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QUALIFICATION OF THE B&W MARK B FUEL ASSEMBLY FOR HIGH BURNUP

First Semi-Annual Progress Report:
July-December 1978

by

T. A. Coleman
E. J. Coppola
P. L. Doss
V. O. Uotinen
NPGD

H. H. Davis
RDD

Prepared for
The United States Department of Energy
by

BABCOCK & WILCOX
Power Generation Group
Nuclear Power Generation Division
P.O. Box 1260
Lynchburg, Virginia 24505

BABCOCK & WILCOX
Power Generation Group
Research and Development Division
Lynchburg Research Center
Lynchburg, Virginia

Submitted by

BABCOCK & WILCOX
Contract Research Division
P.O. Box 835
Alliance, Ohio

DUKE POWER COMPANY
P.O. Box 33189
Charlotte, North Carolina 28482

eb

Babcock & Wilcox

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ACKNOWLEDGMENT

As with any project or program of this magnitude, the work reported herein is the product of many individuals at The Babcock & Wilcox Company, Duke Power Company, and the Department of Energy. The authors gratefully recognize their help and efforts in the extended burnup program.

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Babcock & Wilcox
Nuclear Power Generation Division
Research and Development Division
Lynchburg, Virginia

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Qualification of B&W Mark B Fuel Assembly for High Burnup — First Semi-Annual Progress Report: July-December 1978

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Key Words: UO_2 Fuel, Fuel Cycle, Burnup, Pressurized Water Reactor

ABSTRACT

Five Babcock & Wilcox standard Mark B (15 × 15) fuel assemblies are being irradiated in Duke Power Company's Oconee Unit 1 reactor under a research and development program sponsored by the U. S. Department of Energy. Valuable experimental data on fuel performance characteristics at burnups of >40,000 MWd/mtU will be obtained from these assemblies. This information, at a duty approximately 20% greater than that achieved by typical discharged assemblies, will be used to qualify standard Mark B fuel assemblies for extended burnups. Extending the burnup obtainable from the fuel offers a large potential near-term improvement in uranium utilization, the amount of energy extracted from a given quantity of yellowcake. In addition, the high burnup fuel performance data will be utilized to design fuel assemblies that are capable of substantially higher burnups than durrent design light water reactor fuel.

This report, covering the period from July through December 1978, is the first semi-annual progress report for the program. Efforts during this period included fuel cycle design and reload licensing of Oconee 1 for cycle 5, in which the assemblies are being irradiated, and nondestructive examination of the assemblies during the refueling outage between cycles 4 and 5. An operating license for cycle 5 was granted on October 23, 1979. The Oconee 1 cycle 5 startup tests proceeded in a routine manner, and the reactor has operated with a 92% capacity factor since completion of power escalation testing on November 10, 1978. Irradiation of the fuel assemblies is currently in progress.

SUMMARY

The United States Department of Energy (DOE) is participating with Duke Power Company and The Babcock & Wilcox Company (B&W) in a high burnup program which is a part of the national effort to improve the utilization of uranium in light water reactors (LWRs) by increasing the amount of energy extracted from each ton of available uranium ore. This joint effort includes an irradiation demonstration phase in which five standard B&W pressurized water reactor (PWR) fuel assemblies are being irradiated in Duke Power Company's Oconee Unit 1 reactor to burnups in excess of 40,000 MWd/mtU.

The goal of the B&W/Duke program is to demonstrate that the discharge burnup of typical PWR current design fuel assemblies can be extended safely from the current batch average limit of ∿33,000 to ∿38,000 MWd/mtU. Such an extension of the burnup limit would result in substantial improvements (∿5%) in uranium utilization in PWRs.

During the last half of 1978, a first-of-a-kind reload licensing effort was carried out for cycle 5 of the Oconee 1 reactor. The core design for this reload cycle included five assemblies that were placed in core locations where they would reach burnups in excess of 40,000 MWd/mtU by the end of the cycle. The presence of these higher burnup assemblies required that special attention be given to ensuring that all reload licensing criteria were met. Nuclear, mechanical, and thermal-hydraulic evaluations indicated that core behavior would be satisfactory. Each Final Safety Analysis Report accident was examined in light of changes in core parameters to ensure that thermal performance during hypothetical transients was not degraded. These evaluations demonstrated the safety of the plant under normal and abnormal conditions, and an operating license was granted by the NRC on October 23, 1978.

During the refueling outage between cycles 4 and 5, the five fuel assemblies designated for extended burnup in cycle 5 were subjected to an extensive series of nondestructive measurements. Measurements were taken of assembly and rod dimensions, water channel spacings, holddown spring preload force,

fuel column axial gap, and stack lengths. In addition, sipping of the fuel assemblies, with subsequent radiochemistry analyses, was conducted. The pool-side measurements established the dimensional and structural integrity of the fuel assemblies prior to their reinsertion for extended burnup. A thorough series of measurements (both nondestructive and destructive) will be carried out on completion of cycle 5. Thus, the measurements performed prior to cycle 5 provide a set of baseline data for determining what irradition-induced changes occurred in the fuel assemblies during their extended burnup residence time in the core.

Oconee 1, cycle 5 startup tests proceeded in a routine manner with completion of power escalation testing on November 10, 1978. Since then the reactor has operated with a 92% capacity factor, and coolant radiochemistry analyses have continued to indicate fuel integrity. At the end of 1978, four of the lead burnup assemblies had accumulated assembly average burnups of 32,800 MWd/mtU. The other assemblly had a burnup of 29,300 MWd/mtU.

CONTENTS

List of Tables

List of Figures

Figures (Cont'd.)

1. INTRODUCTION

A major new constraint was introduced into nuclear fuel cycle considerations as a result of the decision of the U.S. government to defer indefinitely the reprocessing of spent fuel and subsequent deferral of plutonium and uranium recycle. For a number of years the traditional practice in the LWR industry has been to discharge fuel after it has been irradiated for three or four cycles and has achieved a batch average burnup in the 25,000-33,000 MWd/mtU range. The discharge batch average burnup limit of about 33,000 MWd/mtU had been established through economic optimization studies based on the assumption that spent fuel reprocessing would make it possible to reclaim and reuse the residual fissile materials that exist in spent fuel. In addition to representing an economic optimum, this burnup limit of $\sim$33,000 MWd/mtU has been demonstrated over the past decade to be conservative from a mechanical performance standpoint and to give ample assurance of cladding integrity and safe operating performance.

In the absence of reprocessing and recycle, however, conventional LWR fuel management strategies no longer represent optimum approaches. The industry must now assume that residual fissile materials in spent fuel cannot be reclaimed and reused. This currently imposed "once-through" fuel cycle has created an economic incentive to look for ways to minimize uranium requirements in this new mode of operation.

One of the more straightforward and most readily employable means of achieving substantial improvements in uranium utilization in LWRs in the near term is to increase the discharge batch average burnup limit. Engineering projections have indicated that uranium utilization improvements in the 15-20% range can be achieved when the discharge batch average burnup is increased from the current $\sim$30,000 MWd/mtU into the 45,000-50,000 MWd/mtU range. This imporoved uranium utilization results in lower fuel cycle costs and also reduces requirements for spent fuel storage space.

The DOE has initiated research, development, and demonstration efforts involving Duke Power Company, Arkansas Power and Light Company (AP&L), and Babcock & Wilcox, with the goal of demonstrating improved fuel utilization, mainly through the successful operation of PWR fuel assemblies to extended burnups.

The overall fuel utilization improvement effort between B&W/Duke/AP&L/DOE is divided into two separate but interrelated programs. In the program reported here, B&W and Duke are seeking to demonstrate that the batch average burnup limit of current PWR assemblies can be increased safely from ∿33,000 to ∿38,000 MWd/mtU. This program, which does not involve any design changes in current fuel assemblies but rather will extend the current fuel performance data base, will pave the way for the wide-scale implementation of the higher batch average burnups beginning as early as 1981. This burnup extension will allow substantial improvements (∿5%) in uranium utilization to begin to be realized within 2-3 years.

In the second program, B&W and AP&L are undertaking the development and eventual demonstration of an improved fuel assembly (FA) design that will be capable of achieving batch average burnups in the 45,000-50,000 MWd/mtU range. The B&W/AP&L program is a longer term effort, which is expected to lead to full-scale implementation of such higher burnup FAs by the late 1980's with an additional 5 to 10% improvement in fuel utilization.

This report is the first semiannual progress report for the B&W/Duke program, "Qualification of B&W Mark B Fuel Assembly for High Burnup," ET-78-C-02-4711. It covers progress that was made during the July 1, 1978 to December 31, 1978 time period. The report includes work under both of the technical tasks that comprise the program: Task 1 — High Burnup Characterization of the Mark B Fuel Assembly, and Task 2 — Operational Limits of a Mark B Core.

2. PROGRAM SCOPE

Implementation of extended burnup for B&W's current design Mark B (15 × 15) fuel assemblies and realization of the subsequent uranium utilization improvement are the primary objectives of this program. To support these objectives, the program scope as currently structured consists of an initial phase which includes irradiation and examination of five PWR fuel assemblies at burnups in excess of 40,000 MWd/mtU and analytical work to identify and quantify those factors limiting fuel assembly lifetime. An additional phase, covering full batch implementation of extended burnup, is planned. The first phase is divided into two tasks:

1. Task 1 — High burnup characterization of the Mark B fuel assembly.
2. Task 2 — Operational limits of a Mark B core.

The specific objectives of each task, together with a breakdown by subtask and a description of the technical work scope, are presented in the following section.

3. PROGRAM DESCRIPTION

3.1. Task 1 — High Burnup Characterization of the Mark B Fuel Assembly

Objectives

1. Develop a fuel cycle design for Oconee, cycle 5 that irradiates selected FAs to burnups in excess of 38,000 MWd/mtU and meets cycle energy requirements.
2. Characterize the high burnup lead assemblies prior to the fourth cycle of operation.
3. Secure an operating license for Oconee 1, cycle 5.
4. Characterize the high burnup lead assemblies after the fourth cycle of irradiation.

Technical Workscope

1. Subtask 1A — Fuel Cycle Design — Design Oconee 1, cycle 5 to meet standard fuel cycle design criteria. Use the PDQ[1] computer code with two-dimensional (X-Y) quarter-core goemetry and a pin-by-pin representation to model the core. Design cycle 4 to have a lifetime of 235 EFPD and a cycle 5 lifetime of 320 EFPD as specified by Duke Power Company. Select five FAs from Oconee 1 for a fourth cycle of irradiation with a target burnup of 38,000 MWd/mtU.
2. Subtask 1B — Licensing — License Oconee 1, cycle 5 as a standard uranium reload in compliance with the requirements outlined in the USNRC document, "Guidance for Proposed License Amendments Relating to Refueling."
3. Subtask 1C — Poolside Characterization — Using nondestructive measurement techniques and B&W post-irradiation examination equipment installed in the Oconee 1 and 2 spent fuel pool characterize the five fuel assemblies selected for a fourth cycle of irradiation at the end of their third cycle. Include visual examination, gross gamma scans, water channel measurements, dimensional measurements, and measurements of holddown spring load deflection characteristics in the planned scope of the examination.

4. Subtask 1D — Irradiation — Operate Oconee 1, cycle 5 for 320 EFPD.

5. Subtask 1E — Nondestructive Testing (Poolside) — Nondestructively examine the five four-cycle FAs at poolside at the end of cycle 5 of Oconee 1. In the planned workscope include spacer grid spring-load deflection measurements and crud sampling in addition to the measurements carried out in subtask 1C.

6. Subtask 1F — Destructive Testing (Hot Cell) — Ship one of the five four-cycle FAs to B&W's Lynchburg Research Center hot cell facility for further detailed nondestructive and destructive examinations based on the results from subtask 1E above.

3.2. Task 2 — Operational Limits of Mark B Core

Objectives

1. Define the maximum reliable operating limits for the B&W standard Mark B (15 × 15) fuel assembly.
2. Identify the constraints which limit allowable FA and component burnup and quantify the burnup limit.

Technical Workscope

1. Subtask 2A — Fuel Cycle Design — Analyze the feasibility of extended burnup fuel cycles through a series of fuel management studies. Perform fuel cycle calculations using the PDQ diffusion theory code and B&W's standard two-dimensional reactor model to develop fuel bundle arrangements utilizing extended burnup fuel. The fuel cycle calculations will yield enrichment requirements, fuel loadings, power distributions, fuel burnup data, selected control rod worths, and isotopics.

2. Subtask 2B — Nuclear, Mechanical, and Thermal-Hydraulic Analyses — Generate key physics parameters such as moderator temperature and Doppler coefficients for use in subsequent safety and control analyses based on input from subtask 2A. Analyze from a mechanical and thermal-hydraulic standpoint input from subtask 2A. Include evaluations of fuel temperatures, fuel rod internal pressure, cladding creep collapse, fatigue, stress, and strain.

3. Subtask 2C — Safety Analyses and ECCS Evaluation — Assess the relative impact of extended burnup on the plant safety analyses based on the input from subtask 2B. Include group and single rod withdrawals, rod drop, rod ejection, four-pump coastdown, feedwater and steam line breaks, and anticipated

transients without scram. Evaluate dose rates for extended burnup fuel. Develop loss-of-coolant-accident (LOCA) limits for extended burnup fuel. Identify the worst time in life for LOCA and generate specific linear heat rate limits as a function of height in the core.

4. Subtask 2D — Control Analyses — Conduct soluble boron shutdown analyses to determine the effects of changes in the physics characteristics of extended burnup fuel cycles due to the higher core average burnup, and changes in the isotopic composition of the core on boron requirements. Review load change capability and control scheme requirements.

5. Subtask 2E — System Evaluation — Review the results from subtask 2D above to ensure compatibility with the capabilities of the auxiliary fluid and chemical addition systems.

4. Task 1 — HIGH BURNUP CHARACTERIZATION OF THE MARK B FUEL ASSEMBLY — PROGRESS TO DATE

4.1. Introduction

The major emphasis during this reporting period was on the fuel cycle design and licensing of Oconee 1, cycle 5 containing the high burnup lead assemblies (batch 4D) and the poolside characterization of these assemblies at burnups of approximately 30,600 MWd/mtU prior to the start of the fourth cycle of irradiation. On completion of Oconee 1, cycle 5, the batch 4D fuel assemblies (FAs) will be discharged and nondestructively examined. One assembly will be selected for additional detailed nondestructive and destructive tests in B&W's Lynchburg Research Center hot cell facility.

4.2. Subtask 1A — Fuel Cycle Design

A fuel cycle design was developed for Oconee 1, cycle 5 which satisfied the design criteria (see Table 4-1) and met energy extraction requirements while attaining burnups in excess of 40,000 MWd/mtU on selected lead assemblies. The PDQ[1] computer code with two-dimensional (X-Y) quarter-core geometry and a pin-by-pin representation (B&W's standard design model) was used for the fuel cycle design. The design result was a cycle with the following characteristics:

1. An out-in shuffle scheme — The fuel shuffle scheme and batch designations for cycle 5 are shown in Figure 4-1. The feed batch of 56 assemblies of 3.02 wt % ^{235}U was loaded primarily on the core periphery. A mixture of once-, twice-, and thrice-burned assemblies was loaded in the interior of the core. Significant parameters associated with the fuel cycle depletion are summarized in Table 4-2.

2. 14% margin-to-design radial pin peaking limit — Calculated assembly relative powers at the beginning and end of cycle 5 are shown in Figure 4-2. The fresh fuel assemblies located in H-4, H-12, D-8, and N-8 display the highest powers. The peaking in these assemblies is suppressed by the

adjacent batch 4D assemblies, and the batch 4D fuel is in turn driven to high burnups by the proximity of the fresh fuel.

3. Batch 4D assembly burnups in excess of 40,000 MWd/mtU — The assembly burnups given by PDQ calculations for cycle 5 of Oconee 1 are presented in Figure 4-3 for the beginning and end of cycle. The maximum assembly burnup at the beginning of cycle 5 is 31,185 MWd/mtU for the four assemblies in core positions H-5, H-11, E-8, and M-8; which are four of the batch 4D fuel assemblies being irradiated for a fourth cycle. A burnup of 40,395 MWd/mtU is projected for these assemblies at the end of cycle 5. They will be the maximum burnup assemblies in the core.

Table 4-3 compares the core physics parameters for cycle 5 with those of cycle 4. The values for both cycles were generated using PDQ. The power deficits from hot zero power (HZP) to hot full power (HFP) differ from those for the design cycle 4 because of the burnup difference. The differential boron worths and total xenon worths for cycle 5 are greater than or equal to those for the design cycle 4 because of fuel depletion and the associated buildup of fission products.

Figure 4-4 identifies the control rods by group, number of rods in each group, group location, and function. Control rod groups are withdrawn according to the numbering sequence of the banks shown in Figure 4-4.

Control rod bank worths were calculated using the PDQ code in two-dimensional quarter-core geometry. Rod worth calculations were performed at HZP and HFP conditions at the beginning and end of cycle 5. Table 4-4 summarizes control rod group worths.

A two-dimensional full core PDQ model was used to determine the value of the maximum worth stuck rod. The results are given in Table 4-5. Verification of shutdown margin assuming maximum worth stuck rod conditions is shown in Table 4-6.

Additional details concerning the design and licensing of Oconee 1, cycle 5 are available in references 2 and 3. The fuel cycle design for Oconee 1, cycle 5 met and exceeded the design goal of 38,000 MWd/mtU burnup on selected assemblies and satisfied all applicable design criteria.

4.3. Subtask 1B — Licensing

Oconee 1, cycle 5 was licensed in compliance with the requirements outlined in the USNRC document "Guidance for Proposed License Amendments Relating to Refueling."

In addition to the nuclear considerations discussed in section 4.2, analyses were performed to address the mechanical and thermal-hydraulic behavior of the core during cycle 5. The mechanical evaluations included FA growth, fuel rod growth, holddown spring relaxation, fuel rod creep collapse, and cladding stress and strain. A FA growth analysis for the batch 4D assemblies using growth data obtained from the poolside examination demonstrated sufficient margin for FA burnups in excess of 48,000 MWd/mtU. Adequate expansion space was shown to be available to accommodate fuel rod growth for these assemblies. Evaluation of holddown force for the batch 4D FAs showed that safe holddown forces are maintained. The guide tube growth which causes additional spring compression adequately compensates for the decrease in holddown force which occurs due to plastic yielding of the spring under cold shutdown conditions.

Creep collapse analyses for batch 4D gave a cladding collapse time in excess of 30,000 effective full power hours (EFPH) which greatly exceeds the maximum projected batch 4D exposure life of 28,469 EFPH. A conservative fuel rod stress analysis showed a margin in excess of 30%. Since the batch 4D fuel is within maximum design local pellet burnup and heat generation rate, cladding strain is within design limits.

Core thermal-hydraulic performance was not impacted by the batch 4D FAs since these assemblies are hydraulically similar to all other FAs in the cycle 5 core. The potential effect of fuel rod bow on DNBR was considered by incorporating suitable margins into the core safety limits and reactor protection system setpoints. Fuel rod internal pressure and temperature were calculated for the batch 4D fuel based on the actual power histories of these assemblies for their first three cycles of operation, and a conservative power history for their fourth cycle. An acceptable combination of fuel temperature and pin pressure was verified.

The radiological consequences of the fuel handling accident are directly dependent on the fission product inventory within a single fuel assembly. Since the five batch 4D FAs will receive higher burnup than normal, an evaluation was performed to determine whether or not the FSAR treatment of the radiological consequences of the fuel handling accident represents a conservative analysis for each of these five high burnup fuel assemblies. The dose results of the evaluation confirm that the radiological consequences of the fuel handling accidents are bounded by the FSAR analysis. Although these high burnup assemblies received extended burnup, the FSAR doses are larger because the FSAR fuel handling accident analysis assumed a larger assembly power than the actual assembly power for the high burnup assemblies. The 2-hour thyroid dose for the worst case FA is 0.29 rem which is a factor of 1.48 smaller than the corresponding FSAR thyroid dose. The 2-hour whole body dose for the worst case assembly is 0.013 rem which is a factor of 2 smaller than the FSAR whole body dose.

Each FSAR accident analysis was examined in light of changes in core parameters to determine the effects of the cycle 5 reload and to ensure that thermal performance during hypothetical transients is not degraded. Table 4-7 presents the key accident analysis parameters from the FSAR and cycle 5. Based on comparison of cycle 5 core thermal and kinetics properties with acceptable previous cycle values, the cycle 5 reload does not adversely affect the Oconee 1 plant's ability to operate safely.

4.4. Subtask 1C — Poolside Characterization (Non-Destructive Exam)

4.4.1. Introduction

The characterization of the five FAs designated for extended burnup was performed at the Oconee site in the Unit 1 and 2 spent fuel pool. No characterization measurements had been made on these assemblies during their previous two irradiation cycles. Therefore, the purpose of this subtask was to provide a baseline data set against which the fuel performance during the fourth irradiation cycle could be compared. The poolside measurements were completed during the fuel shuffle period (September 8-18, 1978) of the fourth refueling outage for the Oconee Unit 1 reactor.

4.4.2. Poolside Operations

The poolside examination consisted of two basic operations: (1) measurement of individual performance parameters, and (2) sipping of FAs to determine fuel rod integrity. Only sound (non-leaking) FAs were to be utilized for a fourth cycle of irradiation. The individual performance parameters were measured using the B&W post-irradiation examination (PIE) equipment described in section 4.4.3. Sipping was performed by the Nuclear Assurance Corporation with a two-can sipping system installed in the Oconee 1 and 2 spent fuel pool. Since fuel sipping was utilized to verify the acceptability of an assembly for incorporation in the high burnup program, it was planned to conduct these tests prior to characterizing the FAs for extended burnup. However, in order to expedite the completion of testing during the short time of the refueling outage, it was necessary to begin the PIE work on two of the assemblies prior to sipping. These assemblies were subsequently declared sound by the sipping test.

4.4.3. On-Site Measurement Techniques

Figure 4-5 shows the PIE test equipment in the spent fuel pool during FA examination. The heart of the PIE equipment system is the line scan tester (LST), which consists of a strong back frame and various test heads for diameter profilometry, assembly bow, and water channel spacing measurements. An assembly is positioned in the LST with its lower end fitting resting in a support socket at the bottom of the LST and its upper end fitting restrained within the confines of a cylinder at the top. The bottom of the LST is about 26 feet below the surface of the water. The bottom support socket is a turntable, permitting FA rotation in the LST so that any face of the assembly is accessible to the measuring heads. The measuring heads are mounted on a cart which travels vertically to scan the axial profile of the assembly. The measuring head for diameter profilometry is calibrated prior to beginning data collection. The probe used for water channel spacing measurements is also calibrated before each pass through the assembly. Thus, highly accurate and precise data on profilometry and water channel spacings are obtained. Precision for these systems is estimated to be ±0.000104 and ±0.00007 inch, respectively, for the two systems at the 1σ level. Controls for measuring heads and the data acquisition packages are located topside at the spent fuel pool. Figure 4-6 is a photograph of this area while work is in progress.

Holddown spring measurements are performed while the assembly is resting in the LST. The holddown spring tester is pole-mounted and attached to the top of the assembly and operated from poolside. The tester consists of a grapple-like fixture which looks onto the FA upper end fitting. A hydraulic cylinder mounted inside the fixture is used to apply force directly to the holddown spring. The displacement of the spring is transmitted to a dial micrometer, thus correlating load-to-spring displacement.

In the lower right of Figure 4-5 is a second FA under test. This assembly is suspended from the fuel handling hoist and is being lowered past the gamma radiation detector to obtain a gross gamma scan of an assembly corner rod. Duplicate scans on each rod are obtained as the assembly is first lowered and then raised past the detector. After the assembly is raised out of the alignment bracket, it is rotated 90° and another corner rod is scanned in the same manner. Figure 4-7 is a schematic illustration showing the precise alignment of the gamma scan system to detect the gamma radiation from only the corner rod of the assembly. Spring loaded guide rollers maintain this alignment during the lowering and lifting motions.

The gamma radiation (E_γ > 0.2 MeV) is measured with a NaI detector. The detector, along with a preamplifier, is housed in a hermetically sealed cask which is positioned just above the fuel storage rack. The front portion of the cask is a shielded adjustable collimator. For this program, the slit opening (vertical) was set at 0.033 inch, which essentially determines the gap size resolution for the system. The fuel assembly's rate of travel during a scan, which also effects the resolution, is established by controlling the speed of the variable speed hoist. Downscans were measured at 16 in./min., while upscans were taken at 6 in./min. Of these two scans, the upscan is the more accurate since it is taken at the slower speed and the hoist motion is more reproducible in the up direction. Therefore, the upscans are generally used for analysis while the downscans are primarily used to establish the signal processing parameters. Gap size measurement precision at the slower scanning speed of 6 in./min. is approximately 0.05 inch, and stack length precision is approximately 0.15 inch (based on 0.033-inch collimator slit height).

The fuel handling hoist is used to transfer FAs from the spent fuel pool storage racks into the LST for testing and to move the assemblies vertically in the gamma scan system. In addition, visual examinations of assemblies are conducted with the assembly suspended from the hoist. Two methods are used for visual inspections: First, an underwater television camera (shown in Figure 4-5) which is mounted on a pan/tilt mechanism at the bottom end of a telescoping shaft is used to scan an entire face of an assembly while it hangs from the hoist. The appearance of each assembly face along its full length is recorded on videotape. The second method of visual examination utilizes an underwater periscope. The assembly is lowered past the periscope and 35 mm color photographs are taken of any areas of interest along the assembly length.

With the assembly suspended by the hoist, a steel tape is attached to the bottom of the assembly and tensioned along its length using a constant force. Relative locations of the tops and bottoms of fuel rods, assembly end fittings, and spacer grid locations are recorded using either the TV camera or the periscope. These measurements provide a base for determining fuel rod and assembly growth.

4.4.4. Results and Discussion

A preliminary qualitative assessment of the results from the high burnup assemblies (1D13, 1D26, 1D42, 1D45, 1D55) indicates that the assemblies have performed well through three cycles of irradiation and that this behavior should continue through the fourth irradiation cycle. A comprehensive analysis of the data from these measurements is currently being performed. A discussion of the results to date for each of the tests is given below.

Sipping

During the end of the cycle 4 refueling outage 57 batch 4 FAs were sipped. Of these, four were determined to be leakers based on their cesium-137 fission product release rate. The distribution of cesium-137 activities for batch 4 is shown in Figure 4-8.

A two-can sipping system installed in the Oconee 1 and 2 spent fuel pool was used for the sipping operations. The procedure utilized was to load an assembly into one of the cans, flush the can with demineralized water, inject air to displace a small amount of water at the top of the assembly, depressurize the can atmosphere for a fixed time, pressurize the can, and then collect a gas sample. After venting the remaining gas, water was circulated in the can for a period, at the end of which, a coolant sample was obtained. The can was opened, the assembly removed, and the procedure repeated on subsequent assemblies. The cesium-137 activities of coolant samples for each assembly were counted with a Ge(Li) diode detector and multichannel analyzer system.

One of the assemblies (1D54), which had initially been designated for reinsertion in cycle 5 of Oconee 1, was identified as a leaker and replaced with a sister assembly (1D55) that had the same previous incore history. Of the four batch 4 assemblies which were identified as leakers, three had the same previous incore histories as the batch 4D FAs that were reinserted in cycle 5 for extended burnup. Investigation of the cause of these leakers is to be pursued as an additional part of the extended burnup program.

Visual Inspections

Visual examination of the FAs showed no significant observable defects. Crud (corrosion product) deposit patterns observed during the visual inspections were typical of patterns commonly seen on PWR fuel. The deposit patterns included (1) areas of uniform color, (2) crud exfoliation, and (3) "banding" at pellet interfaces. Figure 4-9 shows photographs of two areas from assembly 1D13. Note the variation in crud patterns in different areas and the crud bands occurring at a frequency corresponding to the fuel pellet length.

A cladding score was observed on fuel rod A13 below grid 3 of assembly 1D21. The sipping results showed 1D21 to be a sound bundle. Nevertheless, it was replaced in the program by assembly 1D26 because of the unexplained nature of the score.

Dimensional Measurements

Line scan diameter profiles were obtained on a number of peripheral rods for each assembly (see Table 4-8). Orthogonal scans on corner rods were obtained in each case. Figures 4-10 through 4-14 show the orthogonal scans for the four corner rods on each of the batch 4D assemblies. The variations in the two traces for a given rod result from the ovality in the fuel rod. The traces show that the average diameter varies as a function of axial location. Quantitative evaluations of these results are in progress. The overall average creepdown has been shown to be less than expected, about 3.9 mils.

Assembly and rod length measurements were carried out on all four faces of each of the five assemblies. In addition, assembly length measurements were performed on five additional sister assemblies to enlarge the data base on this parameter. Preliminary results indicate that fuel rod growths were approximately 0.5% versus 0.35% $\Delta L/L$ for the guide tube growth at the 31,000 MWd/mtU point. Figures 4-15 and 4-16 present the preliminary fuel rod and fuel assembly (guide tube) growth data, respectively. Fuel assembly growth was within design limits.

Water Channel Spacings

The separation distance (channel spacing) between adjacent fuel rods is measured using strain-gaged leaf springs on a long wand, referred to as a Sulo probe. Figure 4-17 displays the water channel spacing measurement technique. The Sulo probe is inserted through the FA at each channel and every mid-plane level between spacer grids. Repeating the measurements on an adjacent face of the FA results in a full characterization of the rod-to-rod spacings of the assembly. Since the water channel spacings vary with fuel rod bow, the spacing distribution is a measure of rod bow. The values obtained indicate an average gap closure of less than 30% for 95% of the water channels. This gap closure is conservative relative to the gap closure that is assumed for licensing. The axial distribution based on preliminary rod bow data is shown in Figure 4-18.

Holddown Spring

Load-deflection curves for the holddown spring were obtained on each of the five assemblies; Figures 4-19 through 4-23 present these data. Values for the spring preload for the five assemblies averaged about 376 pounds. This is about a 35% decrease from the as-fabricated nominal values. However, since the FA growth results in an increased spring deflection, the net FA holddown force during operating conditions remains essentially unchanged from beginning of life. Thus, the spring holddown force is expected to remain relatively constant with increasing fluence throughout the operating lifetime of the fuel assembly.

Gross Gamma Scans

Gross gamma scans were obtained for all four corner rods on each of the assemblies shown in Figures 4-24 through 4-28. Values for the fuel column length and the location and size of gaps for each rod were obtained. A total of only eight gaps were found in the 20 rods scanned and all of these were <0.1 inch in size. Thus, no significant gaps were observed. The locations of these gaps and the fuel column lengths will be used for comparison of corresponding values at the end of the irradition exposure for these assemblies.

In conclusion, valuable baseline data on the material and structural performance of FAs at burnups of approximately 31,000 MWd/mtU were obtained prior to the insertion of five FAs in Oconee 1, cycle 5 for a fourth cycle of irradiation. Preliminary evaluations of these fuel assemblies' characteristics through three cycles of irradiation indicate that fuel performance was as expected and within design limits. Continued good FA behavior through four cycles of irradiation is anticipated.

4.5. Subtask 1D — Irradiation

An operating license for Oconee 1, cycle 5 was granted by the Nuclear Regulatory Commission on October 23, 1978. Startup tests proceeded in a routine manner with completion of power escalation testing on November 10, 1978. From November 10, 1978 to December 31, 1978 (the end of this reporting period), the reactor had a 92% capacity factor. A cycle burnup of 56 effective full power days (EFPD) was achieved. The coincidental burnups on the batch 4D FAs were 29,300 MWd/mtU for the center assembly and 32,800 MWd/mtU for the four assemblies located on the core axes.

Core follow results for the Oconee 1 extended burnup assemblies exhibit good agreement between measurements and calculations. Figure 4-29 presents an eighth-core power distribution comparison of measured assembly powers from December 18, 1978 at 44 EFPD to 50 EFPD PDQ-calculated assembly powers. The measured power distribution was obtained from rhodium incore detector signals. The measured power for the center assembly (H-8) agrees with PDQ within 2.7% difference. For the other batch 4D location (H-11), the measured and calculated assembly powers are essentially the same.

Analyses of radioiodine activities in the reactor coolant indicate no change in the batch 4D fuel integrity since the beginning of cycle 5.

A projection of Oconee 1, cycle 5 operation at a 90% capacity factor from December 31, 1978 to the design cycle lifetime of 320 EFPD gives an end of cycle 5 date of October 20, 1979. Therefore, batch 4D discharge should occur in the fourth quarter of 1979.

Table 4-1. Oconee 1, Cycle 5 Design Criteria

Cycle lifetime, EFPD	320
Standard design methods	
Standard nuclear, thermal, and mechanical design limits for 2568-MWt plant	
Batch 4D burnup, MWd/mtU	>38,000

Table 4-2. Oconee 1, Cycle 5 Fuel Cycle Depletion

Core Parameters

No. of fuel assemblies in core	177
Type of fuel assembly	Mark B
Fuel loading, kgU/FA	463.7
No. of control rod assemblies	69
No. of full-length control rods	61
No. of axial power shaping rods	8
Type of control rods	Ag-In-Cd

Operating Conditions

Rated power level, MWt	2568
Avg moderator temperature at HFP, F	582
Hot zero power temperature, F	532
Primary system pressure, psia	2200
Core flow, lbm/ft^2-h	2.94×10^6

Fuel Cycle Parameters

Cycle number	1	2	3	4	5
Full power days	310	292	309	235	330
Power level, MWt	2568	2568	2568	2568	2568
Avg kW/ft nucl.	5.81	5.81	5.81	5.81	5.81

Table 4-3. Oconee 1, Cycle 5 Physics Parameters

	Cycle 4	Cycle 5
Cycle length, EFPD	292	330
Cycle burnup, MWd/mtU	9,136	10,327
Average core burnup, EOC, MWd/mtU	19,034	19,027
Initial core loading, mtU	82.1	82.1
Critical boron, BOC (no Xe), ppm		
HZP, group 8 37.5% wd	1415	1458
HZP, groups 7 and 8 inserted	1335	1324
HFP, group 8 inserted	1145	1276
Critical boron, EOC (eq Xe), ppm		
HZP, group 8 37.5% wd	373	343
HFP, group 8 37.5% wd	88	44
Control rod worths, HFP, BOC, % Δk/k		
Group 6	1.07	1.21
Group 7	0.93	1.45
Group 8 37.5% wd	0.50	0.43
Control rod worths, HFP, EOC, % Δk/k		
Group 7	1.16	1.53
Group 8 37.5% wd	0.47	0.48
Max ejected rod worth, HZP, % Δk/k		
BOC (N-12)	0.68	0.57
EOC (N-12)	0.61	0.70
Max stuck rod worth, HZP, % Δk/k		
BOC (N-12)	1.74	2.17
EOC (N-12)	2.02	2.01
Power deficit, HZP to HFP, % Δk/k		
BOC	1.49	1.31
EOC	2.07	2.12
Doppler coeff, 10^{-5}(Δk/k-°F)		
BOC, 100% power, no Xe	-1.45	-1.45
EOC, 100% power, eq Xe	-1.55	-1.62
Moderator coeff, HFP, 10^{-4}(Δk/k-°F)		
BOC (0 Xe, crit ppm, gp 8 ins)	-1.00	-0.45
EOC (eq Xe, 17 ppm, gp 8 ins)	-2.55	-2.64
Boron worth, HFP, ppm/% Δk/k		
BOC (1150 ppm)	109	109
EOC (17 ppm)	101	97
Xenon worth, HFP, % Δk/k		
BOC (4 EFPD)	2.60	2.62
EOC (equilibrium)	2.61	2.73
Eff delayed neutron fraction, HFP		
EOC	0.00593	0.00598
EOC	0.00530	0.00521

Table 4-4. Oconee 1, Cycle 5 Control Rod Worth[a] and Sequential Worth, % $\Delta\rho$

Bank(s) inserted	0 EFPD		330 EFPD	
	HFP	HZP	HFP	HZP
1-4	--	5.41	--	5.23
5	1.11	1.05	1.14	1.07
6	1.21	1.13	1.24	1.13
7	1.45	1.32	1.53	1.36
1-7	--	8.91	--	8.79
8	0.43	0.42	0.48	0.47

[a] All calculations were performed with APSRs (bank 8) inserted except for bank 8 worth calculations.

Table 4-5. Oconee 1, Cycle 5 Stuck Rod Worths

Conditions	Stuck rod location	Worth, % $\Delta\rho$
BOC, HZP	L14	1.99
BOC, HZP	N12	2.17
EOC, HZP	L14	1.95
EOC, HZP	N12	2.01

Table 4-6. Oconee 1, Cycle 5 Shutdown Margin Calculation

	BOC, % Δk/k	EOC, % Δk/k
Available rod worth		
Total rod worth, HZP	8.91	8.79
Worth reduction due to burnup of poison material	-0.36	-0.42
Maximum stuck rod, HZP	-2.17	-2.01
Net worth	6.38	6.36
Less 10% uncertainty	-0.64	-0.64
Total available worth	5.74	5.72
Required rod worth		
Power deficit, HFP to HZP	1.31	2.12
Max allowable inserted rod worth	0.40	0.60
Flux redistribution	0.59	1.20
Total required worth	2.30	3.92
Shutdown margin (available worth minus total required worth)	3.44	1.80

Note: Required shutdown margin is 1.00% Δk/k.

Table 4-7. Oconee 1, Cycle 5 — Comparison of Key Parameters for Accident Analysis

Parameter	FSAR and densification report value	Predicted cycle 5 value
Doppler coeff, 10^{-4} Δk/k/°F	-1.17	-1.45
BOC	-1.17	-1.45
EOC	-1.33	-1.62
Moderator coeff, 10^{-4} Δk/k/°F	+0.5	-0.45
BOC	+0.5	-0.45
EOC	-3.0	-2.64
Total control rod worth, HZP, % Δk/k	10	8.91
Initial boron conc, HFP, ppm	1400	1276
Boron reactivity worth at 70F, ppm/1% Δk/k	75	76
Max ejected rod worth, HFP, % Δk/k	0.65	0.25
Dropped rod worth, HFP, % Δk/k	0.46	0.20

Table 4-8. Oconee 1, EOC 4 Poolside Characterization

Measurement	Planned (per ass'y)	Completed
Visual examination	TV/videotape all four faces, periscope/photograph areas of interest	As planned
Gross gamma scan	All four corner rods	As planned
Assy length and grid locations	All four faces	As planned
Holddown spring	Each assembly	As planned
Water channel spacings	All seven midplanes, all channels, both directions	As planned on four assys, plane 12 on one assy
Diameter profilometry	28 scans (incl. orthogonals)	1D13 27, 1D45 17, 1D42 12, 1D55 32, 1D26 32
Sipping	Each assembly	As planned

Figure 4-1. Oconee 1, Cycle 5 Full Core Loading Diagram

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						7	7	7	7	7					
B				7	7	7	E7 5	R3 6	E9 5	7	7	7			
C			7	M2 6	C6 5	F7 5	L2 6	M5 5	L14 6	F9 5	C10 5	M14 6	7		
D		7	B11 6	O13 6	L1 6	N3 6	D5 5	7	D11 5	N13 6	L15 6	O3 6	B5 6	7	
E		7	F3 5	A10 6	O9 5	K1 6	N2 6	O5 1D55 4D	N14 6	K15 6	K3 5	A6 6	F13 5	7	
F	7	7	G6 5	C12 6	A9 6	K13 5	D6 5	N8 5	D10 5	O7 5	A7 6	C4 6	G10 5	7	7
G	7	G5 5	B10 6	E4 5	B12 6	F4 5	7	K8 5	7	F12 5	B4 6	E12 5	B6 6	G11 5	7
H	7	H15 6	M11 5	7	M13 1D13 4D	H12 5	H9 5	K14 1D42 4D	H7 5	H4 5	E3 1D45 4D	7	E5 5	H1 6	7
K	7	K5 5	P10 6	M4 5	P12 6	L4 5	7	C8 5	7	L12 5	P4 6	M12 5	P6 6	K11 5	7
L	7	7	K6 5	O12 6	R9 6	G9 6	N6 6	D8 5	N10 5	G3 5	R7 6	O4 6	K10 5	7	7
M		7	L3 5	R10 6	G13 5	G1 6	D2 6	C11 1D26 4D	D14 6	G15 6	G7 5	R6 6	L13 5	7	
N		7	P11 6	C13 6	F1 6	D3 6	N5 5	7	N11 5	D13 6	F15 6	C3 6	P5 6	7	
O			7	E2 6	O6 5	L7 5	F2 6	E11 5	F14 6	L9 5	O10 5	E14 6	7		
P				7	7	7	M7 5	A8 6	M9 5	7	7	7			
7						7	7	7	7	7					

G5 1D55 4D	← Previous Cycle Location ← Assembly Ident. ← Batch No.

(shaded box) Location of thrice-burned batch 4D assemblies

Figure 4-2. Oconee 1, Cycle 5 Two-Dimensional Relative Power Distribution

	8	9	10	11	12	13	14	15
H	0.83 0.87	0.93 0.94	0.96 0.97	0.90 0.90	1.37 1.27	1.03 1.02	1.09 1.10	0.87 0.93
K		1.35 1.26	1.07 1.03	1.21 1.14	0.98 0.97	1.09 1.09	0.93 0.96	0.83 0.90
L			1.05 1.01	1.25 1.15	1.03 1.00	0.95 0.97	1.15 1.15	0.67 0.75
M				1.09 1.02	1.23 1.15	0.89 0.92	0.91 0.97	
N					1.21 1.15	0.94 0.97	0.61 0.72	
O						0.70 0.79	- BOC 5 - EOC 5	
P								
R								

x.xx — Relative power density

Location of thrice-burned 4D assemblies

Babcock & Wilcox

Figure 4-3. Oconee 1, Cycle 5 Assembly Burnups

	8	9	10	11	12	13	14	15
H	28479 37208	20488 30108	16053 25982	31135 40395	0 13667	15903 26513	5889 17283	0 9403
K		0 13457	14270 25055	5138 17210	19206 29254	8537 19837	16345 26197	0 9013
L			17336 27844	5853 18130	8262 18694	15846 25743	0 11974	0 7400
M				17341 28175	5011 17261	18348 27717	0 9771	
N					5846 17990	7092 16948	0 6852	
O						0 7699	BOC EOC	
P								
R								

Location of batch 4D fuel

Figure 4-4. Oconee 1, Cycle 5 Control Rod Locations

X

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	
A																
B						3		5		3						
C					1		7		7		1					
D				6		8		4		8		6				
E			1		5		2		2		5		1			
F		3		8		7		6		7		8		3		
G			7		2		4		4		2		7			
H W		5		4		6		3		6		4		5		Y
K			7		2		4		4		2		7			
L		3		8		7		6		7		8		3		
M			1		5		2		2		5		1			
N				6		8		4		8		6				
O					1		7		7		1					
P						3		5		3						
R																

Z

$\boxed{x}$ ← Group Number

Group	No. of rods	Function
1	8	Safety
2	8	Safety
3	9	Safety
4	8	Safety
5	8	Control
6	8	Control
7	12	Control
8	8	APSRs
Total	69	

Figure 4-5. Underwater Equipment Setup in Spent Fuel Pool

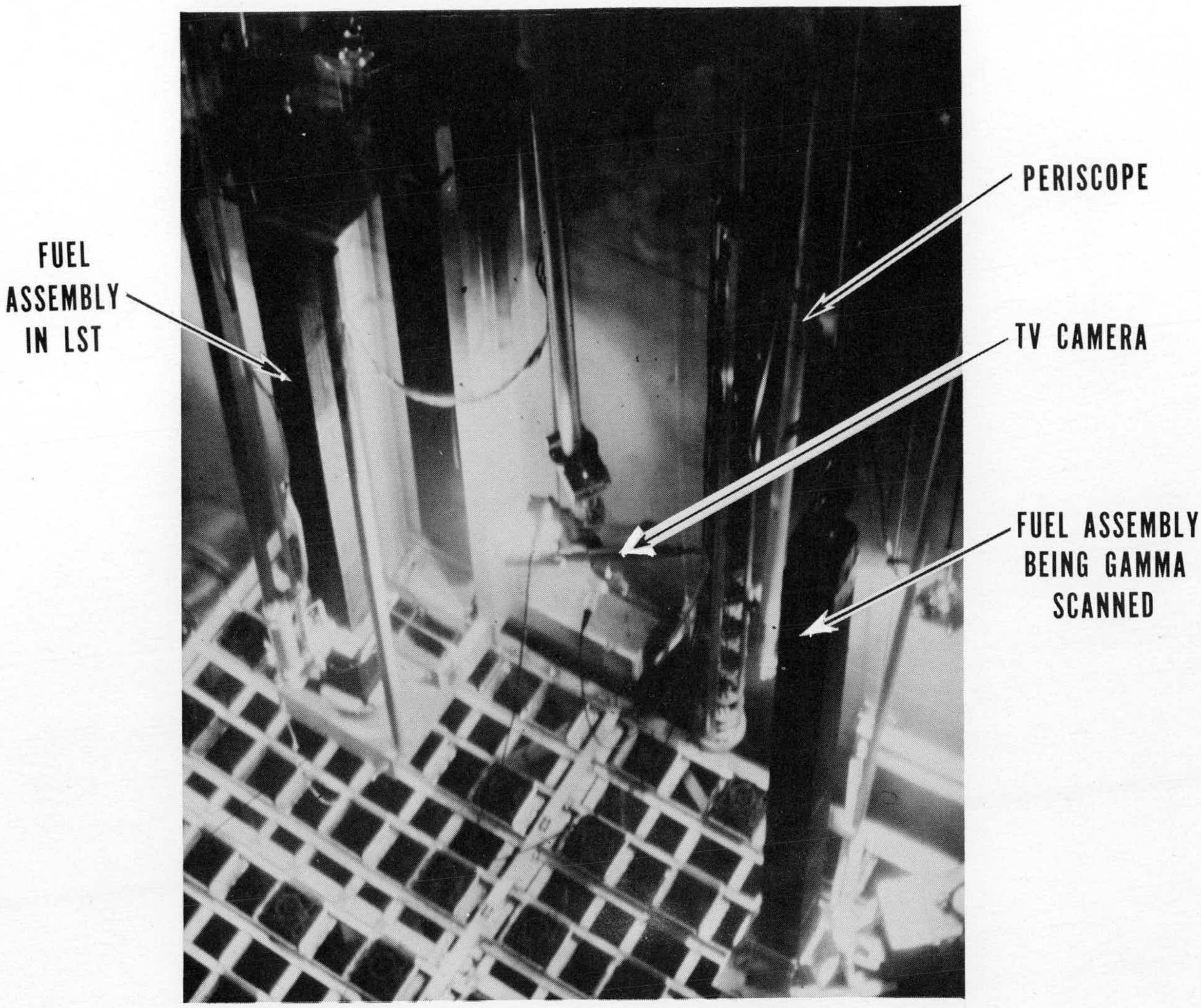


Figure 4-6. Equipment Peripherals at Poolside

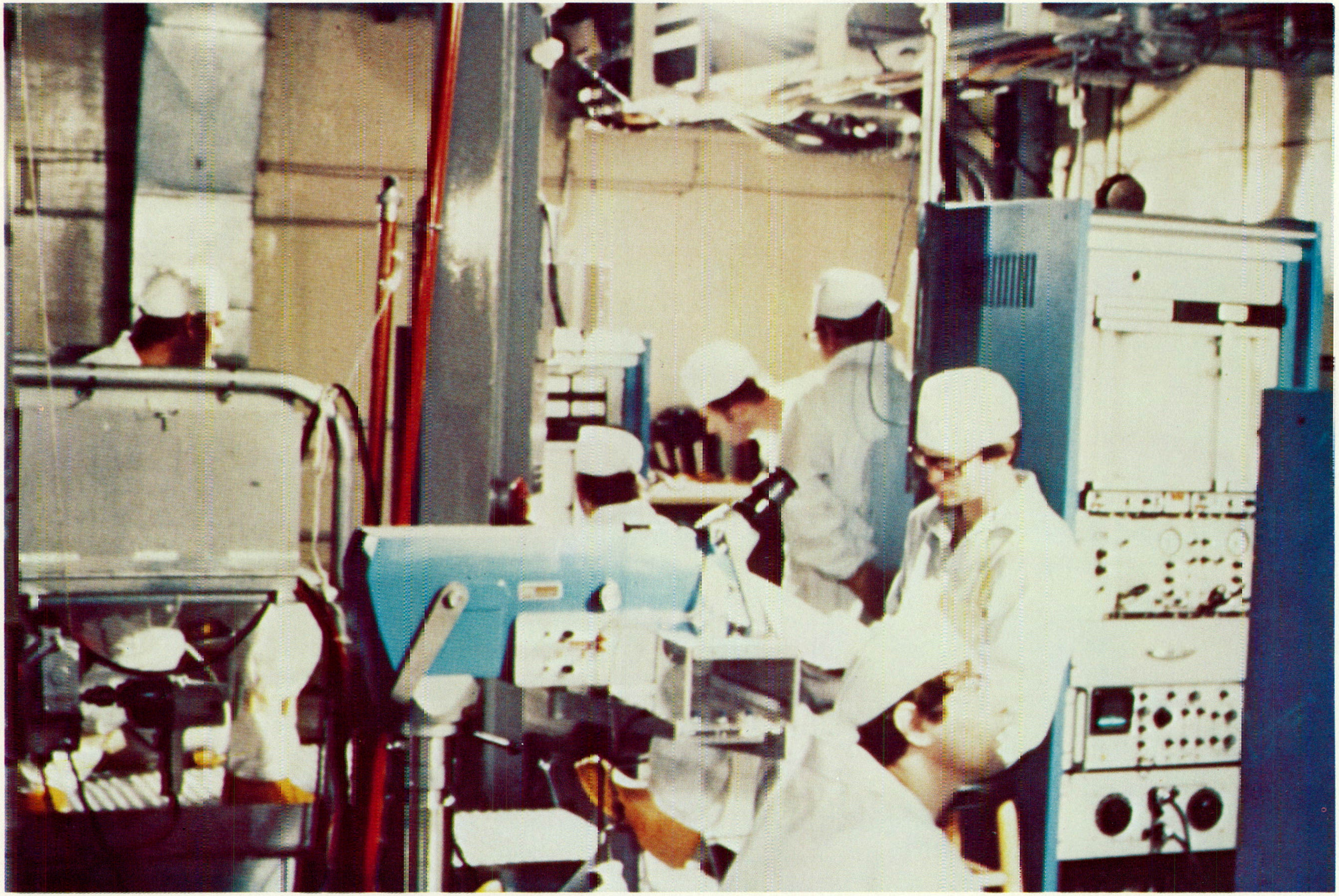


Figure 4-7. Schematic Diagram of Gamma Scan Alignment

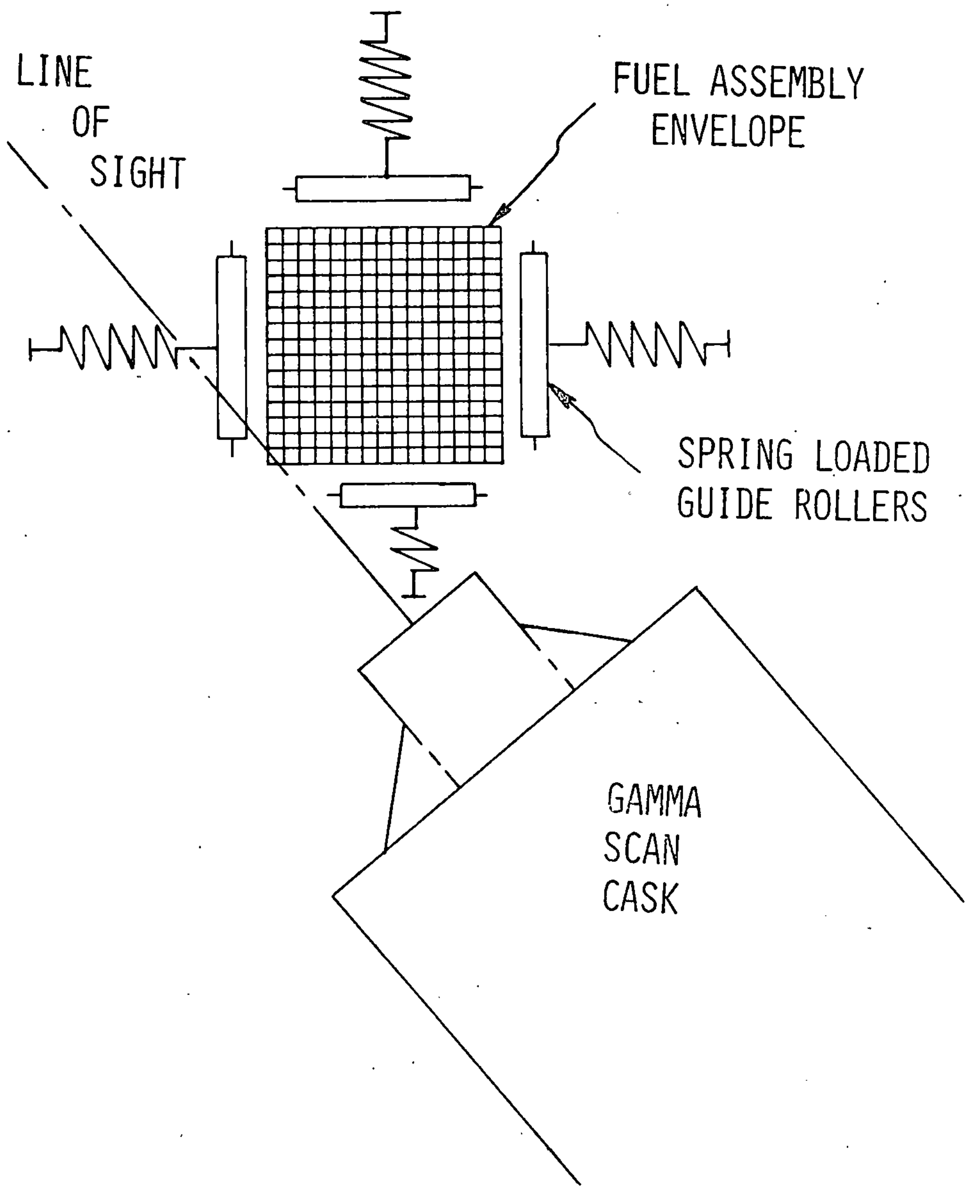


Figure 4-8. Oconee 1 Batch 4 Sipping Results (Frequency Distribution of ^{137}Cs Activity in Liquid Sample)

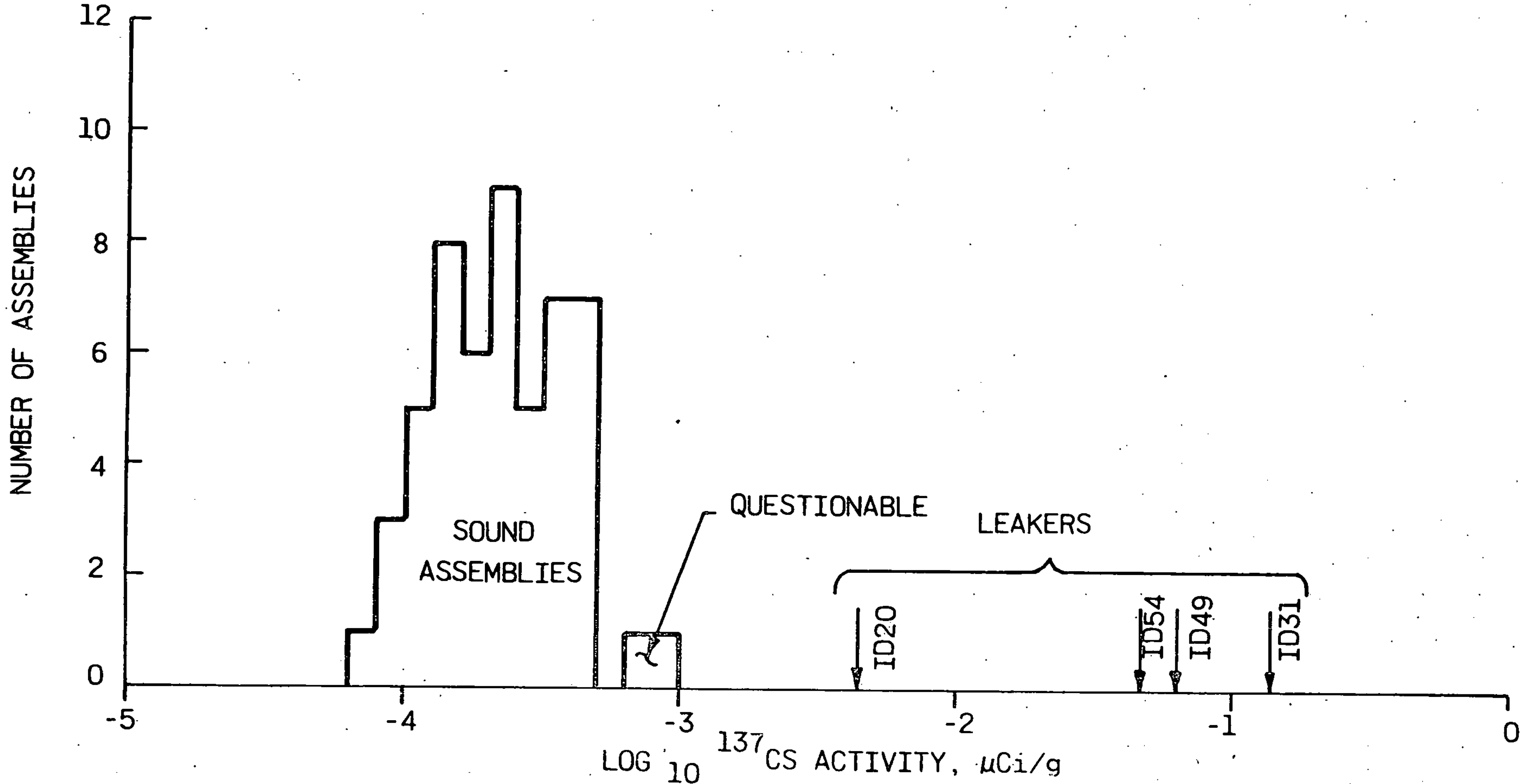
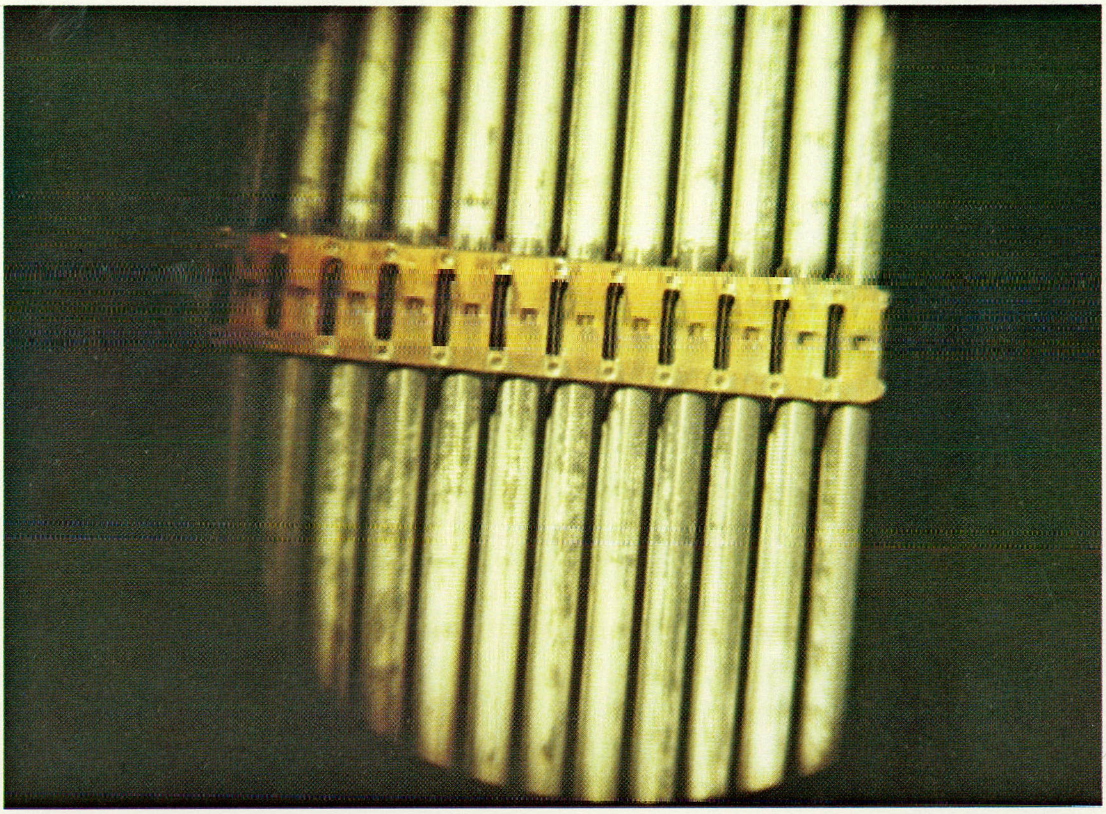


Figure 4-9. Typical Areas on Assembly 1D13



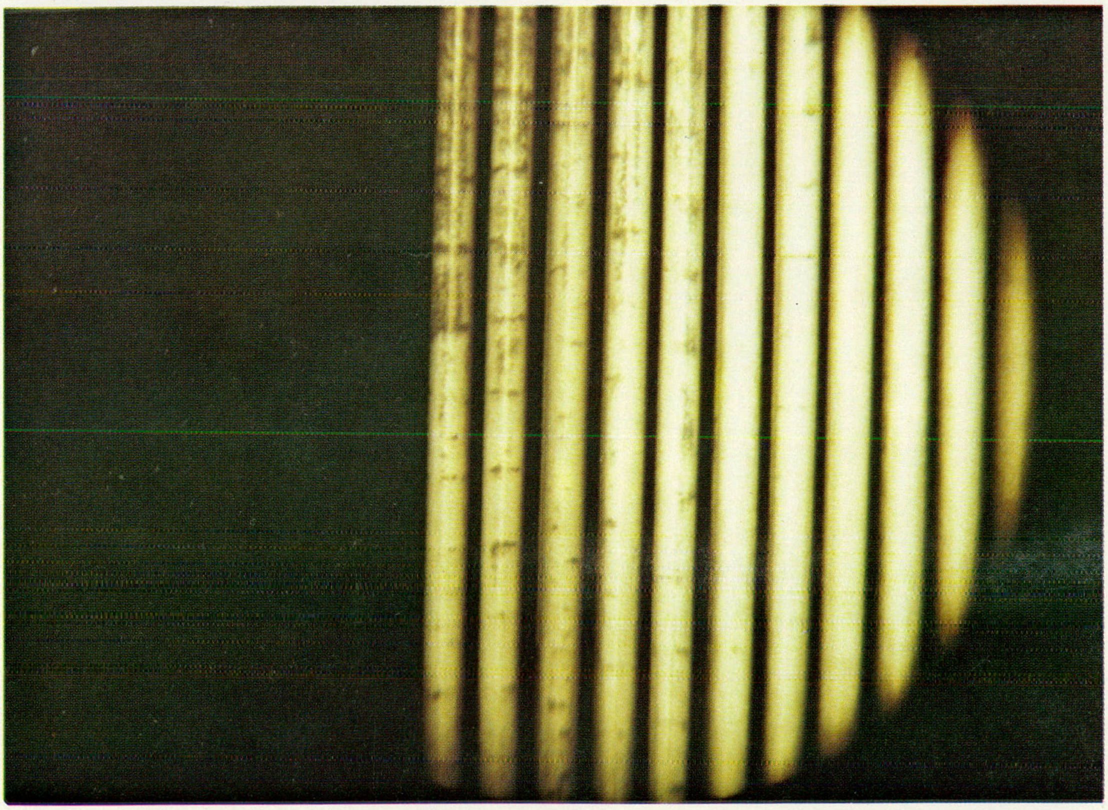


Figure 4-10. Assembly 1D26 Corner Fuel Rods Orthogonal Diameter Profilometry Traces

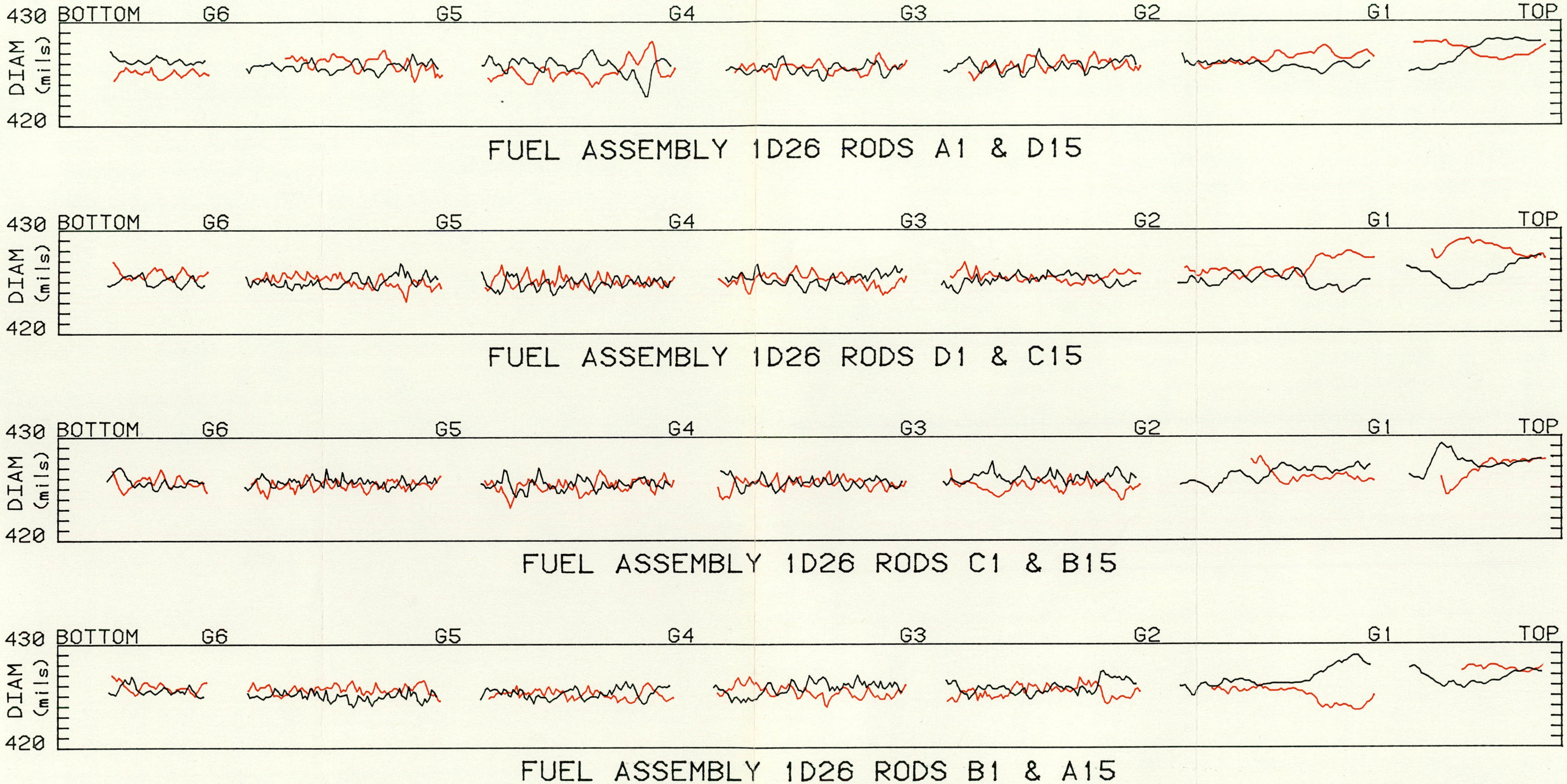


Figure 4-11. Assembly 1D55 Corner Fuel Rods Orthogonal Diameter Profilometry Traces

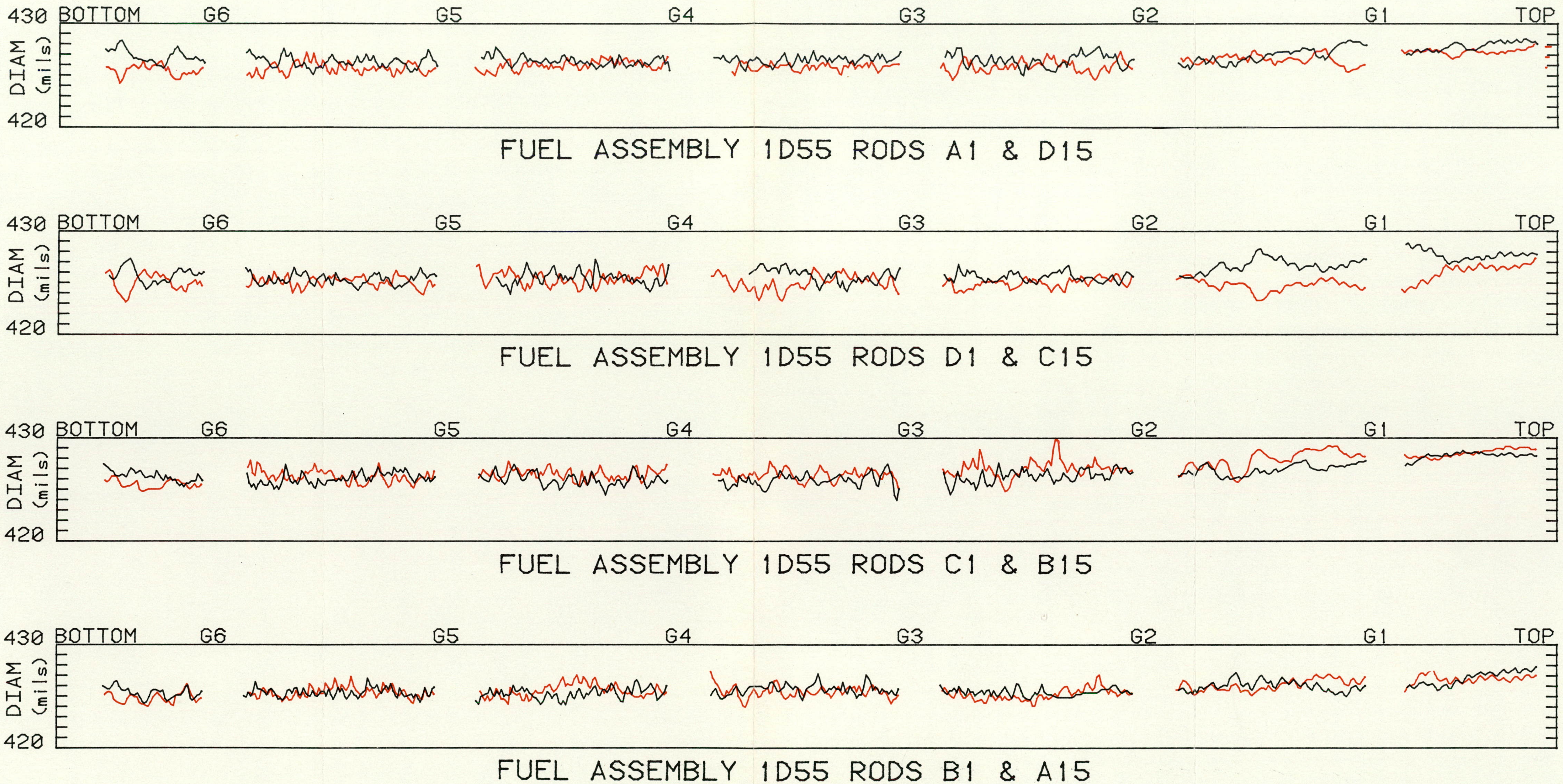


Figure 4-12. Assembly 1D42 Corner Fuel Rods Orthogonal Diameter Profilometry Traces

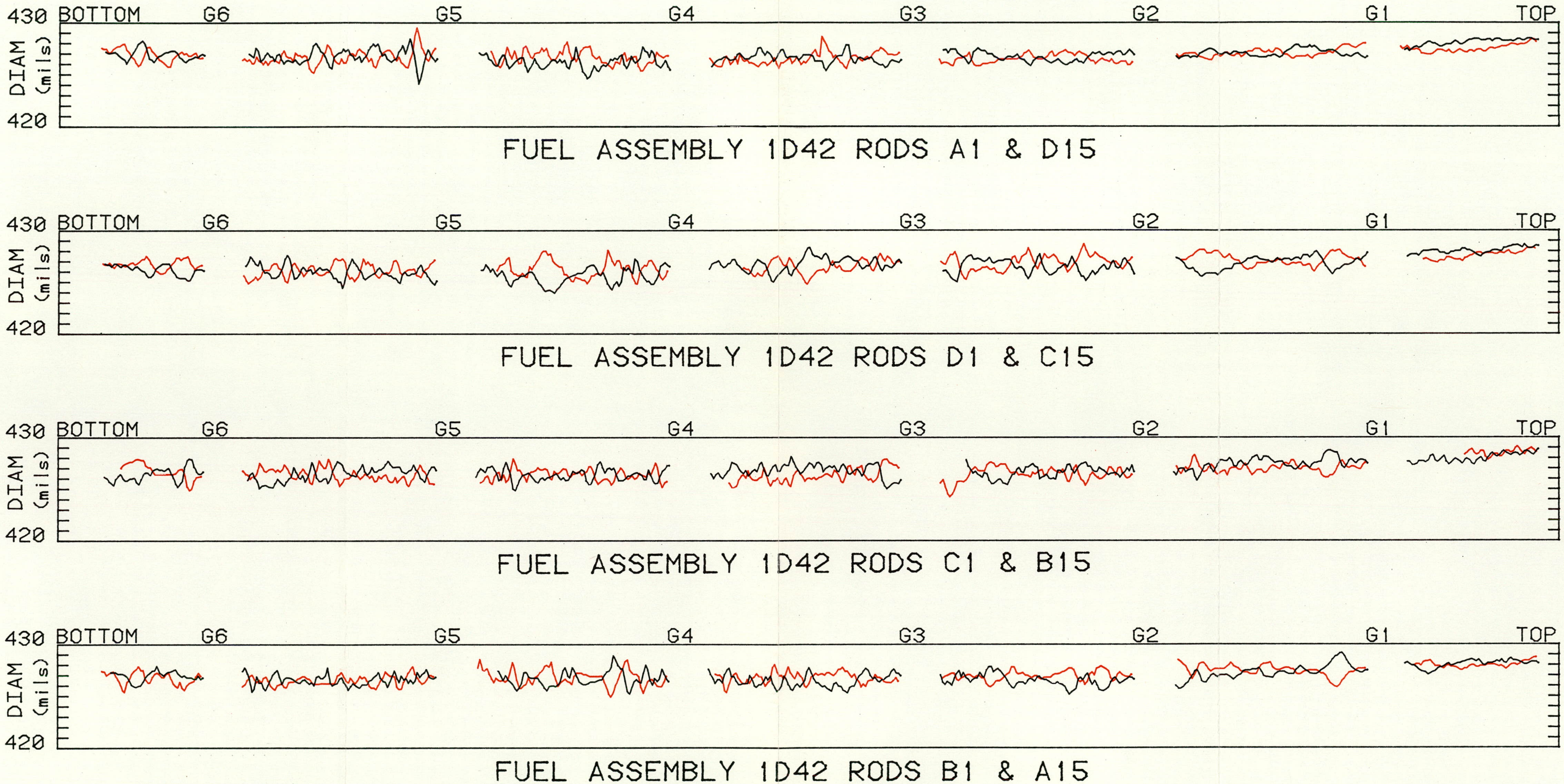


Figure 4-13. Assembly 1D45 Corner Fuel Rods Orthogonal Diameter Profilometry Traces

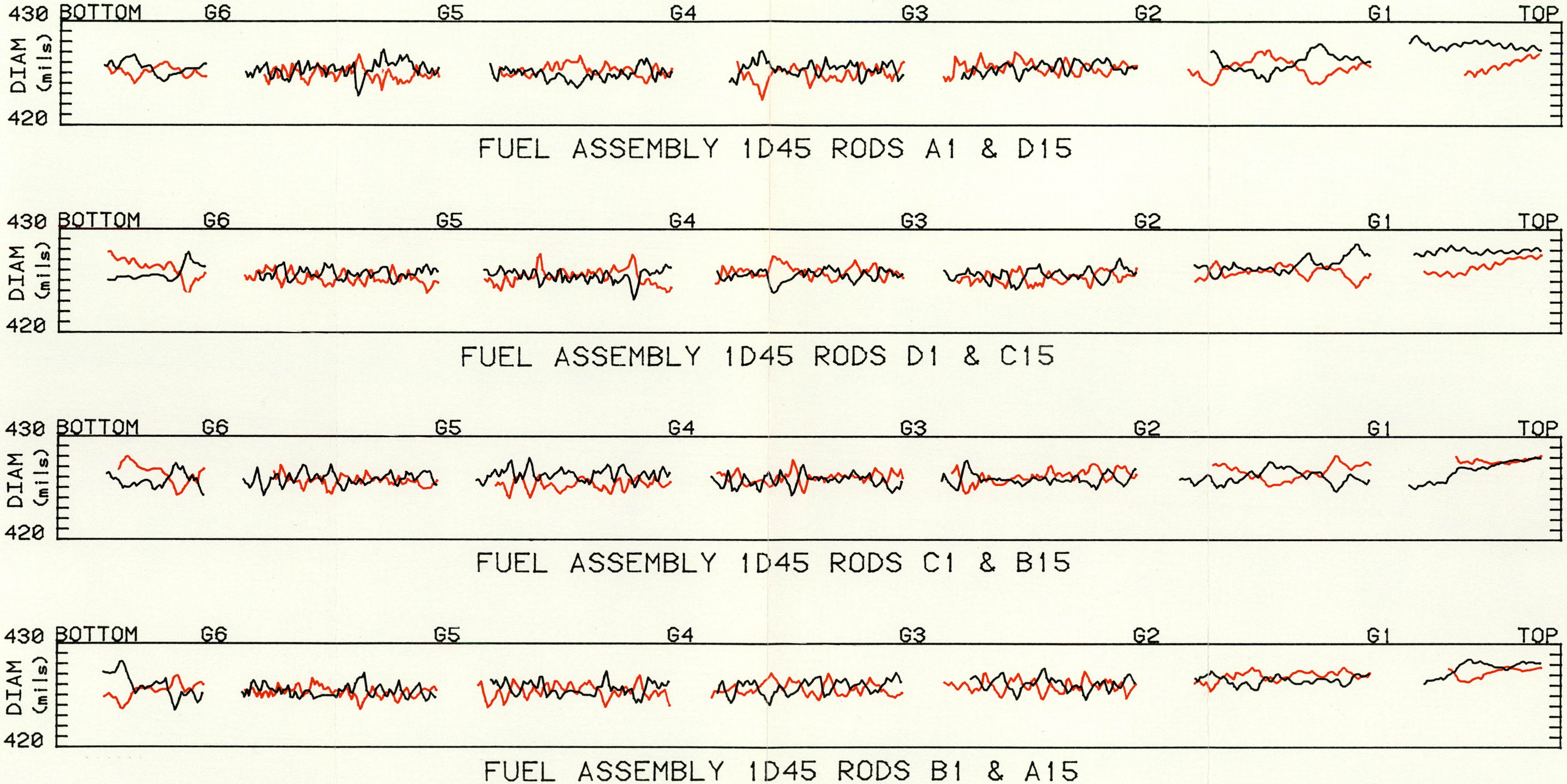


Figure 4-14. Assembly 1D13 Corner Fuel Rods Orthogonal Diameter Profilometry Traces

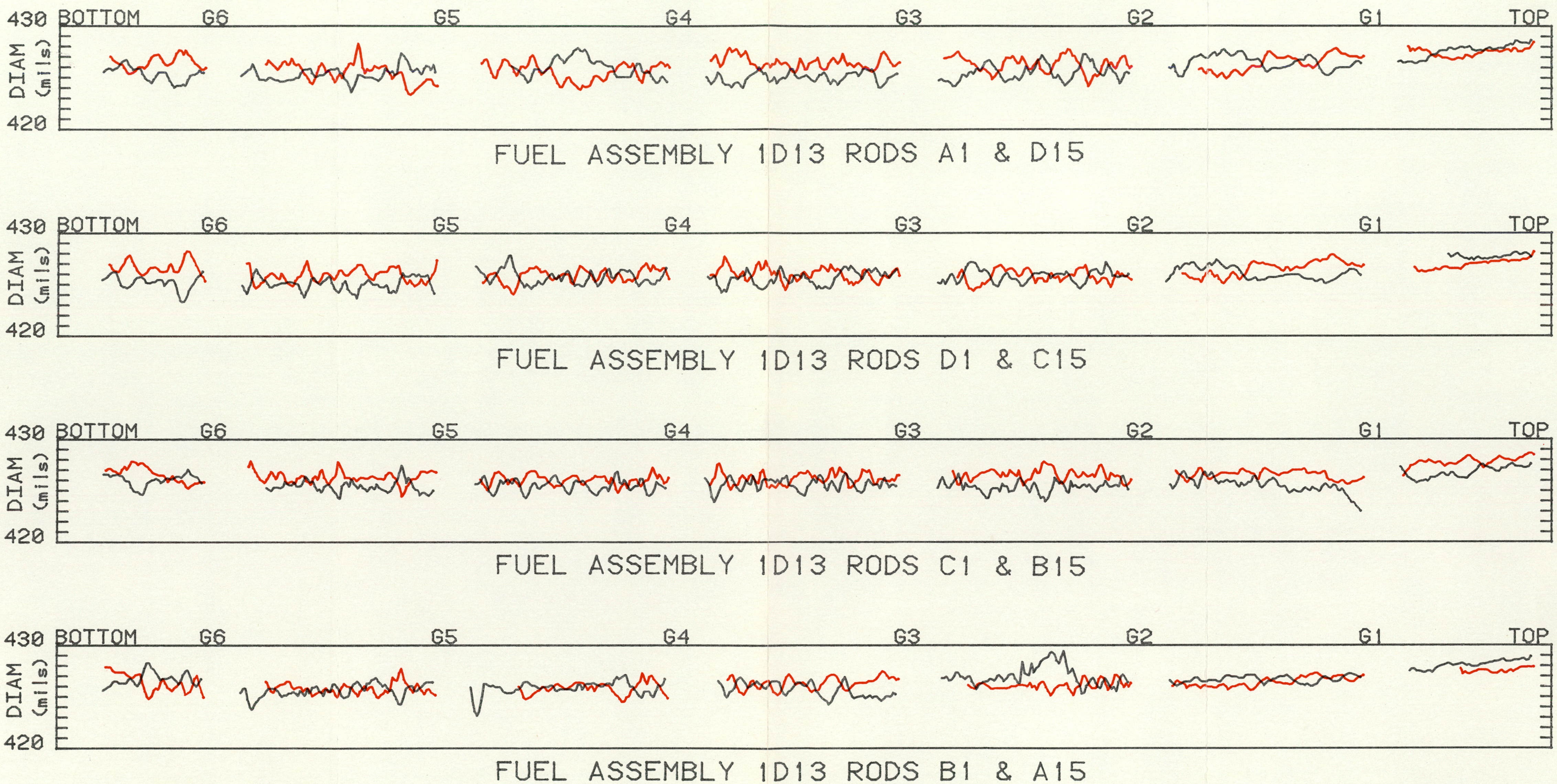


Figure 4-15. Fuel Rod Growth

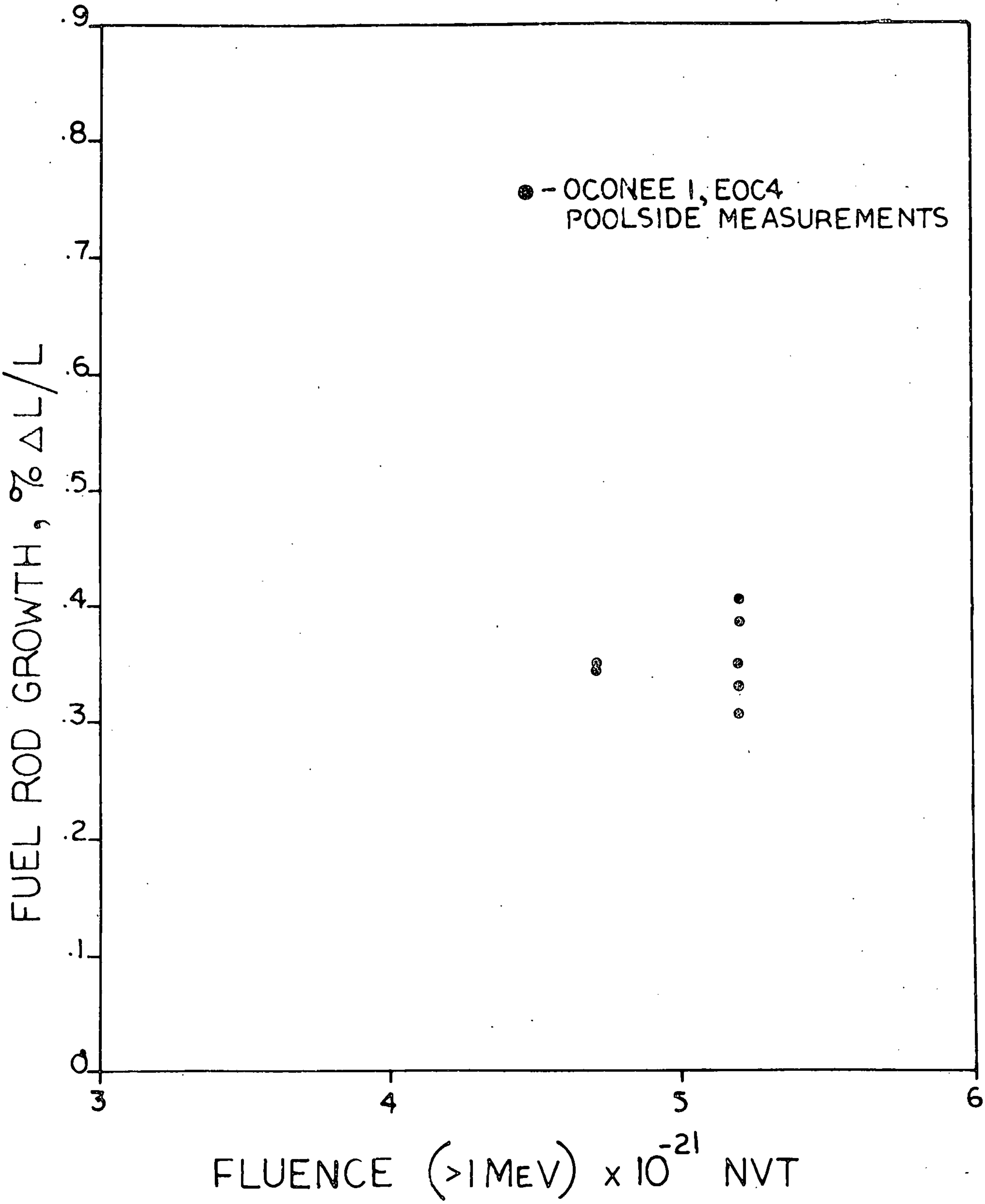


Figure 4-16. Fuel Assembly (Guide Tube) Growth

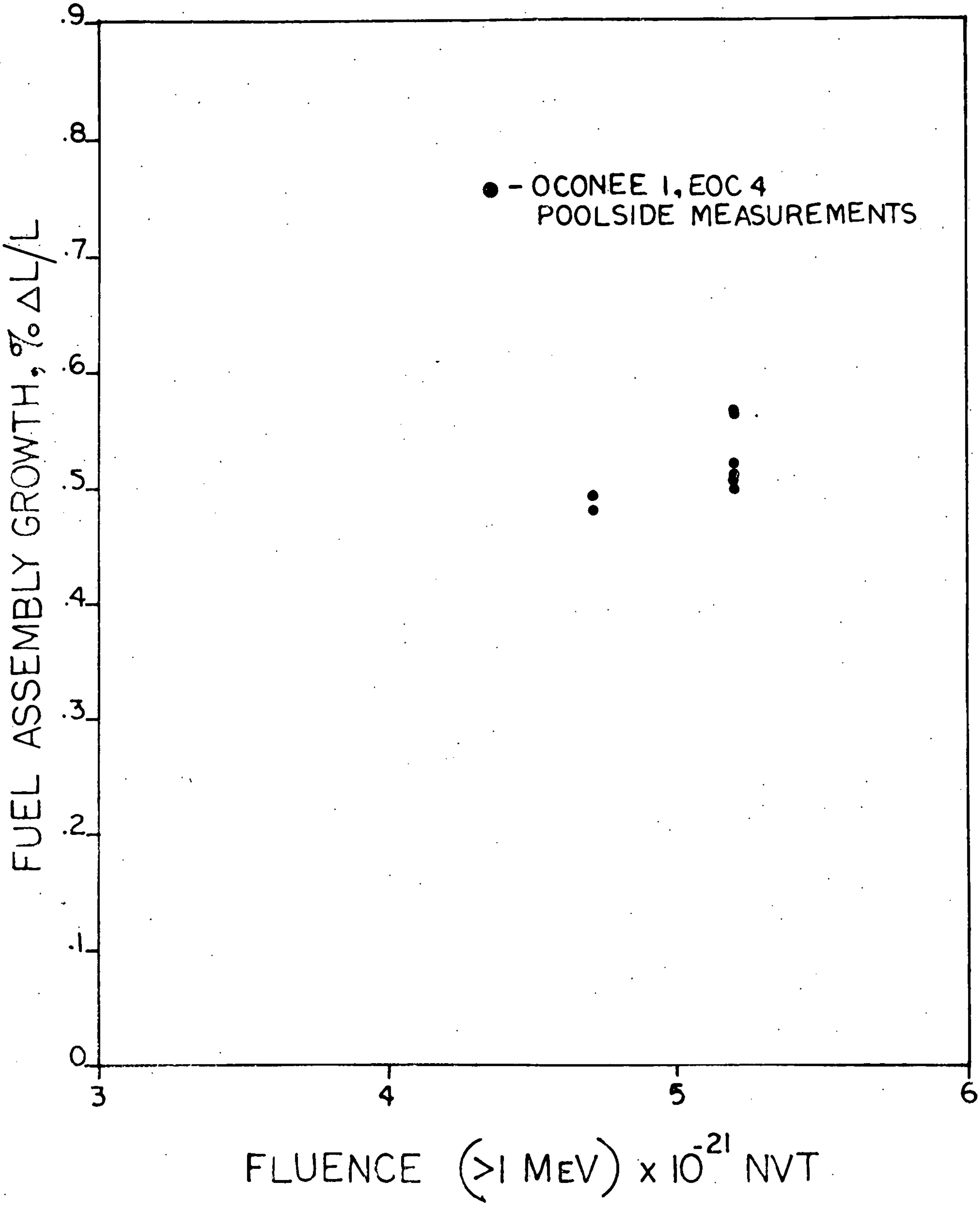


Figure 4-17. Fuel Assembly Water Channel Measurement Schematic

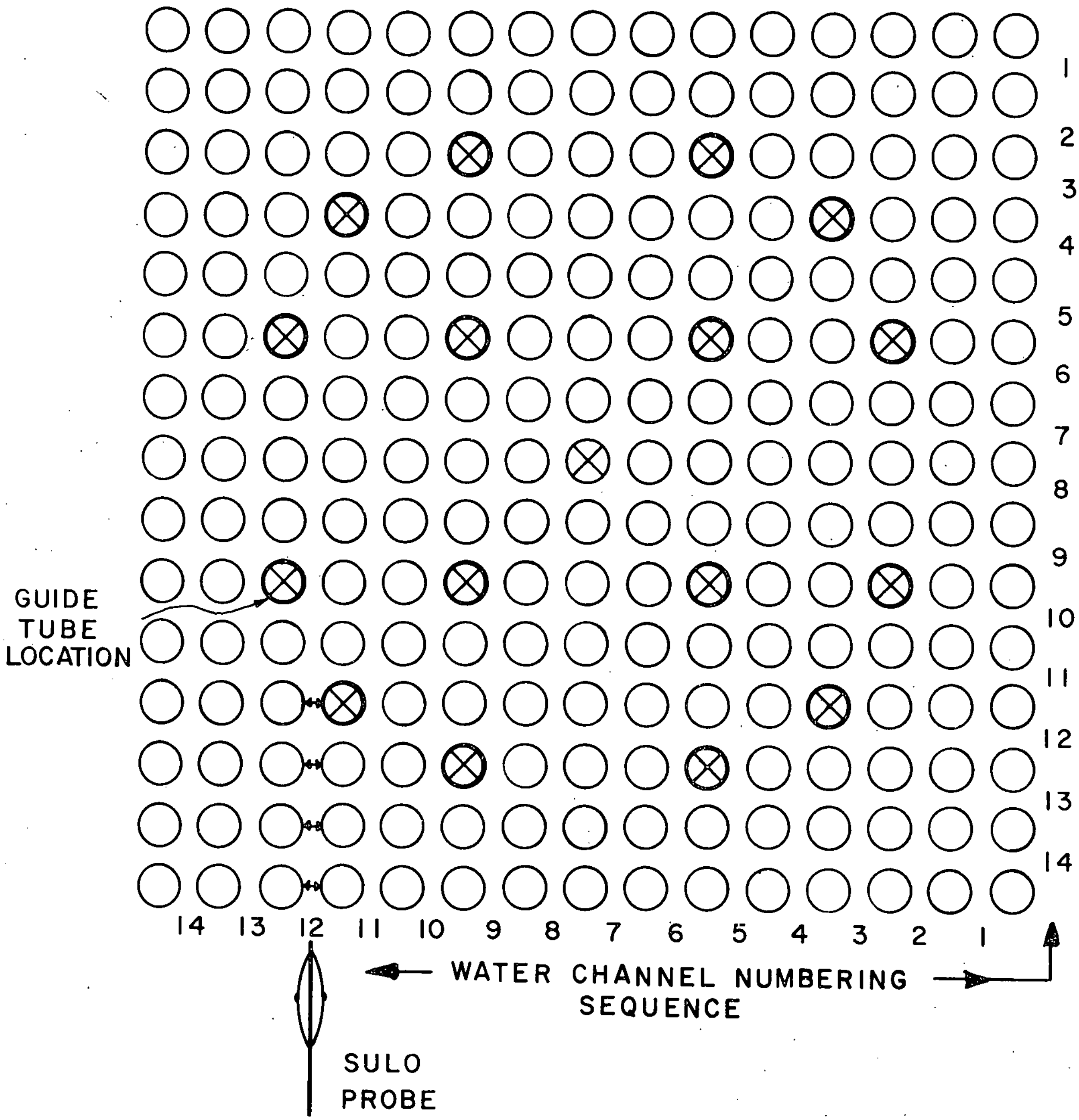


Figure 4-18. Water Channel Spacings, Axial Profile

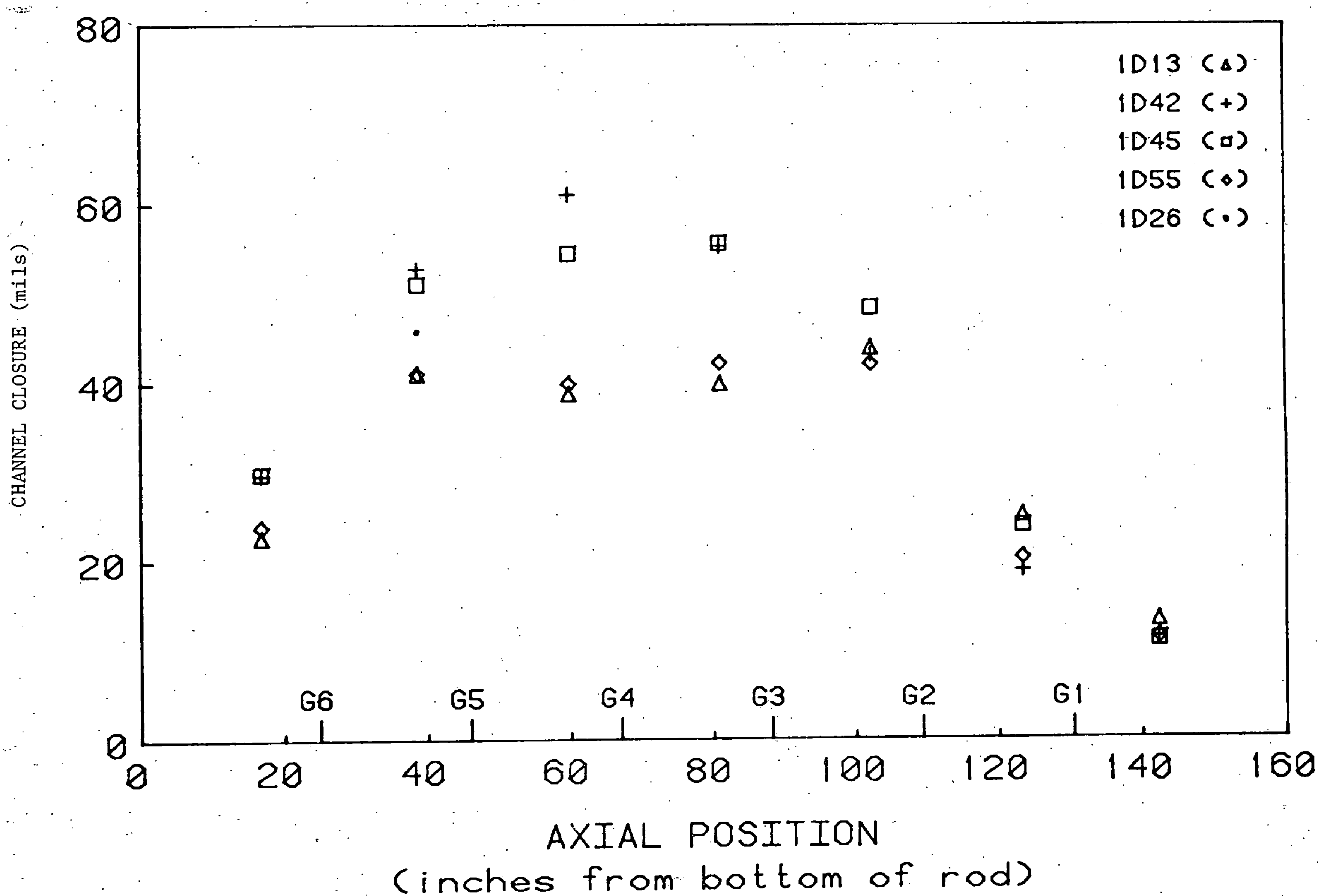


Figure 4-19. Assembly 1D42 Load-Deflection Curve for Holddown Spring

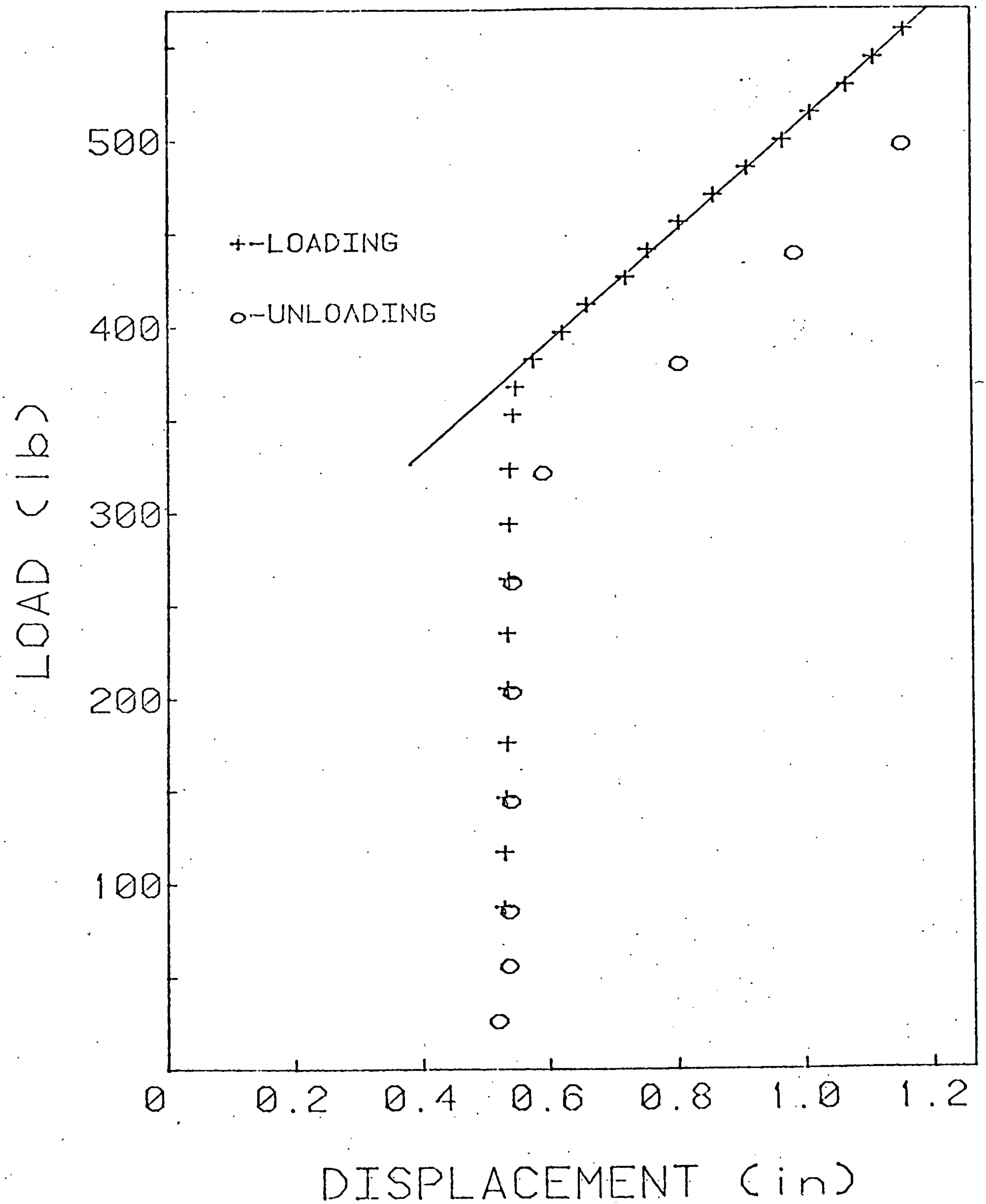


Figure 4-20. Assembly 1D13 Load-Deflection Curve for Holddown Spring

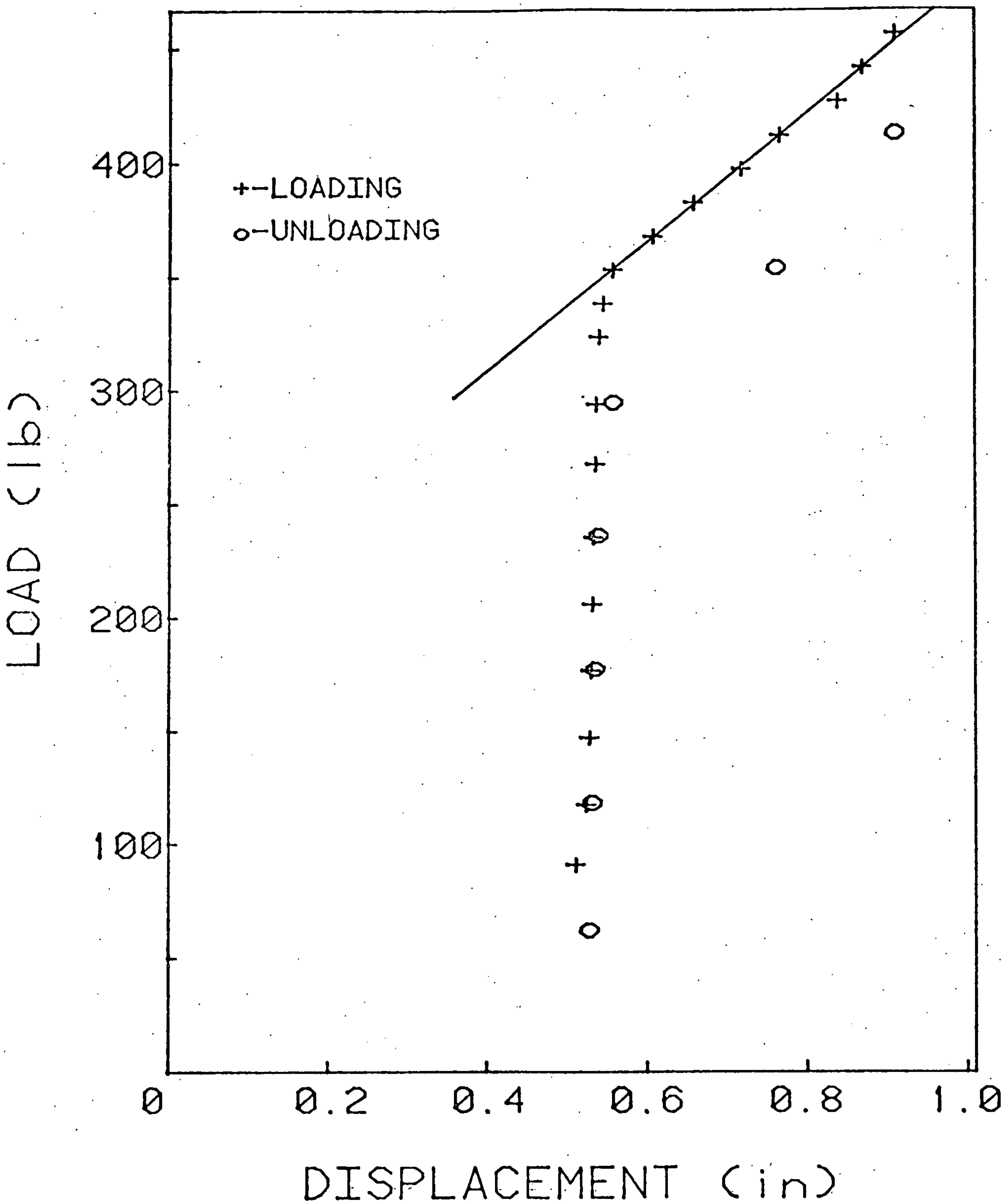


Figure 4-21. Assembly 1D26 Load-Deflection Curve for Holddown Spring

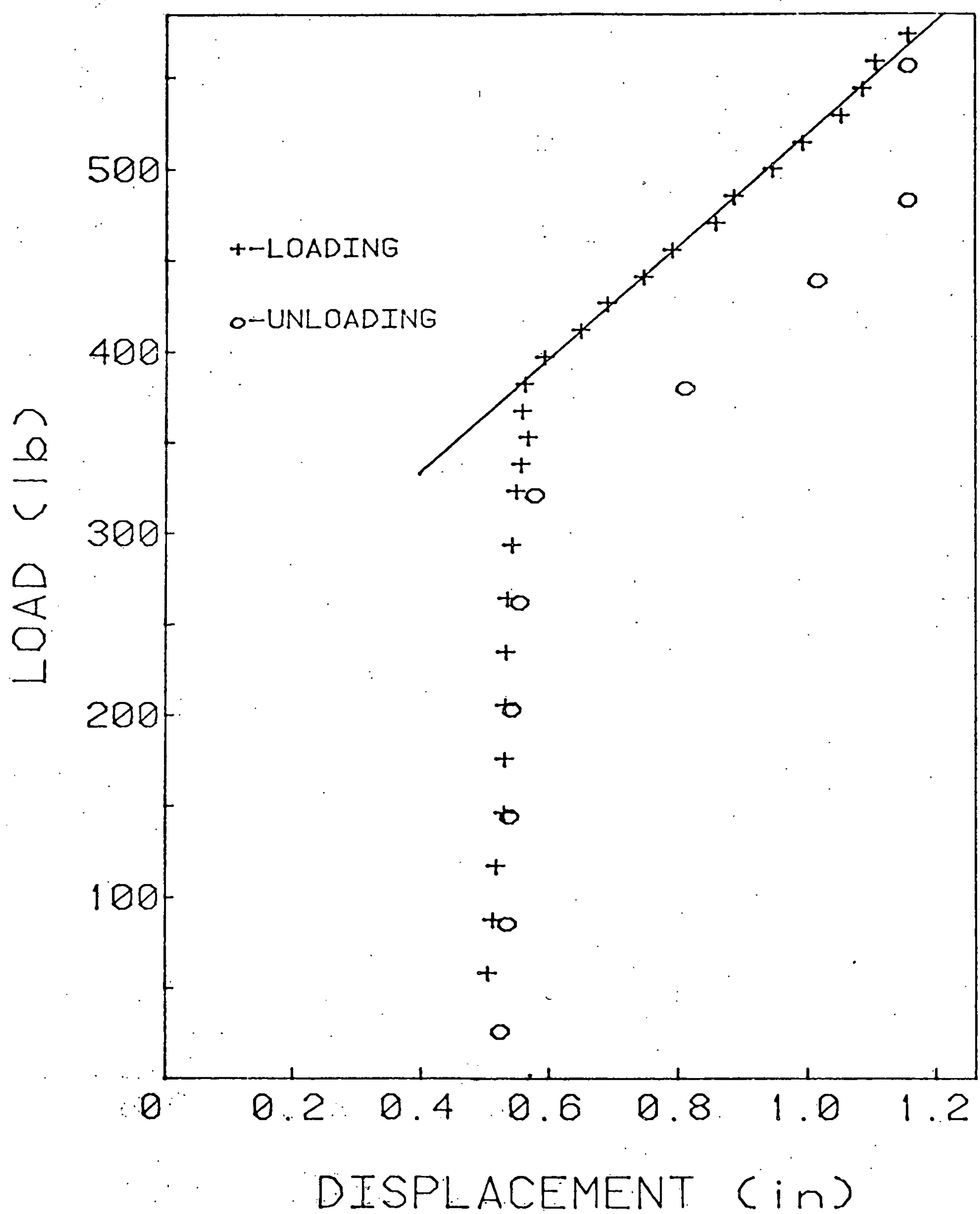


Figure 4-22. Assembly 1D45 Load-Deflection Curve for Holddown Spring

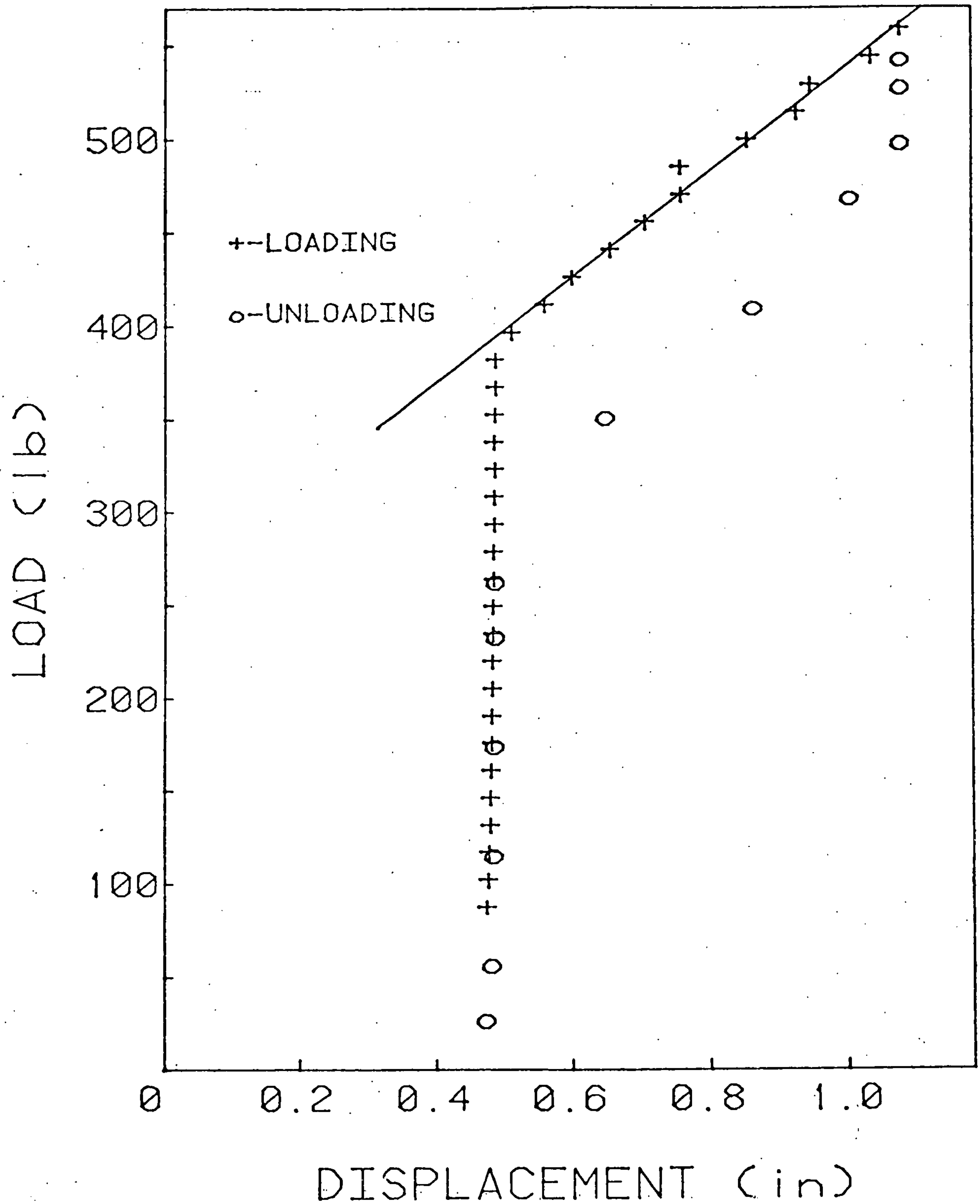


Figure 4-23. Assembly 1D55 Load-Deflection Curve for Holddown Spring

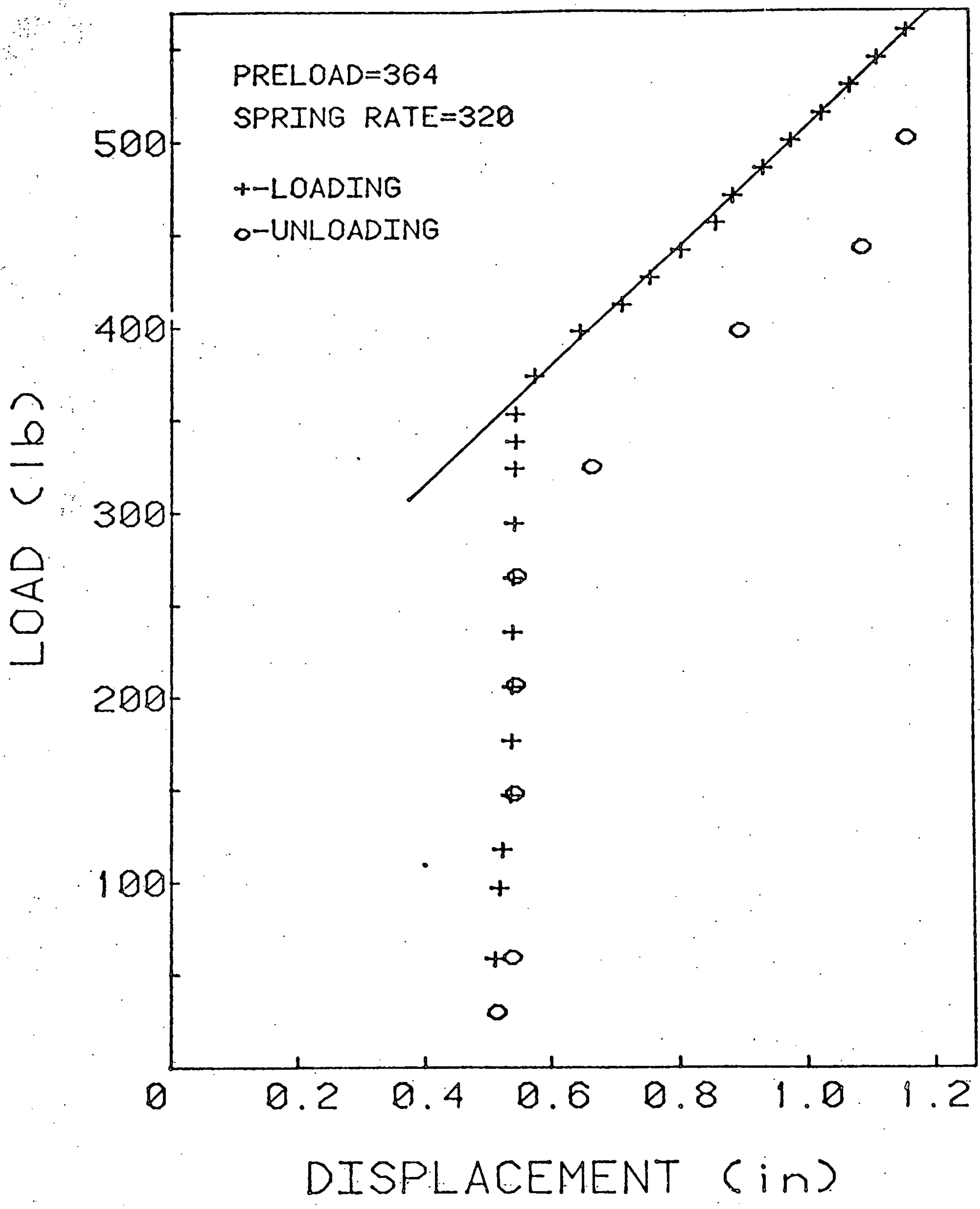


Figure 4-24. Assembly 1D13 Corner Fuel Rod Gross Gamma Scan

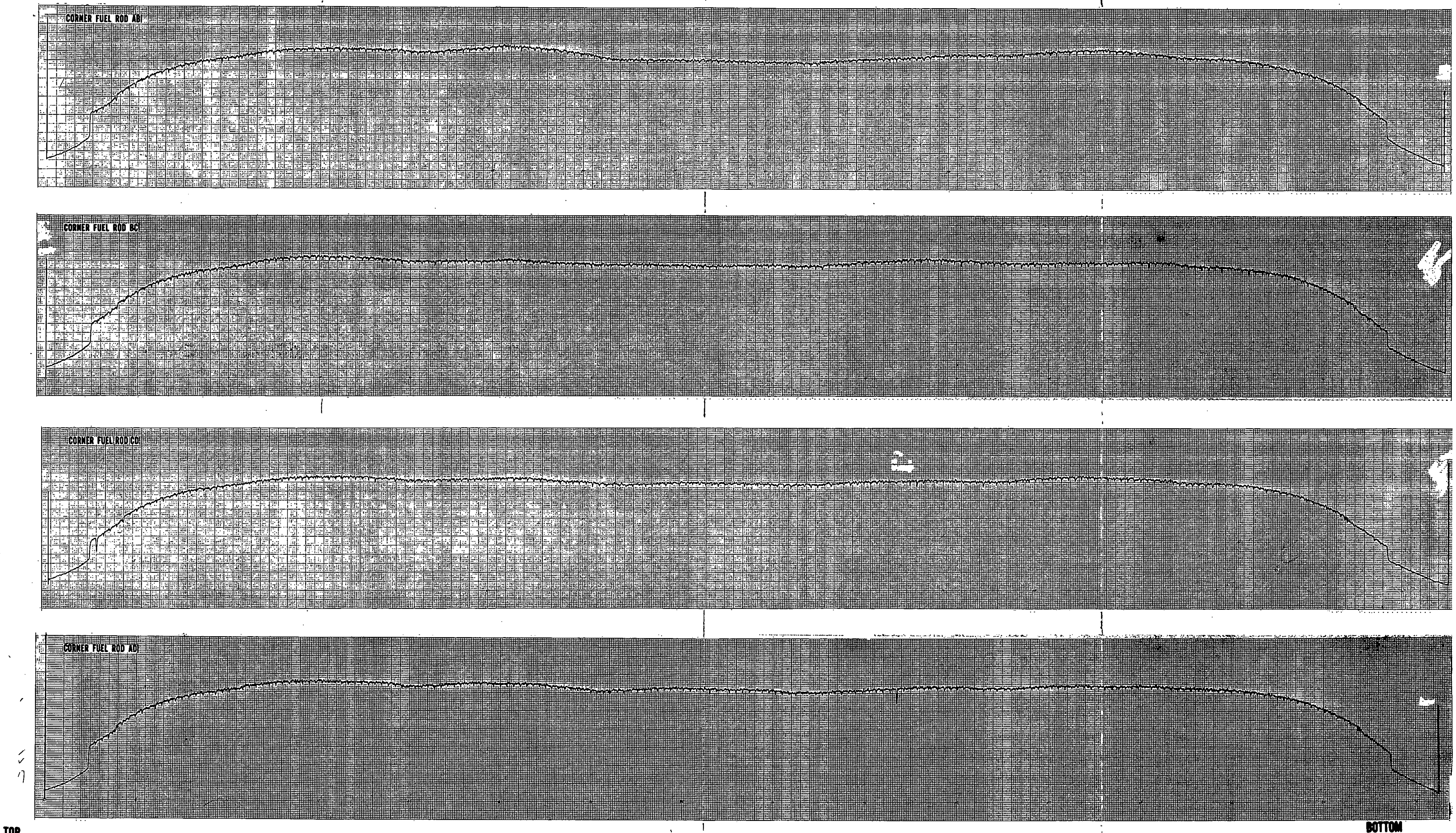


Figure 4-25. Assembly 1D26 Corner Fuel Rod Gross Gamma Scan

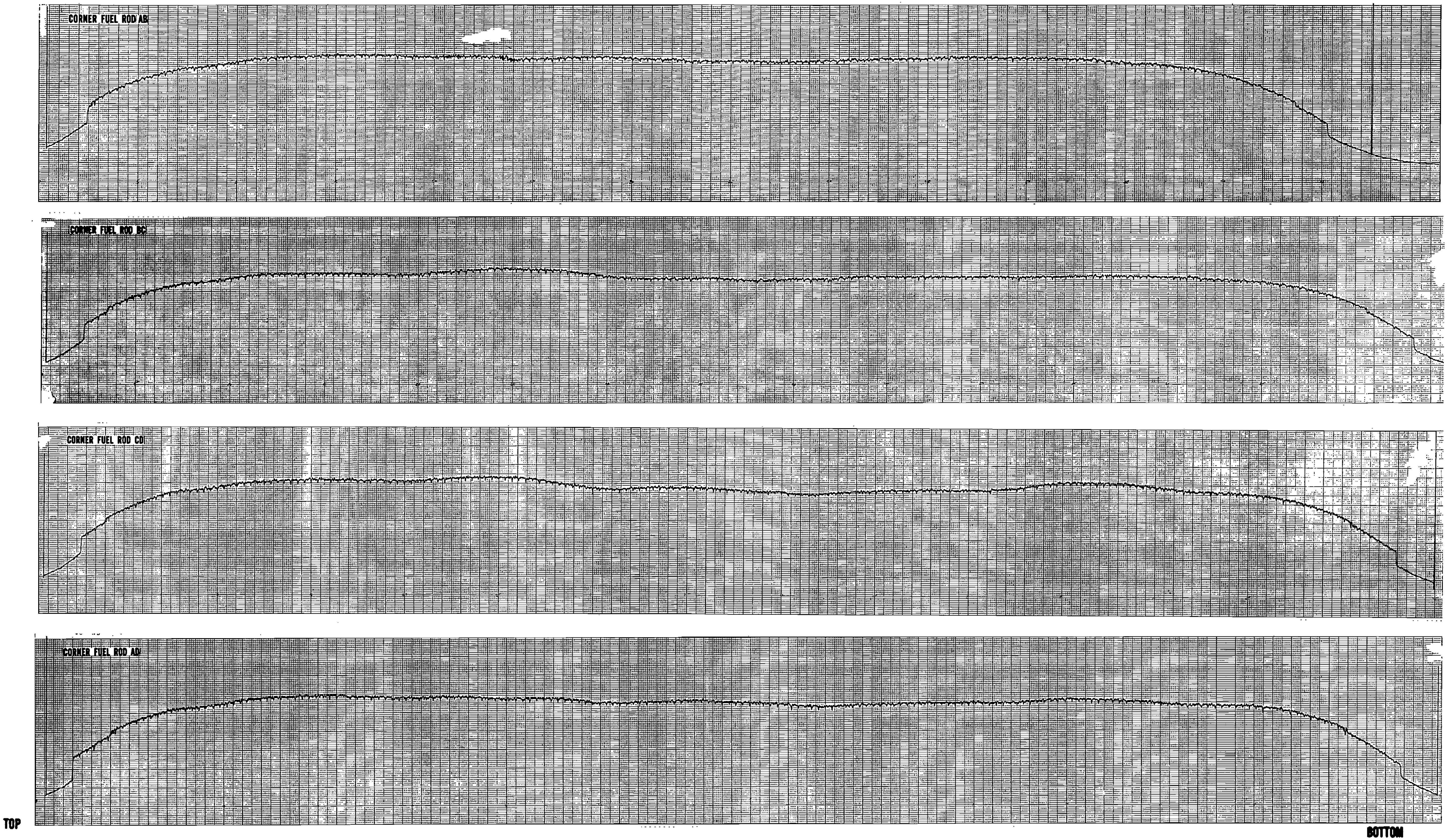


Figure 4-26. Assembly 1D42 Corner Fuel Rod Gross Gamma Scan

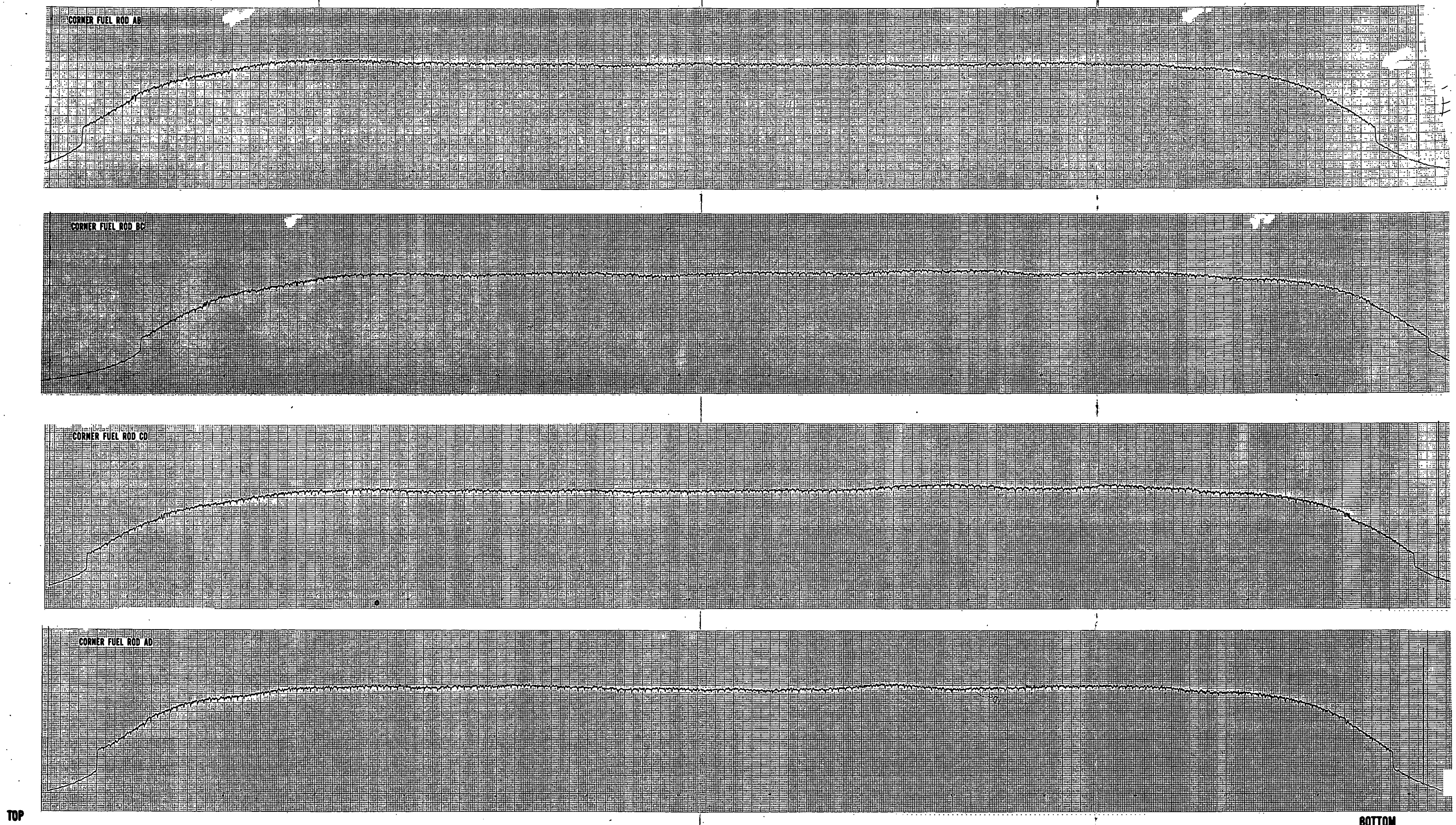


Figure 4-27. Assembly 1D45 Corner Fuel Rod Gross Gamma Scan

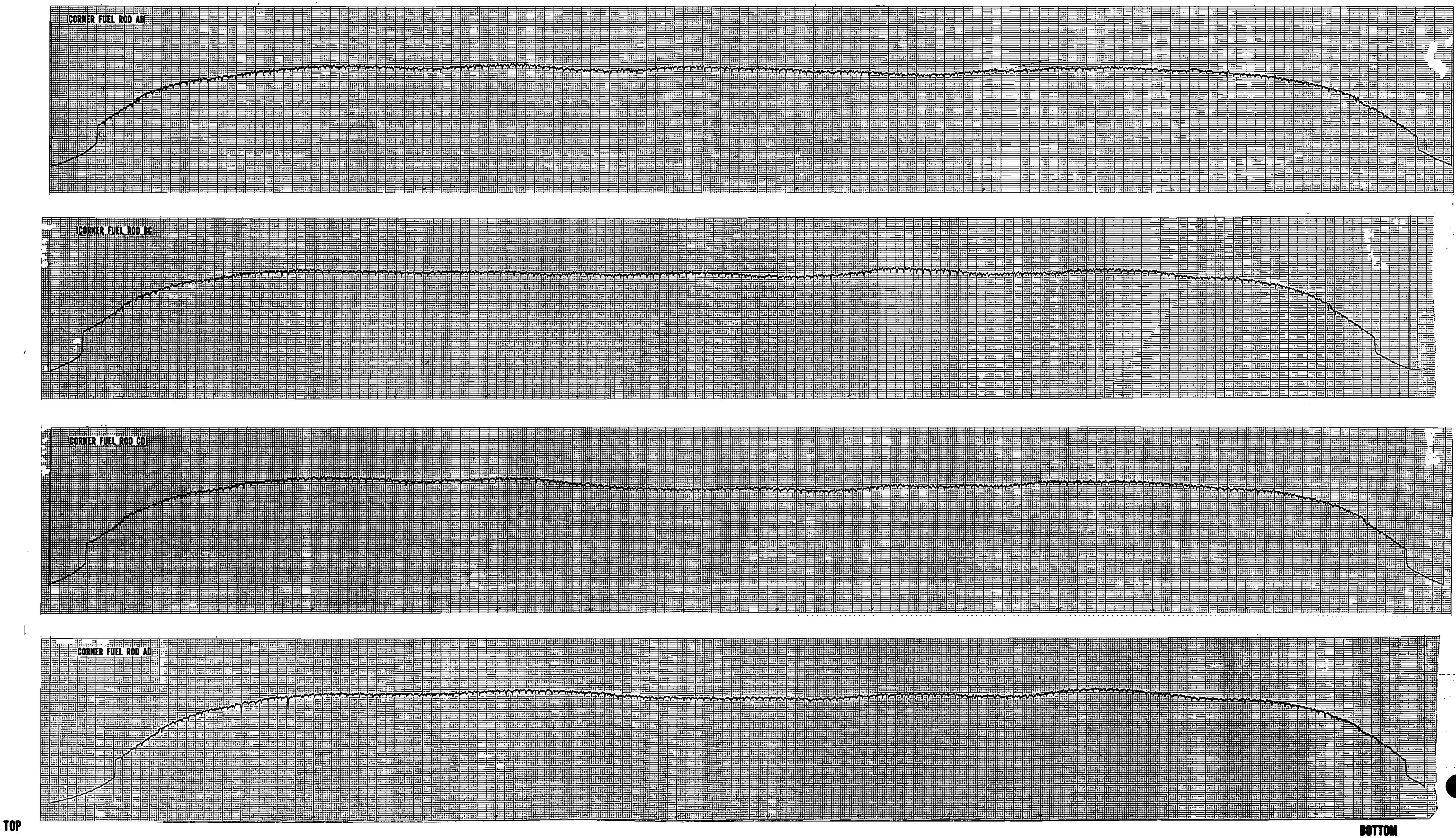


Figure 4-28. Assembly 1D55 Corner Fuel Rod Gross Gamma Scan

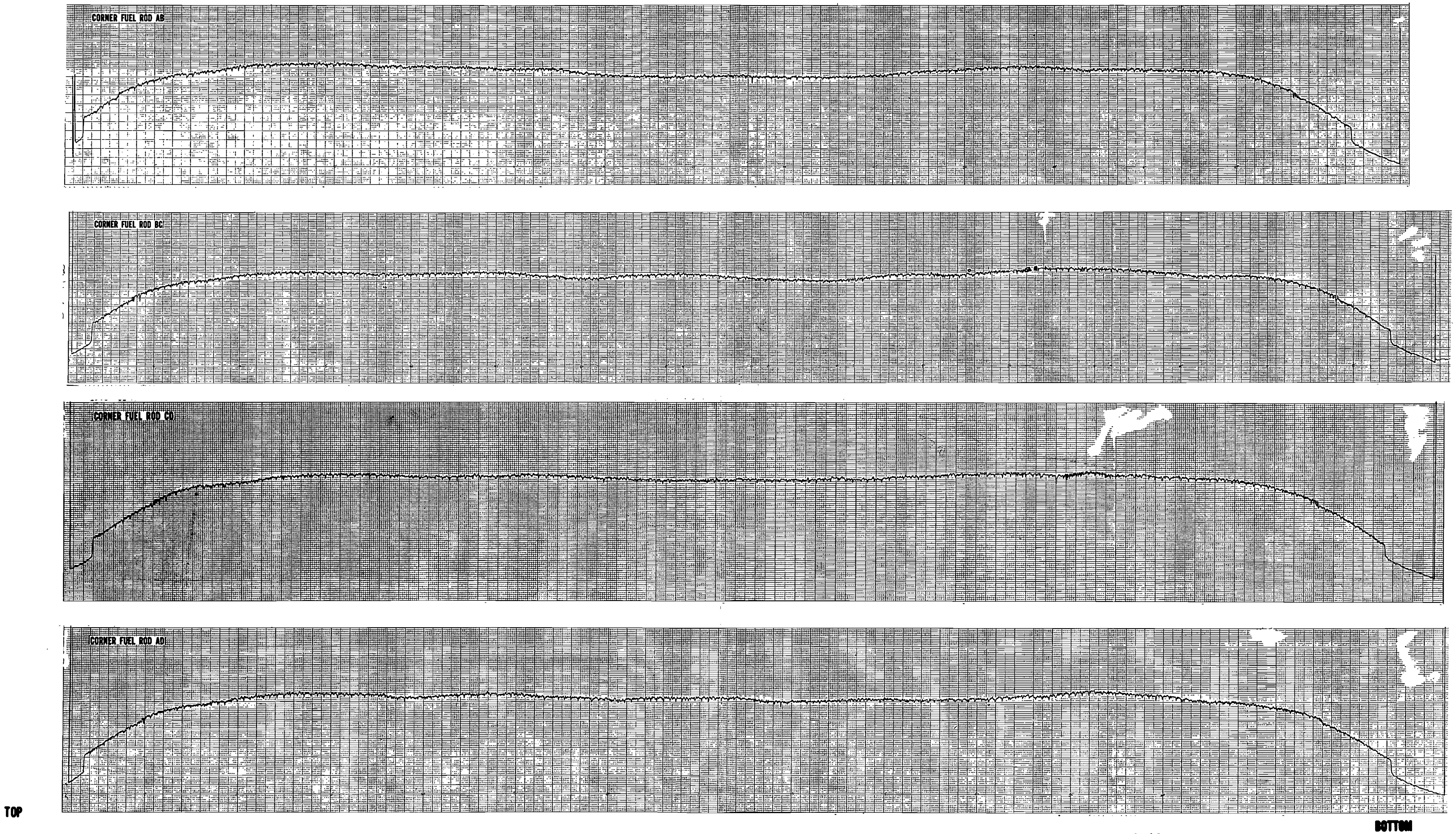


Figure 4-29. Oconee 1, Cycle 5 Power Distribution

	8	9	10	11	12	13	14	15
H	0.81 0.83	0.90 0.93	0.95 0.96	0.90 0.90	1.34 1.36	1.01 1.03	1.09 1.10	0.90 0.90
K		1.32 1.34	1.06 1.05	1.21 1.19	0.98 0.98	1.06 1.09	0.95 0.95	0.83 0.86
L			1.04 1.03	1.23 1.21	1.02 1.02	0.95 0.95	1.17 1.17	0.66 0.70
M				1.10 1.06	1.24 1.21	0.89 0.89	0.92 0.93	
N					1.22 1.19	0.93 0.94	0.65 0.63	
O						0.74 ← Measured 0.72 ← Predicted		

Location of Extended Burnup Assemblies (shaded: H-8, H-11)

	Meas	PDQ
Power, % FP	97.0	100
Exposure, EFPD	44.5	50
Group 6	100%	Out
Group 7	89%	Out
Group 8	28%	In
Boron, ppm	877	820

Measured 12/18/78, 44.4 EFPD
Predicted: PDQ

5. TASK 2 — OPERATIONAL LIMITS OF A MARK B CORE, PROGRESS TO DATE

5.1. Introduction

Progress on this task is dependent on detailed fuel cycle calculations being conducted under a subtask of the AP&L/DOE program, "Development of an Extended Burnup Mark B Design," contract number ET-78-C-02-4712. Progress on these fuel cycles is being reported under a separate document. The fuel cycles will provide information on power peaking, burnup distribution, control rod worths, and reactivity coefficients for use in mechanical, thermal-hydraulic, and safety evaluations of extended burnup fuel cycles. A brief discussion of the status of the analyses follows.

5.2. Subtask 2A — Fuel Cycle Design

The type of fuel cycles, anticipated batch average discharge burnups, and projected fuel enrichments being analyzed are shown in Table 5-1. The 80- and 60-FA cases were completed during this reporting period. Design power peaking criteria for these fuel cycles were met. Cycle lifetime and discharge burnups were verified by the detailed fuel cycle calculations, which indicate that initial projections of the uranium utilization improvement from extended burnup were essentially correct.

5.3. Subtask 2B — Nuclear, Mechanical, Thermal-Hydraulic Analyses

The primary endeavor during this reporting period has been the evaluation of the available post-irradiation data to identify and quantify constraints that will limit the allowable fuel assembly and component burnup.

The B&W Mark B fuel assembly is shown in Figure 5-1. The assembly can be considered in two parts — the fuel rods and the structural cage. The structural cage comprises two end fittings connected by guide tubes. Along the guide tubes are the spacer grids which hold the fuel rods in a coolable array. The two end grids are attached to the end fittings by skirts. The upper end fitting contains a helical holddown spring to prevent FA lift due to coolant flow.

When consideration is given to exposing the FA to higher burnups, the critical factors are neutron fluence and residence time. These factors cause three different structural effects on the assembly: material property changes, geometry changes, and fatigue.

5.3.1. Material Property Changes

The generalized effects of irradiation on the structural metals are to increase strength and decrease ductility. For conservatism, stress analyses for the structural cage design use the beginning-of-life strengths. For those components that may experience plastic strain after significant irradiation, the strain is limited by the design to low values to ensure a conservative margin allowing for ductility loss.

Stress relaxation due to irradiation and to the temperature in the material under constant stress is a concern for the helical holddown spring. Stress relaxation will cause a loss of free height of the spring after irradiation. Thus, this effect can result in loss of the holddown force required to prevent fuel assembly lift. As a counter-effect, fuel assembly growth causes greater spring compression, which increases holddown force. The net effect based on analysis of the PIE data obtained on the high-burnup lead assemblies at the 30,000 MWd/mtU point and from other B&W programs is almost no change in holddown force (∿1%) at hot operating conditions. This conclusion is obvious from the holddown force data presented in Figure 5-2.

5.3.2. Geometry Changes

Growth of Zircaloy under irradiation causes dimensional changes in the FAs, including increases in the length of the fuel rods and guide tubes. The allowable length changes impose limits on the fluence (burnup) to which the FAs can be exposed. The design interface between the reactor vessel internals and the FAs includes a gap between the upper end of the FA and the upper grid plate of the internals. The gap is necessary to accommodate differential thermal expansion and FA growth. Because the FAs expand less than the internals as the temperature increases, the gap is smallest at the cold shutdown condition.

Figure 5-3 shows the mean growth curve and a 95/95 upper tolerance limit curve for FA (guide tube) growth based on post-irradiation growth data from Mark B FAs with burnups ranging from 0 to 31,000 MWd/mtU. A statistical analysis considering the variability in as-built fuel assembly dimensions and the variance of the growth data resulted in a 0.5% probability of gap closure at a fast fluence of 7.3×10^{21} neutrons/cm^2 (E > 1 MeV), which corresponds to a fuel assembly average burnup of 43,000 MWd/mtU. This burnup limitation is conservative but does establish a bounding limit on fast fluence and consequently FA average burnup. Collection of growth data at the 40,000 MWd/mtU level is planned in late 1979. These data will allow better definition of the burnup limit arising from growth considerations and will also define the burnup limits of the batch 4D high-burnup assemblies. Higher fluences and therefore higher FA average burnups can be utilized for assemblies on which actual growth measurements after irradiation or as-built dimensions are available.

The fuel rods also grow due to irradiation; the growth curve is shown in Figure 5-2. This curve is based on PIE data for burnups up to 31,000 MWd/mtU. The distance between the upper and lower end fittings is greater than the fuel rod length, thus providing a gap to accommodate fuel rod growth. The rate at which this gap is closed is reduced by the fact that the guide tube growth increases this gap. The result is that the FA burnup at which there would be a 0.5% probability of gap closure is beyond 50,000 MWd/mtU FA average burnup.

5.3.3. Fatigue

The increased assembly residence time will result in more fatigue cycles. For the purpose of determining the number of fatigue cycles resulting from flow-induced vibration, residence time is defined as time in core with two or more primary coolant pumps running. Analyses also include low cycle events, such as heatup and cooldown and heat removal. However, fatigue is not expected to be a limiting factor on burnup due to large initial design margins for fatigue.

Based on the PIE data available up to the 30,000 MWd/mtU burnup level, it appears that FA (guide tube) growth will be the primary constraint that will limit the standard Mark B FA burnup. The collection of additional high-burnup PIE data planned under this program is expected to better define growth limits.

5.4. Subtask 2C — Safety Analysis and ECCS Evaluation

Although originally scheduled to begin in mid-1979, B&W was able to initiate selected portions of the safety analyses for extended burnup during 1978.

Developmental work has begun on a computer program (RELOAD) which will facilitate evaluation of the radiological consequences of extending fuel burnup. Extending the fuel burnup to approximately 50,000 MWd/mtU will change the fuel fission product inventory, which in turn will affect the dose impact of the accidents analyzed in the Final Safety Analysis Report (FSAR).

The RELOAD program is designed to provide (1) the accident doses for reload cores containing batches or sub-batches of high-burnup fuel, (2) a comparison of the doses for the reload cores with the corresponding FSAR accident doses, and (3) a comparison of the reload doses with the dose limits of 10 CFR 100. The program is being designed to calculate doses for the following major FSAR accidents:

1. Fuel handling accident.
2. Failure of waste gas decay tank.
3. Steam line break accident.
4. Rod ejection accident.
5. Loss-of-coolant accident.
6. Maximum hypothetical accident (MHA).
7. Steam generator tube failure.

All the accident doses are calculated based on an input fuel burnup history and on the plant specific accident parameters and assumptions described in the FSAR. The fuel burnup history includes power produced per fuel batch and by both uranium and plutonium fissions, and the total fuel burnup (full-power days). These fuel parameters are used to generate the fission product inventories for the fuel, fuel gap region, and reactor coolant using calculational methods described in section 11.1 of the FSAR. Specific plant accident parameters and assumptions for the major FSAR accidents will be taken from Table 2.9-1 of the FSAR.

The results of the dose rate calculations and the comparison of these values with FSAR and 10 CFR 100 limits will be utilized in licensing activities supporting high-burnup fuel cycles.

Table 5-1. Extended Burnup Fuel Cycles

Feed batch size	Core power level, MWt	Equil. cycle length, EFPD	Equil. batch burnup, ~MWd/mtU	Equilibrium enrichment, ~wt % ^{235}U
80	2772	460	34,000	3.5
68	2772	460	40,000	3.8
60	2772	460	46,000	4.1
60	2568	497	46,000	4.1
36	2772	292	48,000	4.0

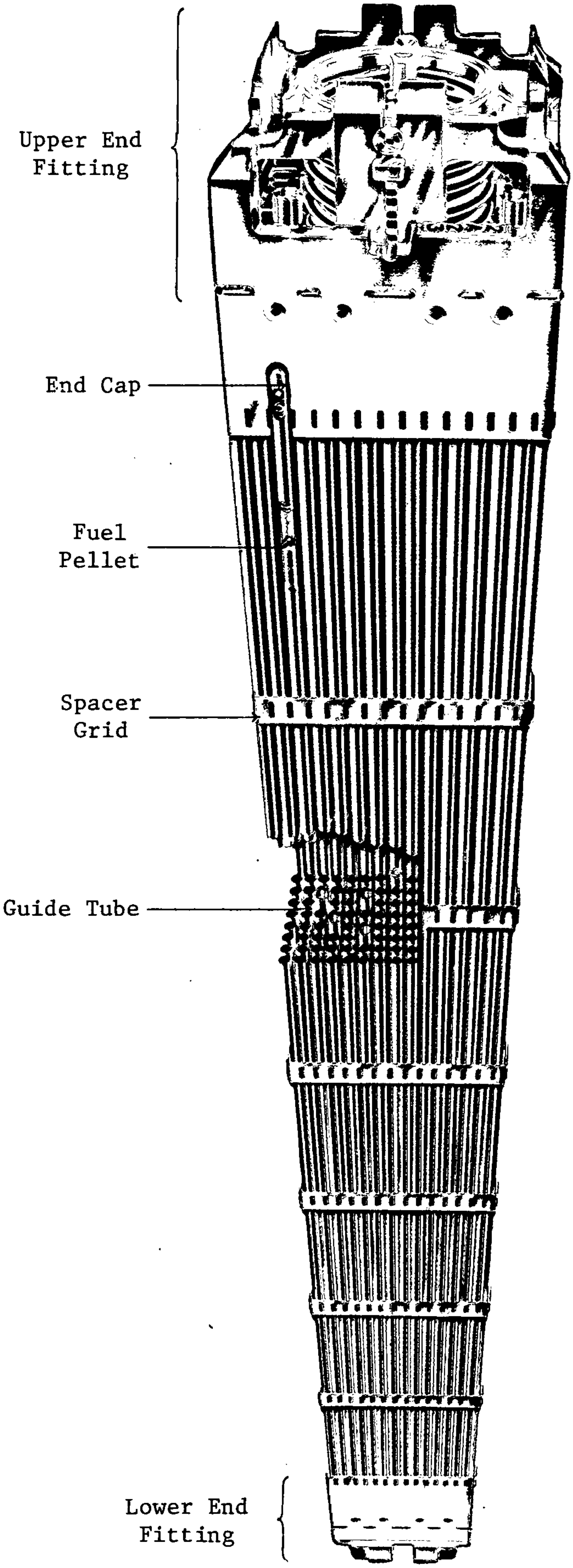


Figure 5-1.
Mark B Fuel Assembly

Figure 5-2. Holddown Spring Operating Force

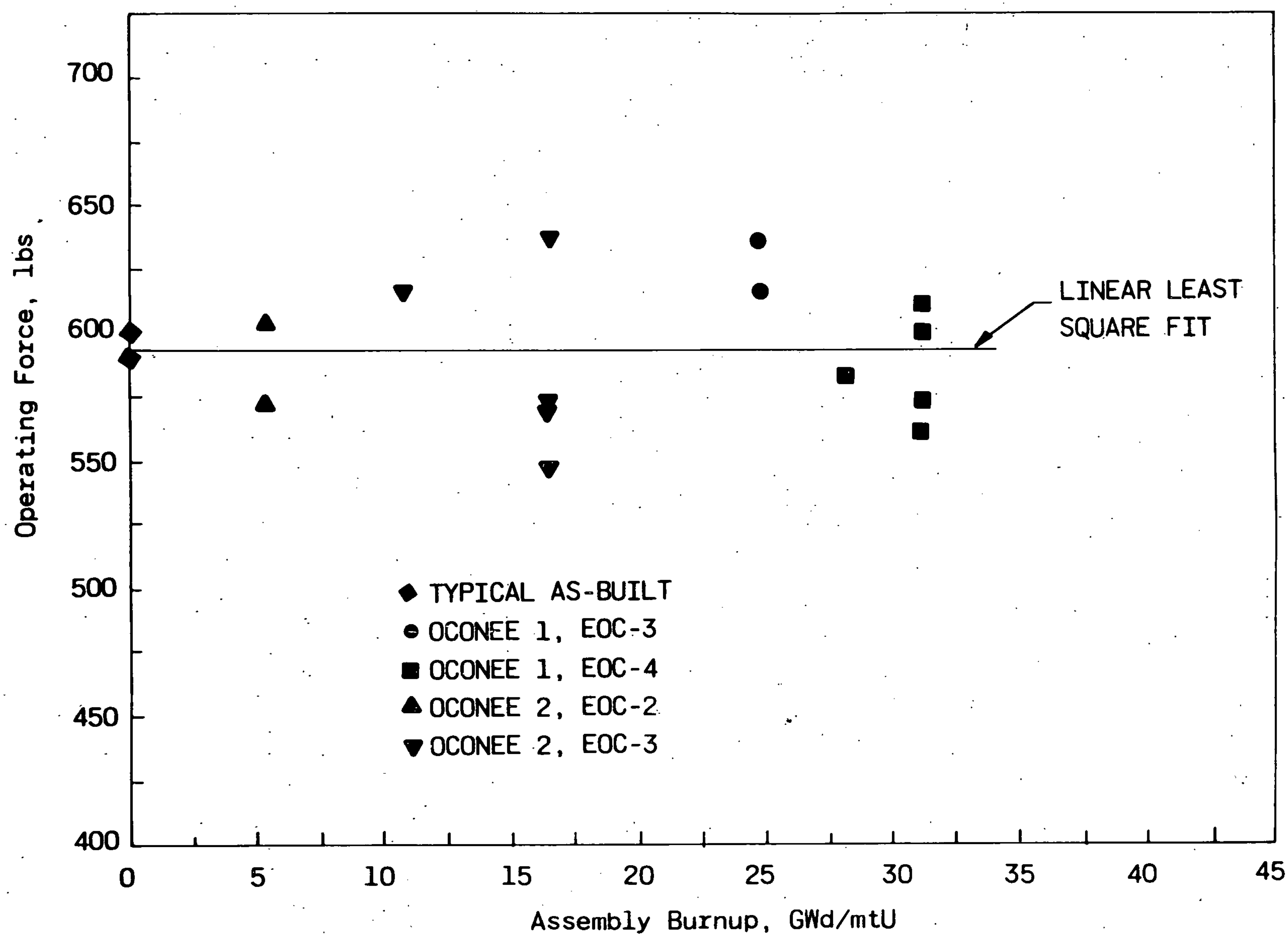


Figure 5-3. Fuel Assembly Growth Curve

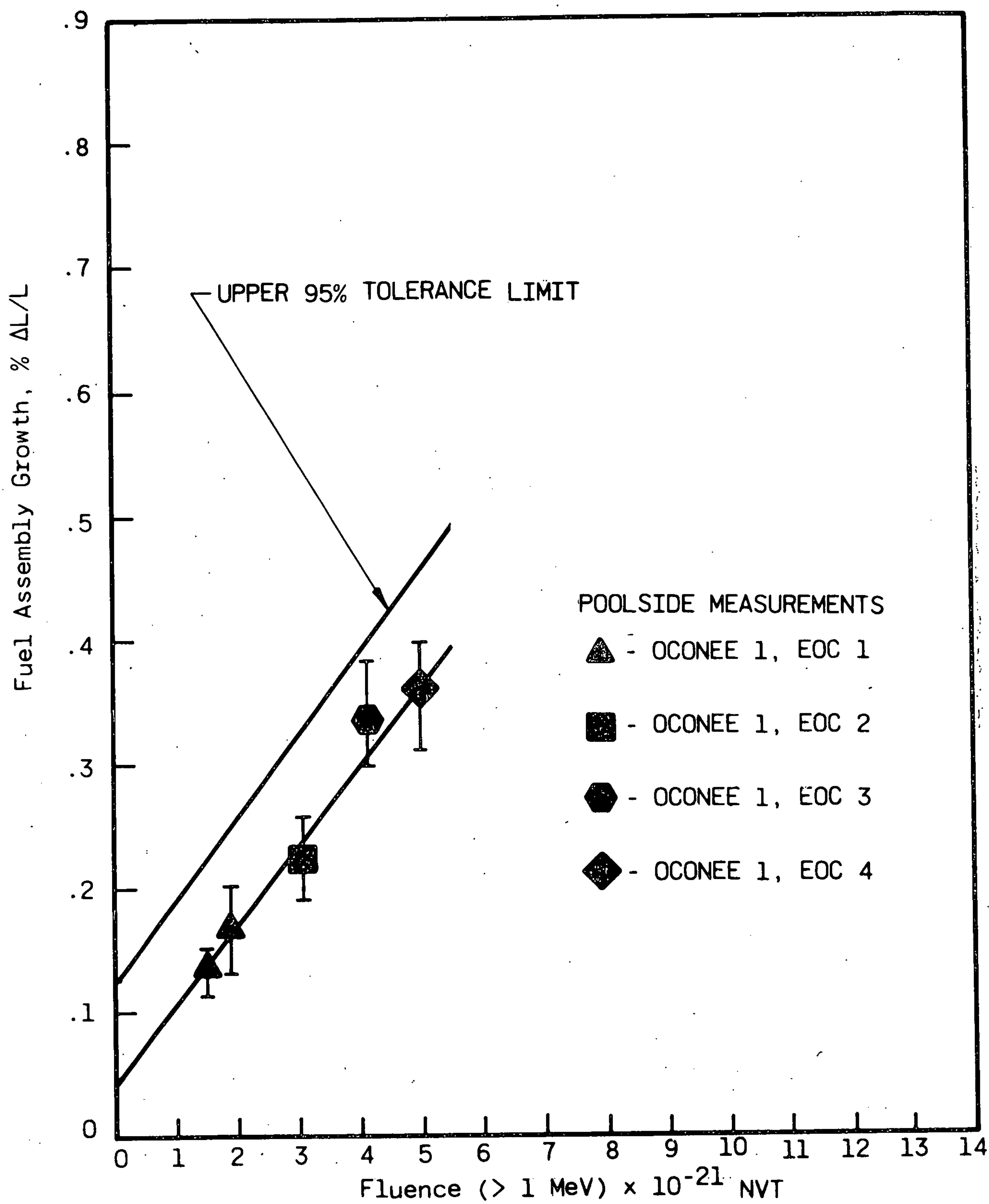


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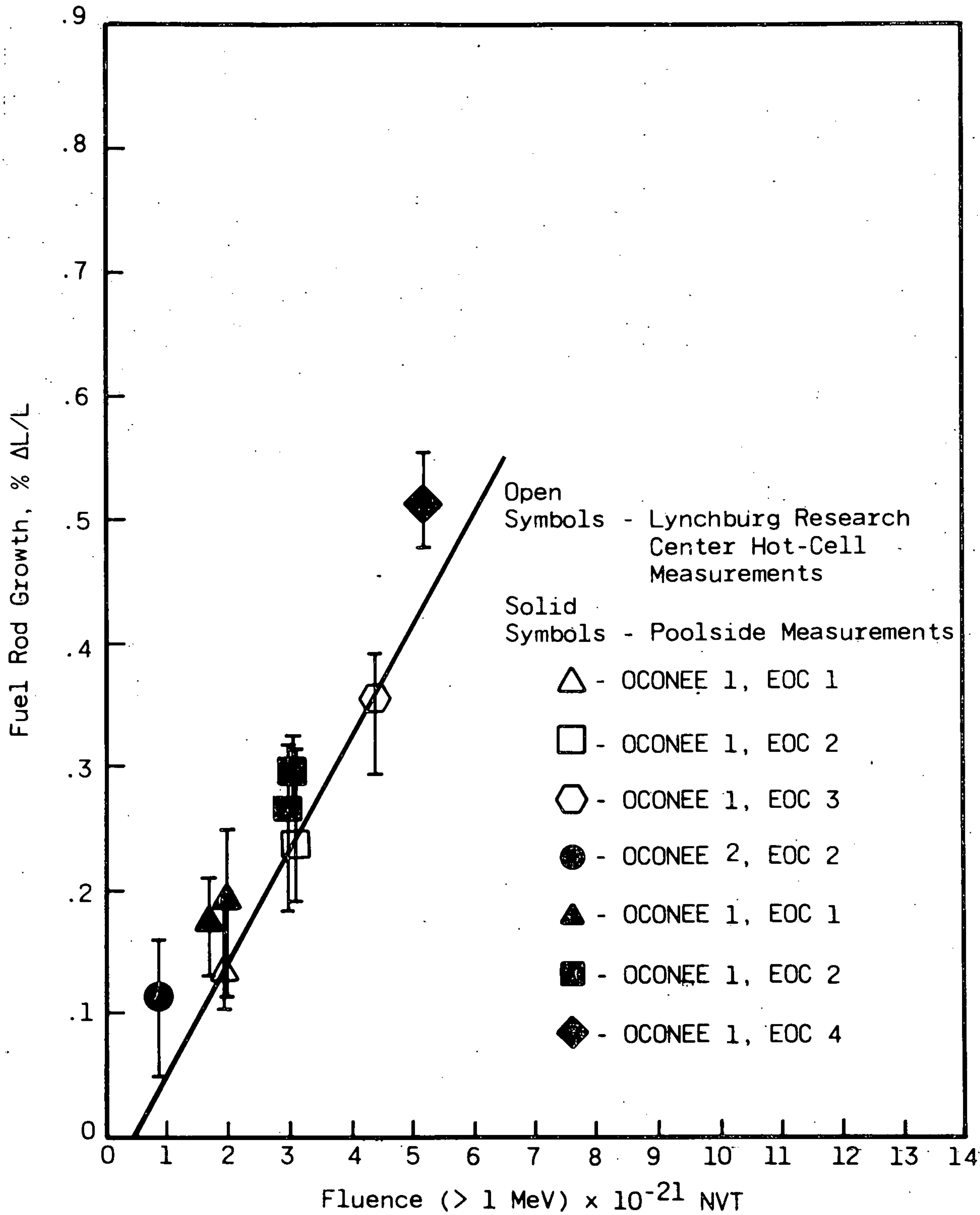
Figure 5-4. Fuel Rod Mean Growth Curve



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1000 Prospect Hill Road
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BATTELLE-NORTHWEST LABORATORY
Richland, WA 99352

Mr. F. E. Gelhaus
ELECTRIC POWER RESEARCH INSTITUTE
P. O. Box 10412
Palo Alto, CA 94303

Mr. S. J. Groetch
GENERAL ELECTRIC COMPANY
Knolls Atomic Energy Laboratory
P. O. Box 1072
Schenectady, NY 12301

Mr. R. M. Grube
YANKEE ATOMIC ELECTRIC COMPANY
20 Turnpike Road
Westboro, MA 01581

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NUCLEAR SERVICES CORP.
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Campbell, CA 95008

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Richland, WA 99352

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Mr. R. S. Miller
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Nuclear Fuel Division
Box 355
Pittsburgh, PA 15230

Mr. R. J. Mullin
TENNESSEE VALLEY AUTHORITY
1410 Commerce Union Bank Bldg.
Chattanooga, TN 37402

Mr. D. O'Boyle
COMMONWEALTH EDISON CO.
P. O. Box 767
Chicago, IL 60690

Mr. R. R. O'Laughlin
PUBLIC SERVICE OF INDIANA
1000 East Main Street
Plainfield, IN 46168

Mr. R. Omberg
HANFORD ENGRG DEVELOPMENT LABORATORY
P. O. Box 1970
Richland, WA 99352

Mr. B. A. Pasternak
BOOZ-ALLEN APPLIED RESEARCH
4330 East-West Highway
Bethesda, MD 20014

Others (Cont'd)

Mr. J. T. A. Roberts
ELECTRIC POWER RESEARCH INSTITUTE
P. O. Box 10412
Palo Alto, CA 94303

Mr. T. Row
OAK RIDGE NATIONAL LABORATORY
P. O. Box X
Oak Ridge, TN 37830

Mr. H. W. Schadler, Manager
Metallurgy Laboratory
GENERAL ELECTRIC CO. R&D CENTER
P. O. Box 8
Schnectady, NY 12301

Mr. T. Snead
DUKE POWER COMPANY
P. O. Box 2178
Charlotte, NC 28342

Mr. Howard Sobel
AMERICAN ELECTRIC POWER SERVICE
Nuclear Materials and Fuel
Management Section
2 Broadway
New York, NY 10004

Mr. G. Sofer
EXXON NUCLEAR COMPANY, INC.
Richland, WA 99352

Mr. I. Spiewak
OAK RIDGE NATIONAL LABORATORY
P. O. Box X
Oak Ridge, TN 37830

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Syracuse, NY 13202

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