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HIGH POWER DENSITY DEVELOPMENT PROJECT

SIXTEENTH QUARTERLY PROGRESS REPORT

January-March 1964

Compiled

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ATOMIC POWER EQUIPMENT DEPARTMENT

GENERAL ELECTRIC

SAN JOSE, CALIFORNIA

1779-TIO
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1. TASK IA - HIGH POWER DENSITY FUEL DEVELOPMENT

Failure of fuel rods in assemblies 1A, 2A, and 1F was verified by visual observation in the VBWR spent fuel storage pool.

The post-irradiation examinations and tests to be performed on Task 1A fuel assemblies were outlined in detail.

2. TASK 1B - FUEL FABRICATION DEVELOPMENT

The post-irradiation examination and tests to be performed on Task 1B fuel assemblies were outlined in detail.

Irradiation of eight Phase I developmental fuel assemblies continued in the Consumers Big Rock Point reactor with the lead assembly reaching an average bundle exposure of 2,660 MWD/t of uranium as of March 30, 1964.

Ten Phase II development fuel assemblies containing fuel rods clad with alternate materials, Zircaloy-2, Incoloy-800, and Inconel-600, were fabricated and shipped to the Big Rock Point reactor for insertion in May.

Poison rods containing B_4C powder in Incoloy-800 tubes were fabricated. These rods will be inserted into several Phase I and Phase II developmental fuel bundles to demonstrate axial power flattening.

A series of tests is being performed in the Trail Cable facility at GETR to verify the dimensional stability of thin clad, low density swaged powder fuel rods, when subjected to a 1500-psi boiling water environment. To date, two specimen failures have occurred which are believed caused by loop and specimen design.

All shop work and testing has been completed on the modified Instrumented Assembly for Big Rock Point. Two flowmeters have been modified with graphite bearings, and loop tested. Probe number three has been repaired and modified for use with the modified flowmeters. Probe, flowmeters, cabling, and recorders have been delivered to the site.

3. TASK II - STABILITY HEAT TRANSFER AND FLUID FLOW

A series of twelve control rod oscillation tests was run at various reactor conditions during the Phase I R and D tests at Big Rock Point.

A data reduction and analysis procedure has been developed which provides emperical reactor system stability data from the results of these tests; a complete reduction and analysis has been made for one of the tests.

4. TASK III - PHYSICS DEVELOPMENT

The fuel bundle placement for the 84-bundle 75 MWe core has been specified, including those bundles which will contain the power flattening B_4C rods.

Calculations are underway to finalize the rod withdrawal pattern from cold shutdown to operating conditions for this core. Calculations are also in progress to predict startup chamber response during the approach to critical, for the selected withdrawal sequence.

A method is being devised to allow a nodal burnout ratio calculation to be coupled to the present operating physics local peak power code. This will allow more rapid evaluation of various power distributions prior to the detailed thermal-hydraulics analysis.

Physics analysis has been completed on the Phase II developmental fuel assemblies, and is continuing on the reload fuel design.

Preparations are under way to implement the on-line computer program for the 84-bundle core, and the Phase II R and D tests. In addition, a number of programs have been modified as a result of the experience gained during the previous period of operation, and will be checked out prior to the May startup.

5. TASK IV - TEST PLANNING AND COORDINATION

Support was given to Consumers Power Company in licensing activities with the AEC to assure compatibility with the Phase II testing under the R and D program.

Discussions were held with the AEC to describe the intended application of instrumented fuel assemblies to Phase II testing at Big Rock Point.

Big Rock Point shut down at the end of the period to conduct annual inspections, special repairs, and to increase the core size for 75 MWe operation.

The reload fuel design recently initiated is being coordinated with development fuel design plans for insertion at Big Rock Point early next year.

Fuel pool equipment at Big Rock Point has been modified to improve performance during R and D testing.

Phase II tests have been detailed and Data Packages have been prepared and submitted to the Big Rock Point R and D Review Committee as required by license.

Core loading with development fuel in selected locations has been defined and analyzed with respect to expected physics and thermal hydraulic behavior.

TASK IA - HIGH POWER DENSITY FUEL DEVELOPMENT

1. FUEL ROD FAILURES

The final irradiation status and tabulation of failures for the Task IA fuel assemblies are given in the 15th High Power Density Development Project Quarterly Report, GEAP-4488.

Failure of fuel assemblies 1A, 2A, and 1F was visually verified by observation in the VBWR spent fuel pool.

Assembly 1A - One peripheral fuel rod has multiple circumferential cracks with a section of the fuel rod completely detached. Figure 1 is a photograph of fuel assembly 1A taken in the VBWR pool.

Assembly 2A - One corner and one peripheral fuel rod have predominantly circumferential cracks. Figure 2 is a photograph of the failed corner fuel rod.

Assembly 1F - Longitudinal cracks are apparent in a corner fuel rod. Figure 3 is a photograph of the failed corner fuel rod.

The appearance of the cracks in the annealed fuel rods of assemblies 1A and 1B is similar to those observed in prior failures of annealed Type 304 stainless steel cladding. The failure of annealed stainless steel cladding in a boiling water reactor environment is characterized by intergranular circumferential cracks rather than the intergranular longitudinal cracks which occur in cold worked stainless steel clad fuel rods.

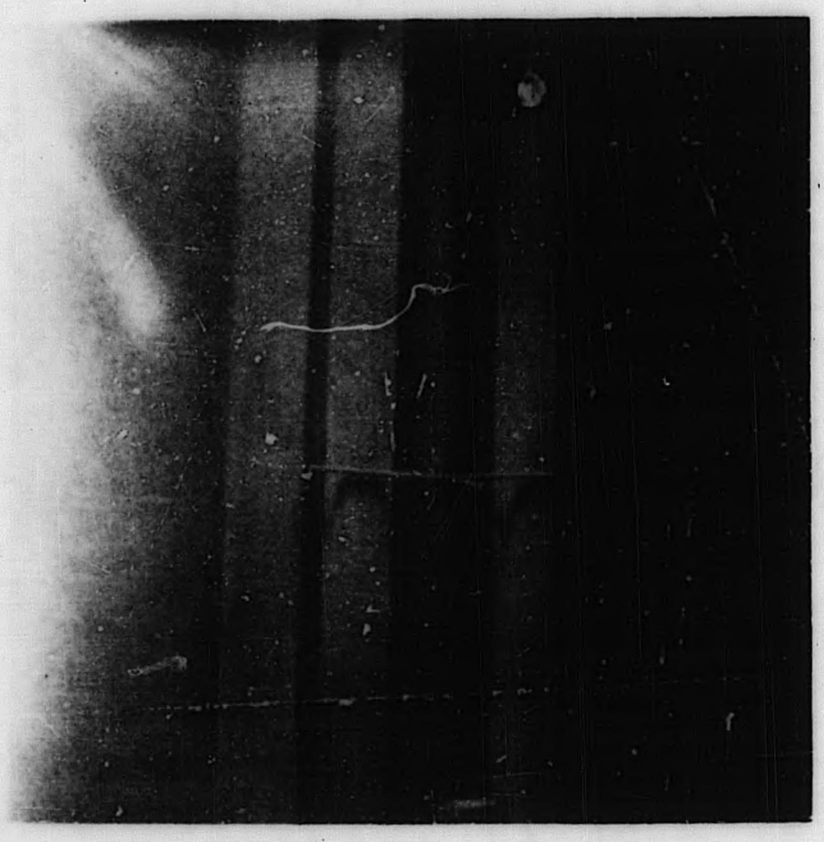
The circumferential cracking made may be related to high local clad deformation at UO_2 pellet interfaces which have been measured previously on an annealed fuel rod from assembly 2E (GEAP-4488).

The longitudinal cracks observed in the 1F assembly failed fuel rod appear typical of the many failures experienced in cold worked Type 304 cladding.

Verification that the failure of the 1A, 2A, and 1F fuel cladding was intergranular will be obtained when the planned post-irradiation metallographic examinations are performed.

2. POST-IRRADIATION EXAMINATION

No post-irradiation examinations other than the VBWR spent fuel pool observations were performed on Task IA fuel assemblies.



Assembly 1-A (Fuel Follower)

16 Fuel Rods: UO_2 Pellet Fuel Clad with Type 304 Stainless Steel
Annealed

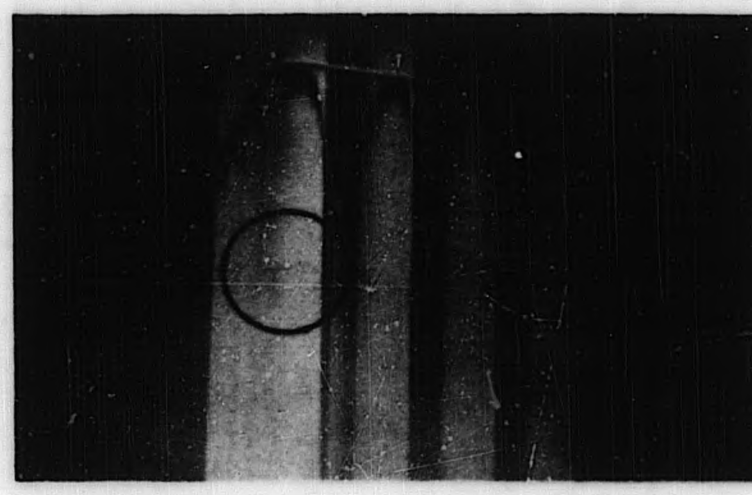
Cladding: Initial Room Temperature Yield Strength - 40,000 psi

0.420-inch OD
0.020-inch nominal wall
0.005-inch initial cold diametral pellet-to-clad gap

Burnup: 8,030 MWD/T of Uranium Bundle Average
12,450 MWD/T of Uranium Fuel Rod Peak

Heat Flux: 361,000 Btu/h-ft^2 Maximum

Figure 1. Fuel Assembly 1-A



Assembly 2-A (Fuel Follower)

16 Fuel Rods: UO_2 Pellet Fuel Clad with Type 304 Stainless Steel
Annealed

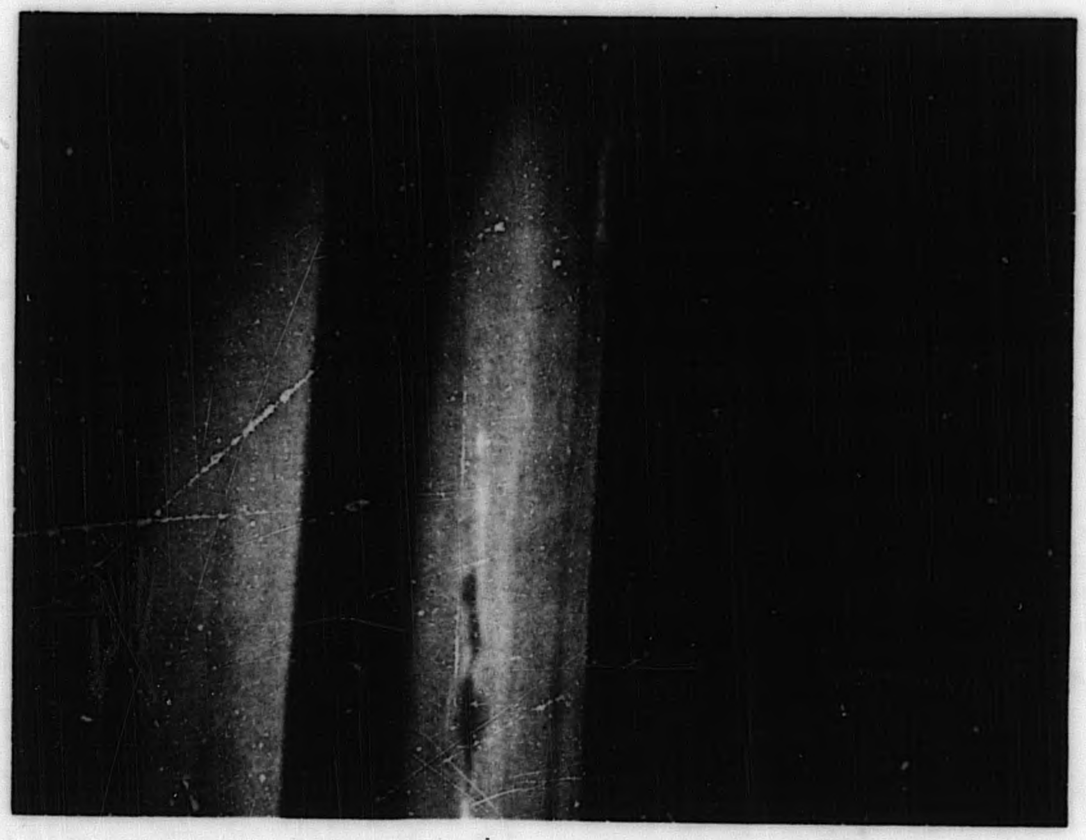
Cladding: Initial Room Temperature Yield Strength - 40,000 psi

0.420-inch OD
0.020-inch nominal wall
0.005-inch initial cold diametral pellet-to-clad gap

Burnup: 8,030 MWD/T of Uranium Bundle Average
12,450 MWD/T of Uranium Fuel Rod Peak

Heat Flux: 358,000 Btu/h-ft^2 Maximum

Figure 2. Fuel Assembly 2-A



Assembly 1F

25 Fuel Rods: UO_2 Pellet Fuel Clad with Type 304 Stainless Steel
Cold Worked

Cladding: Initial Room Temperature Yield Strength - 90,000 psi

0.360-inch OD
0.014-inch nominal wall
0.005-inch initial cold diametral pellet-to-clad gap

Burnup: 10,000 MWD/T of Uranium Bundle Average
15,000 MWD/T of Uranium Fuel Rod Peak

Heat Flux: 480,000 Btu/h-ft^2 Maximum

Figure 3. Fuel Assembly 1-F

Previously obtained post-irradiation examination and test data for the entire program are reported in topical reports issued during the period: "Fuel Failure Examinations and Analysis in the High Power Density Program," by W. H. Arlt and S. R. Vandenberg, GEAP-4360 (September 16, 1963); and "Analysis of Failure of Type 304 Stainless Steel Clad Swaged Powder Fuel Assembly," by E. A. Lees, GEAP-4400 (October 3, 1963).

Recent post-irradiation test results are also given in the Fourteenth and Fifteenth Quarterly Progress Reports, GEAP-4391 and GEAP-4488, respectively.

Post-irradiation examinations and tests to be performed on recently discharged Task IA fuel assemblies were outlined in detail and prepared for submittal to the AEC.

TASK IB - FUEL FABRICATION DEVELOPMENT

1. FUEL ROD FAILURES

The final irradiation status and tabulation of failures for the Task IB fuel assemblies are given in the Fifteenth Quarterly Progress Report, GEAP-4488.

2. POST-IRRADIATION EXAMINATION

No post-irradiation examinations were performed on Task IB fuel assemblies.

Post-irradiation examinations and tests to be performed on recently discharged Task IB fuel assemblies were outlined in detail, and prepared for submittal to the AEC.

3. PHASE I DEVELOPMENT FUEL

Eight Phase I development fuel assemblies continued to operate in the Consumers Big Rock Point reactor during the period. The Phase I assemblies each contain 121 fuel rods. Four assemblies designated PO-1, 2, 3, and 4 contain UO_2 powder fuel clad with 0.010-inch nominal wall Type 304L stainless steel. These assemblies were fabricated by swage compaction.

The other four assemblies designated PE-1, 2, 3, and 4 contain UO_2 pellet fuel and were fabricated by swaging the 0.010-inch wall cladding over UO_2 pellets. PE-1 and 2 are clad with Type 304 stainless steel and PE-3 and 4 with Type 304L stainless steel.

The accumulated average burnup of the lead assembly (PO-4) is approximately 2,600 MWD/t of uranium.

At least six of the Phase I assemblies will be examined visually in the Consumers Big Rock Point reactor spent fuel storage pool during May, 1964.

4. PHASE II DEVELOPMENT FUEL

The basic fuel assembly design criteria for the Phase II (Group II) fuel assemblies are summarized in Table I. The Phase I fuel assemblies and the tentative design of the next group of developmental fuel assemblies for the Consumers Big Rock Point Reactor are also given in Table I.

Ten Phase II developmental fuel assemblies were fabricated and shipped to the Big Rock Point reactor.

TABLE I

BIG ROCK POINT DEVELOPMENTAL FUEL DESIGN CRITERIA

Fuel Assembly		UO_2 Fuel			Cladding			Fabrication Process		Design Lifetime Average Bundle Burnup MWD/t Uranium	Current Status
		Type	Nominal Density Percent Theoretical	Nominal Enrichment Percent U-235	Material	Nominal Wall (inch)	Condition	UO_2 Fuel	Fuel Rod		
Group I	PE-1	Pellet	94	2.7	304	0.010	Annealed	Cold Press and Sinter	Swage over Sintered Pellets	10,000	In Reactor Since July 1963
	PE-2	↓	↓	↓	304	↓	↓	↓	↓	↓	↓
	PE-3	↓	↓	↓	304L	↓	↓	↓	↓	↓	↓
	PE-4	↓	↓	↓	304L	↓	↓	↓	↓	↓	↓
	PO-1	Powder	91	2.7	304L	0.010	Cold Worked	Arc Fuse	Swage compaction	10,000	In Reactor Since July 1963
	PO-2	↓	↓	↓	304L	↓	↓	↓	↓	↓	↓
	PO-3	↓	↓	↓	304L	↓	↓	↓	↓	↓	↓
	PO-4	↓	↓	↓	304L	↓	↓	↓	↓	↓	↓
Group II	D-1	Pellet	94	2.8	Zr-2	0.030	Cold Worked	Cold Press and Sinter	Standard	15,000	Complete for insertion May 1964
	D-2	↓	↓	↓	Zr-2	↓	↓	↓	↓	↓	↓
	D-3	↓	↓	↓	Zr-2	↓	↓	↓	↓	↓	↓
	D-4	Pellet	94	4.1	Incoloy-800	0.019	Annealed	Cold Press and Sinter	Standard	15,000	Fabrication in Process. Complete for insertion May 1964
	D-5	↓	↓	↓	Incoloy-800	↓	↓	↓	↓	↓	↓
	D-6	↓	↓	↓	Incoloy-800	↓	↓	↓	↓	↓	↓
	D-7	Pellet	94	4.5	Inconel-600	0.019	Annealed	Cold Press and Sinter	Standard	15,000	Fabrication in Process. Complete for insertion May 1964
	D-8	↓	↓	↓	Inconel-600	↓	↓	↓	↓	↓	↓
	D-9	↓	↓	↓	Inconel-600	↓	↓	↓	↓	↓	↓
	D-10	Powder	91	3.3	Incoloy-800	0.011	Cold Worked	Arc Fuse	Swage compaction	15,000	Complete for insertion May 1964
	D-11	↓	↓	↓	Incoloy-800	↓	↓	↓	↓	↓	↓
	D-12	↓	↓	↓	Incoloy-800	↓	↓	↓	↓	↓	↓
	D-13	Powder	88	3.3	Incoloy-800	0.011	Annealed	Arc Fuse	Swage compaction and Final Anneal	15,000	Design Verification and Fabrication Experiments in Process
	D-14	↓	↓	↓	Incoloy-800	↓	↓	↓	↓	↓	↓
	D-15	↓	↓	↓	Incoloy-800	↓	↓	↓	↓	↓	↓
Group III	D-16	Pellet	94	3.3	Incoloy-800	0.011	Annealed	Cold Press and Sinter	Swage over Pellets	15,000	Preliminary Design
	D-17	↓	↓	↓	Incoloy-800	↓	↓	↓	↓	↓	↓
	D-18	↓	↓	↓	Incoloy-800	↓	↓	↓	↓	↓	↓
	D-19	Powder	85	3.0	Zr-2	0.030	Annealed	Arc Fuse	Vibratory Compaction	15,000	Preliminary Design
	D-20	↓	↓	↓	Zr-2	↓	↓	↓	↓	↓	↓

Each Assembly contains 121 fuel rods.

109 Fuel Rods are 0.425 inch in diameter by 6 feet long

12 Corner Fuel Rods are 0.320 inch in diameter by 6 feet long

The ten assemblies included:

Three Zircaloy-2 clad pellet fuel assemblies (D-1, 2, 3),

Two Incoloy-800 clad pellet fuel assemblies (D-5, 6),

Two Inconel-600 clad pellet fuel assemblies (D-8, 9), and

Three Incoloy-800 clad swaged powder fuel assemblies (D-10, 11, 12).

Shown in Figure 4 is one of the fuel elements during final assembly of Zircaloy-2 clad fuel rods into an Incoloy-800 support structure.

The removable rod feature of the developmental fuel assemblies enables all fuel rods to be inserted into the structure from the top, simplifying the assembly operation.

The upper end of a fuel assembly with and without the upper handle and fuel rod retainer grid is shown in Figure 5. The handle is capable of being removed in a spent fuel pool to allow individual irradiated fuel rods to be removed and replaced by using simple long handled tools.

The ten Phase II fuel assemblies will be inserted in the Consumers Big Rock Point reactor in May, 1964. The remaining Incoloy-800 and Inconel-600 clad fuel assemblies will be completed during the next period (D-4 and D-7).

The fuel rod fabrication for three Incoloy-800 clad swage compacted fuel assemblies with the fuel rods annealed after swaging has been deferred pending results of design verification tests currently being performed in the GETR Trail Cable facility (D-13, 14, 15).

5. POISON ROD DEVELOPMENT

Poison rods will be inserted into several Phase I and Phase II developmental fuel assemblies to demonstrate axial power flattening in the Consumers Big Rock Point reactor.

The poison rods contain B_4C powder in Incoloy-800 tubes.

The over-all length of the 0.423-inch diameter poison rods is 6 feet; however, the length of the B_4C poison region is 36 inches. The B_4C portion of the fuel rod is located in the peak axial power region of the reactor by means of a 1-foot-long solid Type 304 stainless steel rod attached to the lower end of the B_4C section; and a 2-foot rod attached to the upper end of the B_4C section.

Two types of poison rods were fabricated to simulate possible future control rod elements. The B_4C poison rod design parameters are summarized in Table II. Four poison rods, two of each type, will be inserted into each of six Phase II fuel assemblies prior to insertion into

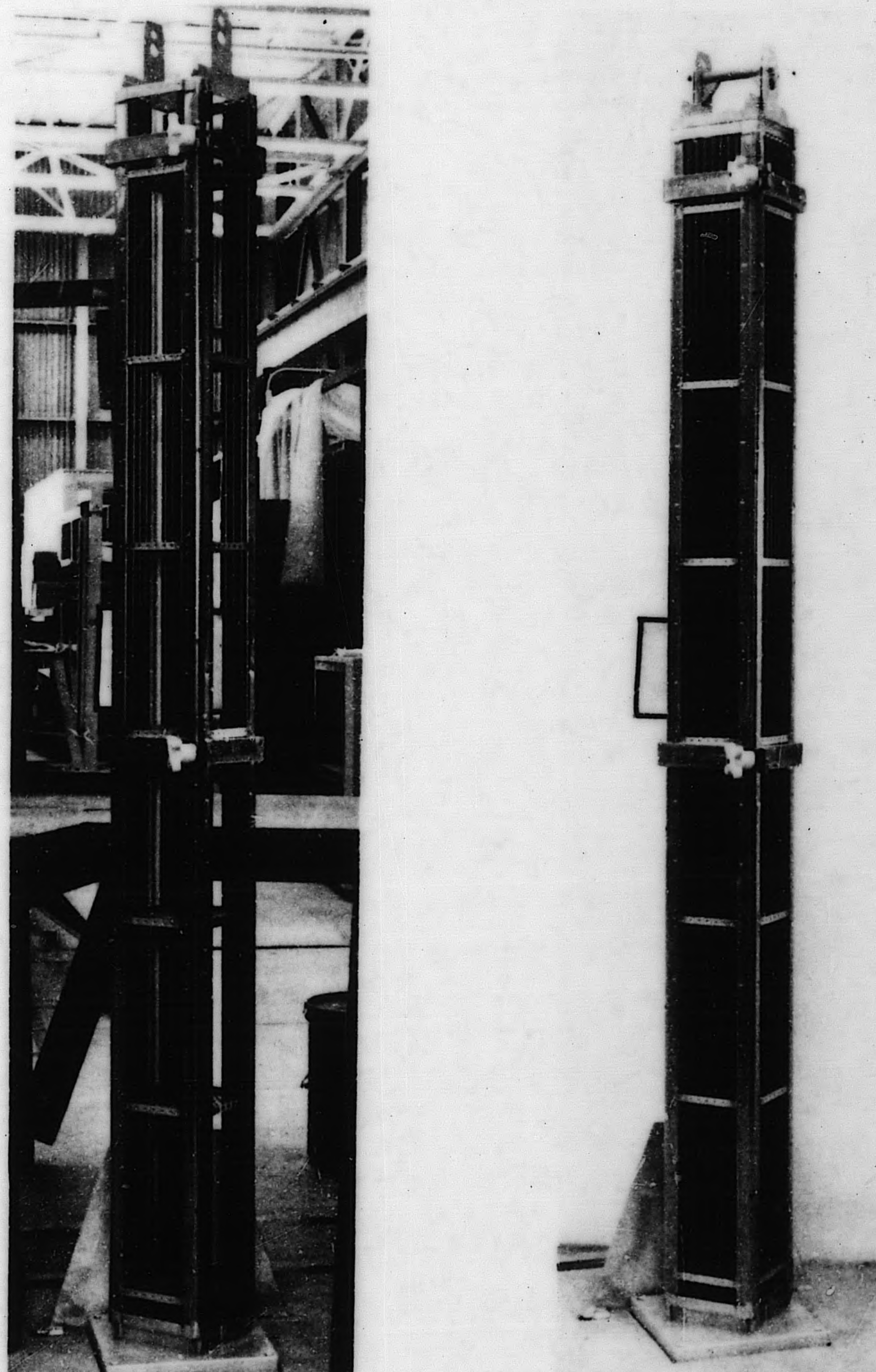


Figure 4. Phase II Fuel Element During Assembly

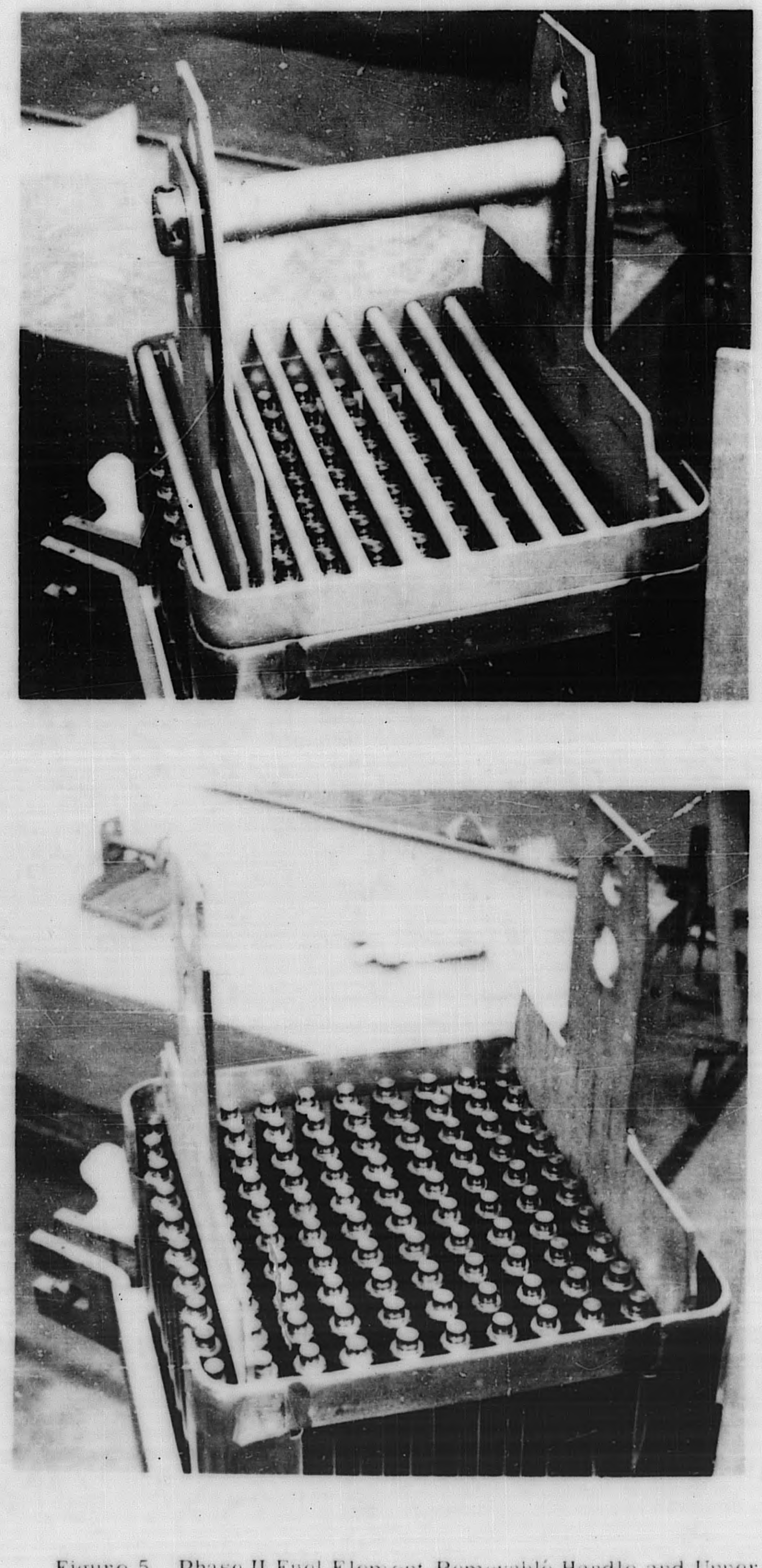


Figure 5. Phase II Fuel Element Removable Handle and Upper Grid

TABLE II

CONSUMERS DEVELOPMENTAL B_4C POISON ROD DESIGN PARAMETERS

Poison Rod	Poison		Poison Section Cladding			Compartments in Poison Section		Poison Rod Fabrication Method
	Material	Density Percent Theoretical	Material	Wall (inch)	Condition	Number	Length (inches)	
Type I	B_4C Powder	85	Incoloy-800	0.020	Cold Worked	1	36	Swage Compaction
Type II	B_4C Powder	70	Incoloy-800	0.019	Annealed	3	12	Vibratory Compaction Swage over compartment spacers

All poison rods are 0.423 inch in diameter and 6 feet over-all length. Poison section is 36 inches long attached to a 1-foot solid Type 304 stainless steel rod at lower end and a 2-foot solid Type 304 stainless steel rod at upper end.

the Big Rock Point reactor in May 1964. In addition, an attempt will be made to remove four fuel rods from each of six irradiated Phase I fuel assemblies and replace them with poison rods. The removal of fuel rods from the Phase I assemblies and replacement with poison rods will be the first significant test of the removable fuel rod design feature of the developmental fuel assemblies.

6. TRAIL CABLE EXPERIMENTS

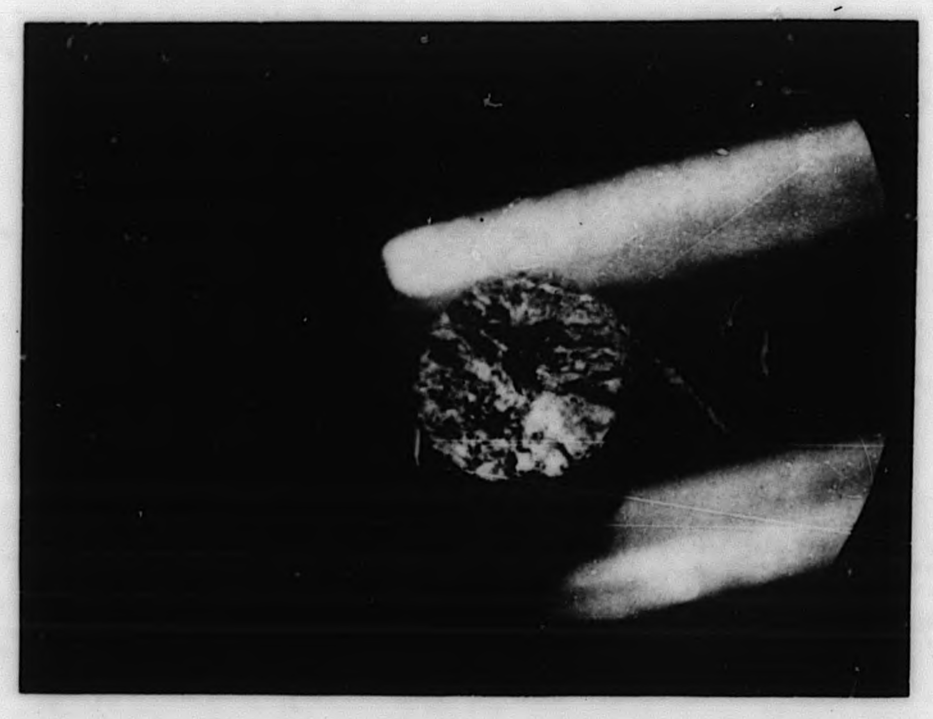
A series of tests is in progress to verify the dimensional stability of swaged powder annealed Incoloy-800 clad fuel rods when subjected to a 1500-psi boiling water environment. The results of these tests will be used to establish the fuel powder density and cladding yield strength required for three Phase II developmental fuel assemblies (D-13, 14, and 15).

These tests were performed in a water-filled, convection-cooled pressurized capsule in the GETR Trail Cable facility. The same capsule was used to irradiate similar thin stainless steel clad fuels (Ref. GEAP-4069) at 1000 psi. The present series of tests was the first attempt to use this capsule design to 1500 psi.

The first specimen to be irradiated operated satisfactorily for 1-1/2 hours at an average heat flux of 468,000 Btu/h-ft^2 before evidence of hydraulic oscillation of the natural circulating 1500 psi boiling water loop was observed. Simultaneously with the onset of hydraulic oscillations the radiation level on the capsule leads increased, indicating that the fuel had failed. The capsule was withdrawn from the Trail Cable facility and shipped to the RML. The fuel specimen was removed from the capsule in two sections. These sections were placed together and are shown in Figure 6. It appears that one-third to one-half of the cladding had melted because of the steam blanketing caused by the hydraulic oscillations.

The appearance of the UO_2 at the break is also shown in Figure 6. It appears that almost all of the fuel sintered, and as a result little fuel was lost from the fuel specimen. A small central void is also evident. If it is assumed that the edge of the void is the edge of the molten UO_2 (~2800°C), and that the thermal conductivity of powder fuel is equivalent to the conductivity of pellet fuel*, the surface temperature of the fuel at this section is about 1300°C. It is quite probable that the edge of the void is not the limit of the molten region and that the thermal conductivity of compacted powder is somewhat lower than pellet fuel which would mean that the fuel surface temperature was somewhat higher than 1300°C. A fuel surface temperature $\geq$ 1300°C is consistent with the damage to the clad after steam blanketing had occurred.

* Transactions American Nuclear Society, Vol. 6, No. 1, June, 1963, page 152.



Fuel Specimen C-5

0.425 - inch OD
0.011 - inch Wall
Clad - Annealed Incoloy-800
UO_2 - Swage Compacted to 87.5 percent

Failed after operating at a heat flux of 468,000 Btu/h-ft^2 average for 1-1/2 hours in a Water cooled capsule in the GETR Trail Cable Facility. The fuel rod was cooled by natural circulating 1500-boiling water.

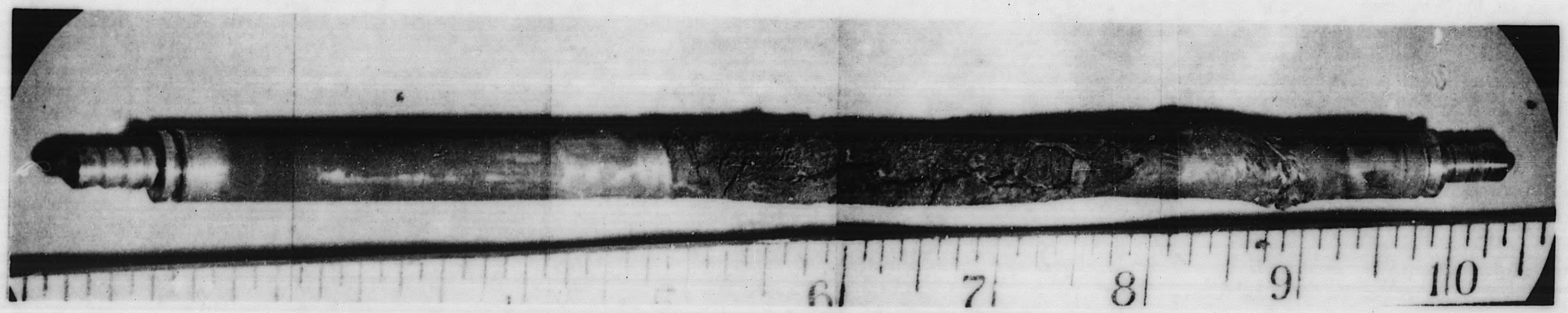


Figure 6. GETR Trail Cable Experiment - Fuel Rod Burnout Failure

For the second fuel specimen irradiated, a different capsule design was used to help reduce the possibility of hydraulic oscillations. This fuel specimen operated satisfactorily for about 1 hour at an average surface heat flux of 530,000 Btu/h-ft^2, when a step increase in power of about 5 percent was observed which was followed by an increase in radiation level on the capsule leads terminating the test. A post-irradiation photograph of this fuel sample is shown in Figure 7.

Excessive internal gas pressure in conjunction with possible local or general overheating of the cladding is believed to be the mechanism for the burst. High apparent UO_2 fuel temperatures are evidenced by the UO_2 visible in the burst region which appears to have been molten.

Further analysis of post-irradiation data will be performed to establish a method of performing these tests successfully.

7. CONSUMERS INSTRUMENTED ASSEMBLIES

All equipment required for reactor installation of one probe with flowmeters, and one probe without flowmeters has been shipped to Big Rock Point.

Two flowmeters were modified to reduce crud access to the bearings and chances of locked rotor failure. The modifications include:

a. Replacing the cantilever shaft with a more conventional double ended shaft to increase bearing length and reduce side loading.

b. Replacing the stellite bushings with high density graphite to obtain nonseizing characteristics.

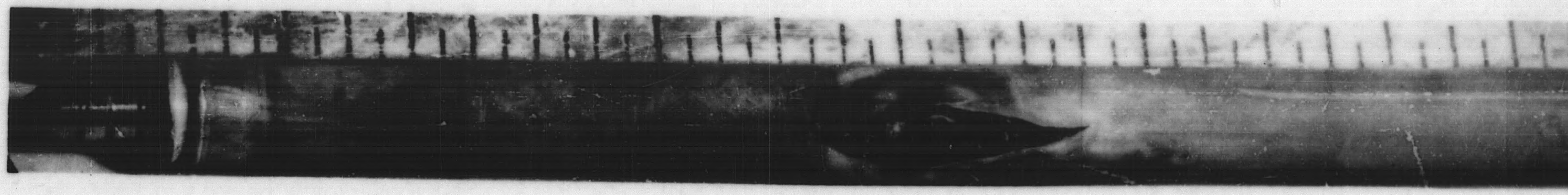
c. Shortening the probe by 1 inch to allow for the new shaft.

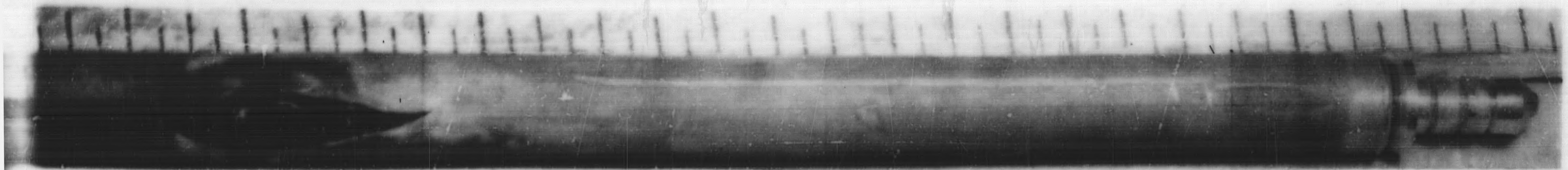
d. Adding an extension to the armature pickup assembly to allow the pickup coil to be moved.

e. Adding a fixed upper thrust pad to eliminate the necessity of the pickup coil to serve as a thrust bearing.

f. Elimination of coolant flow through the bearings to minimize crud access.

Both flowmeters were tested in the Heat Transfer Test Facility at San Jose, and the cold and hot single-phase measurements show no measurable change from previous calibrations of the same meters. Hot two-phase measurements showed slightly lower flowmeter readings for the same calculated steam rate conditions, amounting to about 1/2 percent in quality measurement (i.e., flowmeter indication of 5-1/2 percent versus calculated of 6 percent). This deviation could be either due to errors in calculating the quality, or significant meter bearing friction. Readings at hot single-phase and low quality two-phase do not bear out the likelihood





Test Specimen C-6

0.425 - inch OD
0.011 - inch Wall
Clad - Annealed Incoloy-800
UO_2 - Swage Compacted to 87.5 percent of theoretical density

Failed after operating at a heat flux of 530,000 Btu/h-ft^2 for 1 hour in a water cooled capsule in the GETR Trail Cable Facility. The fuel rod was cooled by natural circulation 1500 psi boiling water

Figure 7. GETR Trail Cable Experiment - Fuel Rod Burst Failure

of bearing drag, however, and bearing wear was within the expected limits for new graphite bushings. Diametral wear was almost identical for both flowmeters: 0.0001 inch on the upstream bushings, and 0.0004 inch on the downstream bushings. Nominal initial clearance was 0.001 inch, with maximum possible diametral wear (for satisfactory flowmeter performance) in excess of 0.020 inch.

One of the existing probes (number three) was repaired and modified for use with the modified flowmeters. This probe was previously damaged at the reactor site and was returned to San Jose for repair. The leads to one of the pickup coils were rewelded, and one thermocouple was repaired.

In addition, several probe modifications were performed which were necessitated by the modified flowmeter design.

Improvements in the pickup coil coupling fingers and armature design have more than doubled the flowmeter output signal from that obtained in prototype tests. Previous tests showed the modified pickup would be a factor of 7 lower than the unmodified pickup. Tests on the flowmeter as built show this factor to be now 3 instead of 7. This means that the output voltage at rated flow will be at least 0.7 volt, or about 30 times the threshold sensitivity of the frequency to analog converter.

Two of the probe connectors were rewired with fiberglass insulated, stainless steel sheathed cable rated at 950°F. This type of cable has been used as thermocouple wire under 3 MeV irradiation approaching 10^9 rads. A stainless steel flexible conduit was installed over the cable and welded to the connector for added physical protection.

An attempt was made to straighten some of the kinks in the flux wire tube, but it is doubtful that satisfactory wire insertion will be possible. The flexible 34-foot-long assembly makes it practically impossible to maintain the straight lengths or smooth bends required for easy wire insertion. A spring temper 1/16-inch OD thin-wall tube has been ordered. An attempt will be made to insert this smaller tube inside the 3/16-inch tube. It is felt the smaller tube will present smoother bends and allow a better fit of the wire to the tube ID. Installation of this smaller tube will take place at the reactor site as soon as it is available.

Parts for two new flowmeters have been ordered for standby use in the event of failure of the two existing meters. Sets of stellite ball bearings are also on order should the graphite prove totally unsatisfactory. After loop test evaluation of the ball bearings, a decision will be made whether to use ball bearings or graphite bearings for the two new meters.

TASK II - STABILITY, HEAT TRANSFER AND FLUID FLOW

1. PHASE I ROD OSCILLATION TEST DATA REDUCTION

During the Phase I tests twelve rod oscillation tests were run at various conditions of reactor power, recirculation flow, reactor pressure, and inlet subcooling. During these tests the following variables were recorded:

a. In-core neutron flux at five locations,

b. Out-of-core neutron flux at two locations,

c. Steam flow and reactor pressure, and

d. Control rod position.

The entire process of recording, reproducing, and analyzing the rod oscillation data is pictured in Figure 8. The process is described below.

Station 1 - The signal from the primary sensor is introduced to the recording system. After the signal is properly adjusted for recording, the recording channel is calibrated by means of a voltage introduced during interruption of the input signal.

Station 2 - The d-c component of the incoming signal is suppressed.

Station 3 - The suppressed signal is amplified.

Station 4 - The amplified signal is put through a noise filter to filter the electrical noise.

Station 5 - The conditioned signal is introduced to the tape recorder.

Station 6 - The FM signal is recorded on the tape.

Station 7 - Later, the tape record is reproduced through the reproduce amplifier at a tape speed eight times that of the original data recording, and played into a Frequency Response Analyzer.* Prior to reproduction, the tape is formed into a closed loop, with the tape splice formed as accurately as possible to provide amplitude and phase continuity of the rod position signal. The tape loop enables the record of rod oscillation to be replayed as an apparently endless experiment.

Station 8 - At the input to Frequency Analyzer, any remaining d-c component is again nulled.

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* Boonshaft-Fuchs Model 711A.

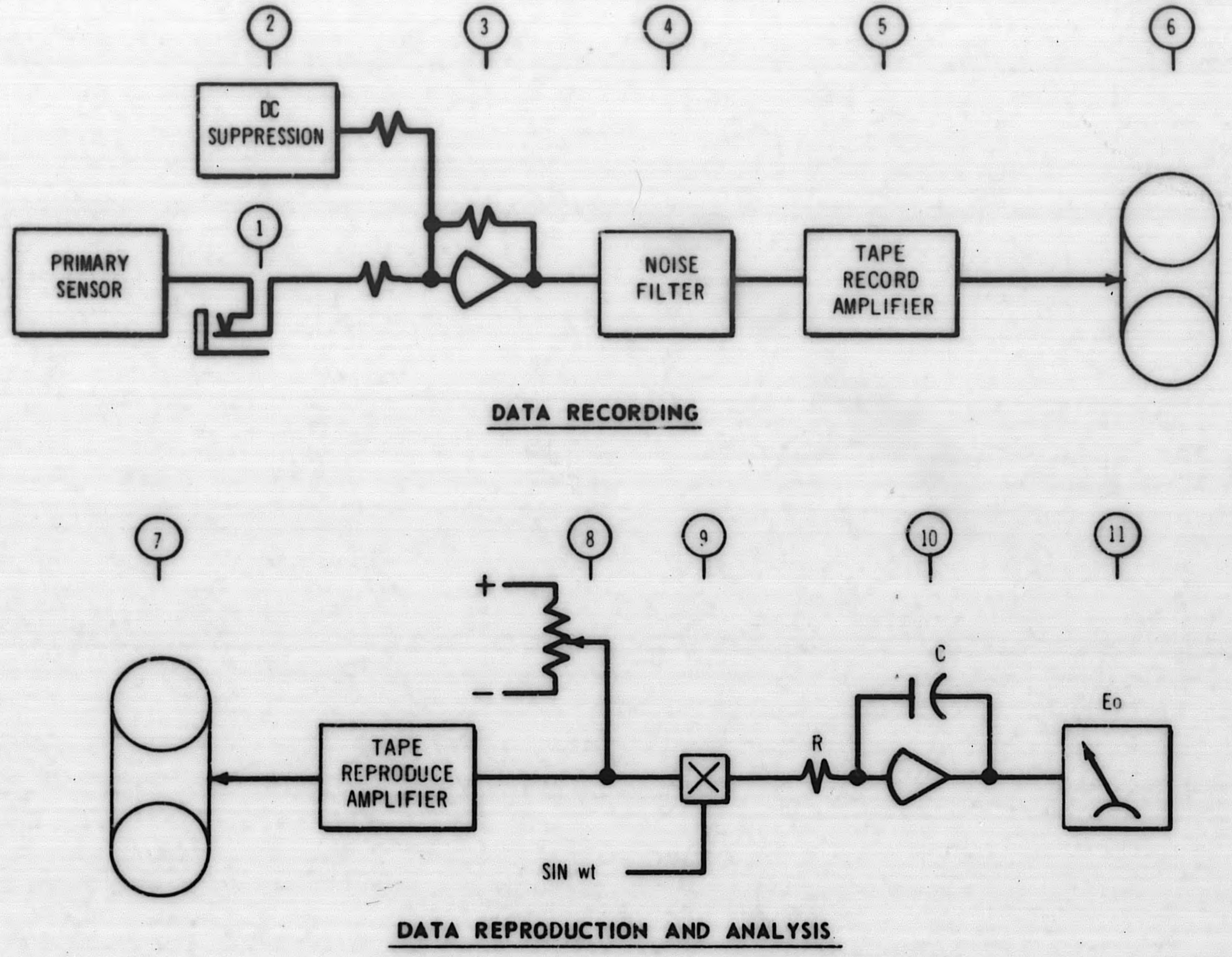


Figure 8. Data Recording and Analysis

Station 9 - The suppressed oscillation signal is multiplied by sin ωt.

Station 10 - The resulting signal is integrated over an accurately specified time.

Station 11 - The signal is read out as a voltage, having magnitude and polarity.

No changes are made in any settings during the recording or reproducing process, once they have been set up. Thus, aside from any d-c drift or gain instability in the equipment, the signals are considered to have the same scale factor for all frequencies.

A recording channel similar to that shown in Figure 8 exists for various signals, including in-core and out-of-core ion chambers and the oscillated control rod position. Only two signals at one time are analyzed on the Frequency Analyzer, the rod position and one neutron flux signal. Each signal introduced into the Analyzer is multiplied by both sin ωt and cos ωt. The products are then integrated as shown in Figure 8, and read out as the in-phase and quadrature components (i.e., real and imaginary components) of the input signal.

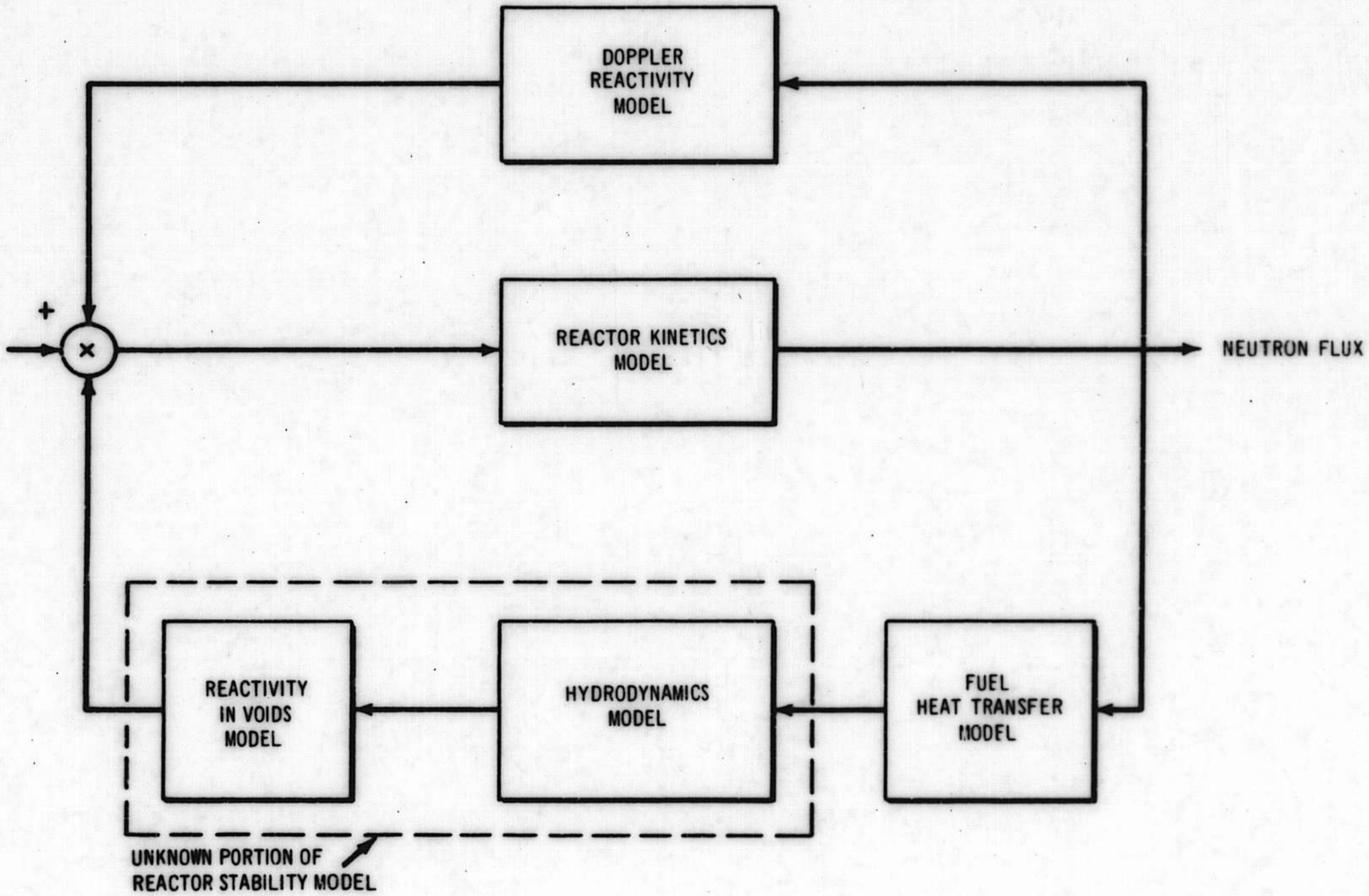
The magnitude and phase response of the system are then obtained from these components, after which the effect of the transient response of the recording system is subtracted, to give the empirical value of the reactor system closed-loop transfer function.

A polynomial curve fit of the closed-loop gain and phase is made on the digital computer. This provides the ability to determine the open-loop reactor transfer function; the gain or phase margins; the emperical power-to-reactivity feedback function from the reactor stability model as shown in Figure 9. It is assumed that the Reactor Kinetics, Doppler Reactivity, and Fuel Heat Transfer models are well established models. The effect of these known models is "subtracted out" of the polynomial fit of the closed-loop reactor system transfer function to give the unknown portion of the stability model. Other items of interest are then computed, such as the open-loop transfer function and gain or phase margins.

An extensive error analysis of the data reduction process has been made, and it has been determined that when reasonable care is taken during the recording and data analysis process, the error in the measurement of the reactor closed-loop transfer function may be held to less than 0.4 db (magnitude ratio) and 2 degrees (phase) with 95 percent confidence.

The data reduction and analysis process was applied to Rod Oscillation Test 45 with the following results:

a. Figure 10 presents the closed loop reactivity-to-flux transfer function, and contains both the emperical results and the pre-test predictions.



<u>Figure 9.</u> **Reactor Stability Model**

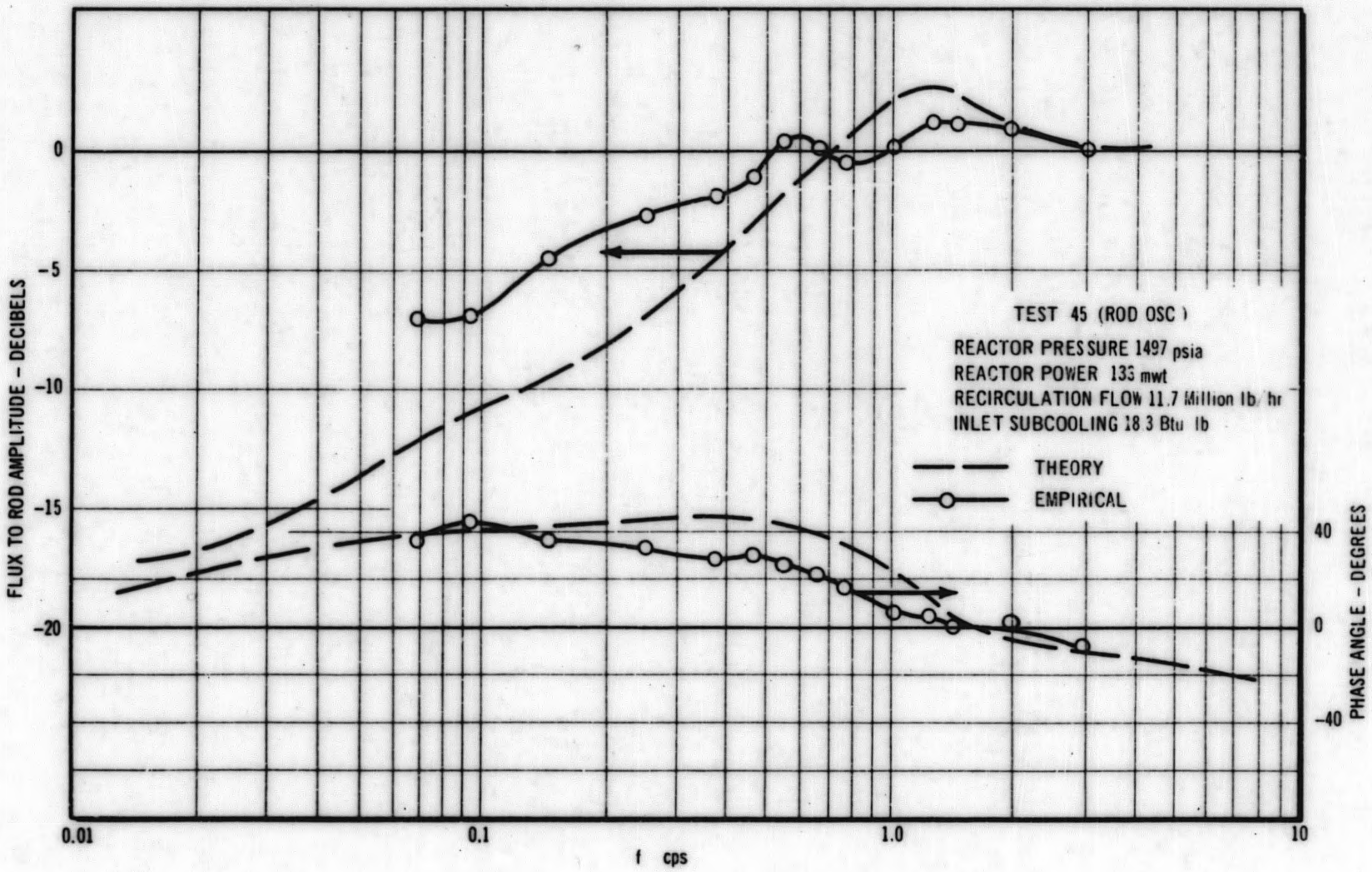


Figure 10. Test 45 - Closed Loop Reactivity-To-Flux Transfer Function

b. The emperical results in Figure 10 were given a polynomial curve fit as shown in Figure 11. The Reactor Kinetics and Doppler Reactivity Models shown in Figure 12, and the Fuel Heat Transfer Model in Figure 13 were then "subtracted" from the polynomial curve fit. The results are shown in Figures 14 and 15 as the empirical open-loop transfer function and the empirical power-to-reactivity-in-voids feedback transfer function. From Figure 14 the empirical phase margin of 104 degrees is obtained

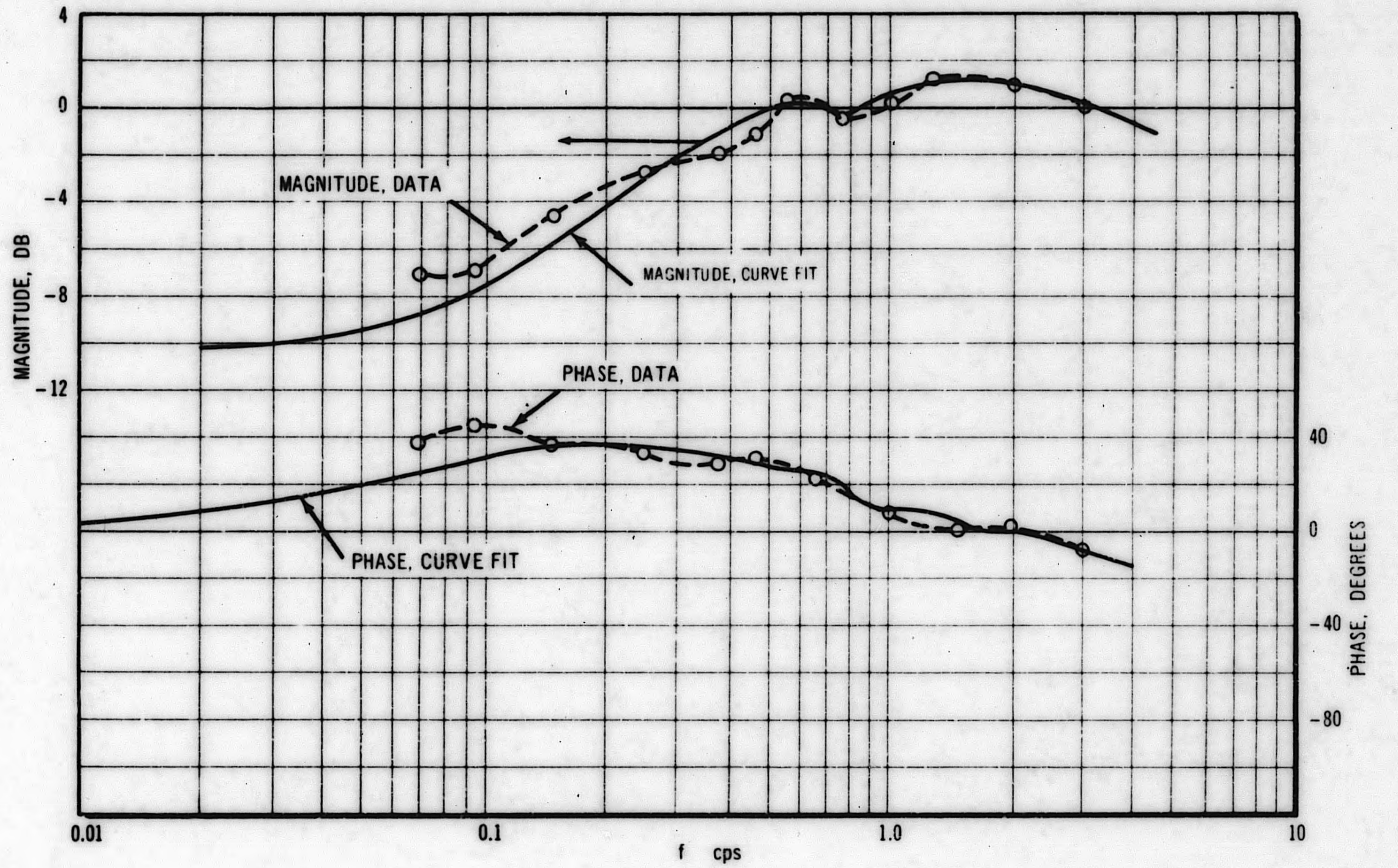


Figure 11. Test 45 - Closed Loop Transfer Function - Curve Fit

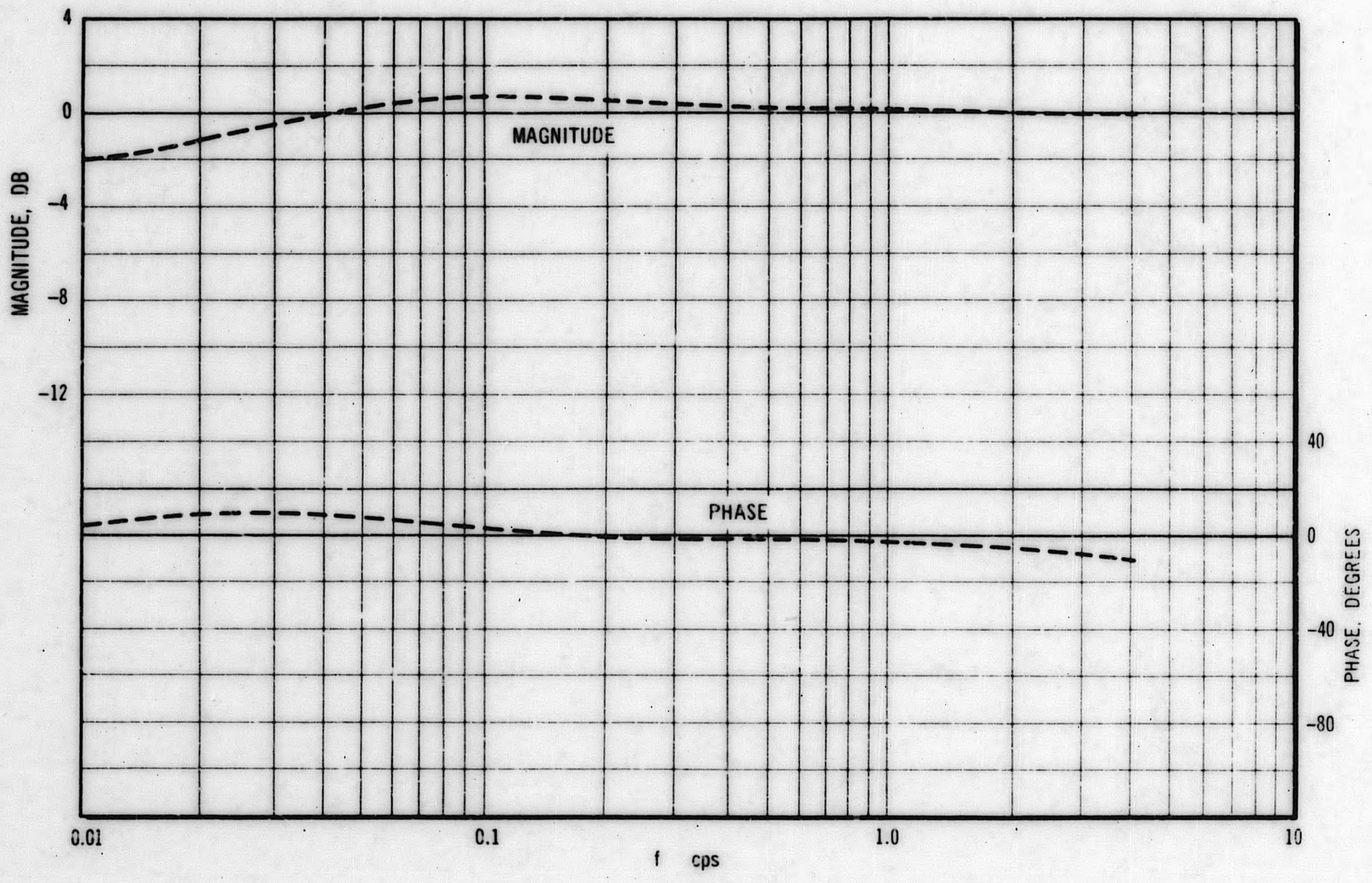


Figure 12. Test 45 - Calculated Forward Loop Transfer Function (Reactor Kinetics and Doppler Reactivity Models)

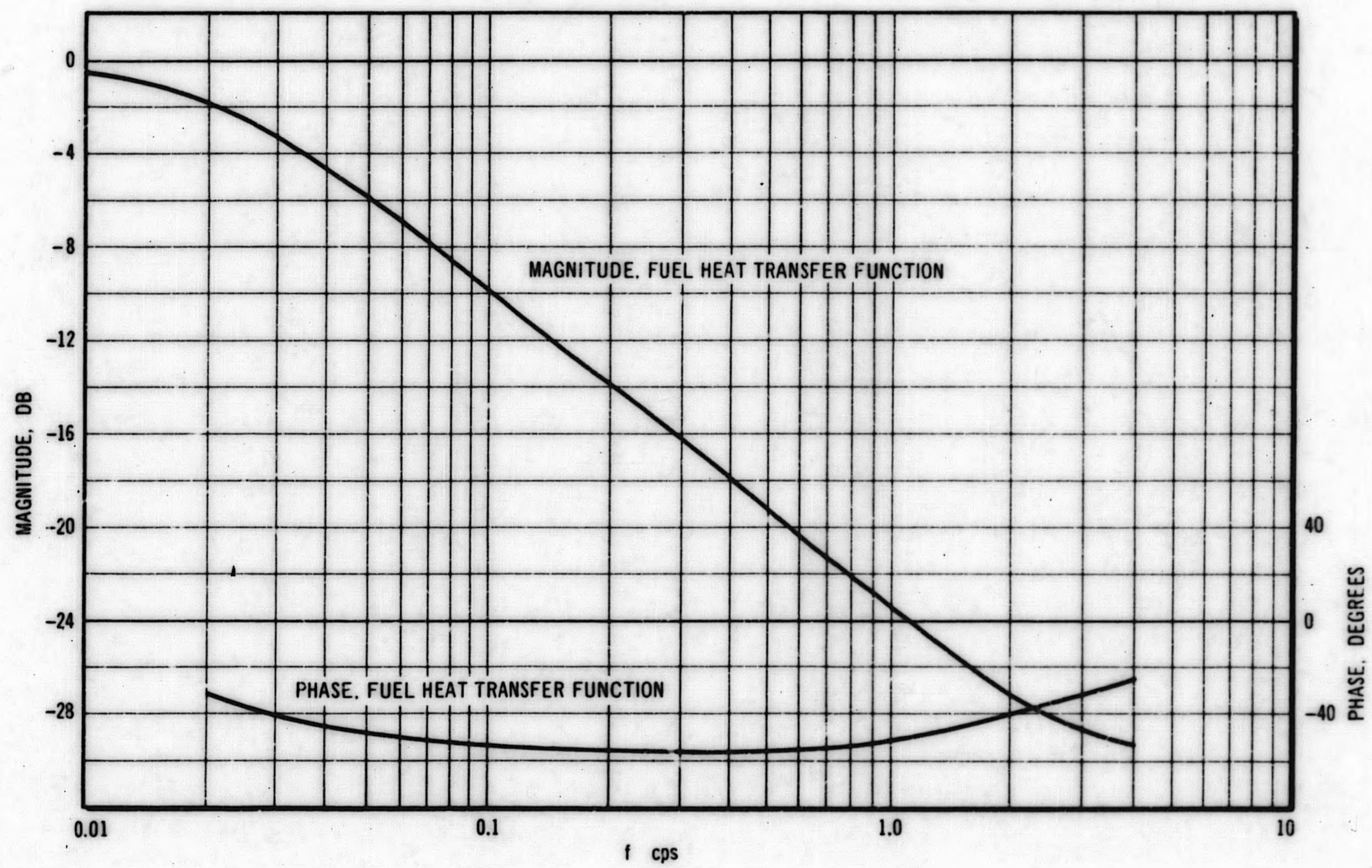


Figure 13. Test 45 - Calculated Fuel Heat Transfer Model

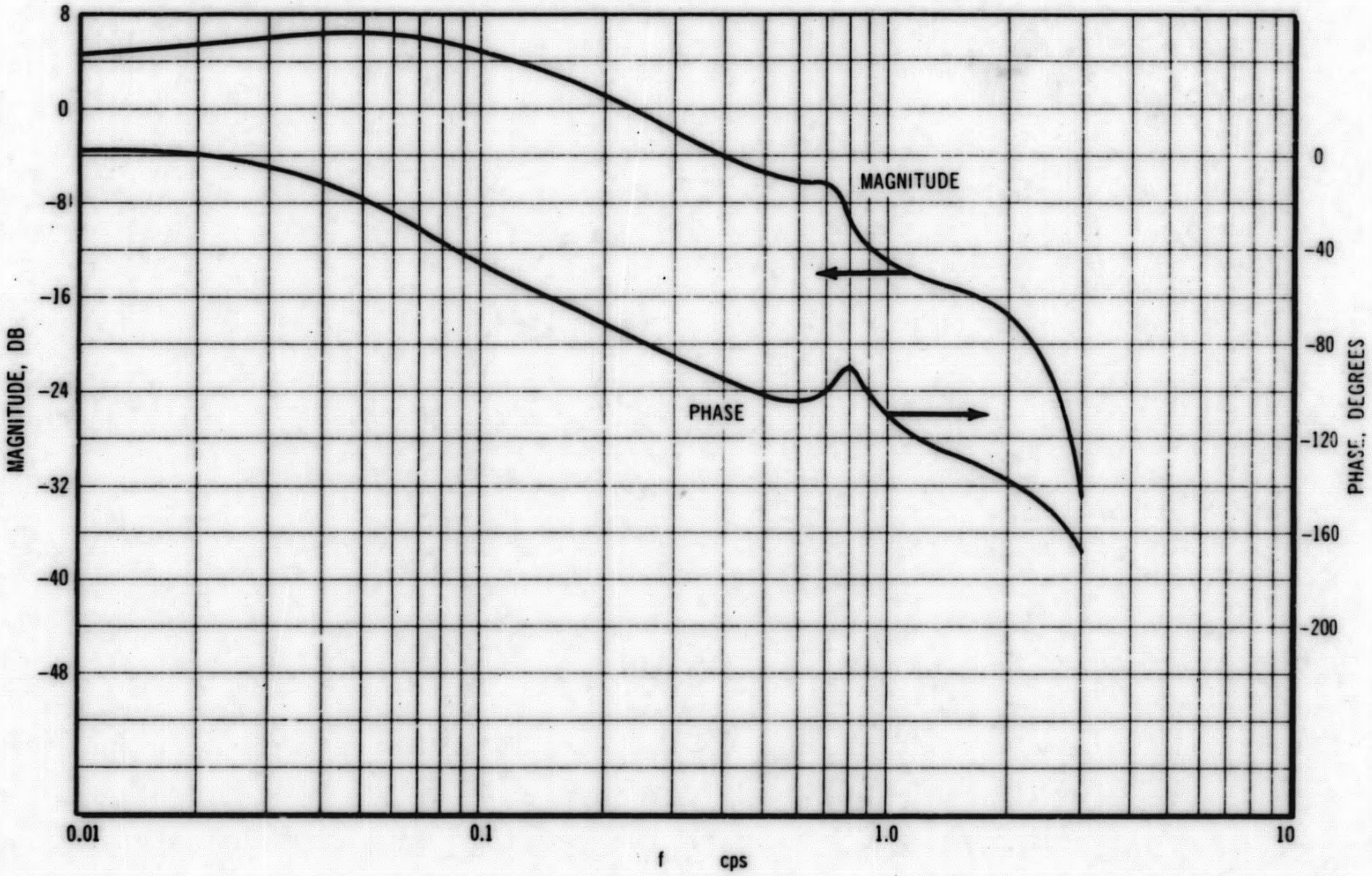


Figure 14. Test 45 - Empirical Open Loop Magnitude/Phase

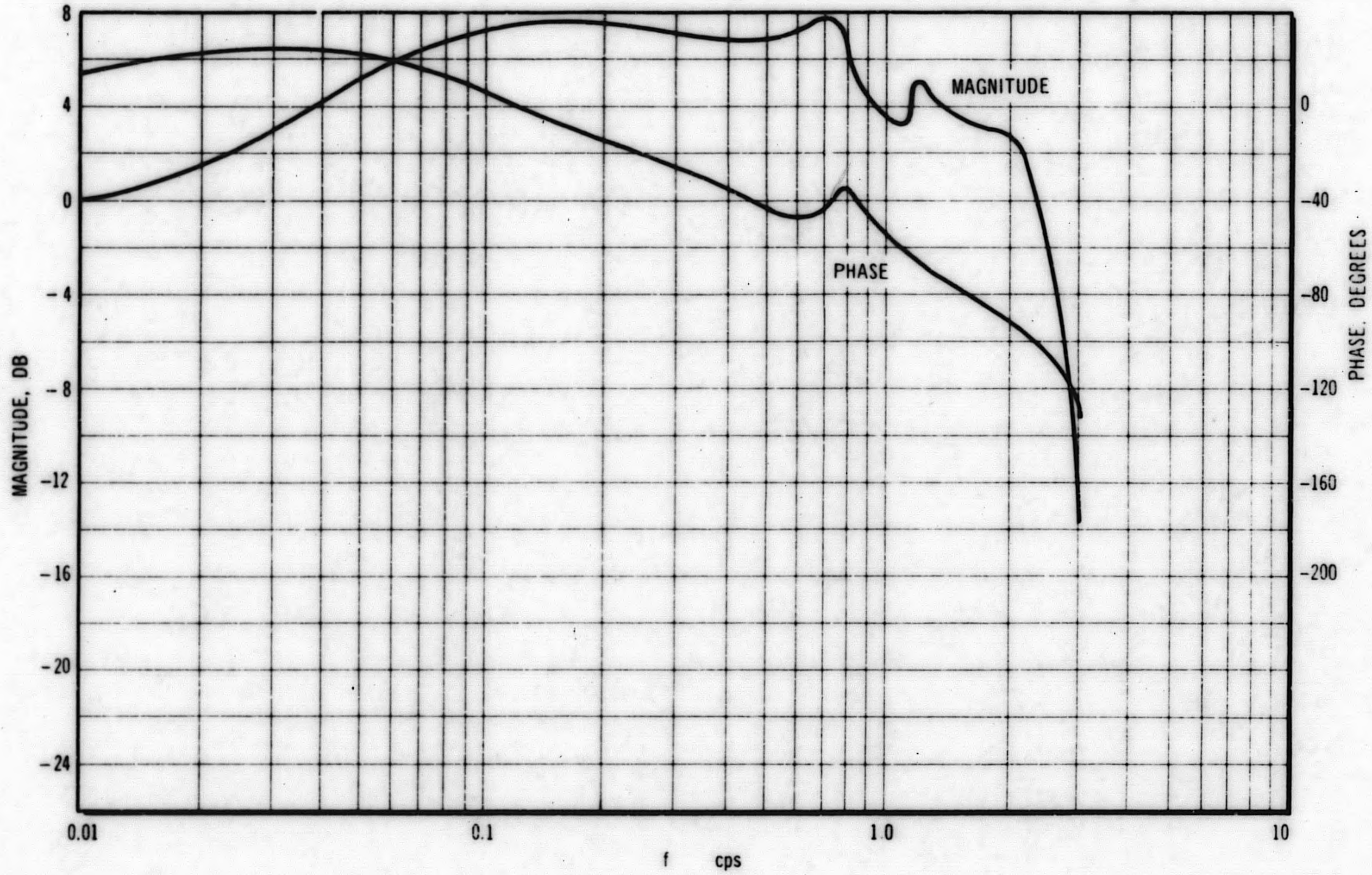


Figure 15. Test 45 - Empirical Hydraulics and Reactivity-In-Voids Relationship

TASK III - PHYSICS DEVELOPMENT

1. 75 MWe OPERATIONS

The 84-bundle core bundle placement has been specified as shown in Figure 16. An attempt has been made to make the transition from the 74-bundle core to the 84-bundle core, followed by the reduction in core size for the 60 kW/l testing to involve as few fuel bundle moves and channel changes as possible.

2. POWER SHAPING B_4C RODS

During Phase II of the R and D Programs selected R and D fuel assemblies will be loaded with rods containing distributed poison. Those bundles selected (six Phase I and six Phase II assemblies), will each contain four poison rods which replace the first large fuel rods diagonally in from the corner of each assembly.

The design characteristics of these rods are described under Task IB. The effect of inserting these rods is to reduce bundle $k\infty$ of the controlled region by approximately 14 percent at operating conditions.

This distributed poison has yielded reductions of core average axial power peaking of approximately 10 percent, providing a significant increase of core operating flexibility.

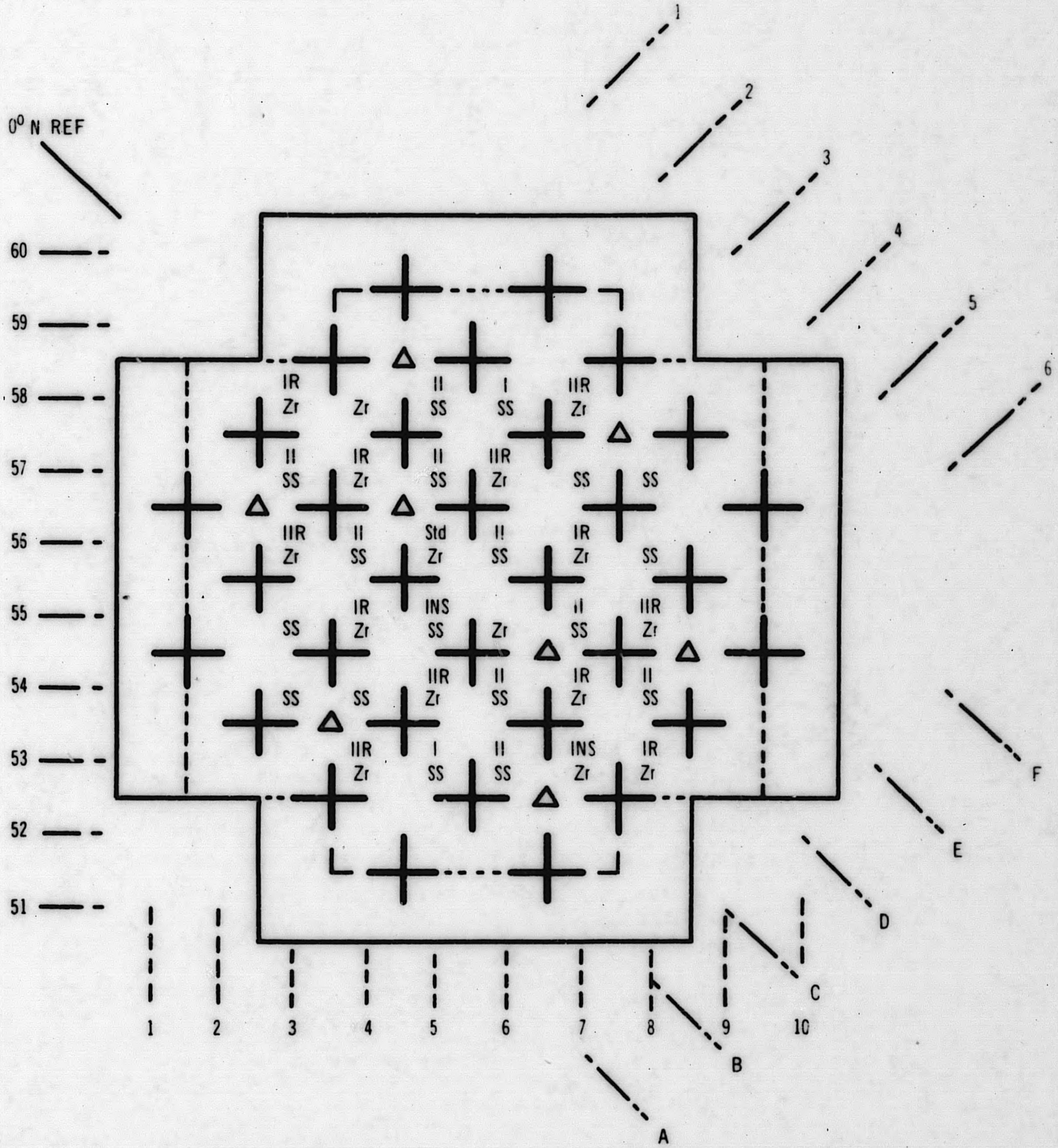
3. SOURCE PLACEMENT AND ROD WITHDRAWAL SEQUENCES

Calculations are now being performed which will provide recommended rod withdrawal procedures and approach-to-full-power rod patterns. These calculations are necessary to provide a pattern with rod worths less than the permitted maximum of 2.5 percent $\Delta k/k$.

Additional calculations are being performed with a one-group, few-node, two-dimensional calculation which is yielding the distribution and magnitude of startup source neutrons in the core during the approach to critical. The attenuation of these neutrons from the core edge to the startup instrumentation may be determined from this calculation, thus providing an indication of control rod withdrawal visibility as a function of rod pattern and source placement. Effort on this project is continuing.

4. NODAL BURNOUT RATIO EVALUATIONS

An addition is being made to the present operating physics three-dimensional power peaking factor program used for the Big Rock Point reactor. This addition will allow the calculation of the local burnout ratio at each three-dimensional node in the core. The resulting burnout ratios will then be used as a guide in selecting operating power distributions. Once the power distribution is obtained, the standard thermal-hydraulics analysis will be applied for purposes of obtaining such items as operating limits.



I	=	PHASE I R&D BUNDLE
IR	=	PHASE I R&D WITH B_4C RODS
II	=	PHASE II R&D BUNDLE
IIR	=	PHASE II R&D BUNDLE WITH B_4C RODS

INS = INSTRUMENTED INITIAL FUEL BUNDLE

Zr, SS DENOTES ZIRCONIUM OR STEEL CHANNEL

VACANT SPACES ARE INITIAL FUEL WITH Zr CHANNELS.

Figure 16. 84 Bundle 75 MWe Core

5. PHASE II FUEL DESIGN

Physics analysis for the Phase II fuels has been completed and is summarized in Table III, which compares all of the fuel types presently specified for Big Rock Point.

6. RELOAD FUEL DESIGN

Reload fuel is being specified for the Consumers Power Company at the present time. The fuel is expected to include the following features:

a. Development fuel design features, including the removable fuel rod feature;

b. Zirconium clad; and

c. An average enrichment of about 3.1 percent with 37 large rods enriched to approximately 4.2 percent and the remaining large and small rods enriched to 2.6 percent.

By placing the 37 higher enriched rods in the center of the 11 by 11 fuel rod bundle, as shown in Figure 17, a significant decrease in local power peaking results, as shown in Table IV which contains the results of some preliminary calculations.

TABLE IV

LOCAL POWER PEAKING OF BIG ROCK POINT FUEL ASSEMBLIES

	Standard Fuel 3.2 percent enrichment	Phase II 2.8 percent enrichment	Phase II 3.3 percent enrichment	Reload
Steel channel	1.27	1.24	1.28	1.25
Zirconium channel	1.32	1.29	1.33	1.21

Exact specification of the reload enrichments is continuing.

7. SCHEDULING COMPUTER

The new set of library stored power shapes for the GE-312 scheduling computer is being developed, including the revised core layout and the different rod withdrawal pattern.

The axial power shape per bundle fits are being recalculated using a one-dimensional technique. The result of these calculations will be fits of axial power shape as a function of bundle power, bundle fuel type, and adjacent control rod position for all the significant fuel types in the reactor.

Peaking factors for each bundle are being included for the additional types of fuel in the core.

TABLE III

BIG ROCK POINT FUEL ASSEMBLIES

	k_∞ at 20° C, 0 MWD/t				k_∞ 1250 psi, 20 percent channel void, 0 MWD/t			
	Steel channel		Zircaloy channel		Steel channel		Zircaloy channel	
		with Phase II B_4C rods		with Phase II B_4C rods		with Phase II B_4C rods		with Phase II B_4C rods
Initial Fuel Bundle	1.105	*	1.219	*	1.100	*	1.208	*
Instrumented Fuel Bundle	1.092	*	1.206	*	1.087	*	1.195	*
Phase I R and D Bundle	1.132	1.072	1.243	1.177	1.135	1.044	1.239	1.140
Phase II, 2.8 Percent Enriched R and D Bundle	1.149	*	1.268	*	1.150	*	1.268	*
Phase II, 4.1 Percent Enriched R and D Bundle	1.181	1.118	1.303	1.234	1.173	1.079	1.294	1.190
Phase II, 4.5 Percent Enriched R and D Bundle	1.181	1.118	1.303	1.234	1.174	1.080	1.295	1.191
Phase II, 3.3 Percent Enriched R and D Bundle	1.167	1.105	1.288	1.220	1.160	1.067	1.279	1.177

* B_4C rods are not planned for these bundles.

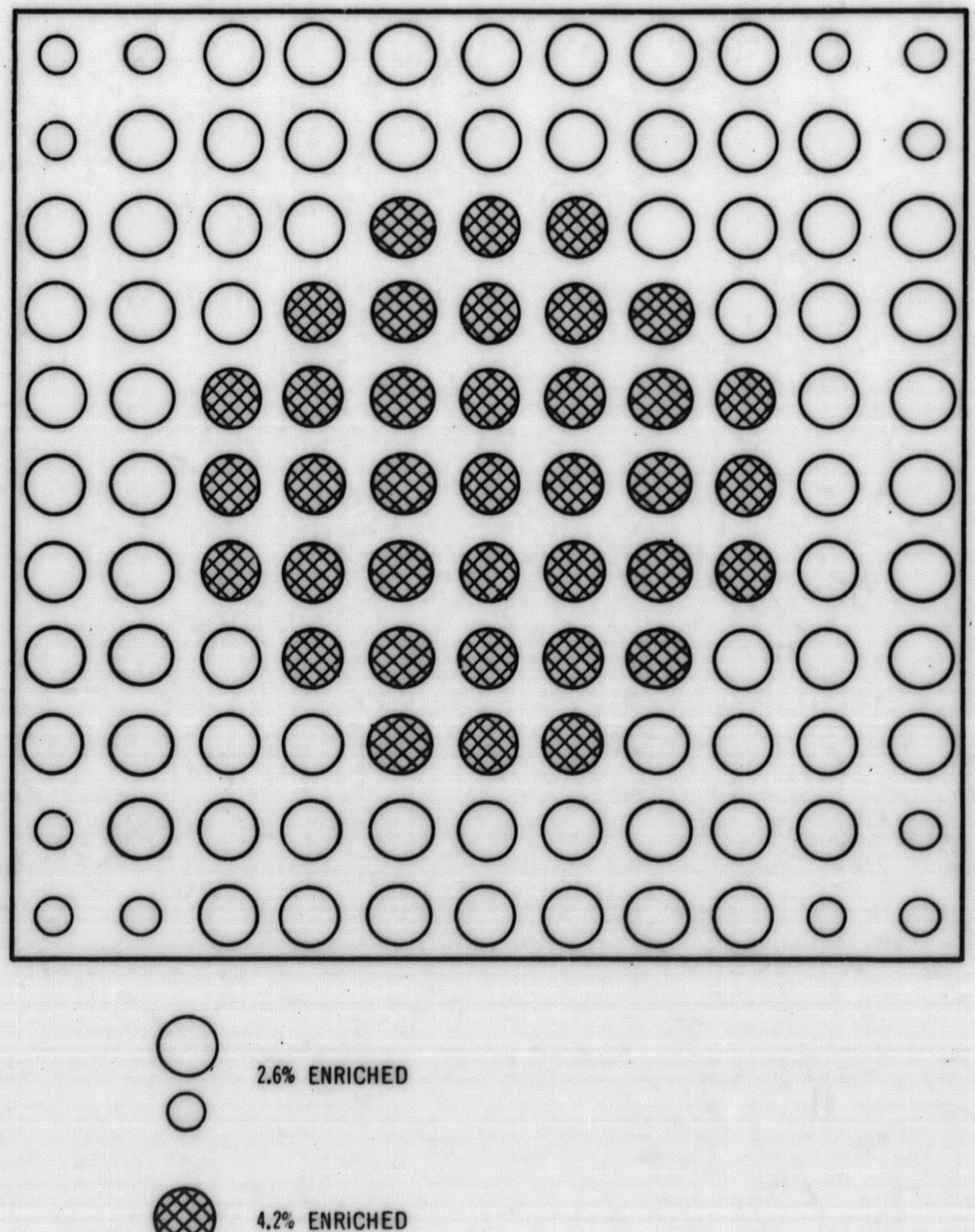


Figure 17. Reload Fuel Assembly

A number of modifications have been, or are being made in the computer system. These are due primarily to the experience gained in the last 6 months of operation.

The Core Thermal Power module has been modified to allow a special reactor heat balance to be performed on demand at any time during the hour cycle. This heat balance makes use of a separate sensor scan, and in no way disturbs the normal calculations. In addition, an on-demand log to be used in conjunction with this heat balance has been prepared. The program and log are currently checked out and operating at the site.

As discussed in previous reports, a method has been devised to allow calculation of the core minimum burnout ratios under conditions of overpower. This is of more interest in the operation of the reactor than the rated power burnout ratio since the license limit is based on the overpower condition. A special calculation is required by the computer, however, since all computed variables are based on actual measured values. Modifications to the channel flow rate, and heat flux and burnout ratio modules required by this method have been programmed. These modules have been reassembled but have not as yet been checked out.

An auxiliary program to allow fuel exposure data to be moved about in the computer memory as fuel is moved about the core has been written and assembled. The operator must prepare a control tape defining each fuel bundle movement, which is then read into the computer. Each piece of data associated with a particular fuel bundle is then moved to a new location in the memory corresponding to the new core location. Provision is also made to move fuel to and from the storage pool, and to insert fresh fuel. This program will be checked out prior to the May startup, and will, in fact, be used to set up the computer for the 84-bundle core.

A revised peak power calculation is presently being prepared which is more consistent with the methods used for operational physics. This method calculates the local peak power as a function of the fuel type, the number of control rods adjacent to the axial position in question, and the direction of the control rod with respect to the fuel bundle. This method will allow the future specification of a program which will normalize the in-core monitors to the local heat flux.

In addition to the changes listed above, improved versions of the channel power, exposure, and plant performance modules have been assembled. All the programs mentioned will be checked out prior to the May startup.

TASK IV - COORDINATION AND TEST PLANNING

1. COORDINATION

In preparation for Phase II of the R and D program, which is related to the increase of Big Rock Point Plant output to 240 MWt, several meetings with AEC and Consumers have been required. A formal presentation was made to ACRS in early January to describe and justify the Phase II development effort. No significant safeguards questions were raised or left unresolved with the Committee as a result of that meeting. Subsequent meetings with the Division of Licensing and Regulation were also held to resolve remaining details of the Technical Specification and other license considerations as related to the R and D Program.

Discussions were held with AEC-SAN with regard to the planned application of instrumented assemblies to the Phase II testing at Big Rock Point. Emphasis is being placed on increasing the probability of obtaining detailed data sufficient to evaluate the individual channel performance. To do this, one instrumented probe will be utilized in each of the test sequences, 75 MWe operation at 45 kW/l and later at 60 kW/l and reduced power output. Two bearing types are being developed and replacement flowmeter assemblies will be available for the second sequence of testing should failure occur during the first.

Also presented to AEC-SAN was the proposed hot laboratory (RML) program intended to evaluate thoroughly the performance of the HPD fuel irradiated in the VBWR. This examination sequence will cover all fabrication process types and designs and will consider the impact of neutron exposure and thermal history on the demonstrated performance of the fuel.

Development fuel has been operating in the Big Rock Point Plant as part of a 74-bundle core configuration at full power throughout the report period. The lead development fuel bundle has accumulated more than 2600 MWD/T exposure through this period.

The Big Rock Point reactor was shut down at the end of the period to carry out the annual turbine inspection, conduct containment tests, make repairs to turbine-generator equipment, and to reconstitute the core for 75 MWe operation. Part of the 15 new development fuel bundles fabricated for Phase II operation will be installed in the core during this outage, as will two instrumented assemblies of which one will include inlet and outlet flowmeters. Development fuel bundles will also incorporate B_4C poison rods to replace four fuel rods in each selected assembly.

Procurement of fuel by Consumers Power Company for reload use at Big Rock Point was initiated and the needs of the R and D program have been factored into this planning. In addition to the fifteen development assemblies now being delivered, it is expected that approximately five more development assemblies will be built for delivery near the end of this year. The swaged pellet fabrication process is currently anticipated to be the choice for all or part of these assemblies.

Modifications have been prepared for the fuel pool equipment at Big Rock Point to improve the utility and accuracy in performing the gamma scans and in inspecting the fuel. These changes will be implemented early next month.

2. TEST PLANNING

Preparations have been made for the Phase II testing which is planned to begin early next quarter. The test program will involve the following major activities:

a. Irradiation of development fuel bundles with cladding of Zircaloy, Incoloy, and Inconel designed for 15,000 MWD/T burnup and for operation in a core up to an average of 60 kW/l.

b. Increase reactor thermal power output to 240 MWt while conducting a sequence of tests at selected operating points with an 84-bundle core configuration.

   (1) Steady state evaluation of thermal hydraulic performance utilizing data from instrumented fuel bundles.

   (2) Transient performance evaluation by pressure set point perturbation, neutron flux noise, and recirculation pump trip tests.

   (3) Stability and transient behavior studies by control rod oscillator tests.

c. Evaluate the power flattening accomplished by the use of local poison rods in selected fuel bundles. Wire irradiation and gamma scanning of the fuel will assist in this study.

d. Increase the reactor average power density to 60 kW/l of core while conducting a sequence of tests at selected operating points with a reduced size core. These tests will be of the type enumerated above in item b.

Coordination of these tests with Big Rock Point supervision is being maintained to assure their compatibility with the plant operating and maintenance schedule.

In support of the planned Phase II activities, four Data Packages, numbered six through nine, have been prepared and submitted to the Big Rock Point R and D Review Committee as required. These Data Packages are as follows:

Data Package No. 6 - Describing the Development Fuel Assemblies for Phase II Research and Development Program,

Data Package No. 7 - Describing the Physics Portions of the Phase II Research and Development Tests, Data Package No. 8 - Describing the Instrumented Assemblies Phase II Research and Development Tests, and

Data Package No. 9 - Describing Phase II of the Steady State Core Performance and Stability Tests on the Consumers Big Rock Point Reactor.

Previous Data Packages (No. 1 through 5) were those associated with Phase I R and D testing.

Appropriate fuel loading patterns incorporating development fuel assemblies have been defined both for high power testing (up to approximately 240 MWt) and for high power density testing (up to approximately 60 kW/l). The test planning has included the definitions of the associated core power distributions and the evaluation of core thermal hydraulic conditions at various planned operating modes to assure feasibility and safety of the tests.

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The Research and Development Program, associated with the Consumers Big Rock Point Plant and sponsored by the AEC, is being conducted by the General Electric Company, with the following personnel contributing to the project during the sixteenth quarter.

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	J. A. Hodde
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	R. L. Holladay