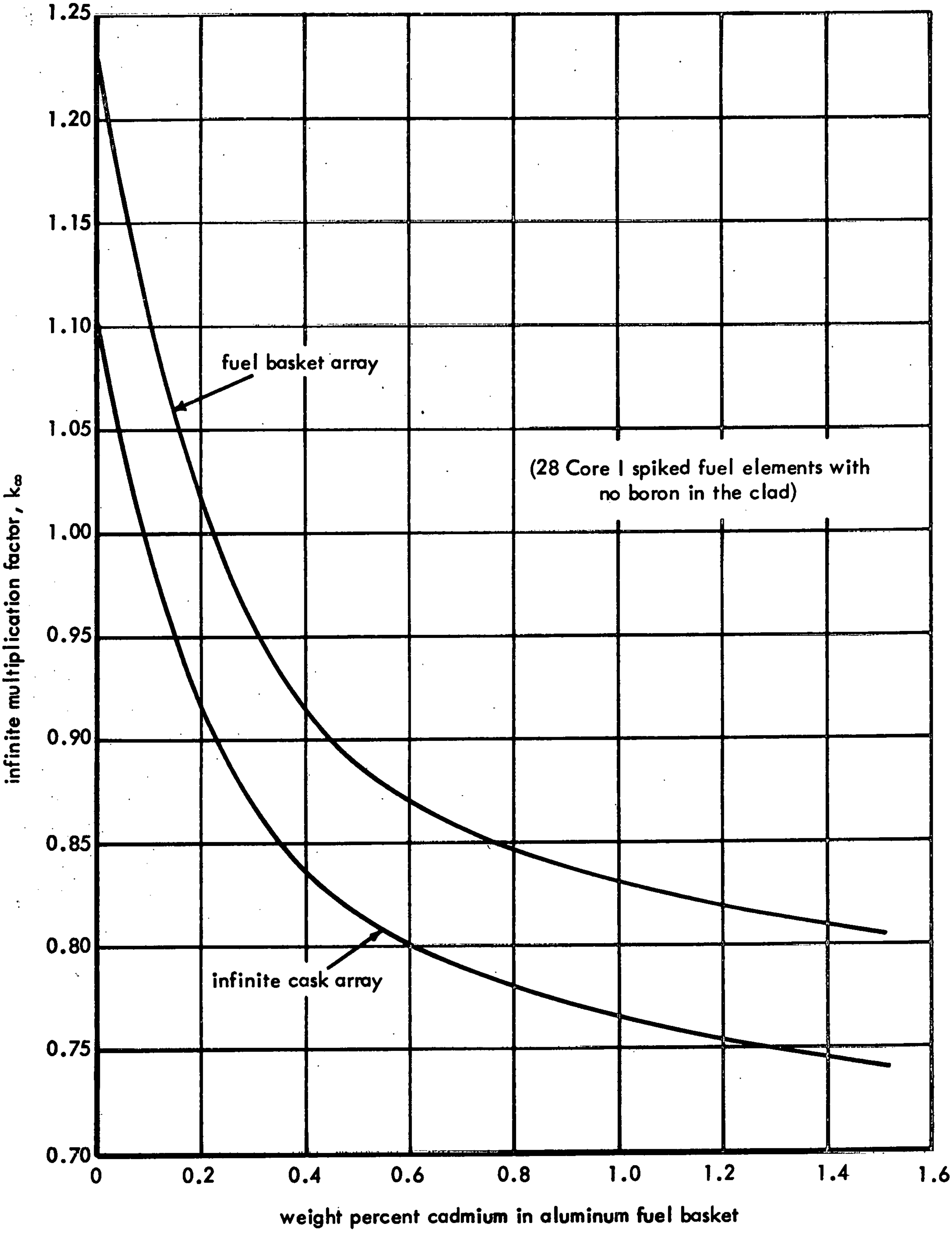
FIG. 7.1

ETCHING REVEALS TRANSGRANULAR AND BRANCHING NATURE OF THE CRACK (TOP EVAPORATOR #1, 100X ETCHED) FIG. 7.2



ERR SPENT FUEL SHIPPING CASK

FIG. 7.3



INFINITE MULTIPLICATION FACTOR VS. AMOUNT OF CADMIUM IN FUEL BASKET

FIG. 7.4

8. REFERENCES

1. J. R. Fisher, E. D. Kenrick, T. P. Kruzic, and A. C. Schafer, Jr., Elk River Reactor Operations Analysis Program Semi-Annual Progress Report, July 1, 1964 to December 31, 1964, ACNP-65532.

2. J. R. Fisher and E. D. Kendrick, Elk River Reactor Operations Analysis Program Topical Report - Fuel Cycle Studies for Second Core, Task 201, ACNP-66522, March 1966.

3. E. D. Kendrick, Method for Calculation of the ERR Fuel Inventory - First Reload, ACNP-66560, August 1966.

4. Marion L. Kennedy, Chieh Ho, and E. D. Kendrick, Methods for Calculation of the Elk River Reactor Fuel Inventory, ACNP-64519, Supplement 2, February 15, 1965.

5. T. P. Kruzic and A. C. Schafer, Jr., Elk River Reactor Operations Analysis Program, Task 401, Elk River Primary Water Makeup, ACNP-65604, November 1965.

6. T. P. Kruzic and A. C. Schafer, Jr., Elk River Reactor Operations Analysis Program, Task 404, ERR Thimble Cooling System, ACNP-65619, November 1965.

7. COO 651-8, RCPA, Elk River Reactor, 25th Monthly Operations Report, November 1964.

8. Elk River Reactor Operations Analysis Program, Interim Report, Task 615, Corrosion Samples and Tests, Evaporator Water Boxes, ACNP-64588, July 1964.

9. Consulting Report on the Inspection of 17-4 pH Mechanism Components, SwRI-1228-48, November 23, 1964.

10. Elk River Reactor Operations Analysis Program, Interim Report II, Task 615, Corrosion Samples and Tests, Evaporator Water Boxes, ACNP-65598, October 1965.

11. Elk River Reactor Operations Analysis Program, Final Report, Task 615, Corrosion Samples and Tests, Evaporator Water Boxes, ACNP-66542, June, 1966.

12. Rural Cooperative Power Association - Elk River Reactor Monthly Operating Reports.

13. Elk River Reactor Operations Safety Analysis Monthly Report No. 6, September 1 to October 31, 1965, NUS-263.

14. Reactor Operations Safety Analysis Technical Report No. 9, February 15 to March 15, 1964, GNEC-388.

15. Elk River Reactor Operations Safety Analysis Monthly Report No. 7, November 1, 1965 to January 31, 1966, NUS-264.

16. E. D. Kendrick, Elk River Reactor Operations Analysis Program Topical Report, Spent Fuel Shipping Cask Criticality Analysis, Task 617, ACNP-66551, July 1966.

17. Knapp Mills Co. Drawings: F-2055 (Rev. 5), Cask Body Machining Details; F-2056 (Rev. 6), Closure Head Machining Details, June 1962.

18. Knapp Mills Co. Drawings: F-2279 (Rev. 0), Elk River Fuel Basket Final Machining (28 Element); F-2280 (Rev. 1), Elk River Fuel Basket Assembly (28 Element); and C-1727 (Rev. 0), Elk River Fuel Basket Support Plate Fabrication and Machining, March 1964.

19. G. D. Joanou and J. S. Dudek, GAM-1, A Consistent P_1 Multigroup Code for the Calculation of Fast Neutron Spectra and Multigroup Cross Sections, GA-1850, June 1961.

20. R. H. Shudde and J. Dyer, TEMPEST-II, A Neutron Thermalization Code, AMTD-111, May 1961.

21. W. R. Cadwell, et al, The PDQ-5 and PDQ-6 Programs for the Solution of the Two-Dimensional Neutron Diffusion - Depletion Problem, WAPD-TM-477, January 1965.

22. G. R. Bond, Elk River Operations Analysis Program Criticality Analysis for the Elk River Reactor Shipping Cask, ACNP-63621, November 1963.

APPENDIX A

SUMMARY OF ELK RIVER REACTOR REPORTS

A-1 MONTHLY OPERATIONAL REPORTS

A-1.1 ELK RIVER REACTOR THIRTY-THIRD MONTHLY OPERATING REPORT; JULY 1965; COO-651-18

The reactor operated 89.7 percent during the month of July. Except for a 24-min outage June 18, the reactor operated continuously for a 55-day period. The reactor operated at a steady load of 49.2 Mwt for 19 days.

On July 28, the nuclear plant was shut down to allow for repairs to be made on evaporator #1. Inspection revealed two sizable tube leaks. Preparations are underway for repair of the existing leaks, after which a final inspection will take place using helium leak detection equipment. During the outage, the evaporator #1 handholes modification will continue, which allows bolted removable closures instead of the welded seal plates.

A technical specification change was submitted to the AEC which would delete the requirements which state that the concentration of radioactive materials in the secondary system coolant not exceed the levels specified in 10 CFR 20, Appendix B, Table II; but compliance with 10 CFR 20 would be required for all discharge from the plant.

The regulating rod was placed in the automatic mode of operation on July 15. This was the first time that the regulating rod was used on automatic during normal operation for an extended period.

A-1.2 ELK RIVER REACTOR THIRTY-FOURTH MONTHLY OPERATING REPORT; AUGUST 1965; COO-651-20

The reactor was shut down for 735.8 hr during the month of August. The major tasks performed during the month were the plugging of three defective tubes in evaporator #1 and the completion of the handhole modifications on evaporator #1.

After completion of the repairs and modifications, the evaporator shell and tube sides were hydrostatically tested at 1100 psig and 1250 psig, respectively. Inspection of the tube sheet during the shell side hydrostatic test, and of the handhole flanges during the tube side hydrostatic test verified that all tube plugs and handhole flanges installations had been successful.

Considerable time was spent on the early phases of the repair program in determining the type of plug to use, developing the technique in installing tube plugs, and fabricating the tools necessary for plug removal and installation.

Some minor modifications were completed on the evaporators, moisture separator, blowdown, and sampling lines.

Other activities during the month included a general overhaul and inspection of various equipment; i.e., sodium pentaborate tank air compressor, superheater, relief valves, etc., and completion of technical specifications tests.

The reactor was started up at 1535 hours on August 31 and primary pressure was increased to 640 psig; at this point a valve packing leak check was performed. The reactor was then shut down at 2347 hours on August 31 in order to perform a quarterly turbine trip scram check as required by the technical specifications.

The no. 3 turbine generator was on line with reactor power at 0158 hours on September 1, 1965 and subsequently returned to power operation.

A-1.3 ELK RIVER REACTOR THIRTY-FIFTH MONTHLY OPERATING REPORT; SEPTEMBER 1965; COO-651-21

The reactor operated for 720.8 hr during the month of September, thereby achieving the highest production month in the reactor's history. Allowing for the one hour gained during the month due to the area's returning to standard time, the reactor had a time operating factor of slightly under 100 percent (99.98 percent). The present run started at 0158 hours, September 1 when the no. 3 turbine generator was placed on the line, and continued through the last day of the month.

No major operating problems were encountered during the month. The next scheduled shutdown is presently planned for January 4, 1966.

A-1.4 ELK RIVER REACTOR THIRTY-SIXTH MONTHLY OPERATING REPORT; OCTOBER 1965; COO-651-22

The Elk River Reactor operated during the entire month of October, thereby achieving new monthly highs for reactor heat generation, electrical power production, and time operating factor. The reactor, as of October 31, has remained on line with the no. 3 turbine generator for 61 days without interruption.

No major operating problems were encountered during the month. The next scheduled shutdown of the reactor remains as previously reported; e.g., January 4, 1966.

A-1.5 ELK RIVER REACTOR THIRTY-SEVENTH MONTHLY OPERATING REPORT; NOVEMBER 1965; COO-651-23

The Elk River Reactor operated for 719 hr in November. The no. 3 turbine generator was on line with reactor power for 717.5 hr. The failure of an electronic tube in the reactor water low temperature scram circuitry resulted in a false reactor scram on November 22, 1965 at 1508 hours. The reactor was started up at 1613 hours, November 22, 1965 and the no. 3 turbine generator was back on line at 1735 hours, November

22, 1965. The plant had been in operation for over 82 consecutive days when the scram occurred. Operation continued for the remainder of the month.

A request for a technical specification change will be submitted to the Commission in the near future to eliminate the reactor low water temperature scram during power operation. A modification to four rod scram interlocks to provide redundancy is being scheduled for the next planned outage.

A-1.6 ELK RIVER REACTOR THIRTY-EIGHTH MONTHLY OPERATING REPORT; DECEMBER 1965; COO-651-24

The Elk River Reactor operated during the entire month of December. The 1,740.6 Mwd produced in the month was the highest in the reactor's history for one month of operation. A plant shutdown, previously scheduled for January 4, 1966 has been rescheduled for approximately April 15, 1966. This rescheduling was possible due to deferment of control rod inspection (Technical Specification Change No. 5) on the basis that periodic rod exercising would be performed during this period. During the April shutdown it is also planned to inspect the reactor vessel nozzles and flanges in accordance with technical specification requirements, and to perform core refueling.

A request for a technical specification change to remove the reactor low water temperature scram from the four-rod scram circuitry was submitted to the AEC during the month.

Several tasks that were postponed until determination of a scheduled plant shutdown date are now currently scheduled for completion in January. These tasks include: (1) installation of temporary sequential monitor for the four-rod scram circuitry; (2) temporary connections for the blowdown effluent monitor; (3) adjustments of the regulating rod automatic control limit switches; (4) off-gas system efficiency test.

A reactor safety survey was conducted by representatives of the Health and Safety Division of the AEC Chicago Operations Office during the month.

A-1.7 ELK RIVER REACTOR THIRTY-NINTH MONTH OPERATING REPORT; JANUARY 1966; COO-651-25

The Elk River Reactor operated during the entire month of January. The 1,767.8 Mwd produced during the month was the highest in the history of the reactor for one month of operation.

Preparations were made to receive 50 Core II fuel elements. These elements are scheduled to arrive on February 7, 1966.

No major operating problems were encountered during the month. The next scheduled shutdown date is April 15, 1966.

A-1.8 ELK RIVER REACTOR FORTIETH MONTHLY OPERATING REPORT; FEBRUARY 1966; COO-651-27

The Elk River Reactor operated for 658 hr during the month of February. On February 1 at 1332 hours the reactor was shut down via a spurious four-rod scram initiated by the reactor water low temperature circuit.

During the outage several tests, as required by the technical specifications, were completed. The detectors for nuclear channels N-1 and N-2 were repositioned. High voltage plateaus were checked, discriminator settings were checked and adjusted, and the preamplifier on N-2 channel was exchanged with a spare unit.

Prior to starting up the reactor, approval was received from DRL to remove the reactor water low temperature scram circuitry from the four-rod scram circuit. This was completed and the reactor was returned to power operation at 0033 hours on February 2.

On February 21 at 2030 hours the reactor was shut down via a four-rod scram initiated by a superheater trip signal. Events that precluded this scram were:

(1) failure of an electronic tube in the secondary feedwater pressure controller which resulted in reduced steam flow and, in turn, a reduced air flow to the superheater, and

(2) imbalance of air flow/fuel flow resulting in an unstable flame and, finally in a loss of flame with a subsequent superheater trip.

The feedwater pressure controller was repaired and checked for proper operation and the reactor was returned to power operation at 2338 hours February 21. The reactor operated for the remainder of the month.

Fifty-one Core II fuel elements and one dummy element were received, inspected, and loaded into the fresh fuel rack on February 7, 8, and 9. Forty-eight additional elements are tentatively scheduled to be received on April 8, 1966.

The off-gas system performance test as per NUS-TM-S-24 was begun on February 23, and is scheduled for completion on March 4.

A-1.9 ELK RIVER REACTOR FORTY-FIRST MONTHLY OPERATING REPORT; MARCH 1966; COO-651-28

The Elk River Reactor operated during the entire month of March.

The off-gas system performances test as per NUS-TM-S-24 was completed on March 5, 1966. The off-gas system operated without interruption throughout the test. Containment building air radioactivity levels were considerably lower than levels during previous off-gas periods; the lower levels were primarily due to improved sampling techniques and considerable maintenance on the system prior to testing.

Fifty-four additional Core II fuel elements are scheduled to arrive on April 11, 1966.

Plant operating records were reviewed by a representative of the AEC Division of Compliance on March 21 and 22.

California Nuclear Corporation began removing spent resins and radioactive thermocouples from the fuel element storage well on March 30.

No major operating problems were encountered during the month. The reactor plant is scheduled for shutdown on April 15, 1966.

A-1.10 ELK RIVER REACTOR FORTY-SECOND MONTHLY OPERATING REPORT; APRIL 1966; COO-651-29

The plant operated continuously during the first half of the month until April 15, when it was shut down as scheduled for partial refueling, inspections, modifications, and preventive maintenance. This run was begun on September 1, 1965 and was interrupted by only three unscheduled outages. These outages totaled one-half day for the seven and one-half month period from September 1, 1965 to April 15, 1966.

Total accumulated reactor thermal power produced as of April 15 was 27,887 Mwd. Twelve control rods were full out with the center rod at full power; the equilibrium xenon position was 49.8 in. of the 56-in. full out position. It was estimated that 1000 Mwd(t) remained in the first core. Average exposure of the first core was 7036 Mwd/MT, and the first core elements to be removed have an average exposure of 8900 Mwd/MT.

As of the end of the month one control rod (rod 9) had been completely inspected and found free of defects. Three other rods had been partially inspected; no defects were observed in the main portion of the rods. Some cracking on two rods was observed near the rod tips where a pin is welded to facilitate rod handling for inspection and transfer purposes. This condition is now being evaluated.

All fuel has been removed from the core and placed in the fuel element storage well. The reactor pressure vessel inspection is in progress. The status of other shutdown work and tests is shown in Sec. III.E and Sec. V.

A-1.11 ELK RIVER REACTOR FORTY-THIRD MONTHLY OPERATING REPORT; MAY 1966; COO-651-30

The plant remained shut down during May as scheduled. The control rod inspection has been completed. Horizontal cracks were observed in the absorber section of the regulating rod, which was located in the center core position. Plans had been made prior to the rod inspection to replace this rod and therefore it was transferred to underwater storage in the fuel element storage well. A new B_4C rod was received at the site and installed in the center position as a replacement.

Some apparent cracking of the 2 percent boron stainless-steel absorber was noted near the rivets at the transition on rods 2 and 5. These two rods will also be replaced with new B_4C rods which will be received next month.

Some cracking was found in the rod lifting pin gussets of 11 control rods. These gussets are being modified by the addition of stainless-steel reinforcing plates riveted to the original gussets. Repairs to three rods were completed at the end of the month.

The reactor vessel inspection was completed. All 8-, 10-, 12-, and 16-in. nozzles and the overlay on the main flanges were inspected. No defects were observed in any of these areas except for some cracking detected with dye penetrant in the overlay cladding of the reactor shell flange. Representative areas were ground out and it was found that the cracks did not extend into the carbon steel flange, but were confined to the stainless-steel cladding.

Testing of evaporator #2 revealed tube wall leaks; tube plugging was in progress at the end of the month. The handholes on evaporator #2 are being converted to a bolted closure at this shutdown as was done previously on evaporator #1.

Work was in progress on the modification to the four-rod scram circuit, superheater conversion and other scheduled tasks during the month. Based on the promised control rod delivery, fuel reloading is scheduled for June.

A-1.12 ELK RIVER REACTOR FORTY-FOURTH MONTHLY OPERATING REPORT; JUNE 1966; COO-651-31

Repairs to the control rod handling pins and gussets were completed during the month. Three new B_4C rods were installed in positions (3), (9), and (R), and original B-SS rods reinstalled in the other ten positions.

The second core was loaded during the month, and room temperature core physics measurements were completed satisfactorily. The reactor internals and head were reinstalled at the end of the month, and preparations were being made for a reactor and primary system hydrostatic test. Operation is scheduled to resume during the first half of July. The plant has been shut down since April 15, 1966.

Modifications to the superheater and preoperational testing were completed so that gas, gas/coke, or oil/coke can be used as fuel. Ultrasonic inspections of primary piping welds and reactor studs were completed; no defects were observed.

A fuel element inspection program revealed some tube bowing. Elements with indicated tube bowing were stored in the storage well pending evaluation. Details of shutdown activities appear in Sec. V.

A-2 REACTOR OPERATIONS ANALYSIS PROGRAM REPORTS

A-2.1 ERR OPERATIONS ANALYSIS PROGRAM PROGRESS REPORT JULY 1965, ACNP-65575, 8 p.

Progress is reported on the following task subjects: fuel cycle studies, technical specification changes, plant energy transfer systems, off-gas system, cask handling and storage, and corrosion test program. The report contains some preliminary results which illustrate the effects of rod programming on power shapes. The results of some radiolytic gas determinations are tabulated and discussed. Trips were made to the Chicago Operations Office and to the ERR site.

A-2.2 ERR OPERATIONS ANALYSIS PROGRAM PROGRESS REPORT AUGUST 1965, ACNP-65582, 11 p.

Progress is reported on the following task subjects: fuel cycle studies, technical specification changes, evaporator repair, and the removal of corrosion samples from the evaporator water boxes. Results are reported on investigations of the relative hot operating characteristics of Core II loading patterns. For a determination of the power distributions, a thermal and hydraulic evaluation of the Core II fuel elements was conducted. To ensure that the proposed Core II loading patterns meet the requirements of the technical specifications for reactivity insertion rates, calculations were performed for the loading pattern with the highest excess reactivity. The use of the ROA recommended Elliott plugs are reported as successfully plugging the leaking tubes of evaporator #1. A discussion of the deviations between the primary and secondary system heat loads is included in this report. Recommendations are made concerning the attainment of a more accurate assessment of the reactor plant heat balances. The report includes a description of the corrosion test specimens removed from evaporator #1 after 6900 hr of exposure in the water boxes. One trip to the ERR site was made in the report period.

A-2.3 ERR OPERATIONS ANALYSIS PROGRAM PROGRESS REPORT SEPTEMBER 1965, ACNP-65595, 10 p.

Progress is reported on the investigation of loading patterns for the first batch of Core II (4.4 w/o U-235) feed elements. Results are reported for patterns based on a three-batch discharge cycle, a four-batch discharge cycle, and on the utilization of Core I spares.

The report includes a diagram showing the location of the leaking tubes encountered in evaporator #1 in April 1964 and July 1965. The restrictions imposed on power operation by tube failures and tube fouling were investigated. The limit for tube failures was determined to be 322 U-tubes per evaporator for full power operation. Curves are presented which show how the reactor pressure operating requirements increase as the number of U-tubes available for heat duty decrease.

Corrosion coupons removed from the evaporators were examined and the corrosion rate data is presented.

A-2.4 ERR OPERATIONS ANALYSIS PROGRAM PROGRESS REPORT OCTOBER 1965, ACNP-65607, 21 p.

The calculated power and void distributions are reported for both the ERR Core I and Core II loadings. This information is presented, both for the present core with approximately 4000 Mwd/MT exposure and for a tentative Core II loading pattern using 48 Core II feed elements and 100 Core I exposed elements. The estimated potential discharge exposures at the end of the second burnup interval are tabulated for a uniform loading pattern.

Results are reported of an investigation of rod programs which used rods other than the center rod as the regulating rod. All patterns investigated gave a total peak-to-average power at least 15 percent greater than the center rod pattern.

Progress is reported on a study to determine what savings, if any, would accrue to the Commission by the addition of more capacity for collecting primary system condensate. Task progress is also reported on a study to determine whether modifications should be made to the thimble cooling system to improve operational control of seal flows and thimble temperature.

A-2.5 ERR OPERATIONS ANALYSIS PROGRAM PROGRESS REPORT NOVEMBER 1965, ACNP-65615, 8 p.

Progress is reported on the analysis of loading patterns based on a three-batch discharge cycle. The burnup characteristics of uniform and zone patterns were investigated for the first partial reload of Core II elements. Potential discharge exposures at the end of the second burnup interval are tabulated for a central loading pattern. A curve of center rod position vs. core exposure is presented for the central loading pattern. Curves are also presented of the peak-to-average power vs. core exposure.

Cold shutdown margins were calculated for the central and uniform patterns with B_4C in-tube-type rods. The results are given in tabular form and are compared with the present boron stainless-steel rods.

Progress is reported on the study of the thimble cooling system.

A-2.6 ERR OPERATIONS ANALYSIS PROGRAM PROGRESS REPORT DECEMBER 1965, ACNP-66504, 15 p.

Progress is reported on the analysis of loading patterns based on a three-batch discharge cycle. The analyses were continued through the second partial reload of Core II elements and the burnup characteristics of both the uniform and central loading patterns were investigated for the third burnup interval. Cold excess reactivities and stuck-rod shutdown margins are given for one example each of the uniform and central loading pattern types. Also given for these loading patterns are: (1) the hot operating excess reactivity and the unrodded power distributions; (2) the burnup characteristics, including the effect of rod positioning during the burnup interval; (3) tentative operating power distribution vs. exposure and rod position; and (4) the average element exposure for the Core I discharge group and for the two Core II feed groups present during the burnup interval.

During the report period, proposed changes in the wording of the technical specification were submitted to the USAEC-CH in draft form. The reasons for the changes and the safety considerations relevant to the proposed changes were also included.

The report contains the results of an analysis which shows that the k_{eff} of Core II elements in flooded storage cabinets will be less than 0.72.

Other tasks during the report period included a review of plant operating data and answers to questions concerning the ERR piping flexibility analysis.

A-2.7 ERR OPERATIONS ANALYSIS PROGRAM PROGRESS REPORT JANUARY 1966, ACNP-66514, 12 p.

Progress is reported on the following task studies: reactivity history, fuel cycle studies, and the plant energy transfer system. Preliminary investigations indicate that approximately 1 in. of indicated position on the center control rod may be due to boron depletion in the rod.

The analysis of central and uniform loading patterns for Core II continued. Data reported include: (1) the hot operating excess reactivity and the unrodded power distributions; (2) the regulating rod worth shown as core k_{eff} vs. rod position; (3) unrodded k_{eff} vs. exposure for the fourth burnup interval; and (4) tentative operating power distribution vs. time.

A-2.8 ERR OPERATIONS ANALYSIS PROGRAM PROGRESS REPORT FEBRUARY 1966, ACNP-66521, 6 p.

Progress is reported on task studies of reactivity history, fuel cycle studies, and the plant energy transfer system. PDQ code calculations were performed to confirm the minimum stuck-rod shutdown margin predicted by the FLARE code for the recommended loading pattern. The results indicate that the FLARE predictions were conservative.

A trip was made to the ERR site.

A-2.9 ERR OPERATIONS ANALYSIS PROGRAM PROGRESS REPORT MARCH 1966, ACNP-66528, 16 p.

The effect of boron depletion in the regulating rod on the indicated rod position was calculated as a function of core exposure. The effective rod position was predicted to be 0.56 in. higher than the indicated position at a core life of 6000 Mwd/MT.

Results are given of a calculation of the reactivity insertion rate for the hot zero power core.

A trip was made to the site to obtain operating data for comparison with data taken in previous years. Overall plant behavior was found to be excellent, and no areas of real concern were evidenced by the analysis of data. Note is made of the fact that the head load carried by evaporator #1 is significantly lower than that for evaporator #2.

The fabrication history of new corrosion coupons and stressed specimens is reported for Task 615 - Corrosion Samples and Tests - Evaporator Water Boxes. These specimens will be used to replace coupons presently installed in the water boxes of the evaporators. These latter coupons are scheduled for removal in April 1966.

A-2.10 ERR OPERATIONS ANALYSIS PROGRAM PROGRESS REPORT APRIL 1966, ACNP-66536, 14 p.

Excellent agreement is reported between the predicted lifetime and the actual lifetime of the core, and lends confidence to the analytical methods being used for predicting core characteristics. The reactivity left (full power operation) at the time of shutdown was negligible.

The results of additional analyses relating to the proposed technical specification changes are reported. Based on these results, it was recommended that the limitation on the maximum reactivity held down by the center regulating rod be no less than that of Core I.

Results of calculations of the auxiliary equipment heat duties are reported as well as a comparison of superheater performance with that of previous years.

Preliminary examination of test specimens removed from the water boxes of the evaporators indicated the presence of some cracks in the stressed specimens.

Work was begun on the criticality analysis of the ERR 28-element spent fuel shipping cask.

Three trips to the site were made during the month.

A-2.11 ERR OPERATIONS ANALYSIS PROGRAM PROGRESS REPORT MAY 1966, ACNP-66543, 12 p.

The establishment of a new technical specification limiting the amount of excess reactivity that may be held down by one rod under operating conditions necessitated the consideration of an initial rod program different from the one used for Core I. As a result, new limit curves were established for the minimum 12-rod bank height vs. power to ensure remaining within the maximum heat flux of 313,000 Btu/hr-ft^2 required by the technical specification.

Three new B_4C in-tube-type control rods will be used in conjunction with 10 of the original boron-stainless-steel-type rods. The necessary input parameters are being generated for the FLARE model.

The report presents the results of an evaluation of the load-carrying capacity of the gussets to which the control rod handling pin is welded. This study was initiated because of the observance of cracks in these gussets. The consequences of dropping a control rod into its channel were also evaluated.

The results of the criticality analysis for the 28-element spent fuel shipping cask are presented.

Two members of the staff attended a meeting at the Chicago Operations Office, USAEC, during the report period.

A-2.12 ERR OPERATIONS ANALYSIS PROGRAM PROGRESS REPORT, JUNE 1966, ACNP-66550, 16 p.

The majority of the pre-startup calculations and predictions were made for the fuel loading pattern which was recommended in ACNP-65522 (see Sec. A-2.20). However, the core was actually loaded to a different pattern; in fact, after initial reloading and critical measurements, the pattern was changed for the final loading to eliminate elements having individual fuel rods which were slightly bowed. A comparison of measured and calculated core nuclear characteristics is reported for both of these core loadings.

The report also includes an estimate of the Nvt exposure of the control rods at the time of shutdown.

A trip was made to the ERR site.

A-2.13 ERR OPERATIONS ANALYSIS PROGRAM SEMI-ANNUAL PROGRESS REPORT, JANUARY 1 - JUNE 30, 1965, ACNP-65602

During the report period, the reactor heat generated was 7434.8 Mwd; and the net electrical generation was 67,362 Mw-hr. The overall plant behavior was excellent, and plant availability was high. The operating authorization was transferred from Allis-Chalmers to the Rural Cooperative Power Association early in June 1965.

The Operations Analysis efforts have been directed mainly to establishing a fuel management program which best meets the objectives of the Elk River Reactor, analyzing the operation of the plant energy systems and the primary auxiliary systems, and studying methods for disposing of spent ion-exchange resins.

The measured position of the center regulating control rod vs. core exposure is compared with the previously reported curve of the predicted rod position. The observed rate of rod withdrawal shows no significant deviation from the calculated rate.

Progress is reported on the fuel cycle studies in establishing fuel loading patterns for Core II which meet the requirements for stuck-rod shutdown margin and excess reactivity. The patterns which have received the most attention are based on a three-zone (most depleted fuel discharged) assumption.

Plant energy transfer system data taken in March 1964 and March 1965 are compared with the design parameters. The differences are small and reflect an increased accuracy in the more recent data which is attributed mainly to better instrument calibration.

Analyses of the primary purification system and the boron poison system are also reported. A study of methods for disposal of spent ion-exchange resins resulted in the recommendation that resins be sluiced into a shielded disposal drum.

Abstracts of reports from all sources which were issued during this report period are included as Appendix A of this report.

A-2.14 T. P. KRUZIC, TOPICAL REPORT TASK 407: BORON POISON SYSTEM EVALUATION, ACNP-65541, JULY 1965, 11 p.

This report presents the results of a study of the ERR sodium pentaborate system. The main emphasis is directed toward eliminating crystallization of sodium pentaborate; such crystallization has previously caused valve and line blockage. A minor modification to the present system is suggested which would provide for immediate dilution of any pentaborate solution leaking past the closure valves. The suggested additions would be simple and inexpensive. The present solution injection time would be maintained, and no changes to the specifications would be required.

A-2.15 T. P. KRUZIC, TOPICAL REPORT TASK 612: PURIFICATION DEMINERALIZER RESIN REMOVAL, ACNP-65562, JULY 1965, 32 p.

This report presents the results of a study of methods for the disposal of spent ion-exchange resins from the primary coolant purification system of the Elk River Reactor. The methods of disposal which were investigated include: (1) removal of the resin cartridge containing resins and disposal of the entire cartridge in a concrete cask; (2) complete removal of the resins only by sluicing them into a shielded drum; and (3) removal of only a portion of the resins by sluicing into a shielded drum. Costs of the various disposal methods are presented, and a recommendation is made to sluice the resins into a disposable container.

A-2.16 C. R. BERGEN, INTERIM REPORT II, TASK 615: CORROSION SAMPLES AND TESTS - EVAPORATOR WATER BOXES, ACNP-65598, OCTOBER 1965, 19 p.

Type 304 stainless-steel coupons and U-bend samples were exposed to the wet steam-water environment in the evaporators of the Elk River Reactor. This report describes the results of evaluation of the corrosion during a 14-month exposure. Small corrosion rates of as-rolled, annealed, and sensitized materials were observed. Evidence of stress cracking was found in one U-bend specimen and tentatively attributed to conditions of wetting and drying and oxygen concentration.

A-2.17 T. P. KRUZIC AND A. C. SCHAFER, TASK NO. 401: ERR PRIMARY WATER MAKEUP, ACNP-65604, NOVEMBER 1965, 16 p.

When the power of the reactor is raised from 70 percent to 100 percent of full power, water is dumped from the primary system to avoid having too high a water level in the reactor. In making such a power change, approximately 250 gal of water are removed from the reactor to the collecting tank. Since the collecting tank volume is only 50 gal, primary water must be discharged to waste via the 3000 gal retention tanks, rather than being returned to the reactor when the power level is reduced.

This report presents the results of a study of the savings that would accrue to the Commission by the addition of more capacity for collecting the dumped water. Some of the important assumptions which were made in the cost study include the following: high quality demineralized water costs \$4.70 per 1000 gal; analytical costs involved in dumping the retention tanks are \$25 per 3000 gal; equipment costs are not capitalized, and no attempt is made to apply present worth techniques to annual savings; the number of years required to recover the capital equipment cost is presented for various operational and equipment schemes.

A-2.18 T. P. KRUZIC AND A. C. SCHAFER, TASK NO. 404: ERR THIMBLE COOLING SYSTEM, ACNP-65619, DECEMBER 1965, 14 p.

This report presents the results of a study to determine whether modifications should be made to the ERR thimble cooling system to improve the operational control of seal flows and to balance thimble temperatures. The operating history of the thimble cooling system is reviewed and the need for improved operational control is verified. Recommendations are made for improvements in control valves and changes in flowmeters which will accomplish the stated objectives. A cost estimate of the recommended changes is included.

A-2.19 J. R. FISHER AND E. D. KENDRICK, ELK RIVER REACTOR OPERATIONS ANALYSIS PROGRAM, TOPICAL REPORT, FUEL CYCLE STUDIES FOR SECOND CORE - TASK 201, ACNP-66522, MARCH 1966, 65 p.

The major emphasis in this study has been placed on a three-batch loading cycle of Core II elements (i.e., approximately one-third of the core is discharged at the end of each burnup period and replaced by fresh feed fuel). A separate phase of the study concerned the effect of varying the size of the discharge group as well as incorporating the 20 Core I spares into the feed patterns. To investigate the three-batch loading cycle, it is assumed that 48 of the Core I regular (4.3 w/o U-235) elements with the highest exposure will be discharged. The average exposure of these 48 elements has been predicted to be approximately 8900 Mwd/MT (assuming a core average exposure of 7000 Mwd/MT). The 100 remaining Core I fuel elements would at this time include 80 regular elements with an average exposure of 5350 Mwd/MT and 20 spiked (5.2 w/o U-235) elements with an average exposure of 9100 Mwd/MT.

All tentative core arrangements were subjected to a determination of the room temperature excess reactivity and minimum stuck-rod shutdown margin. Patterns which exhibited acceptable room temperature characteristics were further examined on the basis of their hot, operating excess reactivity and power distribution. Several patterns which were still acceptable were then subjected to a detailed analysis of the burnup characteristics, in which the effects of rod programming were considered.

Since the choice of a loading pattern for the first batch of the Core II fuel can affect not only the second burnup interval but the entire fuel cycle, patterns which were still attractive at this point were examined further to determine the type of equilibrium cycle which might result; the final recommendation being that which best satisfied the objectives and conditions delineated prior to the study for both the initial and subsequent batch reloads of the core.

A-2.20 C. R. BERGEN, FINAL REPORT TASK 615, CORROSION SAMPLES AND TESTS, EVAPORATOR WATER BOXES, ACNP-66542, JUNE 1966, 27 p.

Type 304 stainless-steel coupons and U-bend test samples were exposed to the wet steam-water environment within the evaporator water boxes of the Elk River Reactor. This report describes the results of evaluations of the test probes after 22 months of exposure. Small corrosion rates of as-rolled and annealed specimens were found. Some intergranular attack of the sensitized coupons showed up during this period. All U-bend samples exhibited stress corrosion cracking.

A-2.21 E. D. KENDRICK, ELK RIVER REACTOR OPERATIONS ANALYSIS PROGRAM TOPICAL REPORT, SPENT FUEL SHIPPING CASK CRITICALITY ANALYSIS - TASK 617. ACNP-66551, JULY 1966, 17 p.

Criticality calculations were performed which considered the effect of various element types and of various amounts of cadmium added to the 28-element fuel basket on the multiplication factor of an infinite cask array. It was determined that the addition of

1.0 wt. % cadmium to the aluminum used in fabricating the fuel basket is sufficient to reduce the multiplication factor to less than 0.80 with even the most conservative assumptions regarding the reactivity of the fuel.

The criticality calculations also evaluated the multiplication factor of an infinite array of fuel baskets (without the cask portion). In this case, the results showed that the addition of 1.0 wt. % cadmium is sufficient to reduce the multiplication factor to less than 0.85 with the same conservative assumptions regarding the activity of the fuel.

It is concluded that 1.0 wt. % cadmium uniformly dispersed in the 28-element fuel basket is entirely sufficient to meet the criticality requirements of 10 CFR 71 for any combination of fuel elements now available, and for any fuel presently envisioned for future use in the Elk River Reactor core.

A-2.22 E. D. KENDRICK, JR., METHODS FOR CALCULATION OF THE ERR FUEL INVENTORY - FIRST RELOAD, ACNP-66560, AUGUST 1966

The Elk River reactor was refueled for the first time in June 1966. This report presents information and procedures which are required to calculate the Elk River reactor fuel inventory. Tables are used to simplify the calculation of fuel burned and fuel remaining, and the isotopic content of the core. Other tables are included which permit the calculation of the isotopic inventory of each element.

A-3 REACTOR OPERATIONS SAFETY ANALYSIS REPORTS

A-3.1 ERR OPERATION'S SAFETY ANALYSIS, MONTHLY REPORT NO. 4, MARCH 1 TO MAY 31, 1965, NUS-246, JULY 16, 1965, 158 p.

A-3.1.1 Health Physics, Waste Disposal and Safeguards

The sensitivity of the fission product monitor has been calculated based on certain assumptions as to its performance. Under the assumed conditions the monitor should be adequate for its intended use. A test procedure is presented for calibration of the monitor and determination of its sensitivity.

Review of monitored data of the turbine air ejector activity has been performed, and analysis indicates that a primary to secondary leak is present. Since the activity levels are below the alarm limits, the leakage presents no health physics problem at this time. This area will continue to receive close surveillance in the future.

A-3.1.2 Chemistry and Materials

Use of a deoxygenating resin in the primary purification system of ERR has been evaluated. The purpose is to reduce the oxygen content of the primary system and thereby reduce the corrosion rate. Its main role is to remove oxygen in the coolant at startup, particularly when the system has been open and the water saturated with air. It should also reduce the hazards of venting the primary system since the overall radiolytic gas content would be reduced. The deoxygenating system consists of an anion resin bed in the sulfite form followed by a strainer, a mixed bed HOH resin column, and after-filter. The two new columns would be piped in series with the effluent of the existing mixed bed columns.

An investigation was made into the radioiodine peaking which has occurred regularly with nearly every plant startup at Elk River during the past year. The peak activity levels are usually several hundred times higher than the previous steady power levels for iodine-131 and ten to 50 times higher for iodine-133. The conclusion of this investigation is that the source of the ERR iodine peaking during startup is due to uranium-235 surface contamination ("tramp uranium") of the fuel, and that no detectable fuel element defect is believed to exist.

A-3.1.3 Instrumentation and Control

Examination of startup data for several recent reactor startups indicated that nuclear instruments N3 and N4 are linear over the six decades of neutron flux to which they are sensitive.

A proposed redesign of the four-rod scram circuit was discussed with Elk River personnel on March 31. It was determined that space is available for all of the additional instruments which will be required to make the four-rod scram circuit operate on the coincidence of 2 out of 3 instrument signals. The all-rod scram circuit output relays will be modified to permit the failure of any single relay without inhibiting a scram or causing a false scram; space is available for the relays needed to make this change, which can be readily accomplished at the same time the four-rod scram circuit modifications are installed.

Tests were performed in the Elk River laboratory on the rod clutch coil circuit to verify design calculations on the replacement of a potentiometer, capacitor, and resistor with a zener diode circuit.

Work was accomplished on the voltage regulator on the fail free bus and the power source for the rod position indicators.

A-3.1.4 Reactor Plant Engineering

An investigation was made into the possible reasons for the bowing of some of the fuel tubes in the two fuel elements removed from the core in November 1964. Although the bowing cannot be attributed at this time to any one mechanism, several possible reasons for this occurrence are given. It is recommended, in view of possible consequences of the observed deformation, that a more comprehensive inspection of the bowed fuel elements presently in the fuel pit be undertaken.

A study was made of the Elk River control rods in order to provide support for a technical specification change to permit operation of the regulating rod beyond the limit of 3.4 percent elongation at fracture. It is concluded that no cracking is expected for operation at least up to a boron depletion of 0.7 a/o.

A-3.2 ERR OPERATION'S SAFETY ANALYSIS, MONTHLY REPORT NO. 5, JUNE 1 TO AUGUST 31, 1965, NUS-256, OCTOBER 6, 1965, 120 p.

A-3.2.1 Health Physics, Waste Disposal and Safeguards

During the first six months of 1965, the turbine air ejector (TAE) effluent activity showed a slow, steady increase due to a primary to secondary system leak. The off-gas system was placed in operation and significantly reduced argon-41 activity in the TAE discharge; but the reduction in fission gas activity was small. In June, the TAE activity increased by about a factor of 2 every 20 days and the last week in July the activity suddenly increased from 40,000 cpm to 130,000 cpm with peaking to 160,000 cpm on July 27, 1965.

The proposed changes to 10 CFR 20 published in the Federal Register of August 1965 were reviewed to evaluate their effect on Elk River Reactor operations. The changes are not expected to have any significant effect on plant limits or on analytical requirements. For example, the proposed limit for krypton-88, formerly not specifically listed in 10 CFR 20, is the same as that established by calculation and in use by the site at present.

The permissible release rates of airborne activity are being reviewed based on diffusion data and source data obtained during the course of the off-gas system test program. Weather data taken at the site and data from the Minneapolis-St. Paul and St. Cloud Stations are to be used to define the diffusion climatology at the site. It is intended that this work result in a change to the technical specifications to permit higher stack release rate and, therefore, allow more flexibility in plant operations.

A-3.2.2 Chemistry and Materials

In May of 1965, the secondary water monitor indicated an increase in the coolant activity level. The secondary water activity continued to increase, ultimately reaching a level of 16,000cpm in late July, which corresponds to about 2.4×10^{-4} μc/ml. The only identifiable contributor to the water activity was fluorine-18 which accounted for essentially 100 percent of the activity in the later stages of operation. Earlier, prior to operation of the off-gas system, the presence of argon-41 was also noted.

Plots of various primary system parameters as determined during the first eight months of 1965 are presented.

A-3.2.3 Instrumentation and Control

Investigation of the proposed modifications to the 4-rod scram circuit indicates that 2 out of 3 logic with all logic blocks in series (single leg circuit) appears to be the most advantageous in minimizing both the extent of modifications and the cost of accomplishing the work.

The sensitivity of N-3 and N-4 intermediate range instruments was investigated; increasing the range by switching methods presents some undesirable features. Means were developed to take full advantage of the trickle current modification.

Modification of the linear nuclear instrument channels N-5 and N-6 to provide automatic range switching and incorporate "averaging type" period protection were investigated. This modification has the potential of decreasing the probability of scram due to range switching errors or spurious period scrams while increasing the range over which period protection is available.

The control rod clutch coil circuit modification has been finalized. The revised circuit employs a semiconductor to limit voltage spikes and results in improved clutch coil circuit reliability.

A-3.2.4 Reactor Plant Engineering

Analyses are being performed to determine whether the 425 F scram could be eliminated without adversely affecting reactor operations or safety. Advantages which would be gained from elimination of the scram include: (1) a simplified reactor startup procedure which would permit criticality to be achieved at relatively low temperatures, followed by plant heatup utilizing nuclear power; (2) improved operational reliability through reduction in the number of spurious scrams; and (3) simplification of the reactor safety circuits. The studies being performed include: an analysis of temperature and void coefficients as a function of reactor lifetime and coolant temperature, evaluation of core safety limits as a function of system pressure, and analyses of steady-state operations, normal operational transients, and accidental transients. It is expected that these studies will show that, if the 425 F scram is eliminated, the reactor will still be safe under all operational conditions. Protection provided by the other scram signals and the inherent self-regulating characteristics of the reactor itself, is tentatively believed to be adequate.

Details are presented of the detection of the leaks in evaporator #1, procedures used for plugging the tube sheet mockup and the evaporator, and the testing of the installed tube plugs. Three tubes in the evaporator were found to be leaking; these leaks were successfully sealed by installation of six plugs. On-site assistance was provided by NUS for these operations.

A-3.3 ERR OPERATION'S SAFETY ANALYSIS, MONTHLY REPORT NO. 6, SEPTEMBER 1 TO OCTOBER 31, 1965, NUS-263, DECEMBER 10, 1965, 117 p.

A-3.3.1 Health Physics, Waste Disposal and Safeguards

The turbine air ejector activity level was reduced from about 130,000 cpm before plugging of the evaporator leaks to approximately 3000 cpm at startup on September 1, and appeared to level off at 4000 to 4500 cpm during the month. There was a slight increase during October, but it is not considered to be indicative of leak enlargement.

Off-gas system performance tests were performed during the week of September 20, 1965. The off-gas system was operated at 40 percent of design steam flow rate from the water boxes to the recombiner. The recombiner effluent required dilution air to reduce the hydrogen content to a safe level. The system did not significantly reduce the primary system oxygen and hydrogen contents, but inert gas content and gaseous activity were markedly reduced. A high sulfur dioxide concentration in containment atmosphere and a high makeup charging rate due to off-gas system operation were the presumed causes of a high primary system conductivity which necessitated stopping the tests.

A-3.3.2 Chemistry and Materials

The activity level of the secondary system was reduced from 24,000 cpm to 400 to 500 cpm as a result of the tube plugging operation. Some increase in these readings has been observed toward the end of October when 900 cpm was detected.

Primary system oxygen concentration has decreased during September and October to less than 0.1 ppm in the water sample and 6 to 7 ppm in the steam sample. Primary system hydrogen content in the steam has varied from 1.0 to 4.2 ppm.

A-3.3.3 Instrumentation and Control

Proposed modifications to the 4-rod scram circuit have been finalized and delivered to the RCPA. These modifications improve reliability both from the standpoint of preventing a false scram and from losing scram protection. A temporary sequence monitor has been designed to replace the existing monitor which operates directly on 4-rod scram circuits. A final design of the sequence monitor will be installed when the 4-rod scram modifications are incorporated in January 1966, and will indicate the first three 4-rod scram signals occurring.

To increase the effectiveness of the reactor automatic pressure controller, a rod limit switch range selector circuit has been designed. This circuit will allow automatic control over all ranges of reactor power without the operator's leaving the control room.

Efforts have been generated to support the proposed automatic range switch and period protection with the linear intermediate range nuclear instruments. Solid state static inverters are being evaluated for application as replacement for, or addition to, the present rotary inverter for the vital bus.

A-3.3.4 Reactor Plant Engineering

Analyses have been conducted to determine the magnitude of the errors in determination of the oxygen content in the main steam system. A maximum error of 65 percent due to the combined effects of hydraulic considerations and analysis procedures was calculated. It was concluded that a more realistic estimate of the maximum error was about 25 percent. The reduced maximum error is considered to be conservative in view of the observed consistency in measured sample concentrations when sampling techniques, sample volume and purge time were calculated.

Even using the calculated maximum errors, the oxygen concentrations are within the range normally experienced in boiling water reactors and should not present any problems.

A-3.4 ERR OPERATION'S SAFETY ANALYSIS, MONTHLY REPORT NO. 7, NOVEMBER 1, 1965 TO JANUARY 31, 1966, NUS-269, APRIL 6, 1966, 111 p.

A-3.4.1 Health Physics, Waste Disposal and Safeguards

The size of the primary demineralizer cartridge necessitates awkward and potentially hazardous maneuvering to remove the spent resin from the unit. Three alternative solutions to this problems are presented along with additional evaluation of resin handling and storage equipment.

Re-evaluation of annual average dilution factors for the discharge of gases to the atmosphere was performed utilizing an NUS computer program (WINDIF-3) with input data from the site and alternatively from adjacent meteorological centers.

A-3.4.2 Chemistry and Materials

Primary system oxygen concentrations have generally remained at less than 0.1 ppm in the water sample and 5 to 7 ppm in the steam sample. Primary system hydrogen content in the steam has varied from 1.3 to 2.3 ppm. Corrosion product activities in the water have decreased somewhat; however, the chromium-51 activity has decreased a factor of 100 during the period of October through December, 1965.

Little change has been noted in the secondary water monitor readings during this period. On January 9 a peaking of the activity level from 750 to 2000 cpm was noted, but the activity returned to its previous level in a couple of days. No concurrent peaking of the turbine air ejector activity was noted.

The relationships between oxygen and chlorine concentrations in the primary system with regard to the corrosion of stainless steel in the Elk River Reactor are discussed.

A-3.4.3 Reactor Analysis

As a check on the reactivity loss rate the position of the center control rod was recorded for an extended period. The actual rod positions were then compared to previous predictions of reactivity loss rate with the result that the recorded information correlated well with the predicted trend.

It was requested that the inspection of control rods normally scheduled on the basis of reactor history be deferred until the scheduled refueling shutdown.

The alternative methods of calculating fuel costs were reviewed, including projected costs for several reactor modes.

The prediction of center rod worth, including fuel burnup effects was derived using the FLARE calculation and compared with beginning-of-life conditions.

A-3.4.4 Instrumentation and Control

A design for modifying the four-rod scram system has been developed to provide redundancy of each reactor plant parameter signal supplied to the four-rod scram relays. This modification also extends the concept of redundancy to the vital scram relay functions (withdraw permit, four-rod insert and four-rod scram).

Additional refinements to the system include a provision for testing secondary plant auxiliary relays with scram functions during reactor operation. The modification includes the installation of a sequence monitor to identify the order of receipt of multiple four-rod scram initiation signals.

A-3.4.5 Reactor Plant Engineering

A tool post holder was developed to facilitate the refacing of the previously seal-welded evaporator handholes. In anticipation of concerting the No. 2 evaporator handholes from seal-welded closures to a gasket and coverplate type, a more adaptable type of tool post holder was designed.

A study was conducted to obtain an understanding of the mechanism and cause(s) of the primary feedwater system hydraulic oscillations. A comparison of observed characteristics and analytical models led to a tentative hypothesis; however, experimental testing did not verify the principle that the oscillations were initiated by evaporator variable heat transfer phenomenon.

Specifications were developed for a network of insulation block support bands and an aluminum exterior weather cover for the containment vessel.

A-3.5 ERR OPERATION'S SAFETY ANALYSIS, MONTHLY REPORT NO. 8, FEBRUARY 1 TO 28, 1966, NUS-283, MAY 25, 1966, 45 p.

A-3.5.1 Chemistry and Materials

Activity in the secondary system remained normal and essentially constant during the month. Activity reductions of about 20 percent in the secondary system were noted during off-gas system operation which reduced the activity level to approximately 750 cpm.

Primary system oxygen levels continued to be low during the month. Primary steam oxygen levels remained at about 7 ppm, except during off-gas system operation when the level increased to about 15 ppm. Dissolved oxygen concentrations in the primary water were below 0.4 ppm throughout the month.

The off-gas system performance test was begun on February 23 after replacement of the recombiner catalyst and instrument checkout.

A-3.5.2 Reactor Analysis

A review of possible means of reducing Core I fuel costs, by extended power operation, was performed. It was concluded that, although extended Core I life would tend to reduce Core I fuel costs, the significantly lower Core II fuel cost obviates further consideration of extended Core I operation.

A-3.5.3 Instrumentation and Control

An additional consideration to improve station safety and reliability was to review the superheater control and protective circuitry. The relationship of the superheater to reactor safety and the adequacy of the present controls indicated that revision of the superheater controls was not warranted.

To prolong the useful life of the life of the N-1 and N-2 source range detectors, which are scheduled for replacement due to degradation of the installed units, it was recommended that an end shield be provided for each unit.

A-3.5.4 Reactor Plant Engineering

NUS reviewed the Allis-Chalmers' study of the control rod drive thimble cooling system and concurred in the conclusions and recommendations of the study. The study recommended a revision of cooling water system valves and flow instrumentation to improve operational control of the cooling water flow.

A review of the Elk River Reactor Containment Leakage Testing Program was performed and draft proposals for changes to the Elk River Reactor Technical Specifications were developed.

Proposed changes to the Elk River Technical Specifications, developed by Allis-Chalmers, were reviewed. The proposed changes were originated to permit the use of Core II fuel elements and a new type of control rod (B_4C in-tube). After some modifications and the addition of several new items to cover operations during refueling and reactor vessel inspection, Technical Specification Change Request No. 7 was prepared and forwarded to RCPA for review, approval, and transmittal.

A-3.5.5 Task Progress Report

The progress of certain tasks assigned to NUS, under subcontract No. 5, will henceforth be the subject of a new section of this report. During the month of February 12 tasks were assigned with various scheduled completion dates as noted on Fig. 6.

A-3.6 ERR OPERATION'S SAFETY ANALYSIS, MONTHLY REPORT NO. 9, MARCH 1 TO 31, 1966, NUS-286, JUNE 8, 1966, 113 p.

A-3.6.1 Health Physics, Waste Disposal, and Safeguards

A review of the allowable containment leak rate under maximum credible accident conditions was performed. It was determined that the two-hour thyroid dose (300 rem) at the exclusion boundary is the governing factor with regard to leak rate criterion. Comparison of expected dosage at MCA conditions, as determined by computation based on various assumptions and operational characteristics, indicated that a request for an increase in the Technical Specification limit for containment leak rate, from the present value of 0.1%/day to 0.45%/day, is warranted.

A-3.6.2 Chemistry and Materials

Secondary water activity levels were constant throughout the month except for the period of March 20 to March 25 when the activity level increased from 750 to 1100 cpm but did not exceed the normal range of activity levels. Off-gas operation reduced the activity to 600-850 cpm where it remained the rest of the month. The primary system oxygen levels remained well below the prescribed upper oxygen level limits throughout the month. Following shutdown of the off-gas system on March 4, oxygen levels in the primary steam decreased to a minimum value of 1.9 ppm while dissolved oxygen levels were as low as 0.14 ppm.

The radiation levels of various areas within the containment increased significantly during the period from March 20 to March 25. Subsequent operation of the off-gas system on March 25 caused the reduction of area radiation levels to the approximate values prior to March 20, 1966.

Chemicals considered for use in the reactor shield cooling system were examined for suitability from the standpoint of corrosion control and potential adverse effects as the result of irradiation. It was recommended that, of the chemicals considered, isotopically enriched lithium hydroxide represented the most suitable additive.

A-3.6.3 Reactor Analysis

A study of reactor fuel burnout safety factors, central fuel temperatures, and heat flux conditions was performed to obtain information which would define the limits of the anticipated Phase III Tests of the Power Escalation Test Series.

A-3.6.4 Instrumentation and Control

A modification of the source range detectors was developed to provide shielding of the detector to reduce the rate of degradation due to gamma flux.

A-3.6.5 Reactor Plant Engineering

In response to a request by RCPA, the weights of various core components, in air and submerged, were determined.

To determine the feasibility of increasing the reactor power from the present nominal value of 58 Mwt, a series of operational tests was developed. The Power Escalation Tests were designed to obtain specific information regarding plant and component performance.

A three-phase testing program was scheduled to be performed prior to the scheduled April 1966 station shutdown. The first test phase was to determine component performance at various pressures, the pressure coefficient of reactivity, and to calibrate the evaporator downcomer water level in response to feedwater valve opening. The second test phase was to measure evaporator and secondary steam moisture separator performance at steam flow rates above the existing nominal values. Subsequent to evaluation of the results of the Phase I and Phase II tests, Phase III tests will be developed to determine the degree of increased power which may be obtained from the existing reactor system.

A-3.6.6 Task Progress Report

Of the 12 tasks assigned under Subcontract No. 5 during February 1966, one has been completed. The scope, action, and reports submitted during the current month are summarized for the remaining tasks. A graphic presentation of the work in progress and approximate status is depicted by Fig. 7.

A-3.7 ERR OPERATION'S SAFETY ANALYSIS, MONTHLY REPORT NO. 10, APRIL 1 TO APRIL 30, 1966, NUS-291, JULY 14, 1966, 166 p.

A-3.7.1 Health Physics, Waste Disposal, and Safeguards

The operational test of the off-gas system was performed during the period February 23 through March 5, 1966. System performance and individual component characteristics were determined at off-gas system flow rates of 40, 70, and 100 percent of design flow. On the basis of the system operation during the test, it was concluded that: (1) certain revisions in operating instructions are desirable, (2) there are questions regarding accuracy of off-gas flow measurement, (3) periodic calibration of instrumentation is desirable, (4) 100 percent recombiner efficiency can be achieved, (5) charcoal traps did not provide xenon or krypton decontamination of the off-gas, (6) there are questions regarding adequacy of the steam dryer, (7) the fission product monitor may serve as an indicator of off-gas activity, (8) the scrubber provides a measurable degree of particulate cleanup, (9) off-gas system operation at 40 percent flow significantly reduced noncondensable gas concentrations in the primary system, (10) evaporator water box concentrations of hydrogen and oxygen increase during off-gas system operation, (11) desuperheater flow control is unsatisfactory, (12) in the event of fuel element cladding failure, operation of the off-gas system will reduce the resulting particulate activity, and (13) startup of the off-gas system is difficult due to the physical location of valves and instrumentation.

A-3.7.2 Chemistry and Materials

The primary to secondary system leak rate appears to have changed very little during 1966 based upon secondary water and turbine air ejector activity level.

Although some minor variation in fission produce concentrations was observed in the primary water during Core I operation, there was no indication of the presence of a failed fuel element. The slight decrease in the gross iodine level with time is also indicative of burnup of "tramp" uranium.

Primary system water was essentially normal during April except for a conductivity of 2.4 μmhos/cm and possibly a chloride concentration of 0.16 ppm on April 23 following reactor shutdown.

A-3.7.3 Instrumentation and Control

Test procedures were developed for determining the operability of the following newly installed or modified instruments and controls:

(1) The four-rod scram sequence monitor and ground indicator chassis, to be bench tested prior to installation.

(2) The four-rod scram sequence monitor, to be operationally tested after installation in conjunction with the Four-Rod Scram System Reactor Shutdown Test.

(3) The four-rod scram system, to be tested following completion of the modifications to the system.

A-3.7.4 Reactor Plant Engineering

The power escalation tests were performed and an initial results summary prepared.

A stress analysis of the primary system piping between the evaporators and the sub-coolers was performed in anticipation of repositioning the evaporator shells to eliminate condensate entrapment in the tube bundles. A procedure for repositioning the No. 1 evaporator was completed.

A-3.7.5 Task Progress Report

Of the 12 tasks assigned under Subcontract No. 5 during February 1966, two have been completed. Two new tasks, 5.22 Control Rod Maintenance Procedures, and 5.23 Plant Shutdown On-Site Assistance, were added. The status of each task is graphically illustrated by Fig. 20.

A-3.8 ERR OPERATION'S SAFETY ANALYSIS MONTHLY REPORT NO. 11, MAY 1 TO MAY 31, 1966, NUS-292, JULY 25, 1966, 108 p.

A-3.8.1 Health Physics, Waste Disposal, and Safeguards

As a result of the off-gas system performance tests performed in September 1965 and February-March, 1966, revised procedures for operation of the off-gas system have been recommended.

A-3.8.2 Chemistry and Materials

Primary system water chemistry during May was essentially the same as previous months except for periods of high conductivity as a result of reactor shutdown operations.

A-3.8.3 Instrumentation and Control

As an aid in determining reactor reliability as affected by electrical line transients, test procedures were developed. These procedures test the response of the startup rate system following transients on the fail free bus.

A-3.8.4 Reactor Analysis

The rate of boron depletion in the new B_4C regulating control rod was calculated. From this value, the lifetime of this center rod was estimated.

A-3.8.5 Reactor Plant Engineering

A fixture was designed and procedures written for manually lifting the Elk River Reactor control rods without use of the control rod drives.

The effect of swelling of the B_4C in the new control rods was reviewed and it was concluded that the swelling will not limit control rod life.

A detailed investigation was performed to correlate the isolation of containment pipe penetrations with respect to accidents involving rupture of the primary system and subsequent possible breach of the containment boundary.

New operators were installed on the containment ventilation inlet and exhaust lines valves to obtain "double valve protection."

Valve RA-127 was fitted with a more rugged and reliable limit switch. This limit switch actuates a four-rod scram if the valve is inadvertently opened and has provisions for testing during reactor operation.

The stainless steel bolts on the core spray piping flanges were found to be in satisfactory condition. These bolts had previously been replaced due to high stress conditions.

A-3.8.6 Task Progress Report

Of the 14 assigned tasks, three were completed in May. Two other tasks were completed prior to May. The scope, action, and reports submitted during the current month are summarized for the remaining tasks. A graphic presentation of the work in progress and approximate status is depicted by Fig. 12.

A-3.9 J. A. THIE, ERR SAFETY ANALYSIS MONTHLY REPORT, JULY 1965; JAT-65-5, 11 p.

During the month of July, center rod positions were analyzed at 58 Mwt and at 49 Mwt. In both cases, the reactivity loss was found to agree with prior expectations. Curves are presented for the worth of the center rod at full power, with applications to the automatic mode of operation. The feasibility and safety aspects have been investigated for operation of the reactor at up to 20 percent above the current 58-Mwt limit. It is believed that the turbine power capability and/or the available head for primary feedwater return will depend on the equipment limitations encountered before significant safety limits are approached. A stability analysis at 48 Mwt showed no change from the previous characteristics of stable operation. A test is proposed to ascertain the extent to which large spontaneous primary feedwater flow can be produced when secondary feedwater is added to the evaporators. The test would be performed at zero power. The desirability of the test is based on an observed 8-sec reactor period experienced on August 24, 1963 under hot zero power critical conditions; this period is believed to have been the result of spontaneous primary feedwater flow.

A-3.10 J. A. THIE, ERR SAFETY ANALYSIS MONTHLY REPORT, AUGUST 1965, JAT-65-6, 16 p.

The reactivity loss of the core, measured at the end of the August shutdown, was found to be as expected. The behavior of the shutdown countrate has been explained in terms of fission product gammas as well as antimony gammas producing source neutrons. A table is presented which predicts the expected countrate under various conditions in order that detector channel malfunctions may be discovered sooner than has been the case in the past. Short period alarms encountered upon withdrawing one rod while all others were fully inserted are explained and found not to be an intrinsic hazard. Calculations of the burnup of the center rod were updated and indicate that the elongation at the bottom of the rod is now 0.4 percent.

A-3.11 J. A. THIE, ERR SAFETY ANALYSIS MONTHLY REPORT, SEPTEMBER 1965, JAT-65-7, 11 p.

No reactivity anomalies were found during the month of September in monitoring the position of the center rod at full power. A survey of the 1965 operating history of the regulating rod position indicates that it would have been necessary to change ranges every two weeks if automatic control had been desired. To avoid this maintenance operation, it is suggested that the 12-rod bank be used in the future to keep the regulating rod in range. Heat flux calculations are given to show that continued compliance with the technical specifications is possible with various proposed rod patterns. A calibration curve of the 12-rod bank position vs. power is presented for the range of 41 in. to 56 in. A number of safety rules are given which were formulated by a former head of ACRS in his analysis of 14 reactor accidents.

A-3.12 J. A. THIE, ERR SAFETY ANALYSIS MONTHLY REPORT, OCTOBER 1965, JAT-65-8, 19 p.

Continued monitoring of core reactivity changes indicates normal behavior. The center rod is coming out only 0.73 as fast as theoretical predictions, and this infers that the core reactivity life could extend a few thousand megawatt days beyond the 28,000 Mwd predicted by theory. A proposal was evaluated which would use a number of limit switches to define the range of the regulating rod on automatic control. Rod worth measurements were recommended as being needed now. An effect of the control rods on the nuclear instruments N-5, N-7, N-8, and N-9, was evaluated quantitatively. A bank factor curve is presented which shows the effect to be 4 percent or less. Primary feedwater oscillations near 10 Mwe are found to be no different than in the past, indicating no unknown changes in the primary flow circuit. Anomalous behavior of the subcooler radiation levels during October was investigated and found to constitute no hazard to operations in itself.

A-3.13 J. A. THIE, ERR SAFETY ANALYSIS MONTHLY REPORT, NOVEMBER 1965, JAT-65-9, 12 p.

Monitoring of the core reactivity by the center rod position continues to indicate normal behavior. Calculations of the burnup of the bottom of this rod which is exposed to the highest flux are given as a function of core megawatt days. A philosophy of equalizing the exposure among all 13 rods is proposed again. From data obtained in November, the calibration of the center rod at 58 Mw, and slightly below, has been extended up to 28 in.

Since refueling is anticipated sometime in 1966, preparations are currently underway for a test program. A summary of the procedures to be written is presented.

A-3.14 J. A. THIE, ERR SAFETY ANALYSIS MONTHLY REPORT, DECEMBER 1965, JAT-65-10, 12 p.

The center rod at full power continues to maintain its expected position; and it is still believed that the core reactivity life will be a little longer than predicted; namely, 31,000 Mwd. A review of the behavior of the BF_3 detectors during 1965 startups indicates their reliability is too low to permit startup after a scram without first undergoing a thorough debugging. The behavior of the compensated ion chambers following a shutdown from full power was studied in order to establish performance norms for operator use. An apparent reactor power surge of the order of 12 percent lasting only several seconds was observed in November, and on analysis found to be statistically likely from the normally randomly fluctuating power; further, this condition constitutes no hazard. Seven physics test procedures were written for use during the next vessel opening during a shutdown.

A-3.15 J. A. THIE, ERR SAFETY ANALYSIS MONTHLY REPORT, JANUARY 1966, JAT-66-1, 13 p.

The position of the center rod up to about 24,000 Mwd at the end of January continues to be normal. Using an empirical curve for its position vs. Mwd, the end of core reactivity life is predicted to be 30,800 Mwd if present operating policies are maintained. The center rod was calibrated by two methods this month, and the results were used to verify that the automatic control ranges now being used are less in worth than the 75-cent safety limit.

A review was made of the status of the reactor shutdown margin, both now and after the contemplated refueling this year. Because of the important part the quantity of this margin plays in refueling safety, the procedures have been modified in order to obtain additional information about subcritical reactivity states.

A-3.16 J. A. THIE, ERR SAFETY ANALYSIS MONTHLY REPORT, FEBRUARY 1966, JAT-66-2, 11 p.

The program of monitoring the core reactivity loss by means of both zero power critical rod positions and the full power center rod position continues to show expected behavior. The most recent estimate for the end of core reactivity life is 28,700 Mwd. This estimate is a little lower than prior estimates due to a slightly faster withdrawal of the center rod due to burnup found in February. Two instrumentation-induced scrams in February afforded an opportunity to further study difficulties in channels N-1 through N-4 due to high gamma backgrounds. A need to change both BF_3's has been added to prior recommendations as a result. Upon return to full power after a scram, the center rod position may be used as an indication of normal core behavior, and a guide for obtaining expected positions has been prepared. Another calibration of the center rod at its full power position gave essentially the expected result. A safety review of all procedures for the spring rod inspection and refueling shutdown has been conducted.

A-3.17 J. A. THIE, ERR SAFETY ANALYSIS MONTHLY REPORT, MARCH 1966, JAT-66-3, 10 p.

The reactivity loss monitoring program using the full power center rod position continues to show expected behavior. The data still show that additional reactivity life will exist in mid-April. Results of calibrations of the center rod at full power have been used to obtain assurance that the reactivity worths of the automatic control ranges being used are less than a 75-cent safety limitation. The possible sensitivity of the ion chambers used for scram protection to control rod positions in Core II has been anticipated; this is discussed from a safety standpoint. Work is still progressing in regard to shutdown margin monitoring.

A-3.18 J. A. THIE, ERR SAFETY ANALYSIS MONTHLY REPORT, APRIL 1966, JAT-66-4, 21 p.

Reactor shutdown took place at 27,877 Mwd on April 15, with reactivity indications from the center rod positions being as expected. Approximately 1000 Mwd more could have been obtained. Extensive data from all nuclear tests during, and after, the shutdown have been analyzed. The results are reported here. The shutdown margin with only the strongest rod 5 out is $11.40. The core is subcritical with any two rods out, including rod 4 and rod 6, the strongest pair. Improvements in the N-1 and N-2 startup channels have been made; as a result, these channels are reliable safety monitors during this shutdown. Temperature coefficient measurements gave values explainably less negative than earlier in core life. Rod calibrations, with one exception, were as expected. Nuclear data indicated that rod inspections may safely proceed with all the fuel present; but when cracks in the lifting webs were found, the core was completely unloaded to facilitate the more difficult than anticipated rod inspection program. The unloading countrate data agreed with those of the last unloading in November 1964.

A-3.19 J. A. THIE, ERR SAFETY ANALYSIS MONTHLY REPORT, MAY 1966, JAT-66-5, 11 p.

Large reactivities associated with single control rods are discussed, and experimental data are given to show that undesirable rod worths can be avoided. Computer results to assist in making shutdown margin measurements in Core II are given; these are still satisfactory. A theoretical interpretation of April temperature coefficient measurements is given in which an overall negative effect between two widely spaced temperatures is probably the sum of a negative and a positive effect. However this is not regarded as any significant change in intrinsic core safety because of the small numbers involved. A summary of the fuel loading steps is discussed.

A-3.20 J. A. THIE, ERR SAFETY ANALYSIS MONTHLY REPORT, JUNE 1966, JAT-66-6, 18 p.

The reloading of Core II proceeded during June according to expectations anticipated from Core I unloading data. Criticality tests also gave expected results with the exception that the center rod, rather than rod 5, was the strongest. The shutdown margin with the center rod out and all others inserted was measured to be 3.95 percent; large enough so as not to present any safety problems. Other data from the core reloading and criticality tests are given, including measurements indicating the increased strength of the boron carbide rods over the boron-stainless-steel rods they replaced. A number of suggestions to improve future refuelings from an efficiency and safety standpoint are given, based on experience during this refueling.

A statistical study of the occurrence of bowed tubes found during the fuel element inspection was made. It was concluded that the probability of finding a regular (4.3 percent enriched) rod bowed is 0.36 now; significantly higher than the probability during the 1964 inspection. It is also significant that no spikes (5.2 percent enriched) are bowed, and that there is no observable correlation between bowing occurrence and either relative burnup or radial core location.

A-4 REPORTS ON THE SERVICEABILITY OF THE ERR PRESSURE VESSEL

A-4.1 W. A. GUNKEL, C. E. LAUTZENHEISER, A. L. LOWENBERG, AND E. B. NORRIS, EVALUATION OF THE SERVICEABILITY OF THE ELK RIVER REACTOR PRESSURE VESSEL, PROGRESS REPORT NO. 12, SEPTEMBER 1965, SwRI-1228-75, 31 p.

Progress is reported on Phase D-2 of the program to determine the influence of neutron irradiation on the nil-ductility transition temperature and fatigue properties of steels and welds typical of those in the ERR pressure vessel. Preparations are described for conducting tensile, Charpy V, and fatigue tests on irradiated specimens.

Progress is also reported on Phase D-3 of the program to develop equipment and procedures for remote nondestructive testing and the detection and monitoring of gross defects in critical areas of the ERR pressure vessel. It has been decided that the inspection mechanism should consist of a multi-transducer head on the end of a boom for insertion into the vessel and into the nozzle. The motor driven inspection head will be comprised of three ultrasonic transducers positioned to scan the areas of interest.

A-4.2 J. McDONALD AND P. D. WATSON, INVESTIGATION OF THE EFFECTS OF FABRICATION ON THE PROPERTIES OF ERR PRESSURE VESSEL MATERIALS. TOPICAL REPORT NO. 2, MARCH 14, 1966, SwRI 1228-4-17, 67 p.

Investigations have been made to determine the effects of fabrication history on the nil-ductility transition temperatures and the low cycle fatigue strengths of the Elk River Reactor pressure vessel steels.

The probable shell forming procedures for the pressure vessel were simulated for A302 Grade B by cold straining and warm (600 F) straining the material an amount equivalent to forming 3-in.-thick material to a 7-ft dia. Standard Charpy V-notch curves and standard S-N fatigue curves were developed for A302 Grade B material for the various conditions.

Low cycle fatigue curves and Charpy V-notch curves were determined for A105 Grade II material trepanned from the ERR vessel flange and for A105 Grade II material simulating the ERR pressure vessel nozzles.

The fatigue and NDTT curves developed in this phase of the program will be used as standards for the post irradiation tests to be performed on the surveillance specimens.

A-4.3 C. E. LAUTZENHEISER AND E. B. NORRIS, EVALUATION OF THE SERVICEABILITY OF THE ELK RIVER REACTOR PRESSURE VESSEL, QUARTERLY REPORT, SwRI 1228-4-18, APRIL 13, 1966, 9 p.

A quarterly progress report covering the period January 1, 1966 through March 31, 1966. The objective of the technical program is to evaluate the serviceability of the Elk River Reactor pressure vessel by determining the effect of fabrication procedures, irradiation, dissimilar weld metallurgy and geometry on the fatigue life and nil-ductility transition temperature of the completed vessel. In addition, a phase of this investigation is directed toward the development of remote nondestructive testing equipment and techniques for the detection and monitoring of gross defects in critical areas of the vessel.

Progress is reported on Phase D-1, Fabrication History Effects and Fatigue Testing Program. The drafts of two typical reports were submitted to the Commission for approval. Progress on Phase D-2 includes the completion of fabrication of ERR surveillance capsules and plans for testing of surveillance specimens to be removed from the reactor. Phase D-3, Development of Remote Testing Equipment, has progressed through the detailed design of all inspection mechanisms and the receipt of most of the components required.

A-4.4 J. McDONALD AND P. D. WATSON, LOW CYCLE FATIGUE STRENGTHS OF DISSIMILAR WELDMENTS, TOPICAL REPORT NO. 3, SwRI 1228-4-20, MAY 23, 1966, 55 p.

Investigations have been made to determine the low cycle fatigue strength of weldments that simulate the A105 Grade II-to-Type 304 stainless steel dissimilar welds now in service on the Elk River Reactor. The weldments simulated were:

1. 8- and 10-in. nozzle-to-external piping welds
2. 16-in. nozzle-to-cap welds (as welded)
3. 16-in. nozzle-to-cap welds (stress relieved)

Low cycle fatigue curves were developed for all three types at both ambient and elevated temperatures (600 F). In addition, the effects of superimposed fatigue-thermal cycling (simulating reactor startup and shutdown) were determined.

The fatigue curves developed in this phase of the program will be used as standards for the post-irradiation tests to be performed on the dissimilar weld surveillance specimens.

The report states that in no case did a fatigue failure occur in the A105 Grade II-to-stainless steel weld metal fusion zone. The fatigue performance of the dissimilar welds or the failure locations indicates no cause for concern regarding the life of the actual ERR pressure vessel dissimilar welds.

A-4.5 C. E. LAUTZENHEISER, EVALUATION OF THE SERVICEABILITY OF THE ELK RIVER REACTOR PRESSURE VESSEL, QUARTERLY REPORT, APRIL 1, 1966 THROUGH JUNE 30, 1966, AUGUST 1, 1966, 17 p. SwRI 1228-4-22

Progress is reported on phases D-1, D-2, and D-3 of the program. Phase D-1 has been completed. The object of phase D-1 was to determine the effect of the Elk River reactor pressure vessel fabrication history on the fatigue life and nil-ductility transition temperature of the steels used to construct the reactor pressure vessel, and to determine the effect of the ERR pressure vessel dissimilar welds on the fatigue life of the reactor pressure vessel.

The object of phase D-2 is to determine the influence of neutron irradiation on the nil-ductility transition temperature and fatigue properties of steels and welds typical of those in the ERR pressure vessel. Progress is reported on the removal of capsules from the reactor, and the removal of specimens from the capsules. An outline of tests to be performed is also included.

The object of phase D-3 is the development of equipment and procedures for remote non-destructive testing to detect and monitor defects in the critical areas of the ERR pressure vessel. The phase objectives were achieved in that the equipment developed could and, in fact, did perform the required inspection of an 8-in. inlet nozzle and a 10-in. outlet nozzle. Difficulty was encountered in the use of the equipment due to the vessel internals' not being as shown on the "as-built" drawings and also due to the eccentricity of the nozzle bores.

A-5 LIST OF PAPERS PRESENTED AT TECHNICAL MEETINGS

A-5.1 Robert Campbell (RCPA) and Joseph A. Signorelli (NUS), Elk River Reactor Operations, Conference on Reactor Operating Experience, July 28-29, 1965, ANS Transactions, Supplement to Volume 8, Page 31.

A-5.2 E. D. Kendrick (A-C) and J. R. Fisher (A-C), Elk River Lifetime - Predicted and Observed, 1965 Winter Meeting, November 15-18, 1965, ANS Transactions, Volume 8, No. 2, Page 523.

A-5.3 E. D. Kendrick and J. R. Fisher, Adequacy of a One-Group Three-Dimensional Model in a Core-Following Program, 1965 Winter Meeting, November 15-18, 1965, ANS Transaction, Volume 8, No. 2, Page 525.

A-5.4 J. R. Fisher and E. D. Kendrick, Comparison of Measured and Predicted Characteristics of the Elk River Reactor, Second International Thorium Fuel Cycle Symposium, Gatlinburg, Tennessee, May 1966.