Westinghouse Electric Corporation Power Systems

PWR Systems Division
Box 355
Pittsburgh Pennsylvania 15230

January 18, 1979

NS-TMA-2018

Mr. D. B. Vassallo
Assistant Director
Light Water Reactors
Division of Project Management
Office of Nuclear Regulatory Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Vassallo:

Subject: Response to Request for Information to Complete Review of WCAP-8278/8279, "Hydraulic Flow Test of the 17x17 Fuel Assembly"

Enclosed are forty(40) copies of a report entitled, "Surry Unit 1 End of Cycle 3 Onsite Fuel Examination of 17x17 Demonstration Assemblies After Two Cycles of Exposure" dated December, 1978.

This report includes operating history and the results of post irradiation examination of two demonstration assemblies, and is being sent to complete our commitment to provide evidence of satisfactory in-pile performance of the 17x17 fuel assembly as requested by the Staff(1) in the review of WCAP-8278/8279, "Hydraulic Flow Test of the 17x17 Fuel Assembly".(2)

It is our position that this report together with responses to previous NRC requests(3-6) provides sufficient supporting information for the NRC to satisfactorily complete the review of WCAP-8278/8279.

Please note that WCAP-8278/8279 has been a frequent reference in past licensing applications and we plan to continue referencing it in future applications. Therefore, we request to be informed of your review schedule.

Very truly yours,

T. M. Anderson, Manager
Nuclear Safety Department

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Enclosures

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References:

1. October 22, 1975 letter from D. B. Vassallo to C. Eicheldinger.

2. February 28, 1974 letter from R. Salvatori to D. B. Vassallo. NS. RS. 157.

3. September 11, 1974 letter from D. B. Vassallo to R. Salvatori.

4. October 22, 1974 letter from R. Salvatori to D. B. Vassallo. NS. RS. 407.

5. January 23, 1975 letter from D. B. Vassallo to R. Salvatori.

6. May 20, 1975 letter from C. Eicheldinger to D. B. Vassallo. NS. CE. 649.

VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

January 9, 1979

Mr. H. R. Denton, Director
Office of Nuclear Reactor Regulation
Attn: Mr. Albert Schwencer, Chief
Operating Reactors Branch No. 1
Division of Reactor Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Serial No. 017
FR/MLS: mvc
Docket Nos. 50-280
50-281
50-338
50-339
License Nos. DPR-32
DPR-37
NPF-4

Dear Mr. Denton:

TOPICAL REPORT ON VEPCO FLAME MODEL

Enclosed are forty (40) copies of the Vepco topical report VEP-FRD-24, "The Vepco Flame Model."

This report describes a three-dimensional, one energy group, modified diffusion theory calculational model. It is our intention to use the Vepco FLAME model to perform three-dimensional reactor physics analysis in support of reactor startup and cycle operation in the near future and in reload licensing submittals by August 1979. Concerning topical report review fees, Vepco is aware that a fee will be charged for the NRC review of VEP-FRD-24. However, we would anticipate that the NRC review effort would be minimized by the fact that the FLAME computer code has already been approved (reference letter from J. F. Stolz (NRC) to K. E. Suhrke (B&W), dated May 28, 1976), and only our application of this code to the Surry and North Anna Nuclear Power Stations remains to be reviewed.

Should you have any questions on the material in this topical report, please contact us.

Very truly yours,

W. N. Thomas
Vice President
Fuel Resources

Enclosures

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SURRY UNIT 1
END OF CYCLE 3 ONSITE FUEL EXAMINATION
OF 17 x 17 DEMONSTRATION ASSEMBLIES
AFTER TWO CYCLES OF EXPOSURE

Reference Document: WCAP-9139*

*WCAP-9139 was prepared by Westinghouse Electric Corporation as an account of work sponsored by the Electric Power Research Institute Inc. (EPRI).

TABLE OF CONTENTS

1.0 SUMMARY

Two prototype 17 x 17 assemblies were irradiated in the Virginia Electric and Power Company Surry Unit 1 reactor for two cycles under the EPRI/Westinghouse Cooperative Project on PWR Fuel Performance. This report details work performed on the two prototype Surry Unit 1 17 x 17 fuel assemblies after two cycles of operation to obtain performance data, and to assess overall integrity of the 17 x 17 assemblies and the removable fuel rods. The two 17 x 17 assemblies (ND 1 and RD 1) irradiated in the Surry Unit 1 reactor were initially loaded into the core as part of the reload region 4 in January 1975. During the third cycle of reactor operation (second cycle for the 17 x 17 fuel), the assemblies achieved an average cycle burnup level of 8945 MWD/MTU with rod average powers ranging from 4.7 to 5.4 kilowatts per foot, and a cumulative two-cycle burnup of 16,125 MWD/MTU. During the cycle operation of the reactor was normal and coolant radioactivity levels were low which is indicative of excellent overall fuel integrity.

The design of the prototype 17 x 17 fuel assemblies in the Surry Unit 1 reactor is identical to that of the standard 17 x 17 design except that seven spacer grids rather than the standard eight grids are present for compatibility of the fuel assemblies with the rest of the Surry 15 x 15 core. One assembly is of the removable rod design to allow removal and inspection of individual fuel rods in the utility spent fuel storage pool.

The two 17 x 17 assemblies were permanently discharged from the Surry Unit 1 reactor after two cycles of operation (end of reactor cycle 3) in October 1976, and the site examinations were performed in April 1977.

Underwater television examinations of the two 17 x 17 fuel assemblies showed no significant anomalies on the fuel assembly structures, grids, peripheral fuel rods, or holddown springs.

The magnitude of permanent grid cell deflection, or stress relaxation, for the 17 x 17 design was determined from both the breakaway force for withdrawal of removable fuel rods and from the direct measurement of grid cell forces in the removable rod fuel assembly using a stepped pin technique. Both methods gave values in good agreement with each other; grid stress relaxation was determined to range between 67 and 75 percent of the initial condition after two cycles of operation compared with 55 to 65 percent after one cycle. This indicates that the rate of grid spring relaxation with increasing fast neutron exposure was less during the second cycle. There was no evidence of clad fretting on the removable rods even with only about 30 percent of the unirradiated grid spring forces remaining.

2.0 INTRODUCTION

2.1 BACKGROUND

Surry Unit 1 17 x 17 demonstration assemblies, designated ND-1 and RD-1, were inserted into the reactor as part of region 4B in January 1975 for reactor cycle 2. They completed their first irradiation cycle in September 1975. At the end of reactor cycle 2, eight of the removable rods from RD-1 were discharged and replaced by depleted unirradiated UO_2 rods for reactor cycle 3 irradiation. Assemblies ND-1 and RD-1 were reloaded into the core for cycle 3 which began in December 1975 and ended in October 1976. The assemblies were discharged at the end of their second cycle of irradiation as planned.

Assembly RD-1 is of the removable rod design with 88 interior removable-type fuel rods and modifications in adaptor plate, thimble tubes, and thimble plugging devices. The fuel rod design for the prototype 17 x 17 assemblies is identical to the standard 17 x 17 design except for the removable rods which have a modified end plug design to facilitate removal of the rods. As a result, the removable rods are approximately 3/4-inch longer than the standard nonremovable rod.

2.2 OBJECTIVES

The overall 17 x 17 demonstration program objective was to demonstrate the adequacy of the 17 x 17 fuel assembly design under actual operation conditions.

Part of the objectives of the Surry Unit 1 end of cycle 3 onsite examinations were the following:

- To assess the overall integrity of the two 17 x 17 assemblies
- To assess the condition of removable fuel rods and interior grid cells from assembly RD-1

The Surry Unit 1 reactor end of cycle 3 onsite examinations were performed in April 1977, and the major objectives of the program were met.

3.0 SURRY UNIT 1 REACTOR CYCLE 3 OPERATING HISTORY

3.1 GENERAL

This section of the report contains the pertinent operational history for Surry Unit 1 Cycle 3, the two 17 x 17 demonstration fuel assemblies, and the test fuel rods removed from the removable rod assembly RD-1. Cycle 3 was the second cycle of irradiation for the two 17 x 17 assemblies.

3.2 TEST ASSEMBLY LOCATIONS

A core map indicating the symmetrical positions of the two 17 x 17 test assemblies during their second cycle of irradiation in reactor cycle 3 is shown in Figure 1. These assemblies were initially inserted into the core as part of region 4B during the end of cycle 1 refueling in core locations K-5 and F-11, respectively (Figure 2).

3.3 REACTOR POWER HISTORY

The reactor power history for Surry Unit 1 Cycle 3 is shown in Figure 3 as a function of time. Cycle 3 of Surry Unit 1 achieved criticality on December 6, 1975, and shut down on October 17, 1976. The core average cycle 3 burnup was 8945 MWD/MTU for 260 effective full-power days (EFPDs) of operation. The cycle length was 317 calendar days which included approximately 35 days at zero power.

3.4 COOLANT CONDITIONS AND ACTIVITY DATA

The coolant activity for I^{131} and I^{133} was monitored during cycle 3 operation. Figures 4 and 5 show plots of the I^{131} and I^{133} coolant activities (μCi/cc), respectively, during the cycle, and Figure 6 shows the ratio of I^{131} to I^{133}. Reactor power histories are also included on the figures.

Activity levels of I^{131} and I^{133} in the primary coolant are important in core performance follow analyses because they are used as indicators of fuel performance. These two isotopes can leak into the primary coolant system through a breach in the cladding. I^{131} activity quantitatively indicates the condition of fuel clad integrity, and the ratio of I^{131} to I^{133} is used to try to infer the nature of the defect or defects which may have occurred in the fuel cladding.

The data in Figures 4 through 6 indicate excellent fuel integrity for the entire core. The activity of I^{131} during Cycle 3 and Cycle 4 was approximately 0.03 μCi/cc indicating that the 17x17 assemblies, removed at the end of Cycle 3 did not contribute significantly to the activity in Cycle 3.

3.5 UNUSUAL OCCURRENCES DURING CYCLE 3

No unusual events occurred during Surry Unit 1 Cycle 3 which affect the operational history of the fuel (core).

3.6 POWER AND BURNUP HISTORY FOR 17 x 17 TEST ASSEMBLIES

The averaged fast fluence (E> 1.0 mev) accumulated during cycle 2 by the 17 x 17 test assemblies was 1.34×10^{21} n/cm^2. An additional fluence of 1.69×10^{21} n/cm^2 was accumulated during cycle 3 resulting in a total of 3.03×10^{21} n/cm^2 during two cycles of irradiation.

4.0 RESULTS OF THE ONSITE FUEL ASSEMBLY EXAMINATIONS

4.1 GENERAL CONDITION OF 17 x 17 FUEL ASSEMBLIES AND RODS

Assemblies ND-1 and RD-1 were examined by binoculars and by the black and white underwater tv camera. The general appearance of both assemblies was good with no mechanical anomalies on the assembly skeletons, grids, peripheral rods, top and bottom end plugs of rods, or holddown springs. Most of the assembly was covered by a thin layer of crud that appeared to be black through binoculars. This crud appeared on fuel rods, grid straps, and on the top and bottom nozzles. In general, only the bottom end plug region of the peripheral rods appeared to be clean.

4.2 DISCUSSION OF FUEL ASSEMBLY VISUAL EXAMINATIONS

As previously observed on the 17 x 17 assemblies from Surry Units 1 and 2 after the first irradiation cycle, the two 17 x 17 assemblies from Surry Unit 1 were in excellent mechanical condition after two cycles of irradiation with no significant anomalies in any respect.

There was more extensive crud deposition on RD-1 and ND-1 after two cycles of irradiation than previously observed after one cycle of irradiation. Despite the more extensive and thicker crud, there were no indications of unusual clad corrosion that might affect clad integrity or that might indicate a limitation on further irradiation exposure had it been planned.

5.0 RESULTS OF THE ONSITE FUEL ROD/ASSEMBLY EXAMINATIONS

5.1 SCOPE AND RESULTS OF FUEL ROD BREAKAWAY AND WITHDRAWAL FORCE MEASUREMENTS

The breakaway force (force required to initiate upward movement of the fuel rod) and withdrawal force through only the bottom grid were measured for each of the 16 rods removed from fuel assembly RD-1. Safety considerations associated with the tool used to measure rod breakaway force precluded withdrawal of the rods through all grids, and therefore, the measurement of withdrawal force through the other six grids was not made.

Table 1 is a summary of the measured rod breakaway and withdrawal forces obtained on the 16 removable interior rods removed from RD-1. The data have been corrected for load cell bias, for rod weight (5.3 pounds (2404 grams) per rod), and for tool weight and drag.

Figure 7 shows a typical load versus axial position trace for rod removal. The initial load or breakaway force is indicated by the first load peak in the trace. The lower load following immediately after the breakaway point is the withdrawal force through the grid. When the end plug weld passes through the grid there is another (fluctuating) increase in load force. Figure 7 shows that the step decrease in force as the rod exits the bottom grid is clearly evident. The locations at which data were taken are labeled, and the region where the rod end plug passed through the grid, was avoided in tabulating the data.

Table 2 shows the change in breakaway forces for those fuel rods for which preirradiation values of breakaway force were measured. The average grid relaxation (rod D-53 excluded) calculated from the decreases of the irradiated grid breakaway forces was 76.9 percent after two operating cycles compared with 57.9 percent after one cycle of operation. The calculated average spring relaxations are plotted in Figure 8 as a function of rod average fast fluence ($E > 1.0$ mev) for the end of cycle 2 and the end of cycle 3. Figure 8 shows that the rate of grid spring relaxation decreased with increasing fast fluence.

5.2 SCOPE AND RESULTS OF GRID CELL FRICTION FORCE MEASUREMENTS

Following removal of the 16 fuel rods from assembly RD-1, grid cell friction force measurements were made on 12 of the cells. The 12 cells measured were cells that were left vacant after removal of the 16 rods. The other four cells had dummy rods immediately inserted in them to prevent fuel rod cross threading.

A summary of the grid cell friction force data is given in Table 3 which presents the grid-by-grid friction forces for the 12 cells for the largest ((0.3740-inch (0.9500 centimeter)) section of the gage. The data have been corrected for load cell bias.

Figure 9 shows the individual grid friction forces for each grid cell versus the fast fluence (> 1.0 mev) for each specific grid location listed in Table 3. As shown in Figure 9, the fluence at each grid position was relatively uniform and as expected for stress relaxation of Inconel, the grid cell friction force generally tended to decrease as the fast fluence increased.

The grid cell friction forces measured by the smaller diameters of the gage pin, 0.3660 inch (0.9296 centimeter) and 0.3620 inch (0.9195 centimeter), were negligible for most of the midgrids indicating the mid-grid cell sizes were relaxed to 0.3660 inch (0.9296 centimeter) or greater. Based upon measured preirradiation grid cell sizes of 0.3500 inch (0.8890 centimeter), this represents an estimated grid spring relaxation greater than 67 percent over two cycles of operation.

On the basis of the average grid cell friction force of the four midgrids the average grid spring relaxation was calculated to be 72.9 percent at the end of cycle 3 compared with 62.7 percent at the end of cycle 2.

The step pin results for the midgrids also agree well with the overall average of approximately 77 percent relaxation determined from the breakaway force data. This similarity of results would seem to indicate that the reduction in breakaway forces is primarily due to grid spring relaxation and the resulting increase in cell size, rather than to the decrease in rod diameter from clad creepdown.

5.3 SCOPE AND RESULTS OF FUEL ROD TELEVISION EXAMINATIONS

After removal from RD-1 detailed high magnification (approximately 10X) television visual examinations were performed on the eight two-cycle removable rods. The examinations were performed in two passes at the zero-degree and 180-degree orientations with stops, and 360-degree rod rotation and examination at each grid position.

A cursory television visual examination was also performed on the eight one-cycle depleted uranium rods. The examinations were also performed in two passes at zero-degree and 180-degree orientations and approximately 10X magnification, but without stops at the grids. All of the eight two-cycle rods appeared to be in good condition with no defects or significant anomalies.

The appearance of the crud on the removable interior rods was generally similar to observations on peripheral rods (paragraph 4-2).

The grid regions of the removable rods appeared to be in good condition with no evidence of clad fretting or unusual corrosion. At spring-clad contact points there was generally a slight roughening of the clad surface, and often a short longitudinal scratch line (Figure 10) or a small dull white oval spot. At dimple-clad contact points there was generally only a slight oval burnish mark on the clad (Figure 11).

The eight one-cycle depleted uranium rods had crud deposits similar to the two-cycle rods. No anomalies were observed.

Examination of the rods at grid contact positions did not indicate any significant change on the clad surface compared with the end of cycle 2. There was also no indication of end plug weld deterioration on the removable rods. All rods, one-cycle and two-cycle, appeared to be in good condition.

6.0 SUMMARY AND CONCLUSIONS

The objectives of the Surry Unit 1 onsite examination were to assess the overall integrity of the 17 x 17 assemblies and selected removable fuel rods and interior grid cells from the removable rod assembly, RD-1, after two cycles of irradiation. The specific examinations performed included television examinations of assemblies and removable fuel rods, fuel rod breakaway and partial withdrawal force measurements, and grid cell friction force measurements.

Calculations utilizing in-core flux maps, which were taken periodically in the Surry Unit 1 core during cycle 3, showed that assemblies RD-1 and ND-1 achieved an average burnup level of 8945 MWD/MTU and a fast fluence ($E > 1.0$ mev) exposure of 1.69×10^{21} n/cm^2 during their second cycle of irradiation and an accumulated average burnup level of 16,125 MWD/MTU and a fast fluence ($E > 1.0$ mev) exposure of 3.03×10^{21} n/cm^2 for both cycles. Rod average power levels of the eight enriched two-cycle test rods removed from assembly RD-1 varied between 4.7 kilowatts per foot and 5.4 kilowatts per foot during reactor cycle 3 with rod average end-of-life burnups of 16,870 to 17,800 MWD/MTU. The peak local linear power level and end-of-life peak pellet burnup were 5.9 kilowatts per foot and 19,580 MWD/MTU, respectively. Operation of the Surry Unit 1 reactor was normal, and coolant activity (I^{131} and I^{133}) levels indicated excellent overall fuel integrity during reactor cycle 3.

Underwater television examinations of RD-1 and ND-1 showed both assemblies to be in good condition with no signficant anomalies on the assembly skeletons, grids, peripheral fuel rods or holddown springs. Crud deposition on the peripheral fuel rods was thin but more extensive than observed in either Surry Unit 1 or the sister Surry Unit 2 assemblies after one cycle.

The magnitude of grid relaxation for the 17 x 17 design after two cycles of irradiation was determined from both rod breakaway force and grid cell drag force measurements. the breakaway force measurements showed an average grid spring relaxation of 76.9 percent after two cycles. Compared with the 57.9 percent relaxation after one cycle of operation, this indicates that the rate of grid spring relaxation with increasing fast neutron exposure was less during the second cycle. Grid cell friction forces were measured in 12 removable rod grid cell positions in assembly RD-1 with a three-diameter step gage pin. The results indicate that in general the midgrid cell sizes had relaxed to a diameter of at least 0.3660 inch (0.9296 centimeter). Based on measured preirradiation cell sizes of 0.3500 inch (0.9841 centimeter), this represents a grid spring relaxation of 67 percent or greater, consistent with the breakaway forces results. As expected, the two-cycle grid cell friction forces typically decreased with increasing fast neutron exposure. There was no evidence of clad fretting observed on the removable rods despite operation in grid cells with only 30 percent the initial grid spring force remaining.

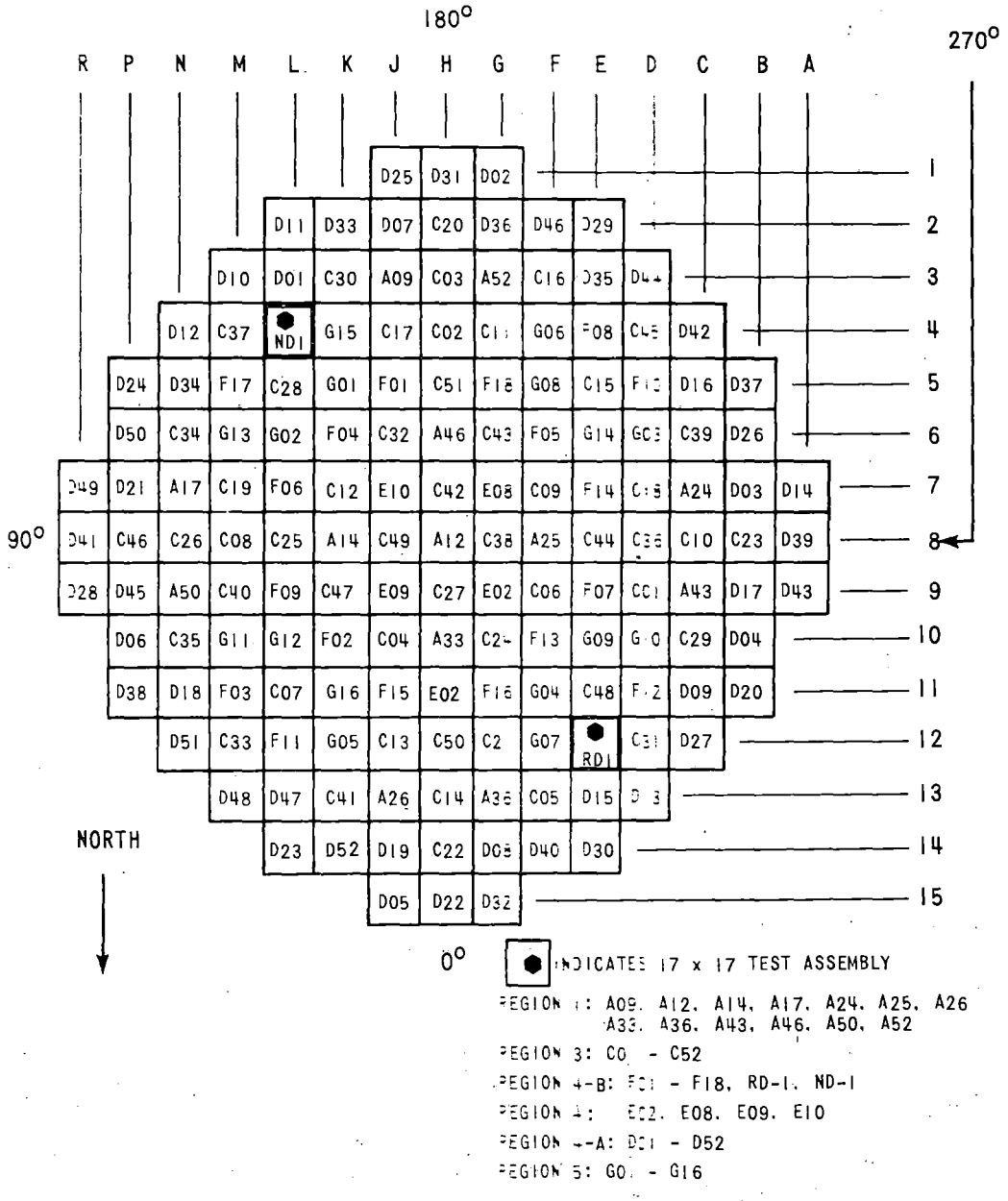

Figure 1 Surry Unit 1 Cycle 3 Reference Loading Pattern

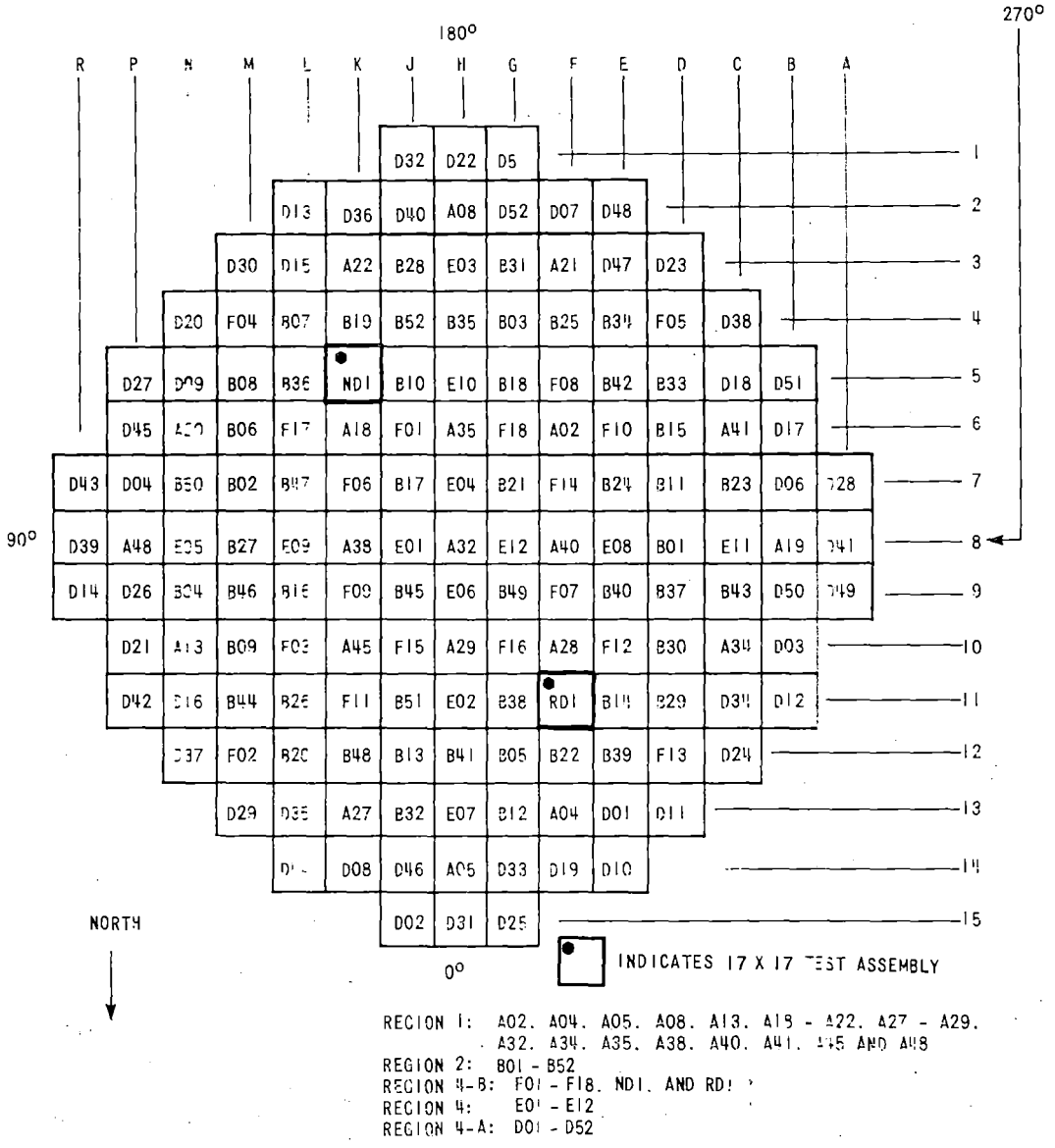

Figure 2 Surry Unit 1 Cycle 2 Reference Loading Pattern

12,089-107

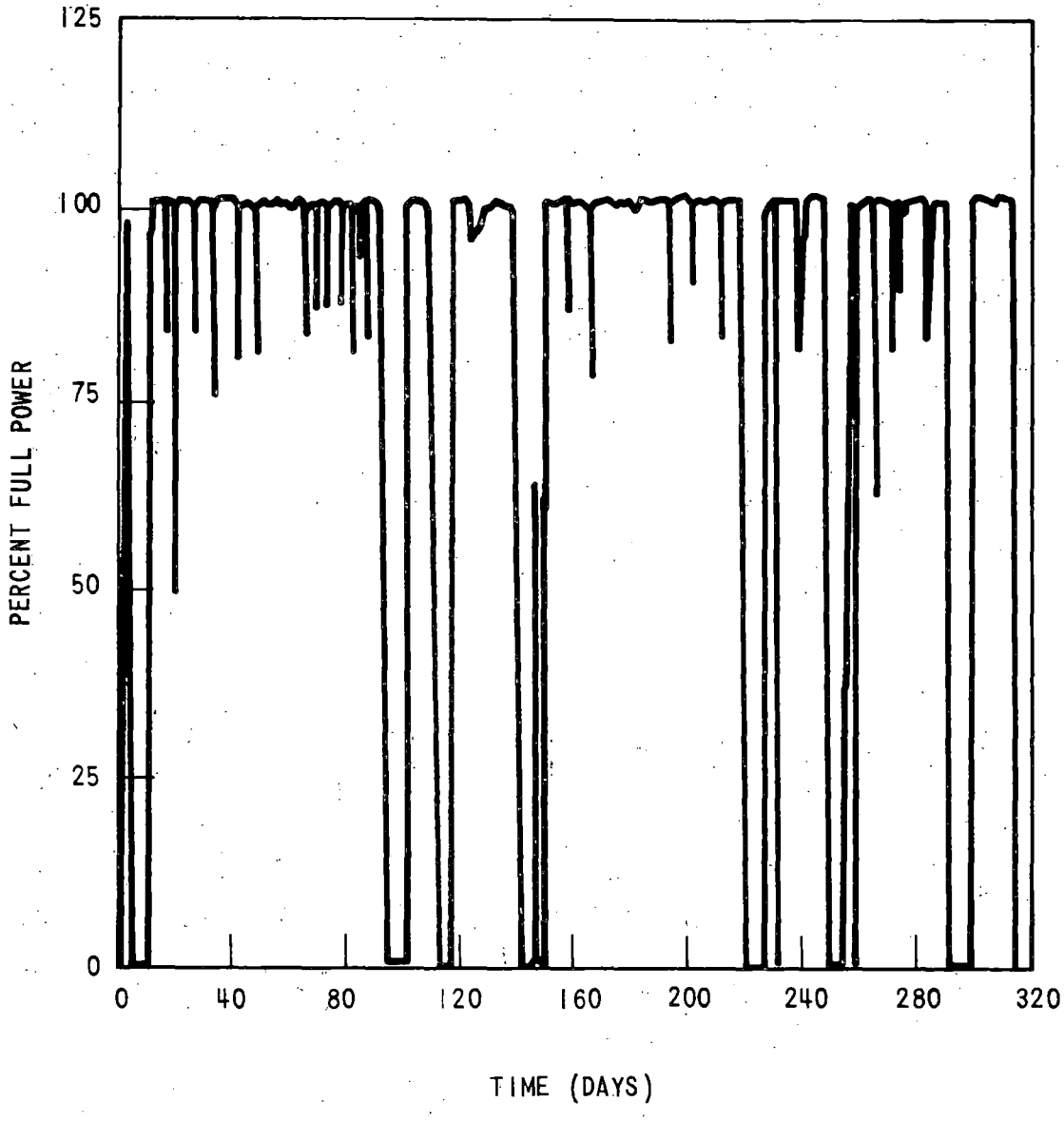

Figure 3 Surry Unit 1 Cycle 3 Power History for Period 12/6/75 to 10/17/76

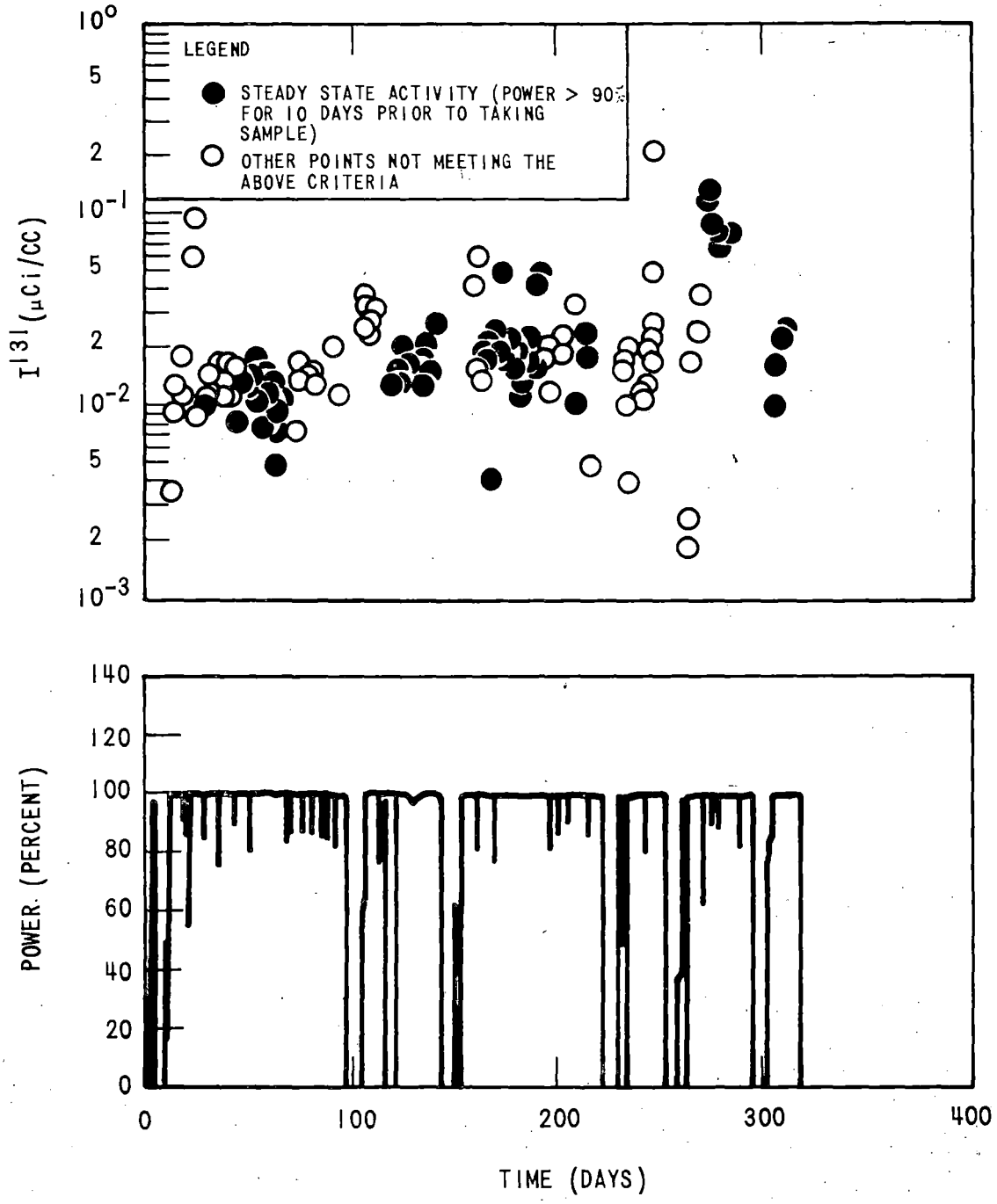

Figure 4 Surry Unit 1 Cycle 3 Coolant Activity (I^{131}) and Reactor Power History

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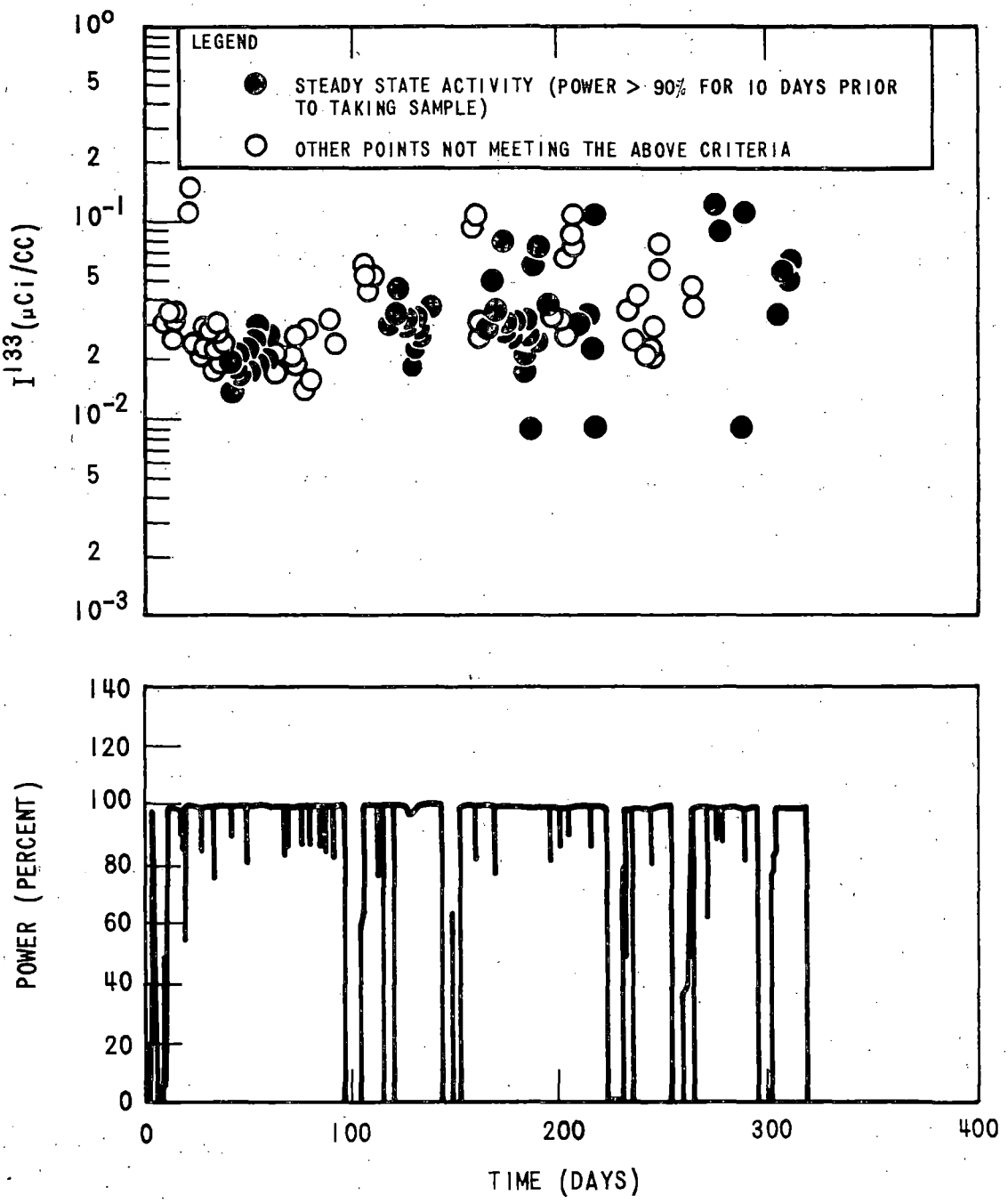

Figure 5 Surry Unit 1 Cycle 3 Coolant Activity (I^{133}) and Reactor Power History

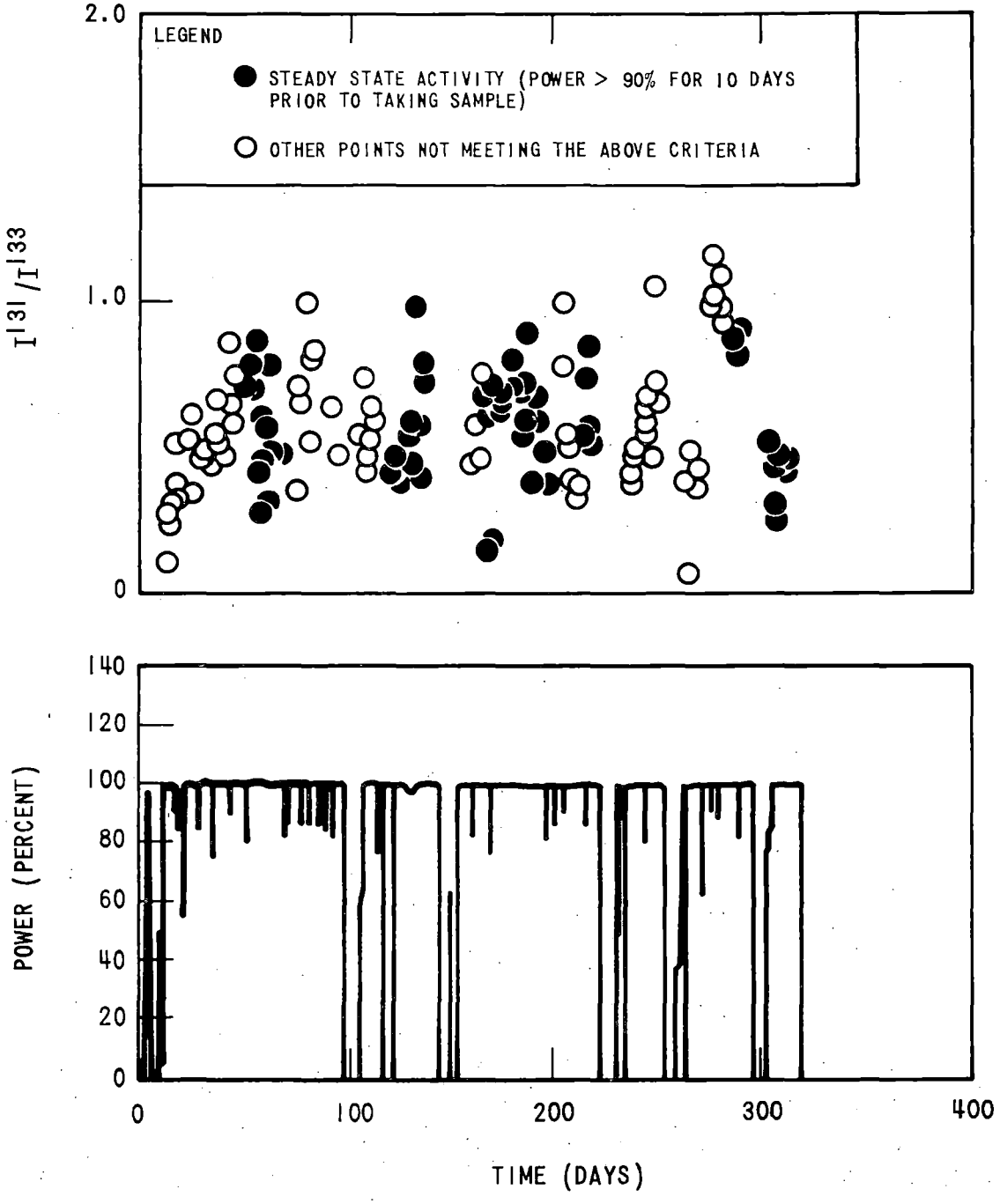

Figure 6 Surry Unit 1 Cycle 3 Coolant Activity Ratio (I^{131}/I^{133}) and Reactor Power History

TABLE 1

17 X 17 REMOVABLE ROD ASSEMBLY RD-1 FUEL ROD BREAKAWAY AND WITHDRAWAL FORCE SUMMARY

Rod No.	Grid Cell	Postirradiation B/A Force (lb)[a]	Withdrawal Force (lb)		
			Avg After Breakaway	Avg After Grid No. 1[b]	Force Through Grid No. 1[c]
114	I-10	16.7	15.7	12.0	3.7
D-54[d]	M-13	14.7	14.5	11.5	3.0
D-51[d]	I-5	16.7	16.3	10.8	5.5
101	I-8	20.3	14.5	12.7	1.8
110	I-13	27.7	18.7	11.8	6.9
103	E-5	18.1	15.5	12.8	2.7
D-53[d]	E-13	33.1	26.2	22.4	3.8
D-57[d]	M-9	18.3	21.5	19.3	2.2
D-59[d]	M-5	16.2	14.5	10.7	3.8
888	E-9	20.3	19.2	15.3	3.9
D-58[d]	H-8	16.8	21.4	16.6	4.8
888	K-13	22.4	19.2	17.2	2.0
888	G-13	24.9	17.1	13.9	3.2
D-56[d]	J-8	14.5	15.5	13.2	2.3
D-52[d]	H-10	20.5	23.1	17.7	5.4
888	M-11	18.9	15.0	9.0	6.0

a. Indicated force minus rod weight (~5.3 pounds)

b. Grid no. 1 is the bottom grid

c. Force through grid no. 1 = (avg force after breakaway) - (avg force after grid no. 1)

d. One cycle depleted uranium rods inserted at end of reactor cycle 2

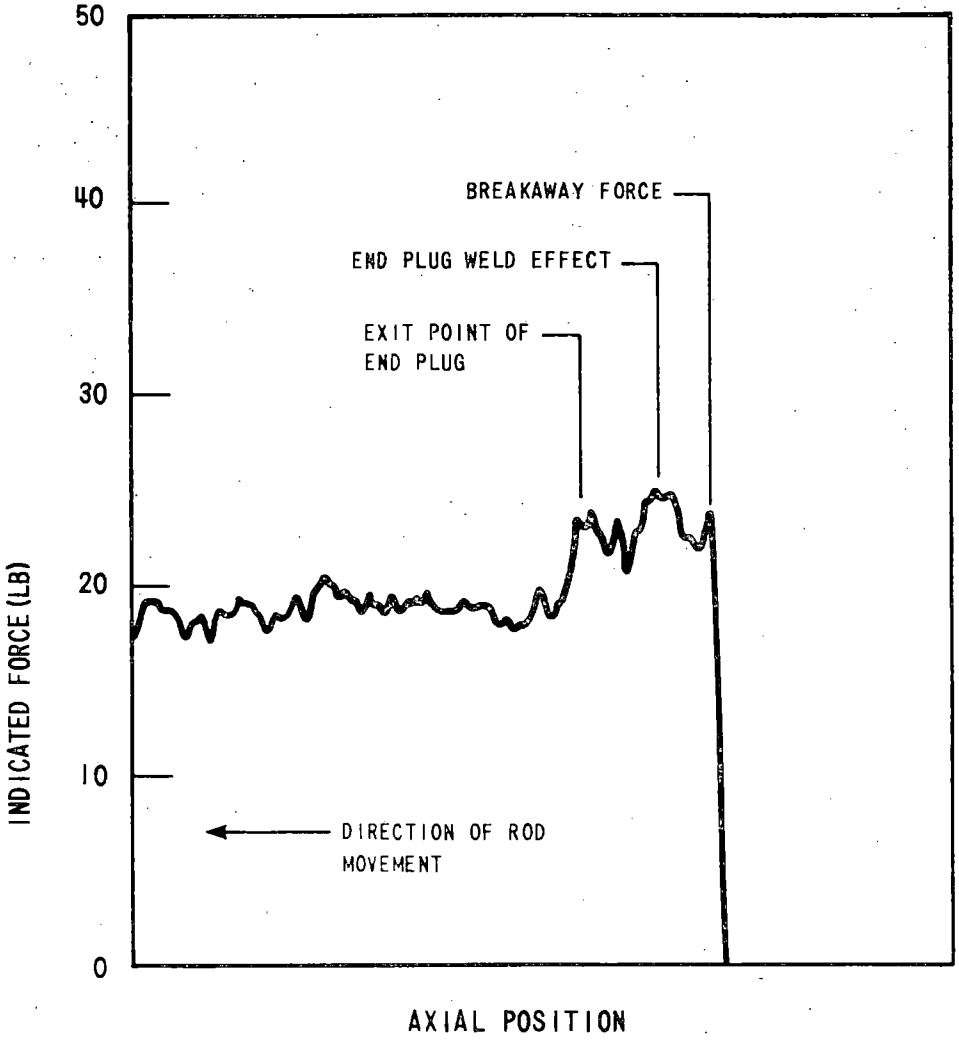

Figure 7 Typical Indicated Load Versus Axial Position Trace for RD-1 Rod D-59, Breakaway and Withdrawal Force, Surry Unit 1, EOC-3

TABLE 2

17 X 17 REMOVABLE ROD ASSEMBLY RD-1 PRE- AND POSTIRRADIATION FUEL ROD BREAKAWAY FORCES[a] AT END OF CYCLES 2 AND 3

Rod No.	Grid Cell	Postirradiation Breakaway Force (lb)		Preirradiation Force (lb)	Decrease in Grid Force (%)	
		EOC-2[b]	EOC-3		Total Reactor Cycle 2 and 3[b]	Reactor Cycle 3 only[c]
114	I-10	31.9	16.7	76	78.0	20.0
D-54[f]	M-13	29.1	14.7	82	82.1	17.6
D-51[f]	I-5	31.1	16.7	[d]	[d]	[d]
101	I-8	36.3	20.3	73	72.2	22.0
110	I-13	32.8	27.7	[d]	[d]	[d]
103	E-5	[d]	18.1	71	74.5	[d]
D-53[f]	E-13	28.1	33.1	73	54.6[e]	[e]
D-57[f]	M-9	32.9	18.3	79	76.8	18.5
D-59[f]	M-5	28.6	16.2	77	78.9	16.1
888	E-9	38.2	20.3	[d]	[d]	[d]
D-58[f]	H-8	29.5	16.8	73	76.9	17.4
888	K-13	35.2	22.4	[d]	[d]	[d]
888	G-13	39.0	24.9	[d]	[d]	[d]
D-56[f]	J-8	29.1	14.5	75	80.7	19.5
D-52[f]	H-10	27.0	20.5	74	72.3	8.8
888	M-11	32.3	18.9	[d]	[d]	[d]

a. All postirradiation force data is indicated force minus rod weight (~ 5.3 pounds).

b. Total percent decrease in grid force =

$$\frac{\text{Preirradiation force - EOC-3 force}}{\text{Preirradiation force}} \times 100$$

c. Reactor cycle 3 decrease in grid force =

$$\frac{\text{EOC-2 force - EOC-3 force}}{\text{Preirradiation force}} \times 100$$

d. No data

e. Anomalous EOC-3 data point

f. One cycle replacement rods

TABLE 3

17 x 17 REMOVABLE ROD ASSEMBLY RD-1 GRID CELL FRICTION FORCE DATA SUMMARY AFTER SECOND CYCLE OF IRRADIATION

Grid Cell	Fuel Rod No.	Test No.	Friction Force (lb)[a]							Accumulated Rod Average Fluence[b] (10^{21} n/cm^2 (E > 1.0 Mev)
			Grid 1	Grid 2	Grid 3	Grid 4	Grid 5	Grid 6	Grid 7	
E-5	103	1	2.4	1.3	1.4	1.3	1.4	1.5	0.8	2.98
		2	2.0	1.9	1.7	1.5	1.4	1.8	0.8	
		3	2.5	2.4	2.1	1.8	1.7	2.0	0.8	
I-13	110	1	2.8	1.8	4.1	1.9	1.4	1.5	0	2.98
		2	2.8	2.2	2.2	2.2	1.9	1.8	0.4	
		3	3.8	2.0	2.0	1.8	1.7	1.8	0	
M-13	D-54	1	2.1	1.1	1.0	1.1	1.2	1.4	0.5	3.03
		2	2.4	1.9	1.4	1.5	1.6	1.9	0.5	
		3	2.9	2.0	1.8	1.9	1.8	1.8	0.6	
I-5	D 51	1	3.8	1.9	2.1	2.1	2.5	2.1	0.7	3.01
		2	4.3	2.2	2.4	2.2	2.5	2.6	1.1	
		3	3.1	2.0	2.5	2.1	2.2	2.3	1.1	
M-5	D-59	1	5.2	2.4	2.2	2.1	2.1	2.5	0.6	3.07
		2	2.6	4.7	2.2	2.1	2.2	4.2	0.7	
		3	3.9	2.6	2.0	2.2	4.6	2.6	1.1	
M-9	D-57	1	4.1	2.0	1.6	4.9	2.5	2.6	2.6	3.04
		2	5.0	2.6	2.0	1.8	2.5	2.7	2.2	
		3	3.9	2.6	2.1	1.9	2.4	2.7	2.6	
E-13	D-53	1	4.8	3.9	3.0	2.9	1.6	2.2	3.0	2.93
		2	5.3	4.9	2.5	3.6	2.4	2.5	3.0	
		3	4.7	5.1	1.9	3.2	1.8	2.5	2.4	
J-8	D-56	1	2.9	1.9	1.3	2.0	1.6	1.8	0.6	3.02
H-8	D-58	1	–	2.5	2.4	2.1	2.0	2.5	4.8	3.01
H-10	D-52	1	4.8	2.5	2.6	2.2	1.3	5.2	4.2	2.99
I 8	101	1	6.2	3.3	3.5	3.1	3.5	3.3	2.5	3.01
I 10	114	1	4.4	2.7	2.9	2.8	2.7	2.2	1.1	3.00

a. Forces corresponding to 0.3740 in. diameter gage pin, corrected for FRHT load cell bias and buoyancy of extension rod and gage pin.

b. Fluence is calculated by integrating the fast flux over the cycle lifetime and multiplying by appropriate constant (~0.26) to obtain the fast fluence for energy levles greater than 1.0 Mev ($nvt = 0.26 \int_0 t\phi, dt$).

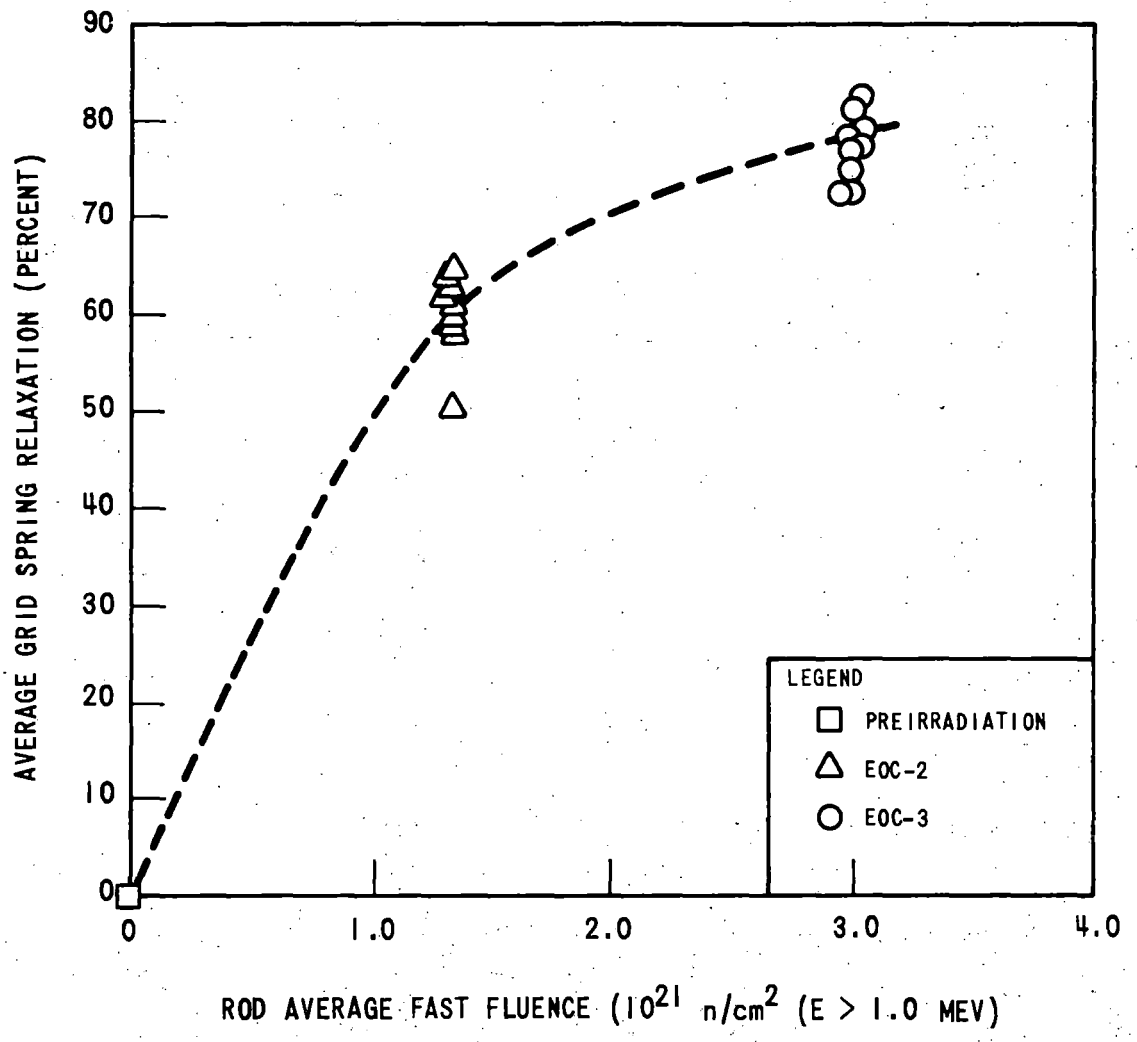

Figure 8 Surry Unit 1 Average Grid Spring Relaxation Versus Rod Average Fast Fluence at EOC-2 and EOC-3 and Prior to Irradiation

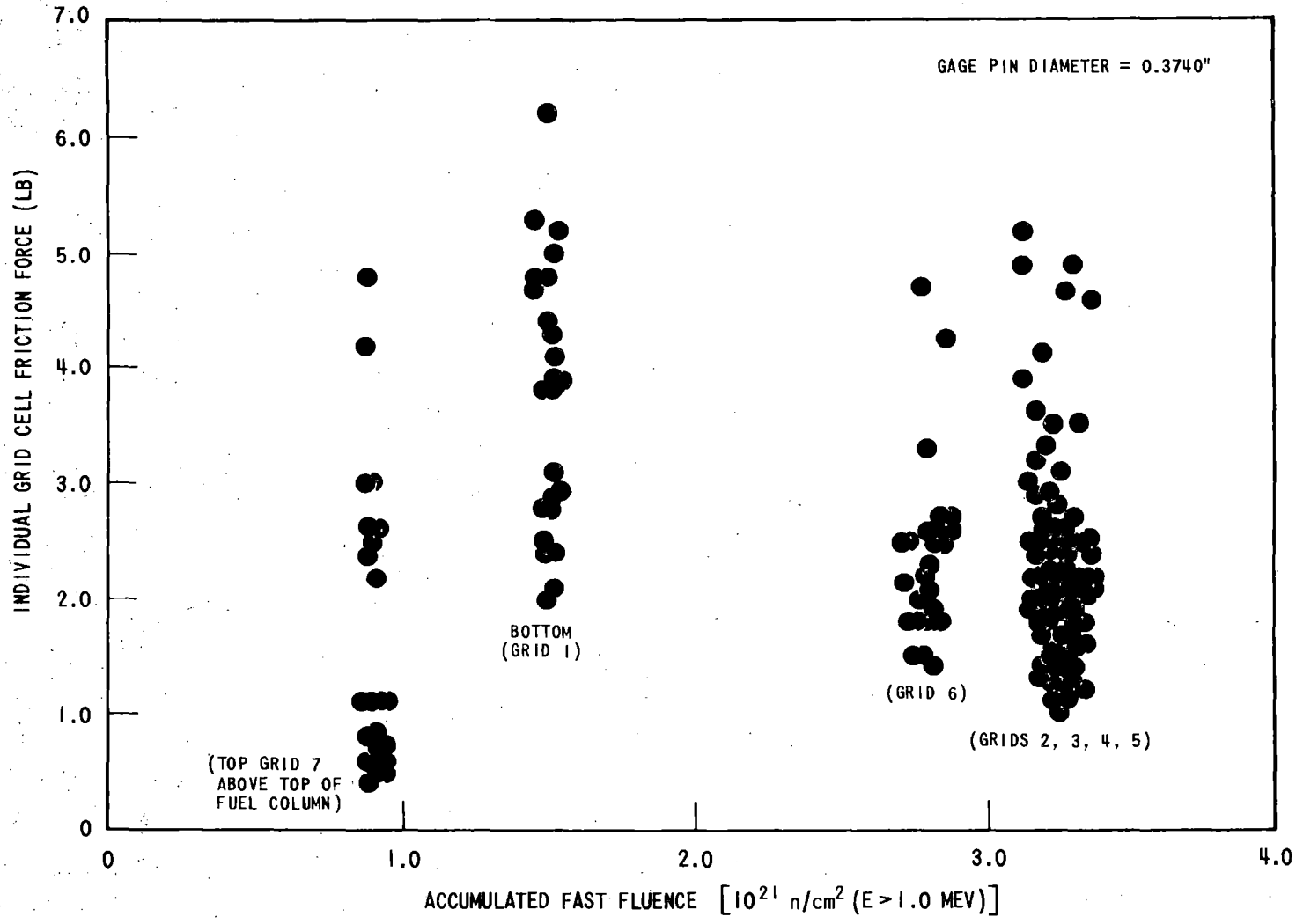

Figure 9 Surry Unit 1 End of Reactor Cycle 3 Grid Cell Friction Force Versus Fast Fluence

12.089-30

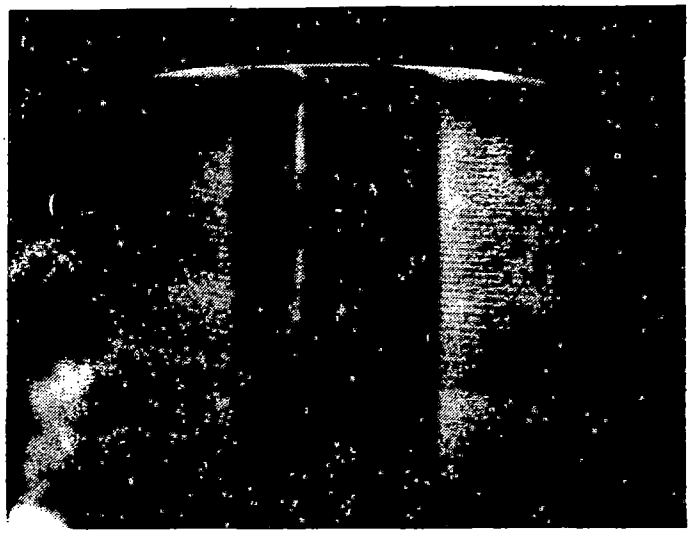

Figure 10 Contact Mark at Spring-Clad Contact Point, Rod 888 (M-11), Grid 1

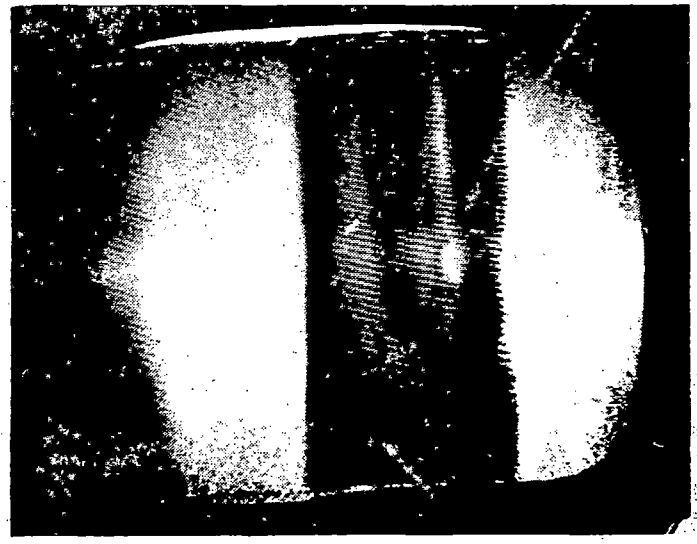

Figure 11 Burnish Mark at Dimple-Clad Contact Point, Rod 888 (E-9), Grid 3