GEAP-3652
INFORMAL AEC RESEARCH
AND DEVELOPMENT REPORT
UC-25, METALLURGY &
CERAMICS
(TID-4500, 15th Ed)

MARITIME LOOP IRRADIATION PROGRAM

SAVANNAH I FUEL IRRADIATION

PROGRESS REPORT

FIRST AND SECOND QUARTERS

JULY, 1960 - JANUARY, 1961

BY

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PREPARED UNDER

CONTRACT AT(04-3) - 189

PROJECT AGREEMENT NO. 20

FEBRUARY 13, 1961

VALLECITOS IRRADIATIONS SERVICES OPERATIONS
ATOMIC POWER EQUIPMENT DEPARTMENT
GENERAL ELECTRIC COMPANY
PLEASANTON, CALIFORNIA

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I. INTRODUCTION

The General Electric Company is proceeding with an irradiation program to proof test a representative array of Savannah I fuel rods. Irradiation of a test assembly containing Savannah I fuel rods has begun and it is proposed that the results of this irradiation will permit an advance evaluation of the fuel performance and fuel burnup in the Savannah I reactor.

The test assembly is composed of 9 fuel rods in a 3 x 3 array. The rods are held by end fixtures and spacer ferrules are provided for lateral rigidity. The fuel is 4.6% enriched U-235 - UO_2 pellets. The fuel cladding is 0.5" O.D. 304 stainless steel tubing. Two test assemblies with zircaloy shroud cans were delivered to General Electric by the Nuclear Division of the Babcox and Wilcox Company. One is being irradiated and one is retained as a spare.

The test assembly is being irradiated under reference nuclear and plant operating conditions in a pressurized water loop at the General Electric Test Reactor. A water chemistry program was established to insure a desired loop water chemistry and a crud deposition program set up to study the effect the loop water chemistry has on crud deposition on materials with different surface finishes. Also, a data analysis program has been initiated to reduce test data to a form which will provide a ready reference to cross check loop and fuel performance and to keep the Maritime Administration advised on operating results.

A pre-irradiation examination was performed on the test assembly prior to insertion and minor modifications were made. Several interim irradiation examinations and a complete post-irradiation examination are planned.

This report covers the first two quarters of the reporting period. All aspects of the subject program have been consolidated (program objectives, specifications, etc.) and applicable portions are discussed in some detail.

II. SUMMARY

The Maritime Loop Irradiation Program was initiated in June, 1960. In accordance with accepted proposals, a Boiling Water Loop at the General Electric Test Reactor was modified to operate as a pressurized water loop. Incorporated in the modification was equipment for water chemistry analysis and control which could be operated by the loop operators with a minimum of participation by chemistry personnel, and crud deposition coupons, which were secured by removable holders. Modifications to the loop proper included installation of a new temperature control valve, new ion exchange columns, new sample heat exchangers, an aluminum "flux window", which was placed between the facility tube and the reactor pressure vessel to displace the water, and modification to the test assembly hold down rod.

Two test assemblies and two zircaloy shrouds were shipped to General Electric's Vallecitos Laboratory by the Nuclear Division of the Babcox and Wilcox Company, which manufactured both the fuel rods and the test assemblies. Preliminary examinations w re performed at Vallecitos and minor modifications were made. One of the test assemblies was used to establish flow versus pressure drop data. As a result of these and other tests, the test assembly shroud was modified to provide additional bypass flow area. The shroud adapts the square test assembly to the round facility tube.

After an extensive operational loop shakedown and instrument calibration, the loop was termed ready for operation. The test assembly (NMSR-GETR #2) was first inserted with flux wires and the reactor brought up to low power to obtain a flux distribution for power output predictions. Final insertion of the test element was then made and the loop was brought up to operating temperature and pressure. Full nuclear loop operation was achieved on automatic control without incident or a reactor scram.

At the end of this reporting period, the test assembly had been irradiated for 10 days and received approximately 265 MWD exposures. The test element power level was approximately 78% of that predicted with the facility tube in its innermost position. It has been subsequently determined that the facility tube had not been positioned to the desired location. During shutdown, at the end of this period, the facility tube was repositioned. Several cross checks have subsequently been initiated to check the power level.

II. SUMMARY (Cont'd)

Water chemistry was generally held within specification with the exception of hydrogen, which was only within specification intermitantly. The loop was operated continuously on automatic control and all loop parameters were held to the desired values.

III. PROGRAM

A. Program Objectives

1. Irradiation Program

The announced objective of the Maritime Loop Irradiation is to proof test a representative array of Savannah fuel rods under reference nuclear and plant operating conditions in a pressurized water loop. An important aspect of the test is to lead the operation and burnup of the fuel in the Savannah reactor sufficiently to permit an advance evaluation of the fuel performance from test results. It is understood that in the event that the cladding on any one of the fuel rods becomes defective, the test will normally continue. However, if the resultant activity levels endanger the normal operation of the GETR .in the opinion of General Electric Co.) the test will be terminated.

2. Crud Deposition and Coolant Chemistry

a. Water Chemistry

The objective of the water chemistry program is to maintain a loop water chemistry paralleling that proposed for the Savannah reactor.

The water chemistry program consists of controlling hydrogen gas within desired limits, chlorides and dissolved oxygen below maximum allowable concentrations, water purity (resistivity) as high as possible, pH, and an estimation of total solids.

In the event of a defect, the continuation of the analysis and control of the loop water chemistry will be dependent upon the radiation levels resulting from the defect. All water chemistry analysis and control will be discontinued if a large defect occurs. An iodine analysis will periodically be performed on water samples to check for defects.

The majority of water chemistry analysis and control will be accomplished at the loop by loop operators. Spot checks by Laboratory sampling methods are made at regular intervals by chemistry personnel. Besides providing more reliable data and control, the above method provides information relating to the abilities of operators and technicians to make measure-

a. Water Chemistry (Cont'd)

ments and exercise control of water chemistry under ship board type conditions. Present techniques and procedures are being evaluated with relation to accuracy and expedience and recommendations for the best combinations of methods and personnel will be prepared.

b. Crud Deposition

The objective of the crud deposition program is to study (utilizing the loop coolant) the effects that different surface finishes have on the amount of crud deposited on these surfaces over measured time periods. A minimum of 15 coupons of each surface roughness will be used and the exposure periods will vary from 3 to 18 months.

c. Data Analysis

The objective of the data analysis program is to maintain hourly records of loop parameters significant to the water chemistry program and to heat generation in the fuel zone. This information will be compiled in periodic reports.

3. Irradiation Effects

The objective of the irradiation effects program is to compile a complete history of the test assembly performance under conditions similar to the Savannah reactor. This will be accomplished through pre-irradiation examinations, interim examinations during the irradiation period, and post-irradiation examinations. The post-irradiation examinations will include metallographical studies, burnup studies, fission gas release data, determination of flux patterns and temperature profiles, and a detailed visual examination accompanied by a photographic record of all phases of the examination.

B. Proposed Programs

1. Irradiations Program

The irradiations program consists of the irradiation of a prototype Savannah I fuel assembly to a burnup of 15,000 MWD/tonne U in the GETR Boiling Water Loop, which is to be operated as a pressurized water loop. The loop parameters, mechanical considerations, and irradiation conditions were agreed upon and/or calculated to be as follows:

a. Maximum heat flux - 300,000 Btu/hr-ft^2.

b. Average heat flux - 130,000 Btu/hr-ft^2.

c. Operating pressure - 1300 psig.

d. Operating temperature

(1) Inlet to test section - 510°F.

(2) Outlet of test section - 533° (approx.)

e. Average thermal neutron flux for 36" length - 1.86 x 10^{13} nv.

f. Peak thermal neutron flux - 4.30 x 10^{13} nv (approx.)

g. Loop flow at the beginning of test - 45 gpm with 42 gpm through the test assembly.

h. Burnout ratio at rated loop flow - (45 gpm) 4.3

i. Maximum cladding temperature - 625°.

j. The test assembly will be subjected to a 40 lb. hold down force throughout the test.

k. The fuel bundle will be indexed to the shroud. The hold down rod will be indexed to the fuel bundle grapple, and in turn to the facility tube flange to permit the orientation of the fuel bundle to the reactor.

l. Neither the test assembly nor the shroud can will contain any instrumentation.

The actual thermal neutron flux and corresponding heat flux for any cycle will vary depending on the position of the reactor control rods and the reactor fuel loading for the cycle.

In the event of a defect, General Electric will continue the experiment, but has reserved the right to terminate the experiment if activity levels caused by the defect approach limits which preclude normal operation of the GETR.

2. Corrosion and Coolant Chemistry (Danielson)

a. Water Chemistry

The water chemistry program is designed to control the loop coolant within the specifications established for the Savannah reactor. The specifications have been established to control the pH range, hydrogen, oxygen, and chloride concentration, and to keep the resistivity of the water as high as possible.

The total solids will be estimated periodically, based on estimates of ionic dissolved solids from conductivity measurements and the suspended solids from suspended iron analyses. Such a process will require occasional determination of the suspended iron to the total suspended crud ratio. The final accuracy of the total solids estimate would be limited. The total solids determination by ASTM specification D1069-54T was not recommended for application to the loop because of the high expense and long time required for an analysis, the large sample size required for the accuracy desired, and the limited usefulness of the data.

pH will be measured by the colorimetric technique. The specification of a pH range of 6.5 to 8.5 presents no operational problem, but only because the range is sufficiently broad to cover measurement errors. The measurement of pH in very high purity water presents problems which have not been solved. Control of pH is inherent to the loop design, and the corrective action in the event of loss of specification is to change the ion exchanger demineralizer resin.

The oxygen specification of less than 0.01 ppm can be attained based on pressurizer blowdown and the radiolytic recombination of H_2 and O_2 that occurs in an excess hydrogen pressurized water-reactor system. A thallium column technique similar to that developed by Wright at the Westinghouse Bettis Plant will be used for dissolved oxygen determination. In the event that the dissolved oxygen concentration is above specification at startups, provision has been made to add hydrazine if necessary to bring the oxygen rapidly within specification.

a. Water Chemistry (Cont'd)

The hydrogen specification of 1.8-3.6 ppm will require frequent or continuous addition of hydrogen since the apparent gas leak rate is high. Equipment for continuous hydrogen gas addition has been installed but was not operated during the reporting period. A special gas sampler has been designed and installed in the sample station for use in hydrogen concentration determinations.

The chloride specification of 0.1 ppm normal maximum with allowance for 1 day excursions to 1.0 ppm can be obtained with normal operation procedures. The chloride determinations will be made by turbidity measurement utilizing samples treated with $AgNO_3$, with appropriate data quality control using such procedures as those provided by Maritime personnel (B&W 1047 T59).

Every attempt will be made to keep the water purity as high as possible. In-line electrical conductivity instrumentation along with auxiliary sample station conductivity flow cells will be used for conductivity determination. Table II summarizes the chemistry program specifications.

b. Crud Deposition

A total of 30 coupons (approximately 3" x 3/4" x 3/32") will be exposed in the loop environment for the purpose of comparing the rate of crud buildup on two different surface finishes (125 RMS and 250 RMS microinches surface roughness). Three coupons of each surface roughness will be removed after 6 months exposure with new coupons reinserted after the withdrawal (total of 18 coupons in place at all times). Weight loss information before and after descaling will be obtained in addition to a gross activity count. (The total number of coupons was reduced from 40 to 30 to minimize the pressure drop in the main flow through the coupon station.)

c. Data Analysis (Danielson)

Hourly records will be kept by the loop operators covering the loop parameters significant to the water chemistry and to the heat generation in the fuel zone. A copy of each completed data sheet will be kept on file for customer use if desired, but the information on the data sheet will be consolidated and summarized for purposes of the periodic reports to the customer.

The summary reports will be presented in terms of graphs of data such as loop conductivity, pH, dissolved oxygen concentration, dissolved hydrogen concentration, differential temperature across the test section, heat flux, and reactor power level. These reports will be prepared following every GETR cycle. Additional data will be plotted or tabulated on chlorides, total solids, flow, pressure, etc. Additional graphs on loop operational characteristics will be prepared for proper operation of the loop and will be kept on file for customer examination if requested.

Comments on estimated accuracy of the data presented will be included with the data transmitted to the customer. Where significant, any unusual excursions in the data will be discussed.

The heat generation in the test section will be calculated on the basis of main flow and differential temperature measurements across the test section (with estimated corrections for heat losses in the facility tube and for gamma heat generation in the fuel assembly, shroud, and pressure tube structure). The total heat generation as calculated from the experimentally observed thermodynamic conditions will be compared with that calculated from physics calculations.

3. Irradiation Effects (Mathay)

The program provides that the two test assemblies be given the following examinations: (1) two pre-irradiation examinations, (2) six interim pool examinations, and (3) one post-irradiation examination. One assembly was to be selected for insertion in the Boiling Water Loop, and the other assembly retained as a standby unit (spare).

The specific breakdown on each examination is described here:

a. Pre-irradiation examinations

(1) Each of the two assemblies will be given a careful visual inspection. All four sides, top and bottom ends, and areas of specific interest will be photographed.

(2) The lengths will be taken of each outer rod to the nearest .001 inch.

(3) The bow of each outer rod will be determined from series of measured points taken longitudinally on each outer rod.

(4) The twist angle will be calculated for each of the four sides of the two assemblies. Two points in a plane at each end of a side will be measured. The angle between will be the twist angle.

(5) Inter-rod spacing measurements (not included in the original scope but requested by telegram during the actual pre-irradiation measurements).

(6) A pre-irradiation report will be issued at the completion of the pre-irradiation examination. The report will contain information on the fuel assemblies as received and visually observed, photographs, methods of measurements, and tables of results.

b. Interim Examination

Six interim examinations will be made on the test assembly being irradiated. The first interim pool examination will be made in the General Electric Test Reactor pool at the period when approximately 2500 MWD/MTU* burnup have been accumulated. The succeeding exams will be made at 5000, 7500, 10,000, 12,500 and 15,000 MWD/MTU intervals.

(1) The assembly will be first visually examined through the underwater periscope for any possible defects. Any suspected areas of distortion will be photographed by using a camera coupled to the periscope.

(2) Length dimensions will be taken on each outer rod.

(3) The outside diameter will be measured on each outer rod at 2 inch intervals.

(4) The bow of each outer rod will be measured; however, it may be necessary to substitute inter-rod spacing instead. (Difficulty may be encountered in trying to make successive readings longitudinally along each rod. However, inter-rod spacing will tell whether one rod has bowed with respect to another rod.)

(5) An interim report will be issued on the completion of each examination and the results willbe compared with the pre-dimensional measurements.

c. Post-irradiation Examination

One post-irradiation examination will be made on one test assembly. On termination of the accumulated radiation exposure of 15,000 MWD/MTU, the test assembly will be shipped to the Radioactive Materials Laboratory (RML) for inspection, disassembly, and examination.

(1) The fuel bundle will be visually examined through the Kollmorgen periscope for any surface distortions that may have resulted from the long irradiation time. Areas of interest will be photographed as observed.

(2) The bow of each outer rod will be calculated from measured points to compare with prior examinations.

*Metric Ton Uranium Metal

(3) The twist of each side of the test assembly will be calculated from measured points. The post-twist measurements will be compared with the pre-twist measurements.

(4) The assembly will then be separated into individual rods.

(5) Each individual rod will then be visually inspected and photographed through the Kollmorgen for any defected areas not apparent when the entire assembly was observed.

(6) A gamma traverse will be done on each rod to show the active fuel length, the flux pattern and the point of maximum burnup.

(7) Length measurements will be made on each rod at 0° and 90° positions. It will now be possible to measure the center rod (rod E). The center rod could not be measured when it is part of the fuel assembly. In the pre and interim exams the lengths taken on the middle rods will be the overall lengths whereas the corner rods will be only the distance between guide tube shoulders. The pre and interim lengths on the middle rods can be compared with the post measurements.

(8) The outside diameters of each individual rod will be measured to 0° and 90° positions at two inch intervals and then compared with the prior measurements.

(9) Each rod will be punctured and the total fission gas released measured.

(10) The void volume will be determined for each rod immediately after the fission gas has been removed.

(11) Each rod will be sectioned at two inch intervals and photographed in order to detect the presence of a central void and grain growth. The sectioning will continue until the void and grain growth has terminated.

(12) One transverse sample will be selected from each rod and a detailed metallographic examination with photomicrographs will be performed. The temperature profile across the

(12) Cont'd)

radius of the fuel will be reconstructed from this examination. Transverse microhardness shall be taken on the cladding section of each specimen. A cold sample specimen will be prepared to compare with the irradiated specimen.

(13) Two samples of clad and fuel will be removed from each rod for burnup analysis based on cesium-137. The samples will be dissolved in nitric acid and aliquot samples will be analyzed by radiochemistry techniques.

(14) Uranium and plutonium isotopic analyses will be performed on two selected hot fuel samples and compared with a similar analysis run on cold fuel samples.

(15) Carbon tetrachloride displacement technique type densities will be run on two selected fuel samples. The hot densities will be compared to the densities of representative cold samples.

(16) All radioactive waste and fuel will be temporarily stored until instructions are received from the customer for final disposal.

Note: All pre and post dimensional measurements are made on a marble surface plate; the in-pool measurements are taken on an aluminum surface plate. Accuracies for the measurements are $\pm$ .001 inch.

IV. FACILITIES AND EQUIPMENT

A. GETR (Reed)

1. Description

The General Electric Test Reactor (GETR), an integral part of General Electric, Atomic Power Equipment Department's Vallecitos Atomic Laboratory, has been operating on a regular full power basis since May, 1959.

The unique design of the GETR combines the high specific-power of the pressurized-vessel-type reactor with the large, easily accessible experimental space of a pool type reactor. Inherent qualities of the GETR are extreme flexibility, capacity, and wide flux range. Therefore, GETR irradiation test facilities are adaptable to most irradiation requirements without a major effect on reactor performance.

The GETR operates at a steady state power level of 30 megawatts and is light-water cooled and moderated. The reactor core is housed in a 24-inch diameter cylindrical aluminum pressure vessel, 23-feet in length. The core is a 2 foot by 2 foot matrix with an active length of 3 feet. A typical core consists of 20 flat-plate type fuel elements utilizing fully-enriched uranium fuel. The core loading is limited to 11.5 percent excess reactivity. Surrounding the core and within the pressure vessel is a beryllium-aluminum reflector cooled by circulating primary water. Reactor shielding is provided by both the pool water and approximately eight feet of concrete.

The reactor is controlled by six bottom-entry control rods penetrating the core with a total design worth of approximately 17 percent.

The reactor pressure vessel sits on the bottom of a 9 foot diameter pool. The design of the pressure vessel permits a high neutron leakage through its walls, thereby utilizing the external experimental space in the pool. The pool design is based on refueling through the top of the reactor vessel. Eleven feet of water is provided between the top of the vessel and the surface of the pool. Connected to the pool is a storage and

1. Description (Cont'd)

service canal used for storing depleted fuel elements and serves as a gamma facility providing a flexible arrangement for gamma irradiations. Facilities are provided for safe disposal of all radioactive wastes.

The GETR is particularly adaptable to accommodating loop experiments, both in-core and in-pool. Three 3 inch through positions are provided in the core. In the reflector pool there are essentially no size or area restrictions for loop facility tube arrangements. Capsule tubes may be replaced by single or multiple loop facility tube arrays. Both re-entrant and hairpin type facility tubes can be used effectively in the pool. The boiling water loop has a hairpin type facility tube.

In addition to loop experiments, GETR versatility provides for core capsule positions, pool capsule positions, hydraulic shuttle facility, trail cable facility, bulk irradiation space, beam-port facility, and a gamma irradiation facility.

2. Nuclear Characteristics

Figure 3 shows the flux profile over the active core length for different periods of the cycle. As may be seen the maximum peak to average flux value occurs at the beginning of the cycle and near the bottom of the core. As time progresses the maximum peak to average flux is reduced and moves toward the top of the core, the maximum value being approximately at mid-core at the end of the cycle.

The flux ratios depicted by Figure 3 are for a reactor power of 30 MW. For most experiments, it is desirable to keep the peak flux below the maximum obtainable value throughout the cycle. This may be done through "power programming". Instead of bringing the reactor up to full power at the beginning of the cycle, only the power necessary to bring the initial peak flux to the desired value is used. As time progresses, the value tends to decrease and the reactor power is continually increased to offset this tendency. Although the peak flux is maintained at a more constant level through "power programming", a smaller maximum peak flux is realized.

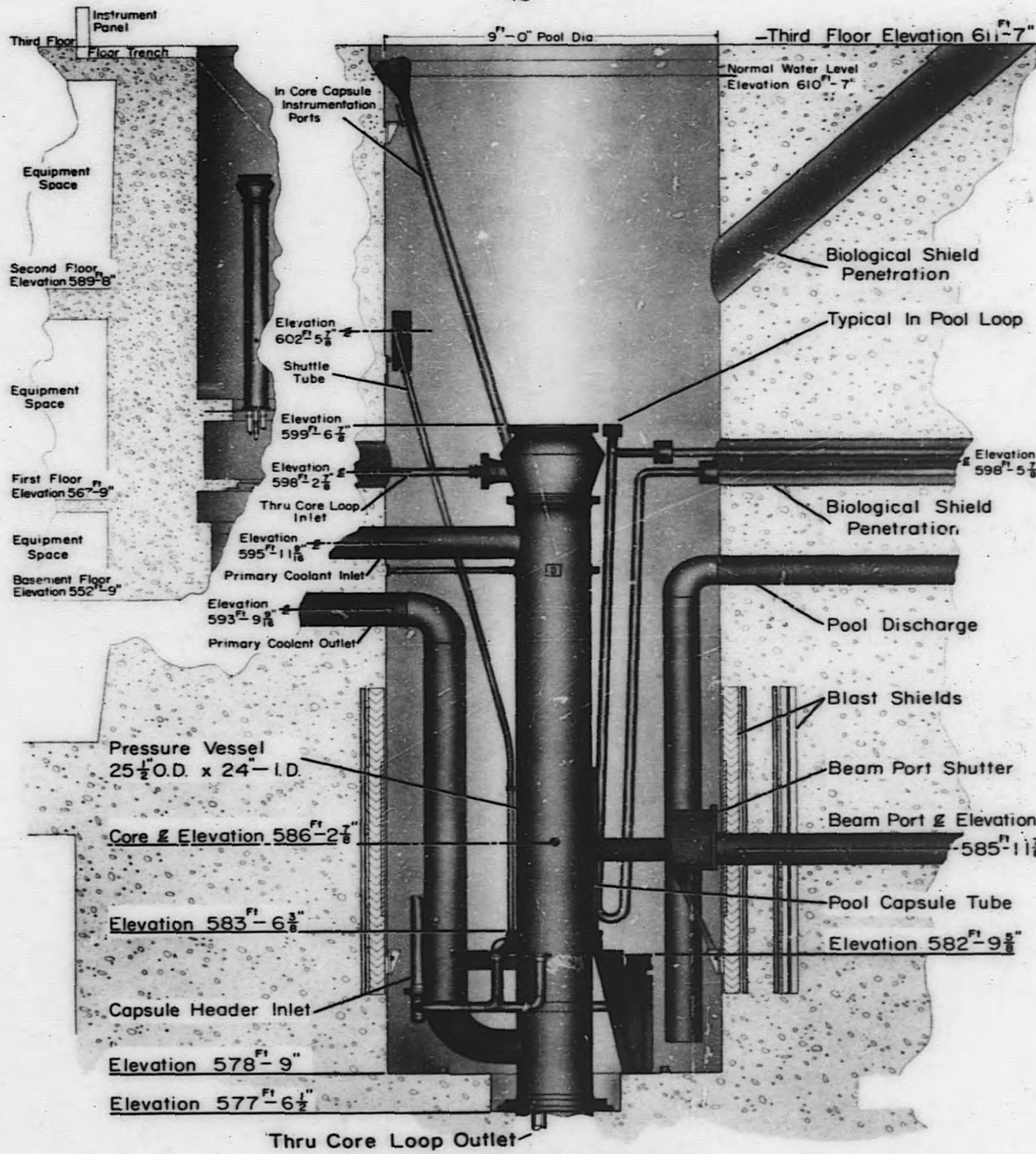

Figure 1. G.E.T.R. ELEVATIONS

VAL 1139

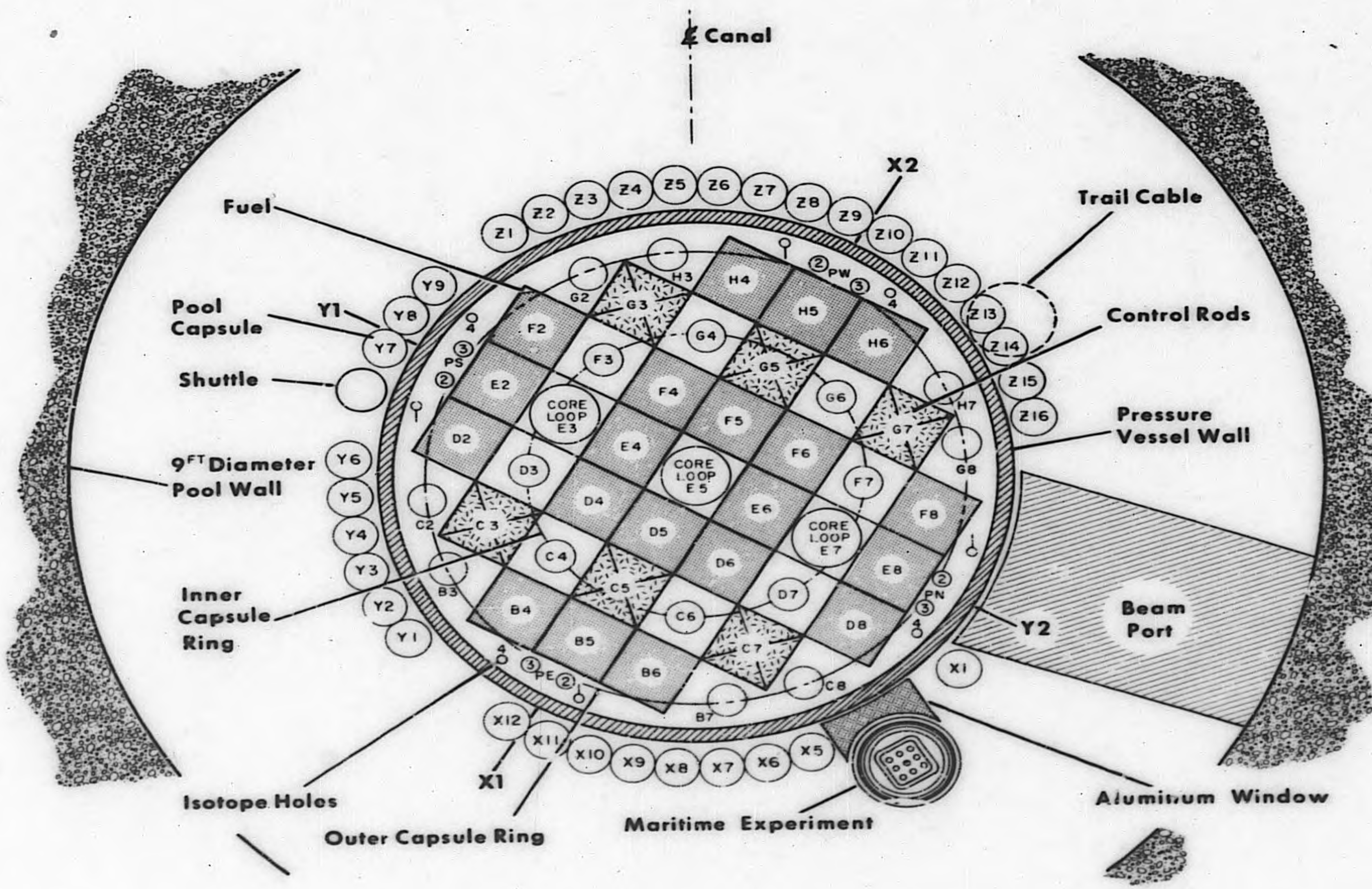

Figure 2. RADIAL LOCATIONS OF G. E. T. R. EXPERIMENTAL FACILITIES

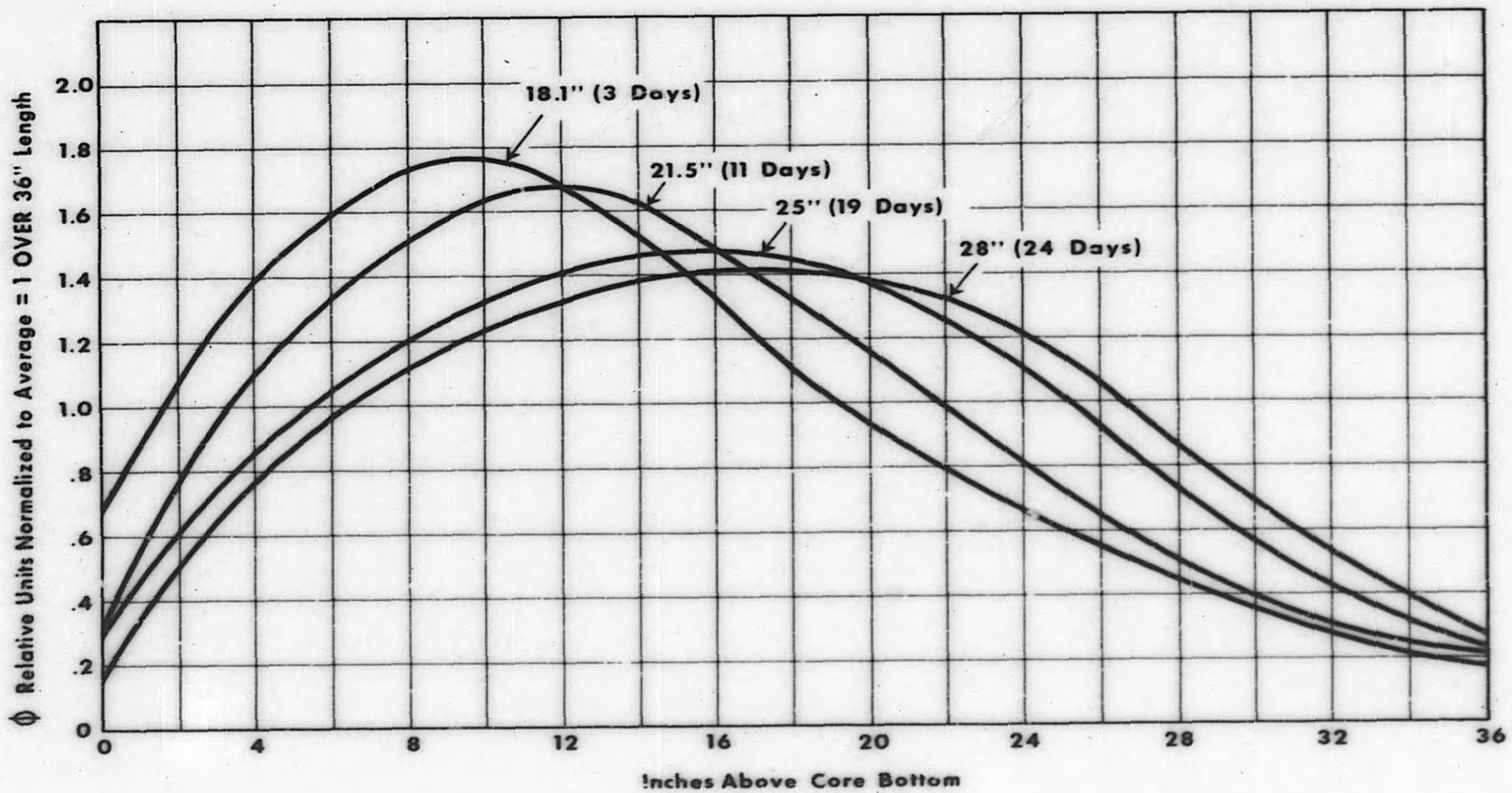

Figure 3. AXIAL FLUX SHAPE AS A FUNCTION OF ROD BANK IN INCHES

VAL 1075

3. Operation

The GETR is operated on a regular basis with scheduled shutdowns for reactor refueling and maintenance, and to install or remove experiments. Operating time is maximized for the benefit of experimental programs. An operating period designated as a reactor cycle is normally five weeks. At the start of each cycle, experiments are loaded and unloaded and routine refueling is accomplished. This down time is minimized and usually amounts to seven days, but, due to complicated operations (such as facility tube installation), the outage may be extended. There are no routine or scheduled shutdowns during the operating period of a reactor cycle. However, due to equipment malfunctions and experimental scrams, unscheduled shutdowns occur.

B. Boiling Water Loop (Howell)

1. Description of Loop

The Boiling Water Loop is a stainless steel facility which is designed for boiling or non-boiling heat transfer conditions in the test section. The loop consists of the facility tube, containing the test section, and the related equipment which is necessary to maintain coolant flow, coolant purity, system pressure, and system temperature.

The loop is designed for an operating pressure of 1300 psi at 600°F. The material in contact with the circulating loop coolant is stainless steel with the exception of the fuel test piece shroud can, which is zircaloy. Loop components are designed and fabricated in accordance applicable portions of the ASME Boiler and Pressure Vessel Code.

The Boiling Water Loop may be considered as three separate, but interconnected systems; these are: (1) the main loop, which provides the pressurization system, main pumps, heaters, controls, and contains the test element, (2) the cleanup loop which contains ion exchange columns designed to assure water purity, and (3) the makeup and transfer system, which provides both high and low pressure deaerated water for makeup or for filling the loop.

A simplified flow diagram of the loop is shown in Figure 21 and a complete Piping and Instrumentation Diagram in Figure 4. Briefly, the flow cycle in the main loop is as follows: Subcooled water from the main coolant pumps, which are connected in parallel, flows through the flow control valve to the heater where the degree of subcooling of the water is automatically regulated. From the heater the water flows through two flow meters in series to the facility tube. In the facility tube test section, the coolant is heated by heat produced in the fuel element. The water exits from the facility tube, enters the coupon station and then the steam separator.

Figure 4

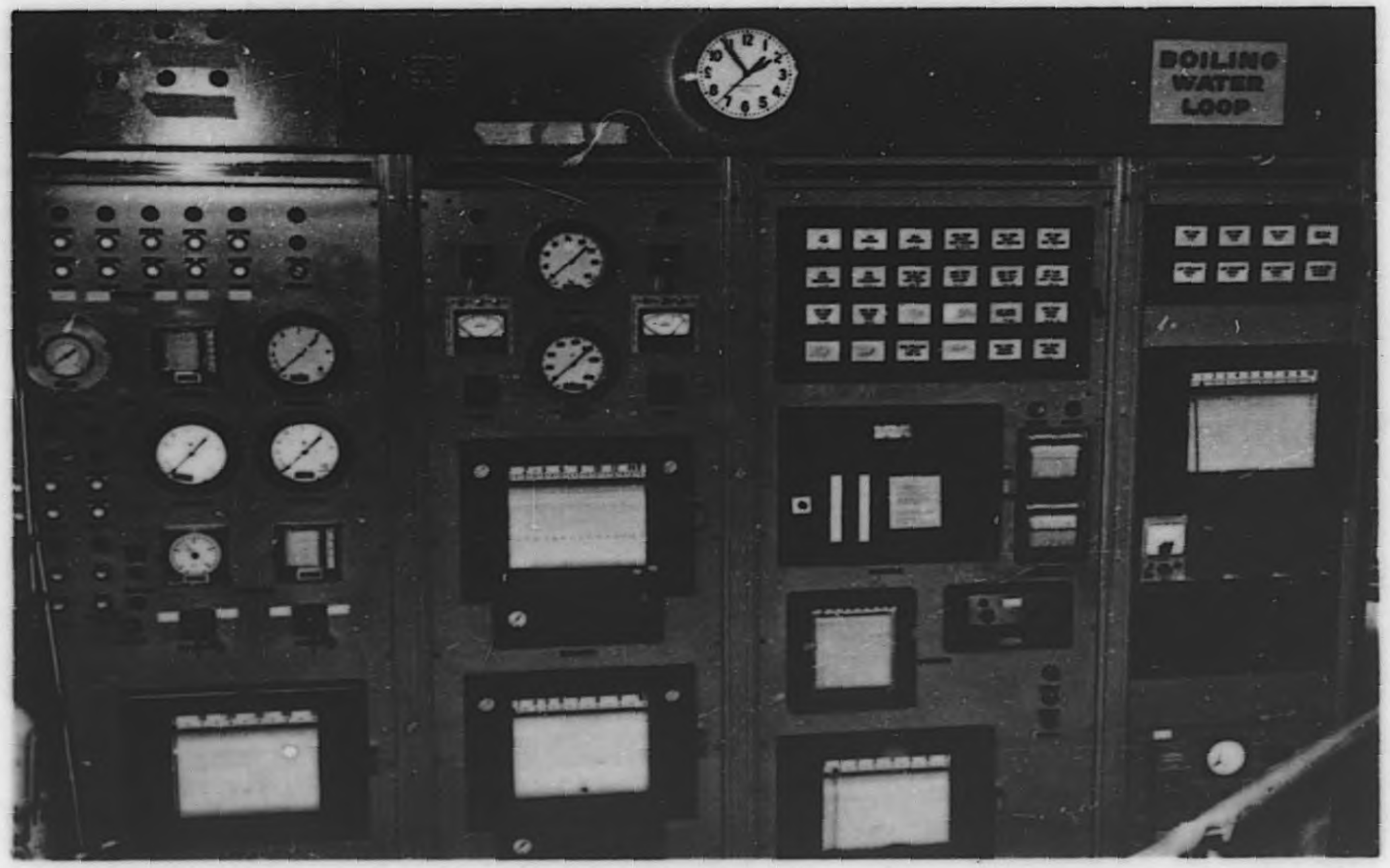

Figure 5. CONTROL CONSOLE

VAL 1118

1. Description of Loop (Cont'd)

The water from the steam separator and the condensed spray flow from the pressurizer combine to flow through the main heat exchanger, which serves the major heat sink in the system. From the main heat exchanger subcooled water returns to the main coolant pumps, thereby completing the main "loop". Bypass flow around the main heat exchanger is automatically controlled by a 3-way flow control valve (TTV-3), and a temperature recorder controller (TRC-251) located on the loop control panel.

System pressure is maintained by the condensation of steam in the pressurizer. The steam is produced by electrical heaters within pressurizer vessel. Condensation in the pressurizer is obtained by diverting part of the subcooled water from the pump discharge through an automatic pressure-regulator control valve and then through a spray subcooler to the spray nozzles in the pressurizer. The test assembly inlet temperature is controlled by the amount of heat rejected in the main heat exchanger and the amount of heat added by the main electrical heater. During normal operation the main heater serves as a final temperature trim just upstream of the facility tube.

A parallel loop connected to the main circulating pumps contains purification equipment. Main loop water from the spray line on the discharge of the main pumps flows through the secondary side of the regenerative heat exchanger and a cooler where the temperature of the water is reduced to 100-120°F prior to flowing through one of two ion exchangers in the purification system. The ion exchangers remove ionic impurities from the water, and in addition, act as filters to remove solid corrosion products. The flow leaves the ion exchangers, is metered, passes through the "cold" side of the regenerative exchanger, and is returned to the suction side of the main pumps. The normal flow rate in the purification loop is on the order of 1 to 2 percent of the main loop flow.

1. Description of Loop (Cont'd)

The make-up system consists of a make-up water storage tank in which demineralized water from the GETR supply system is deaerated by boiling, and stored at 2 psig (maintained by automatically controlled electric heat) to prevent air from contacting the water. Hot water from this tank flows through a heat exchanger where its temperature is dropped to below 120°F before entering the suction of the make-up or transfer pumps which are arranged in parallel. The transfer pump is a low pressure high flow rate pump for filling or flushing operations, and the make-up pump is a high pressure low flow rate pump for make-up operations. The discharge of the pumps is connected to piping which goes to the inlet of the ion exchange columns.

Sampling lines are installed which run to a small sampling station located outside the shielded cubicle housing the loop components. At this sample station it is possible to sample (1) steam after the separator, (2) water after the separator, (3) water before the cleanup loop ion exchanger, (4) water after the ion exchanger, and (5) steam from the pressurizer dome. Both steam and water samples from the steam separator are taken through ASME type sample probes. All other samples are taken through single pipe taps.

The sample station is housed in a closed, ventilated hood into which samples of loop coolant may be drawn without danger of spreading contamination. Any gases which may be released are exhausted to the GETR stack by a blower which holds a slight negative pressure within the hood. Water waste is caught in a stainless steel sink at the bottom of the hood and is carried to the contaminated drain by appropriate piping. All valves and stopcocks are contained within the hood and are fitted with stem extensions which penetrate the sides of the hood and permit manipulation from the outside. A shelf around the sink portion of the hood is provided to support any lead brick shielding which may be required.

2. Description of Facility Tube

The facility tube, shown in Figures 6 and 7, provides a flow channel between the equipment area on the third floor of the reactor building and the fuel element test section. The facility tube is a "hairpin" type with the inlet flow moving down the leg away from the reactor pressure vessel. The test section, containing the fuel test element is in leg nearest the reactor pressure vessel, with the coolant flow moving from bottom to top of the test element. Basically, the facility tube consists of three sections. The first section runs from the equipment area downward through a 45 degree penetration in the biological shielding to a point in the pool above the core centerline where it connects to the inlet leg of the hairpin loop. The second section is the hairpin portion described above. The third section connects the outlet leg of the hairpin loop with the equipment area, going through the same 45 degree penetration as the first section.

The facility tube is anchored to a support ring in the lower portion of the pool and is supported laterally from pool liner pads above the core centerline. The facility tube foot contains a mechanism which will allow slight adjustment of the facility tube position away from and toward the reactor. This mechanism is actuated with a removable long handled tool by an operator standing on the reactor refueling bridge. The movement of the facility tube is in the radial direction. The tube can be moved a total of 1-1/8" which causes a maximum adjustment estimated to be 37% of the average thermal neutron flux.

Each section of the facility tube consists of double-walled tubes. Special flanges and light wire wrapped around the inner tube with a helical pitch maintain the annular space between these two concentric tubes. Nitrogen at about one atmosphere gauge pressure in the annulus serves as insulation between the loop coolant and the pool water. Packing glands seal the annulus between the two tubes to allow for differential expansion.

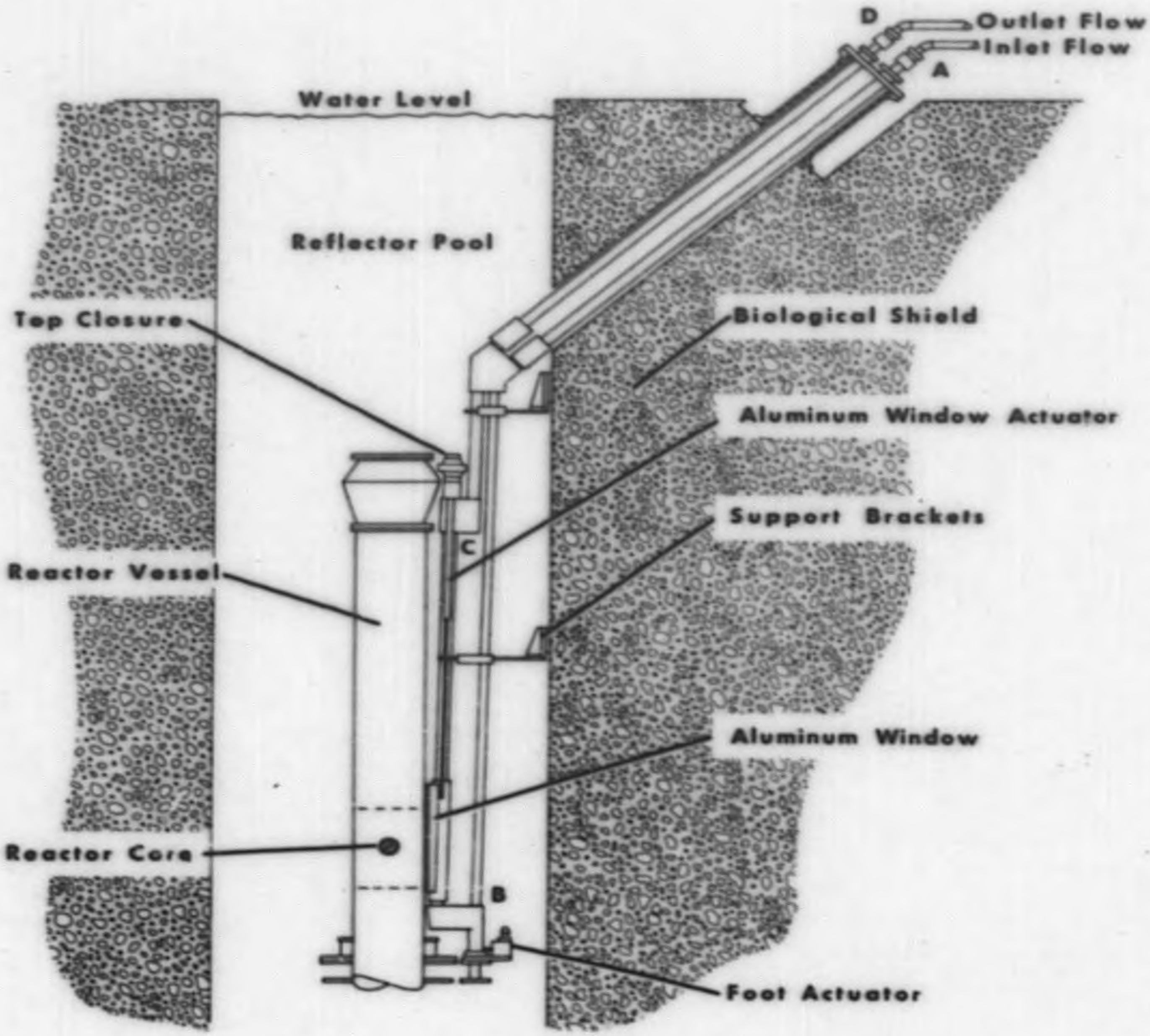

Figure 6. IN-PILE FACILITY ELEVATION

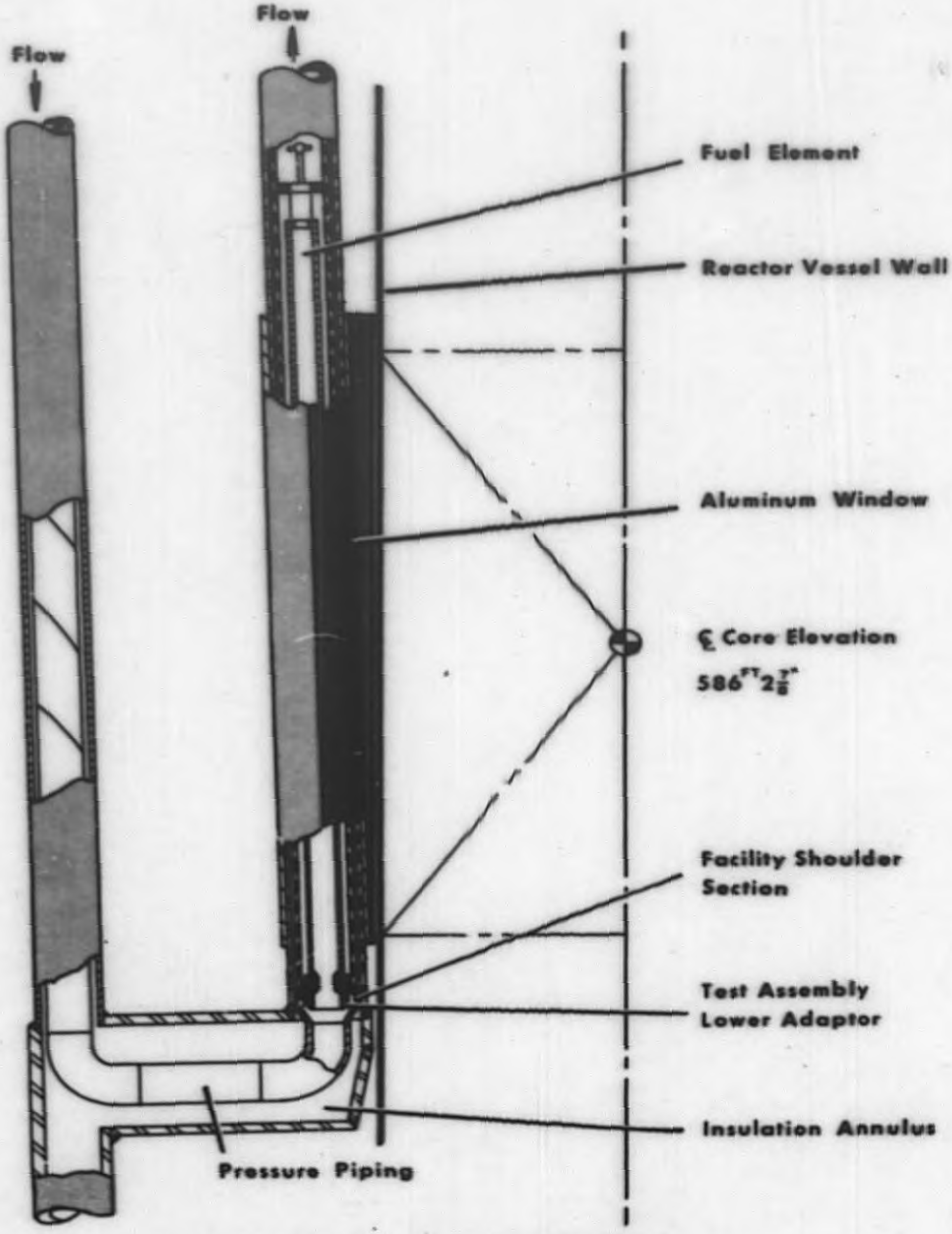

Figure 7. FUEL TEST PIECE ARRANGEMENT

2. Description of Facility Tube (Cont'd)

The facility tube test section is designed to accommodate a square fuel rod array up to 2 inches square and 36 inches long. A round fuel rod array can also be used. A "Marmon Conoseal" type access flange in the facility tube located above the test section permits loading and unloading test fuel elements during reactor shutdown. A 3/8 inch stainless steel hold-down rod is used in the facility tube, running downward from the access flange to the test fuel element, to hold the test fuel element in place. This hold-down rod is spring-loaded to supply a 40 pound axial compressive load to the test fuel element and permits positive orientation of the test assembly.

The test section portion of the hairpin loop adjacent to the reactor core is equipped with an aluminum "window" (see Figure 6). The "window" displaces the water (neutron moderator) which would normally be between the facility test section and the reactor pressure vessel, thereby enabling a higher power level to be reached by the fuel element. The theoretical increase in the power is 36%. The aluminum "window" may be raised or lowered remotely by utilizing a hydraulic drive cylinder.

C. Test Assembly (Mathay)

1. Description

a. Test Assembly

The NMSR-GETR Test Assemblies were fabricated by the Babcock and Wilcox Company. Descriptive information on the assemblies are contained in B&W Drawing #3271-F and their "Specification for Fabrication of the NMSR-GETR Test Fuel Element".

Each assembly is composed of nine fuel rods in a 3 x 3 arrangement. The fuel is 4.6% enriched U_{235}-UO_2 pellets which have theoretical density of 90-95 percent. Each rod contains approximately 851.11 grams of UO_2 and 33.88 grams of U_{235}. The fuel rods are 38.19 inches long by 0.50 inches in diameter. The cladding is 304 type stainless steel tubing with a .035 inch thick wall. The pellet diameters vary from .416" to .426," with resultant diametral pellet to cladding gaps of .002" to .007." The column of fuel in each rod is approximately 36 inches long. A 17-4 pH type stainless steel spring is contained within the approximate .70 inch long plenum zone located at the top end of the end rod. The rod ends are concave type 304 stainless caps welded in a helium atmosphere, by the helium arc process.

The fuel assembly was brazed together using the Kanigen high temperature process. The outer rods are spaced .163 inches apart and have 0.75 inch long 304 stainless steel external ferrules at 7.88 inch intervals. The center rod is positioned with the use of 1.00 and 1.50 inch long by .438 inch diameter stainless ferrules located adjacent to the outer rods at spaced intervals similar to the external ferrules. The stainless steel end fixtures are welded to three tabs extending from each guide tube on the corner rods. The guide tubes cover about 1/2 inch of length at each corner rod end and are brazed.

a. Test Assembly (Cont'd)

in place. A handle is attached to the top end hardware for a handling device and is notched to serve for rod oritentation. Three rods with fuel pellet gaps of approximately .003 inches, .005 inches and .007 inches, respectively, were placed in the row which is nearest the reactor. The remaining six rods were randomly spaced. Overall dimensions of the assembly are 43.50 inches long by 1.876 inches square.

b. Shroud

A separate shroud as shown in B&W Drawing 59919E-0 was fabricated from Zircaloy-2 alloy for each test assembly by the Babcock and Wilcox Company. The shroud adapts the square test assembly to the round facility tube. The Zircaloy-2 shrouds were made from .094 inch thick material and the dimensions are 45.00 inches long by 2.038 inches square (internally). The Zircaloy conforms to B&W Specification No. AEM-106-01480459. All four welded corners were stress relieved. Each shroud had been corrosion tested and inspected prior to shipment to the Radioactive Materials Laboratory. Both ends of the shroud are open to allow for the flow of water around the test assembly. At the top end each of the four sides are bent out to an angle of 30 degrees to facilitate insertion of the test element. There are four 1 inch holes; one on each side at 2 inches in from the top end on center for handling purposes. The assemblies were inserted in their respective shrouds four times by rotating the bundles 90 degrees each time to check for any restrictions. The No.2 shroud had six .128" holes drilled in the sides in a 1-2 pattern at 1-1/2 inches up from the bottom shoulder. These additional holes were necessary to provide sufficient flow of water around the shroud. The shroud will not be removed with the test assembly at the various interim examination inspection periods, but will remain in the loop facility tube, according to present plans.

P12214

Shroud for #2 Assembly Showing Drilled Holes for Flow.

P12187

Shroud

P12188

Assembly #2 and Shroud

Figure 8. NMSR — GETR FUEL ASSEMBLY #2

V. PRE-OPERATIONAL MODIFICATIONS, TESTS AND ANALYSIS

A. Loop Modifications (Howell)

1. Mechanical

Several modifications were made to the Boiling Water Loop in order to meet the Maritime Administration test specifications.

Facility tube modifications consisted of: (1) the replacement of a cadmium shutter with an aluminum window, (2) the addition of a mechanical indexing device which assures proper orientation of the fuel bundle, and (3) the replacement of the fuel bundle hold-down rod with a longer unit.

The major portion of the electrical and instrumentation modifications were range changes for the necessary instruments, and recalibration of all instrumentation. All instruments are to be recalibrated on a routine basis throughout the course of the irradiation.

Other electrical and instrumentation modifications were:

a. The addition of a watt-hour meter to measure total loop power.

b. Modifications to the scram circuit to include reactor scrams in the event that the standby cooling system isolation valves or facility tube bypass valves were opened.

c. The addition of facility tube differential temperature instrumentation.

d. The elimination of several switching circuits which are used for boiling operations.

Several equipment modifications were made to the main loop. A new three-way control valve (TTV-3 on Piping and Instrumentation Diagram) was installed in order to obtain the necessary main loop flow rate. A pipe run between the pressurizer and steam separator was cut and blocked thereby eliminating flow of steam from the pressurizer to the separator. An intermediate cooling water system, new ion columns and new sample line heat exchangers were installed.

2. Corrosion and Coolant Chemistry (Danielson)

Prior to the loop shakedown run some additional equipment was installed at the sample station to be used in obtaining the water chemistry data. Items installed include a thallium column for dissolved oxygen analysis, a conductivity flow cell for improved accuracy of conductivity measurement, and a special permanently installed gas sampler designed for total gas determinations. A hydrogen addition bomb was also permanently installed with appropriate valving to add hydrogen to the loop. Additional equipment for chloride analyses (Model 9 Coleman Nepho-Colorimeter) was procured and located at the sample station location along with a colorimetric pH test kit and a Beckman Model GS pH instrument (for data control).

3. Shakedown (Ulrech)

The Boiling Water Loop shakedown phase of the Maritime Irradiation Program following the loop modification was designed to prove out the satisfactory operation of the loop. The more detailed objectives of the loop shakedown were to: (1) determine the operability of all components in the system, (2) determine loop process characteristics for the purpose of the loop control, (3) the opportunity to set control, alarm, and scram instrumentation and check the operation of such instrumentation, and (4) the opportunity to treat the water flowing within the loop to bring it to desired specification prior to the irradiation phase of this program. In the shakedown portion of this program, which extended about 1-1/2 weeks, all of the objectives were met satisfactorily.

The shakedown operation of the loop indicated that the ion exchangers were working satisfactorily. The hydrogen could be added and total gas sampled as would be necessary in efforts to operate the loop within the hydrogen concentration specification. The thallium column appeared to be operating satisfactorily for dissolved oxygen determination. Chloride procedures were checked in the laboratory standardized with the B and W procedure (B&W 1047T59).

3. Shakedown (Ulrech) (Cont'd)

Prior to the shakedown phase, the loop operators attended classroom sessions conducted by the Design Engineer and the Test Engineer. All phases of loop design and operation were covered in these sessions.

On-the-job training was also conducted during shakedown by the Design Engineer and the Test Engineer. Included in this training were startup and shutdown procedures, emergency procedures, normal operating procedures, and trouble shooting.

B. Test Assembly

1. Pre-irradiation Examinations (Mathay)

The Radioactive Materials Laboratory received two NMSR-GETR Test Fuel Assemblies from the Babcock and Wilcox Company on October 27 and 28, 1960, for pre-irradiation inspections and dimensional measurements. Both assemblies were visually examined with the aid of a stereo microscope. The inspection revealed that some of the guide tube-end fixture weld joints on the No. 1 and No. 2 fuel bundles were cracked. Rod H of the No. 2 assembly had a deep weld spatter on the end cap. Rod C of assembly No. 1 had a deep dent in the cladding approximately 3 mils near the top end. However, both assemblies were helium leak checked but no evidence of any leaks were produced. The weld attachments between the guide tubes and end fixtures were repaired on both test assemblies prior to the pre-irradiation inspections, with permission granted by the AEC. A welding specification supplied by B&W Co. was used. NMSR #2 assembly was selected for the irradiation since the general fuel rod surface condition was better than the NMSR #1 Assembly.

Both fuel assemblies were photographed on all four sides and at both ends from 0.5x to 1.5x magnifications. Typical areas of interest were photographed at 5.5x magnification. All dimensional measurements were made on a marble surface plate utilizing dial gauge calipers accurate to ± .001 inch.

In the length measurements only the middle rods could be measured from end to end. The length of the corner rods had to be measured as the distances between the inside shoulders of the guide tubes. The lengths on the middle rods of NMSR #2 Assembly varied about .016 inch and the corner rods about .025 inch. Inter-rod spacing measurements gave a minimum gap of .157 inch and a maximum gap of .167 inch between rods. The maximum bow measured between ferrules was .009 inch. The greatest angle of twist was on side 2 (Rods C-I) of the NMSR #2 assembly recorded as 3.54 degrees counter clockwise.

At the completion of the pre-irradiation examination work, the two NMSR fuel assemblies were thoroughly cleaned with isopropyl alcohol.

2. Data and Analysis

a. Nuclear Analysis (Worthington)

Thermal flux calculations were performed at the outset of the project to confirm the design requirements for the irradiation of the NMSR-GETR test assembly. The flux calculations formed a basis for a choice of fuel element shroud can material, and determined whether or not the desired flux level could be attained by loop modifications.

The results of these calculations indicated that a peak heat flux of 300,000 Btu/Hr/Ft^2, which was the goal stipulated early in the program, could be attained, but it would be necessary to use Zircaloy in preference to stainless steel for the shroud can, and also necessary to install a flux window whose purpose would be to minimize the amount of moderating water between the test facility test section and the reactor pressure vessel.

The relative effect of both Zircaloy and stainless steel shrouds is summarized by Figure 9. The peak heat flux using a Zircaloy shroud can is approximately twelve percent higher than the flux that could be attained using a stainless steel shroud can.

Addition of an "aluminum window" between the reactor pressure vessel and the facility tube increased the heat flux by a calculated 36 percent.

A brief explanation of the general calculational methods follows:

(1) Calculation of Unit Fuel Cell

The entire region within the inner pipe of the loop was considered to be the fuel region. A single fuel rod with its associated cladding, water and flow shroud were mocked up in cylindrical geometry in the HDP3 Code. The output of this code yields energy hardened cross sections for 3 groups (0 to 0.17 ev., 0.17 ev to 0.18 Mev., and 0.18 Mev to 10 Mev) and a 1 group (thermal) transport theory flux profile.

The results of the unit fuel cell P-3 approximation are shown in Figure 10.

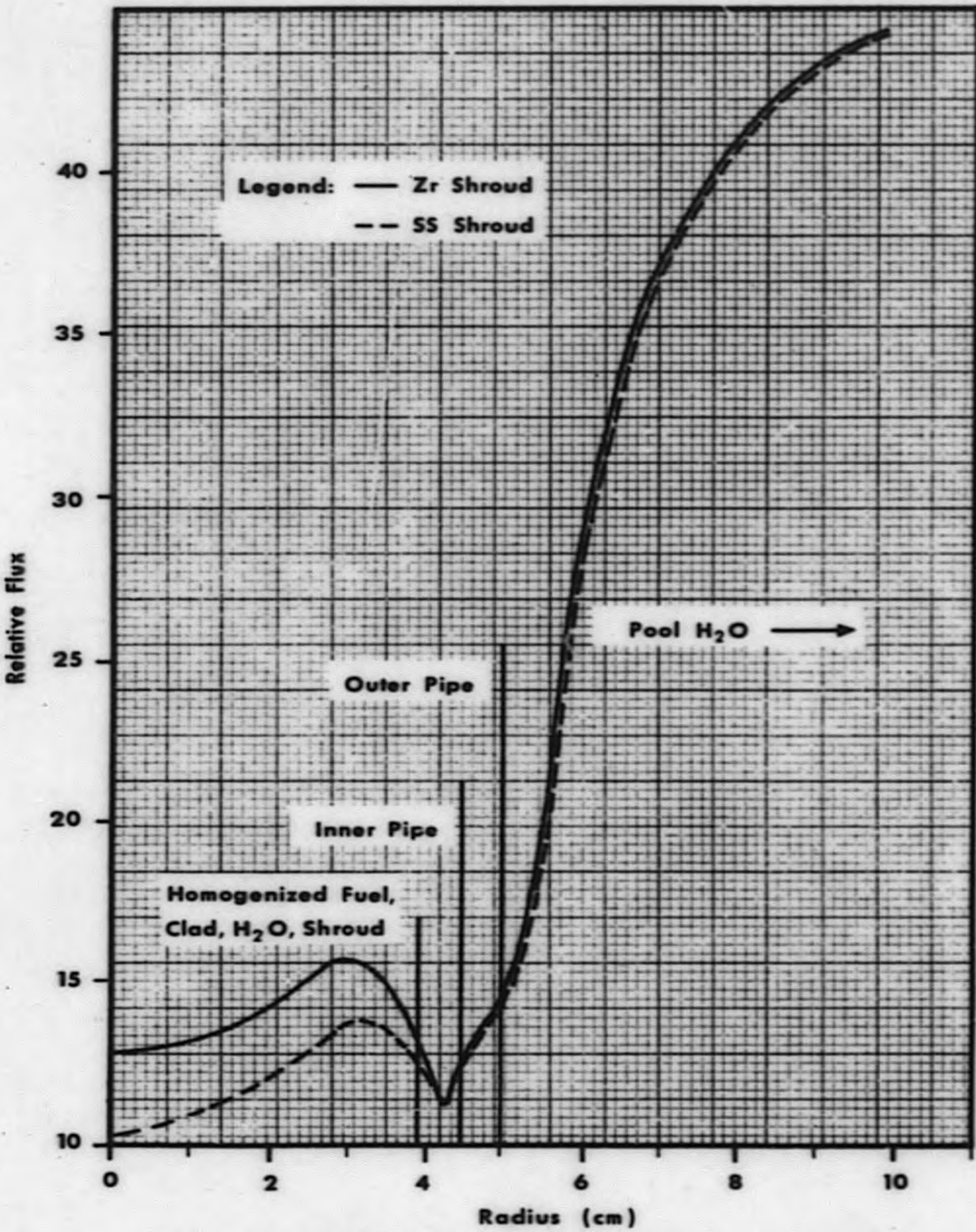

Figure 9. ENTIRE LOOP WITH HOMOGENIZED FUEL REGION

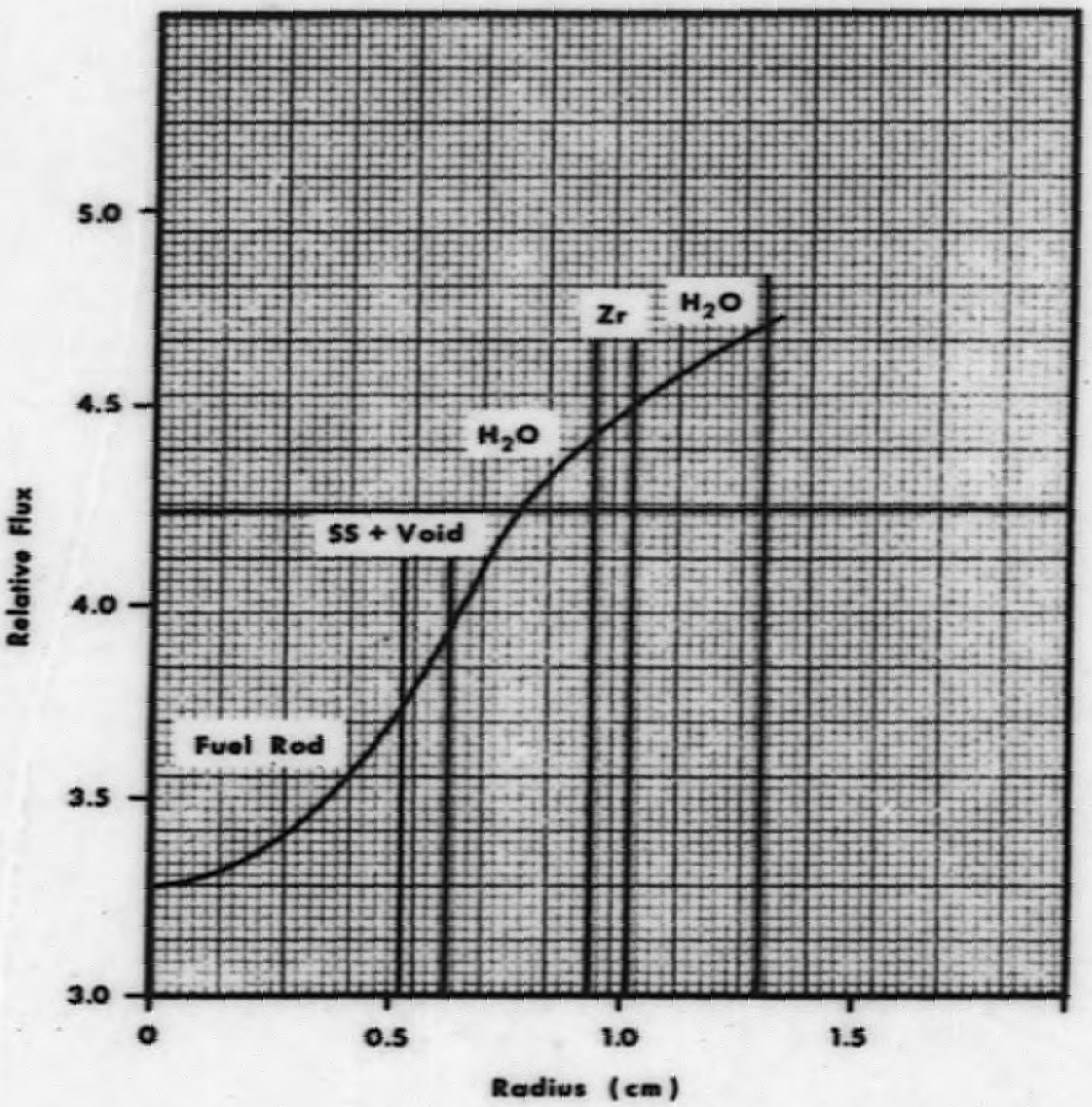

Figure 10. UNIT FUEL CELL P-3 APPROXIMATION

(2) Calculation of Average Flux in Fuel Region

Two 3 group, 2 dimensional core mockups of the GETR including the Maritime Irradiation facility tube were calculated using the PDQ-2 neutron diffusion code. The first calculation was made with the facility tube in its full-in position with respect to core mid-plane. The second was similar except for the addition of an aluminum "flux window" between the facility tube and the GETR pressure vessel.

In order to determine the flux variation from rod to rod it was necessary to investigate the gross flux variation within the homogenized mockup of the fuel region. Comparison of flux values in three sections of the loop mesh starting from the edge closest to the reactor pressure vessel showed the following rod-to-average values:

First Row	1.27
Second Row	0.96
Third Row	0.77

(This distribution taken from the calculation with the aluminum window. Radial peaking without the window indicates a maximum-to-average of 1.20).

(3) Vertical Flux Distribution

Vertical flux measurements have been made at various rod-bank positions using copper-titanium wire at full reactor power in the GETR Trail Cable Facility. This facility is located in a similar position to the Boiling Water Loop. Analysis of the data has indicated that the average flux over the 36 inch active core region does not change, although the peak-to-average value changes as does the position of the peak. These measurements were performed in an assembly perpendicular to the reactor core. The Boiling Water Loop, however, is not perpendicular and in additional correction to the average flux in the loop must be made.

(3) Vertical Flux Distribution (Cont'd)

This correction is on the order of 6 percent from the start of a GETR cycle to its end. Considering the time averaged value of the flux in the loop to be 1.00, the start of cycle average is 1.03, and the end of cycle average is 0.97.

b. Hydraulic Analysis

Two requirements were stipulated as the basis for hydraulic investigations. The first was that sufficient bypass flow must be provided to prevent boiling in the area between the outside of the shroud can and the pressure tube surrounding the test assembly and shroud, and the second that a burn-out ratio of 2 or greater must be maintained.

Bypass flow sufficient to absorb the gamma heating generated in the shroud can and the pressure tube surrounding the test assembly and shroud can without boiling was calculated. The heat generated was calculated by determining the weights of the materials involved and multiplying by the peak gamma heating value. Since the gamma heating values were calculated values, the peak value instead of the average was used to insure a factor of safety. The calculations indicated that a flow greater than that which could be obtained with the bypass area provided in the test assembly seat was needed. It was decided that the shroud can would be modified to provide the necessary area.

In order to obtain an accurate picture of the flow phenomenom in the area of the test assembly, flow tests were conducted. The flow tests were designed to determine the extent of modification of the shroud can necessary to provide additional bypass flow.

The flow experiment provided the following information: (1) pressure drop across the test assembly seat versus flow rate for a progressive number of 0.128 in diameter holes in the base of the shroud can and (2) pressure drop through the test assembly at various flow rates.

b. Hydraulic Analysis (Cont'd)

The results of the flow tests are presented on Figures 11 and 12. The desired bypass flow was approximately 3 gpm. With reference to Figure 11, the pressure drop across the test assembly at 45 gpm would be approximately 1.07 psi. Entering Figure 12 with this pressure drop and interpolating between the curves, it may be seen that a bypass flow of approximately 3.2 gpm would be realized if the shroud can was modified by the addition of six (6) 0.128 in diameter holes. Six holes of this size were drilled in the area between the junction of the shroud can proper and the shroud can seat.

The burn-out ratio for the fuel element was calculated for steady state conditions at various flow rates. The calculations were for the start of cycle with the "aluminum window" in place. This is the worst case. Channeling and mixing were both investigated as follows: The heat flux necessary to produce burn-out was first calculated using the Jens-Lottes correlation (UCLA data). This heat flux was then ratioed with a corresponding local heat flux. The resulting value was defined as the burn-out ratio. The burn-out ratio varies over the length of the fuel element because the local heat flux varies over the length of the fuel element. The burn-out ratio was calculated for each of 18 two-inch nodes.

The results of the burn-out ratio calculations are plotted on Figures 13 and 14.

Using a steady state flow of 42 gpm through the test assembly, the minimum burn-out ratio was calculated to be 4.3 for a hot rod at the start of cycle.

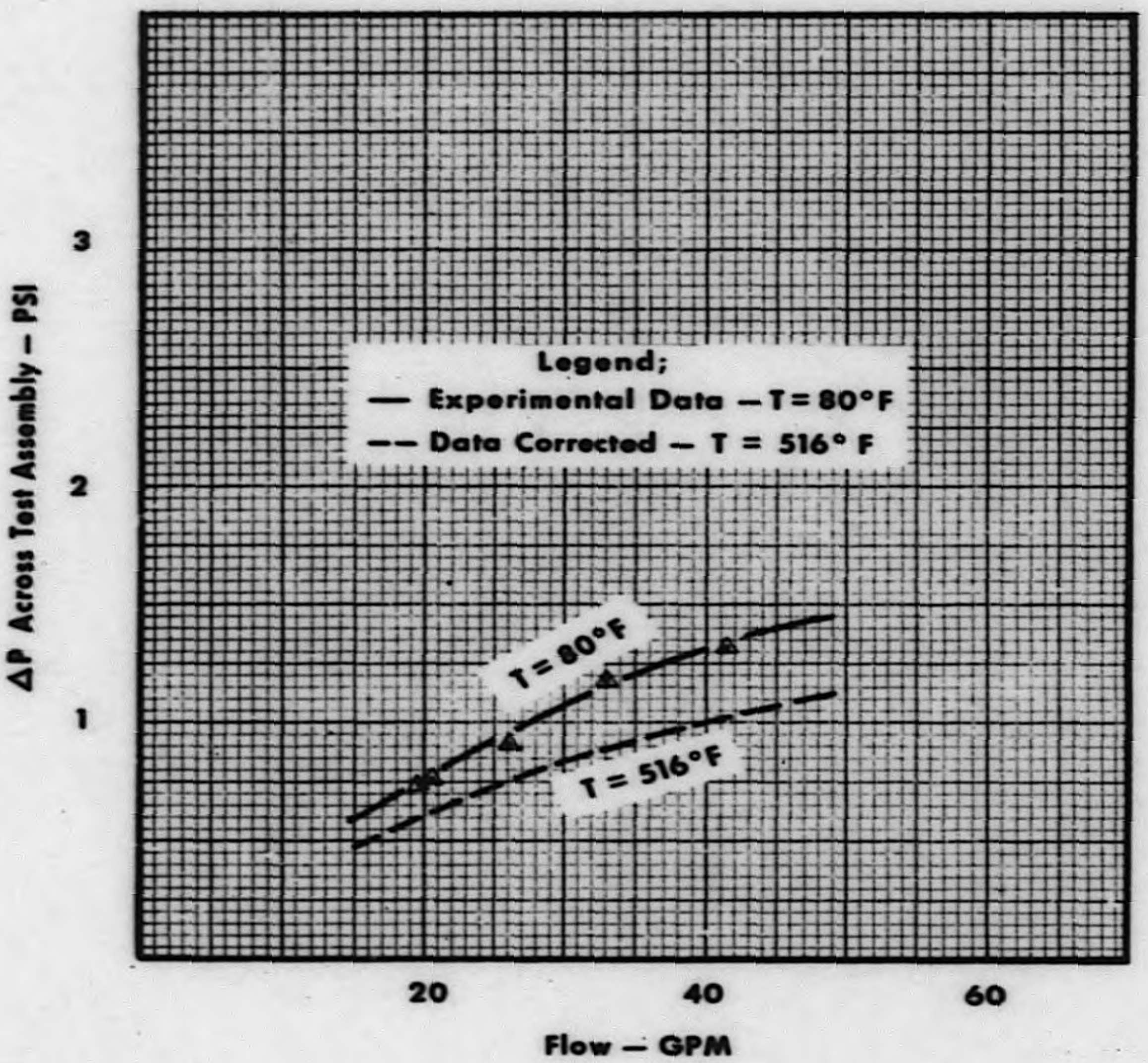

Figure 11. PRESSURE DROP ACROSS TEST ASSEMBLY AS A FUNCTION OF FLOW

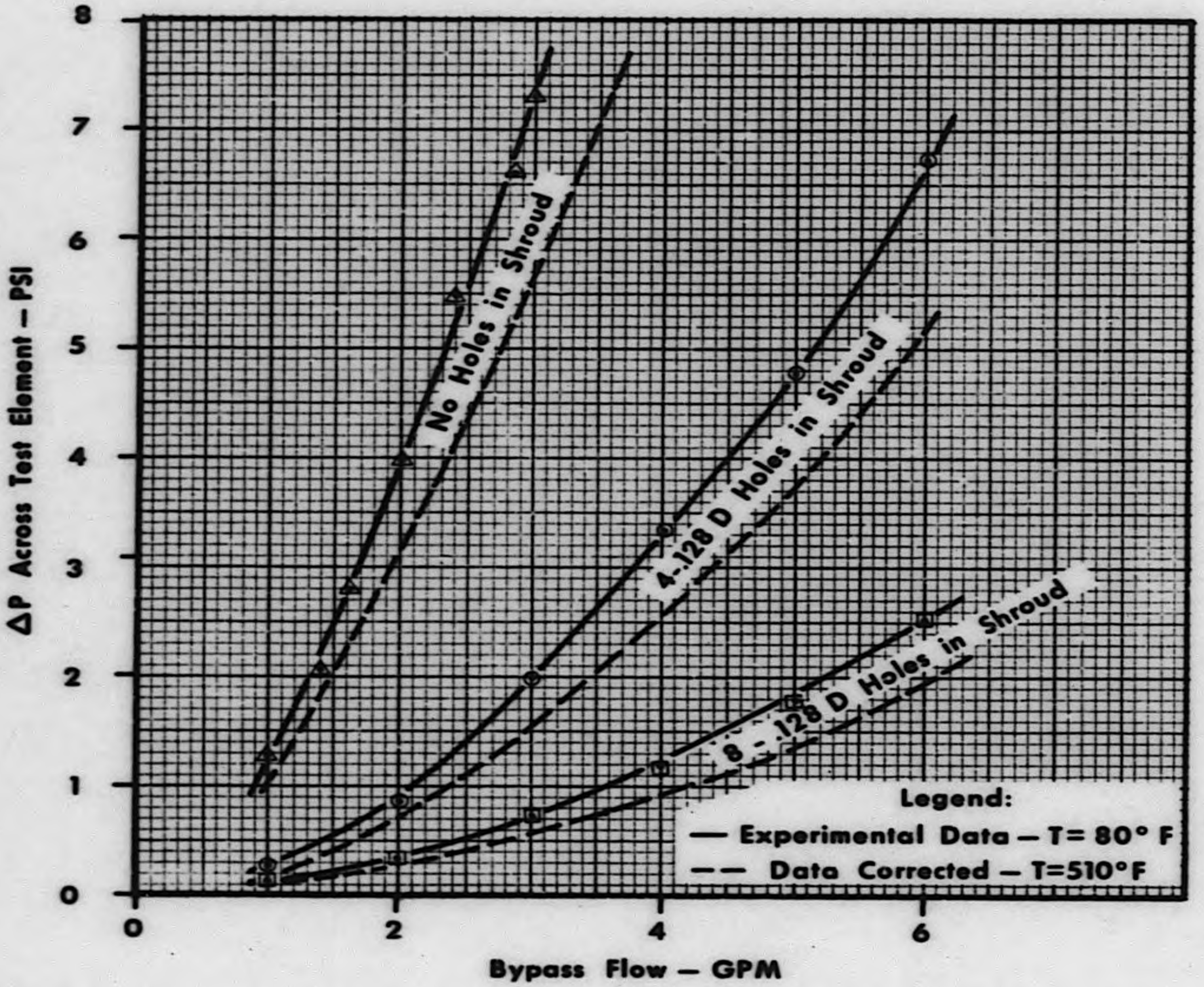

Figure 12. BYPASS FLOW VERSUS ΔP ACROSS TEST ELEMENT

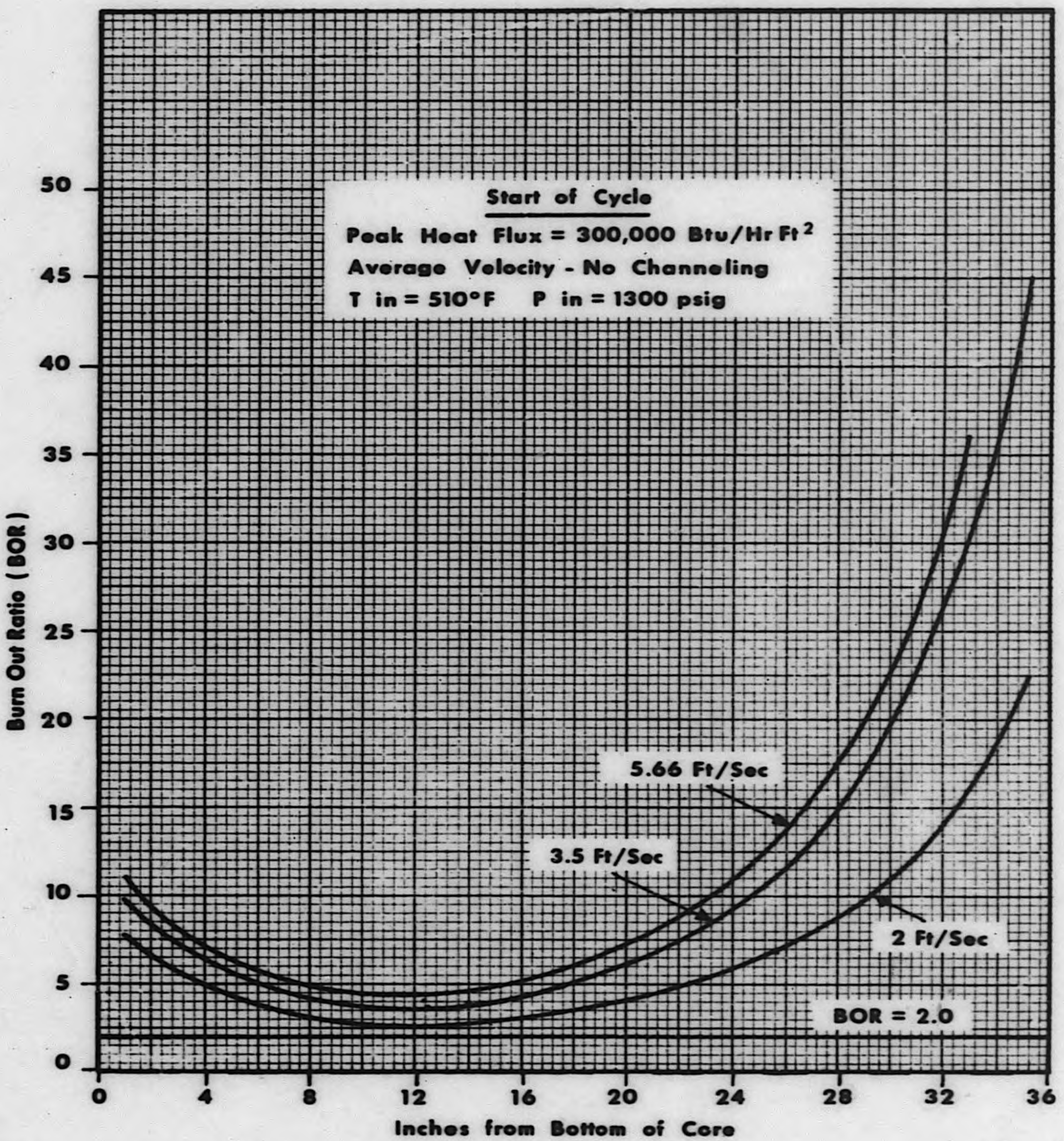

Figure 13. HOT ROD BURN OUT RATIO VERSUS LONGITUDINAL POSITION

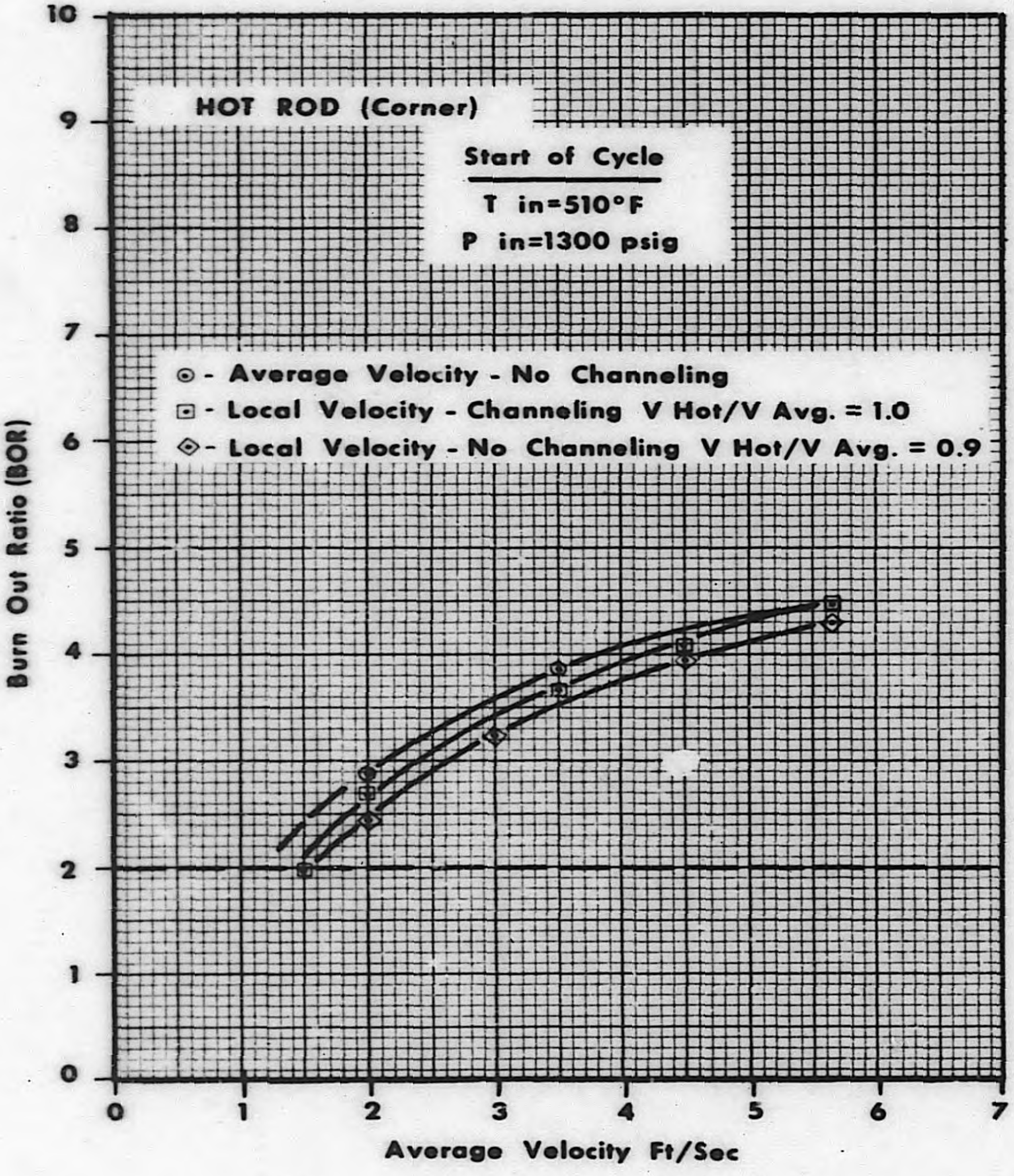

Figure 14. BURN OUT RATIO VERSUS AVERAGE VELOCITY

2. Data and Analysis

c. Thermal Analysis (Liffengren)

Thermal calculations included: (1) hot rod cladding and water temperature axial distributions at start of cycle with different flow rates, (2) cladding and water temperatures for the hot rod, 2nd rod and cold rod at the design flow rate (through the test assembly) of 42 gpm, (3) center fuel temperatures for the hot rod at different flow rates, and (4) center fuel temperature distribution for the hot rod, 2nd rod, and cold rod at design flow.

The cladding surface temperature was determined using film heat transfer coefficients calculated with either the Dittus-Boelter equation or the Jens-Lottes correlation, depending on which was applicable for the existing conditions. The center fuel temperature distributions were calculated using the General Electric Coffi Code Computer program. The center fuel temperature was calculated using the film temperature drop, cladding temperature drop, gap temperature drop, and pellet temperature drop. These values, for the hot rod at the point of maximum heat flux, were calculated to be approximately 64°, 84°, 300°, and 2772°, respectively. Values for rod to rod temperatures were taken directly from the above calculations.

The cladding temperature distribution for the start of cycle conditions were plotted on Figure 15. The maximum surface temperature was calculated to be approximately 590°F for both average velocity (no channeling) and local velocity (channeling) conditions. The flat surface temperature profile is due to nucleate boiling existing on the surface. On Figure 15 it may be seen that nucleate boiling increases with an average velocity of 5.66 ft/sec., the exit water temperature is approximately 533°F, or 46°F subcooled.

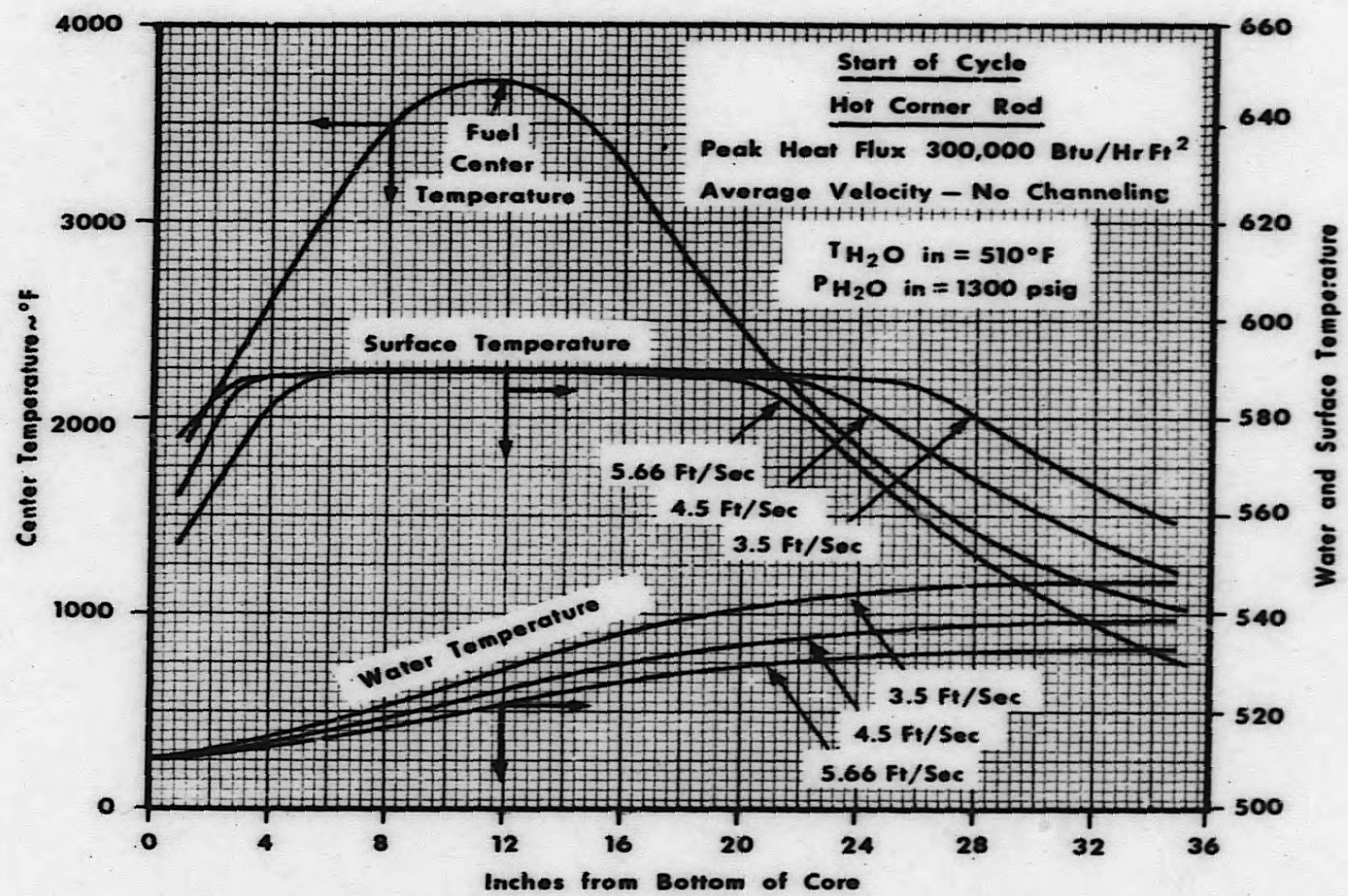

Figure 15. EFFECT OF VARYING VELOCITY ON FUEL CENTER, BULK WATER, AND CLAD SURFACE TEMPERATURE DISTRIBUTIONS

VAL 1127

c. Thermal Analysis (Cont'd)

At a steady state flow of 42 gpm and using local velocities, the surface and water temperatures for the hot rod, 2nd rod, and the cold rod were calculated. The calculated maximum surface temperature gradient was approximately 21°F between the hot rod and the 2nd rod.

The hot rod maximum center fuel temperature was calculated to be 3745°F. The 2nd row edge rod maximum center temperature was calculated to be approximately 3000°F., and the 3rd or cold corner rod maximum center temperature was calculated to be approximately 2500°F.

The calculated maximum clad temperature gradient would be approximately 84°F at a peak heat flux of 300,000 Btu/hr-ft^2. This value exceeds the specification, which calls for 80°F.

The maximum center fuel temperatures for the aluminum window up and down were calculated to be 2880°F. and 3745°F respectively. The peak surface temperatres for the aluminum window up and down were calculated to be 588.5°F and 589.5°F. respectively. As can be seen on Figure 16, the maximum temperature gradient for the center temperature occurs at the location of the peak flux. The maximum surface temperature gradient would occur at approximately 20 inches from the bottom of the core.

The exit water temperatures were calculated to be 527°F and 533°F for the aluminum window up and down, respectively. These water temperatures correspond to 52°F and 46°F subcooled respectively.

These temperature distributions were based on an average velocity (no channeling condition). Assuming local velocity, channeling has only slight affect on the temperature distributions.

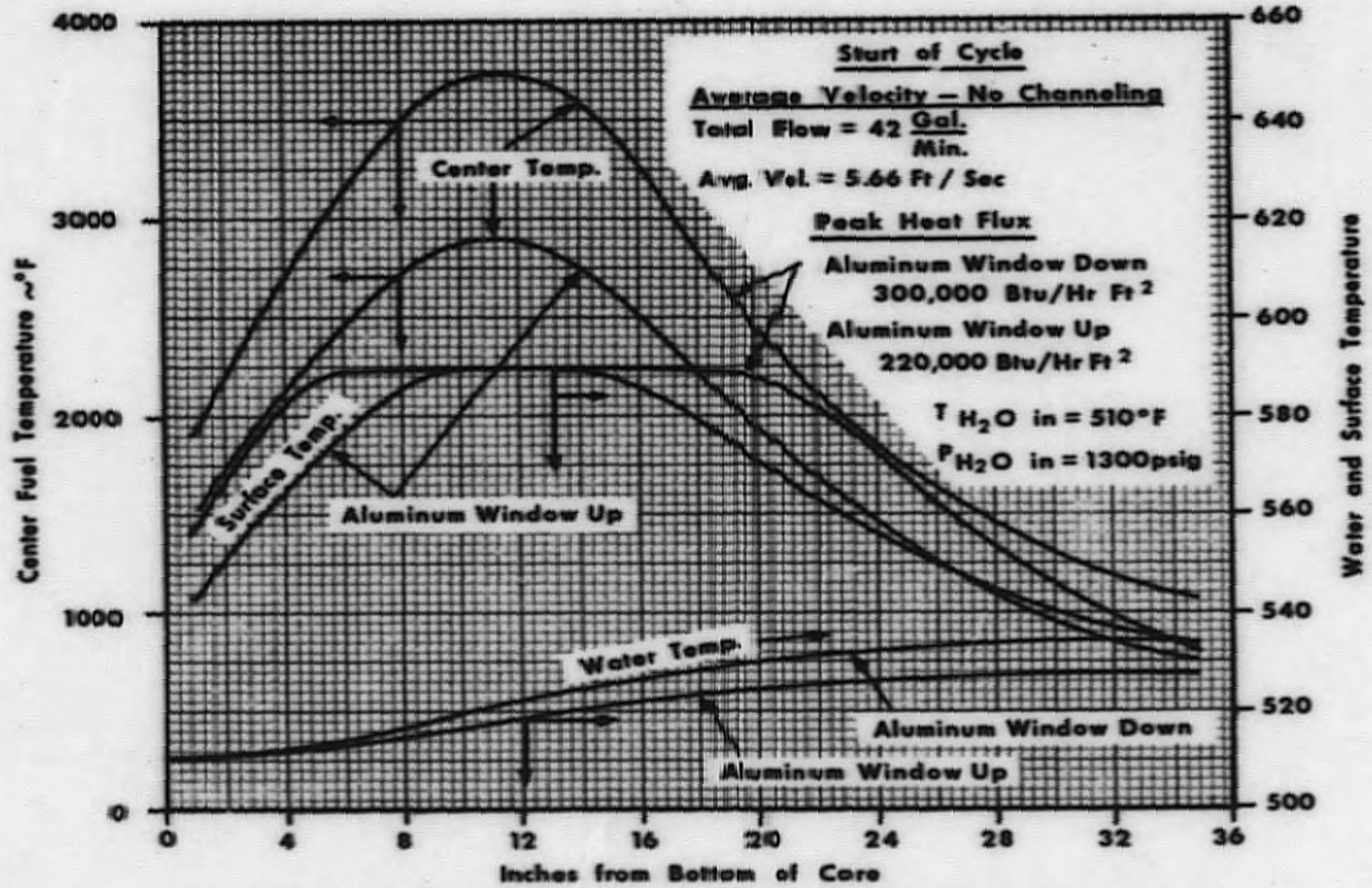

Figure 16. EFFECT OF ALUMINUM WINDOW ON FUEL CENTER, BULK WATER, AND CLAD SURFACE TEMPERATURE DISTRIBUTIONS

VAL 1128

C. Physics

1. Flux Run (Worthington)

In order to confirm the calculated radial and axial peaking factors in the Maritime Loop Test Assembly, neutron flux distributions were measured in the GETR Boiling Water Loop. This measurement was performed during the scheduled outage prior to GETR Cycle 18, on November 12, 1960.

The flux measurements were accomplished by placing detectors in various positions in the test assembly, mounting the test assembly in its normal position in the loop, and then subjecting it to 25 KW reactor power for 30 minutes. The detectors were then scanned to determine activity levels.

Figure 17 shows the positions of the various detectors used with respect to the Test Assembly and the GETR pressure vessel. Flux distributions were measured with natural uranium wire.

Relative fluxes obtained by counting fission product activity in the uranium wires were presented in curves normalized to the average flux of unity over the 36 inch active fuel region of the GETR core. A curve for one of the flux wires is shown on Figure 18. The average position (effective bank) of the control rods during the flux run was 17.2 inches withdrawn.

The arithmetic average of all the peak to average values of the flux wires was 1.66.

Measurements made in the GETR Trail Cable Facility during April, 1960, at 30 MW reactor power, indicate a peak-to-average of approximately 1.8, interpolating to an effective control rod bank of 17.2 inches. The two values are consistent within the expected 10 percent experimental error.

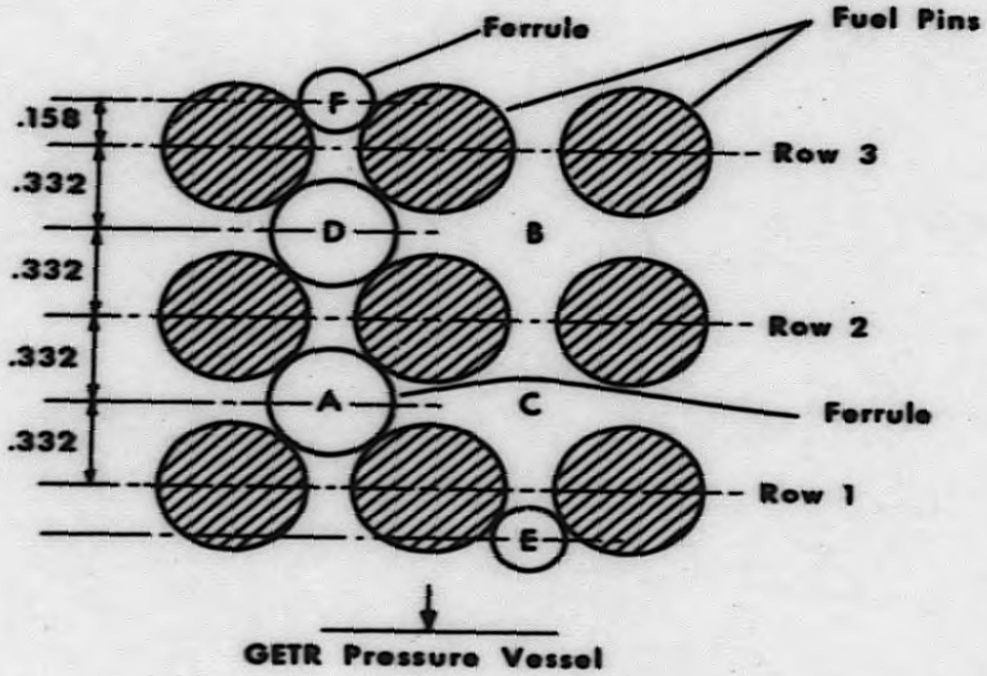

Figure 17. TEST ASSEMBLY WIRE AND FOIL POSITIONING

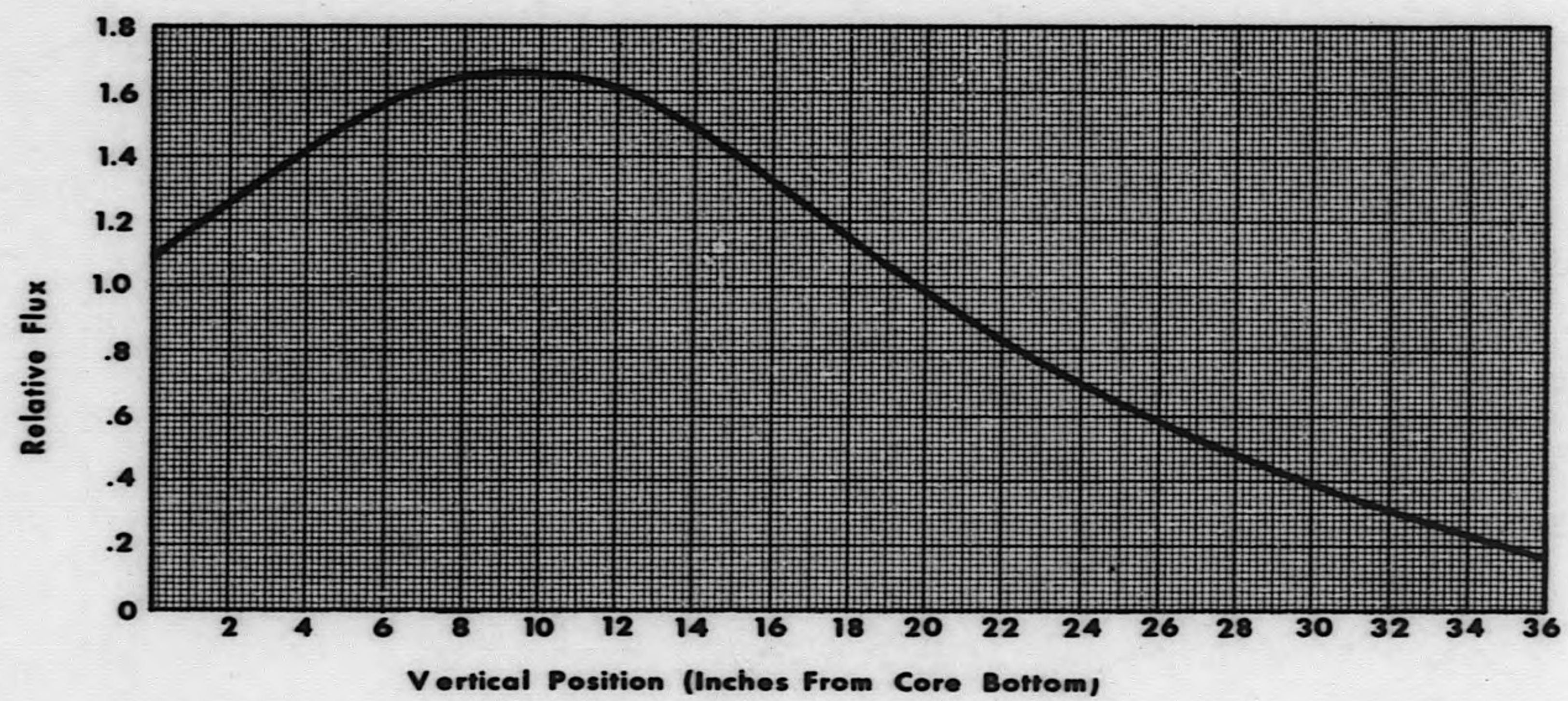

Figure 18. FLUX RUN WIRE "C"

VAL 1130

1. Flux Run (Worthington) (Cont'd)

Average decay, corrected counting rates on the wires are plotted in Figure 19 as a function of distance from the inside edge of the fuel bundle radially outward from the center of the reactor core to determine the radial flux variation in the test assembly. Taking the average flux over the bundle as unity, fuel pin-to-average values are given in Table I.

TABLE I

	Radial Flux Relative to Average in Entire Bundle	
Fuel Pin Position	Measured	Calculated
Row 1	1.39	1.27
Row 2	0.93	0.96

2. Aluminum Flux Window

Initial flux calculations indicated that the desired thermal flux could not be obtained because of the moderating effect of the water between the facility tube and the reactor pressure vessel. To achieve the desired flux, a hollow aluminum "window" was designed to displace the water between the facility tube and the reactor. The aluminum window is hydraulically actuated and may be moved up and down remotely.

3. Burn-up

Figure 20 describes burn-up as a function of time and GETR cycles. Each cycle is approximately 23 irradiation days. The burn-up curve was arrived at using the General Electric Purple Code Computer Program. With reference to Figure 20, a fuel burn-up of 15,000 MWD/Tonne will be accomplished in approximately 34 cycles or 782 days of irradiation.

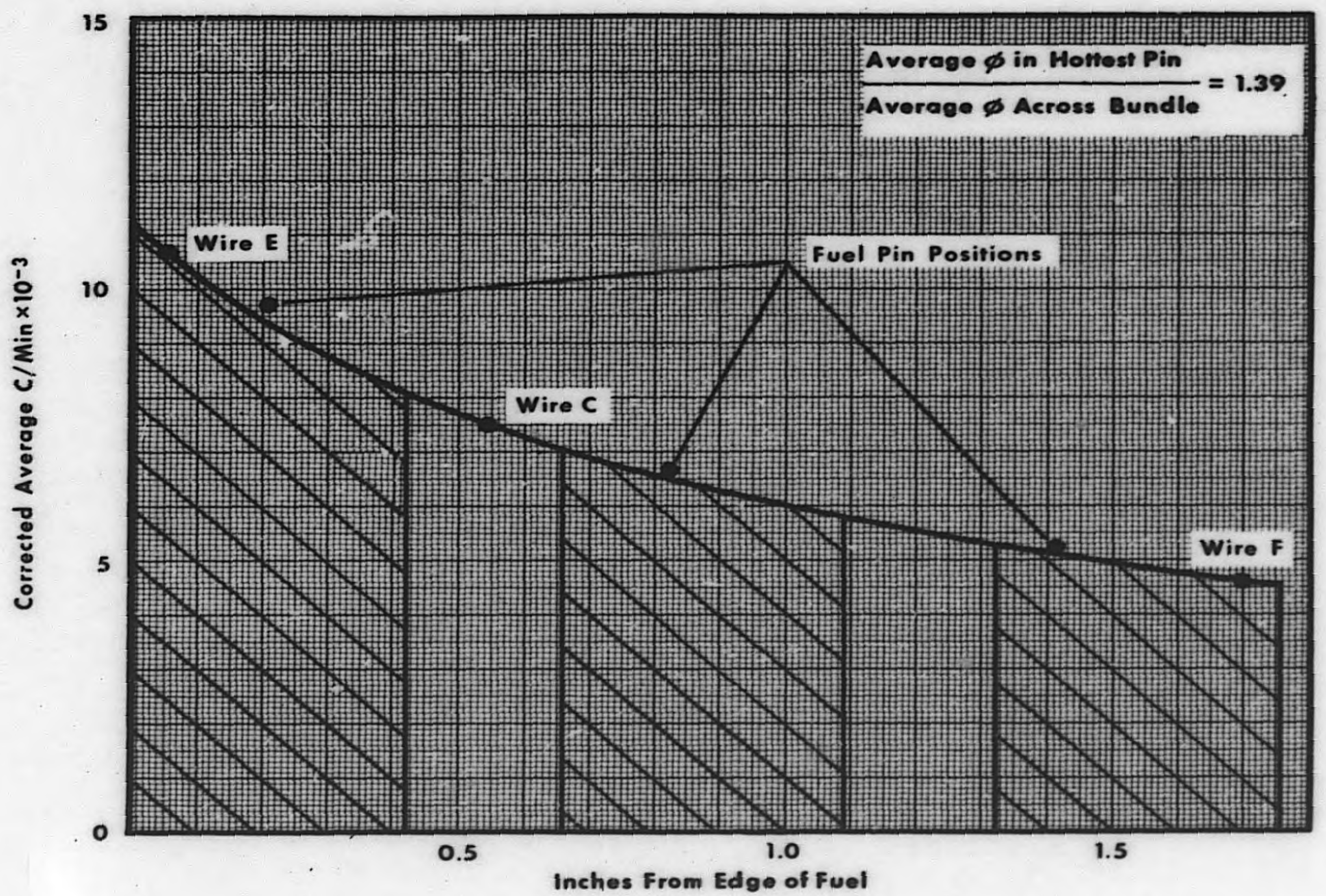

Figure 19. RADIAL FLUX DISTRIBUTION ACROSS NMSR - TEST ASSEMBLY

VAL 1131

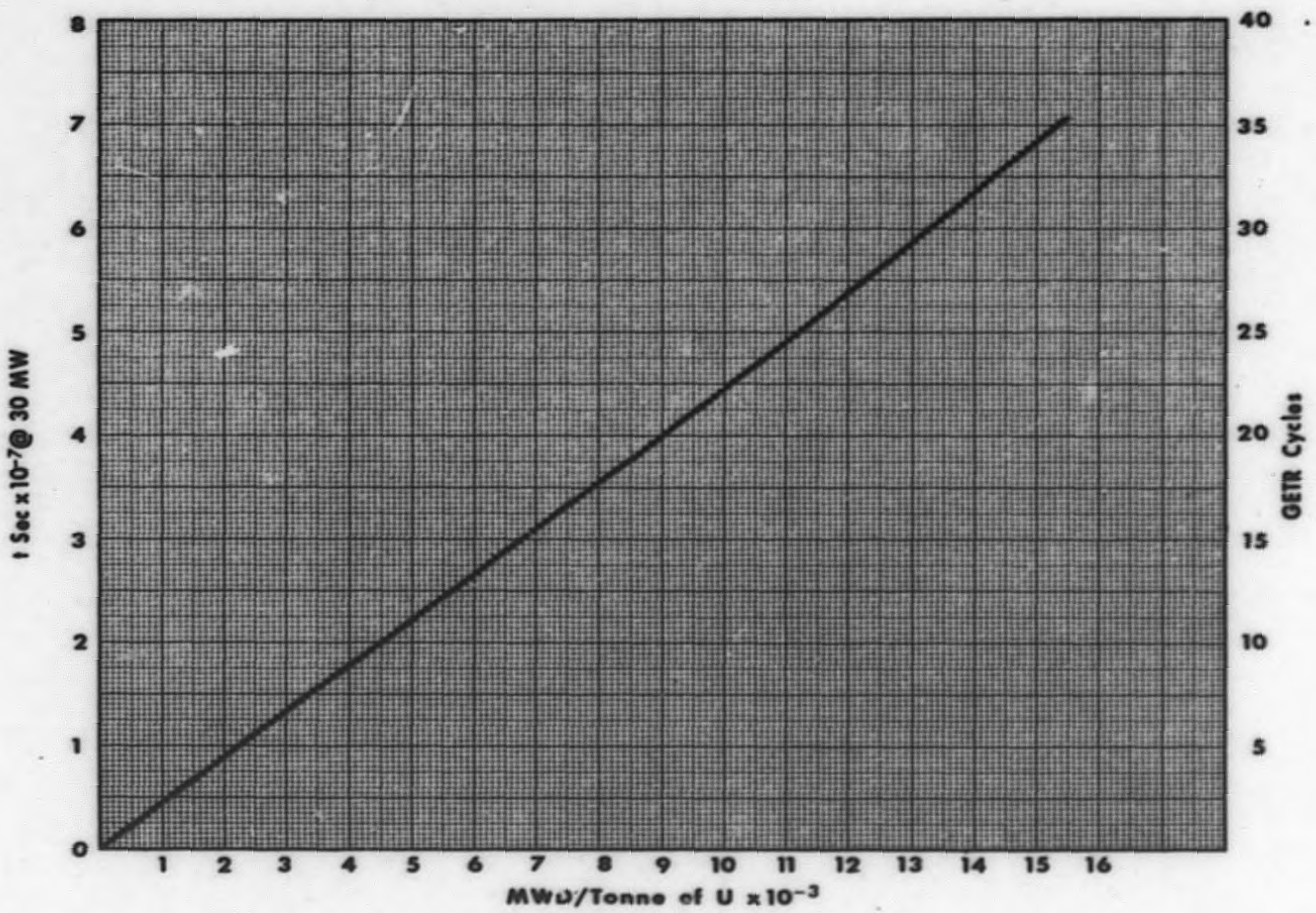

Figure 20. FUEL BURN-UP (AVERAGE) NMSR - G.E.T.R. TEST ASSEMBLY

VAL 1132

VI. LOOP OPERATION - DATA AND ANALYSIS

The shakedown phase of this program was completed on November 29, 1960. The NMSR-GETR test assembly #2 was inserted on the same day and final preparations for nuclear operation were accomplished.

Irradiation of the test assembly began on December 2, 1960, and continued until December 12, 1960, which was the scheduled GETR Cycle 18 shutdown date. No difficulties were encountered during startup (full nuclear operation was achieved without a reactor scram). Startup was accomplished on automatic control and the loop remained on automatic control throughout the irradiation period.

Approximately 265 MWD of exposure were received by the test assembly during the ten days of irradiation. The following sections summarize the loop operation during this period.

A. Design Operating Conditions (Howell)

1. Operating Parameters

Figure 21 shows desired parameter values at various points in the loop. The set points for automatic instrumentation were adjusted to these values.

2. Corrosion - Coolant Chemistry Conditions

Table II shows a summary of the desired water chemistry specifications. A discussion of the actual loop water chemistry during initial nuclear operation is presented in the next section.

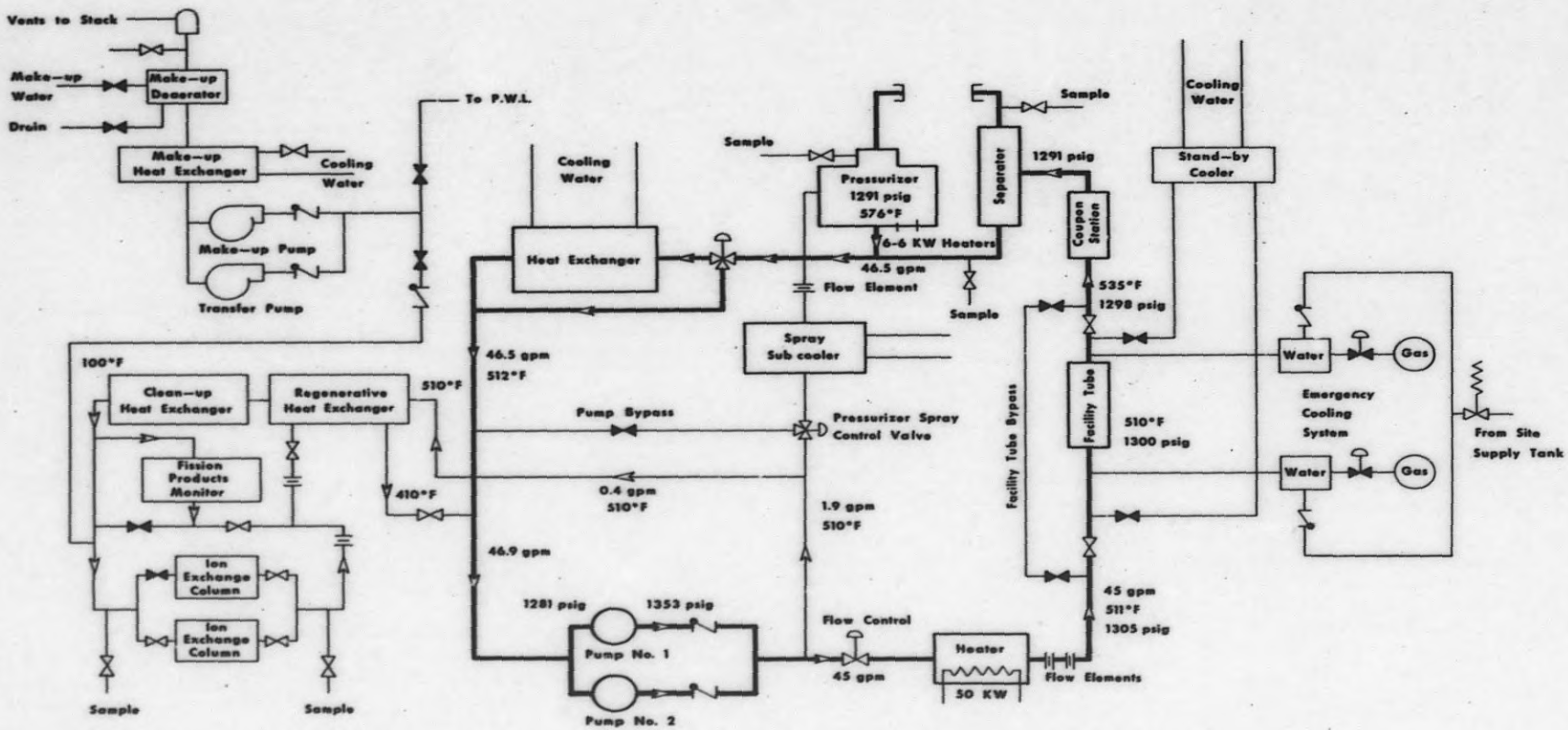

Figure 21. MARITIME IRRADIATION SIMPLIFIED FLOW DIAGRAM - LOOP OPERATING CONDITIONS

TABLE II

CHEMISTRY SPECIFICATION SUMMARY - MARITIME LOOP

Parameter		Measurement Technique	Location of Test	Control Range	Detection Range	Expected Accuracy	Minimum Sampling Frequency
Total Solids	Dissolved	Relate with conductivity		---	---	---	
	Suspended	1. Millipore filter - Fe Determination	Lab	none	0.01 ppm	± 20%	Weekly*
		2. Turbidity		---	---	---	Daily
pH		Colorimetric	loop	6.5 - 8.5		± 0.5	Every 8 hours
Conductivity		In-Line Monitor	loop	best attainable	0-5 μmho/cm	± 0.5 μmho/cm	Continuous
		Conductivity bridge	loop	---	---	± 0.1 μmho/cm	Every week
Chlorides		Turbidity of treated sample (AgCl procedure)	loop	normal max. <0.1 ppm	down to 0.01 ppm	± 0.01 ppm	Daily
Oxygen		Thallium column	loop	<10 PPb	0-100 PPb	± 3 ppb	Daily
Hydrogen		Pressurizer or loop water sample for gas chromatograph	Lab.	1.8-3.6 ppm	---	± 10%	3 per week
		Total gas (correlated to hydrogen)	loop	---	---	± 10%	Every 2-4 hrs. or as necessary to control H_2 conc

*Crud by wt. (if 1 ppm) 1 per cycle
Data may be reported as much as one cycle late to permit multiple processing of samples.

B. Summary Operational Data (Danielson)

1. Chemistry Data Summary

The following discussion summarizes the loop water chemistry for the period December 2, 1960, through December 12, 1960.

a. Water Purity

The water conductivity decreased from an initial measurement of 70 mho to less than 3 mho during the first four days operation and held below 3 mho for the remainder of the cycle. The ion bed effluent held at less than 0.1 mho conductivity. The specification for the loop coolant (consistent with the N.S. Savannah Water Quality Program Fig. 2.1) is 2 mho. Conductivity is measured on an Industrial Instruments Inc. Model RC-16B2 bridge utilizing a Model CEL-D001 glass flow cell (cell constant 0.010). The results from this cell are frequently checked since the oxygen measurement includes an independent conductivity determination utilizing a high pressure cell.

b. Chloride

Initial readings of as high as 2 ppm chloride were obtained immediately after loop start-up, but rapidly decreased due to ion exchange clean-up, to less than 0.1 ppm within the first day's operation. A high purification rate was used initially to reduce the chloride and decrease the loop water conductivity. The high initial chloride level was due to the inadvertent introduction of potable water containing chloride into the loop through the emergency cooling system during the preceding loop shutdown. Steps have been taken to minimize the possibility of a recurrence of this incident.

b. Chloride (Cont'd)

The measurement of the chloride concentration is being investigated in order to provide a simple direct procedure applicable at the loop rather than transport samples to the chemistry laboratory. The nephelometric chloride procedure using $AgNO_3$ as the indicator is the least complicated method applicable to the measurements, and has been used for the routine analyses in the program to date. Repeated standard runs have shown that the indicated nephelos value is roughly linear with the chloride concentration in the range from 0.1 to 1 ppm, with a slope of about 160 nephelos units per ppm of chloride. Since the range of this slope value is 80 to 200, samples indicating less than 8 nephelos are conservatively within the specified limit of 0.1 ppm.

A more exact nephelometric procedure ("Colorimetric Determination of Non Metals", Baltz, Interscience Publishers (1958)) involving controlled temperature and timing has been investigated in the range of 0.1 to 5 ppm. The results, presented below, show excessive variation at the lower concentration.

ppm Cl

Date	0.1	0.2	0.4	0.6	0.8	1.0	2.0	4.0	5.0
	Observed Nephelos Units								
1/13	17.3	27.5	47.0	52.4	72.6	103.7	290.5	616.5	695.8
1/12	17.4	28.2	41.4	79.8	83.7	101.7	-	465	674.4
1/10	3.6	14.2	21	53	89.5	148	334	743	972
Ave	12.6	25.0	36.5	61.7	81.6	117.6	306.2	606.2	755.6
Range	13.8	14.0	26.0	26.8	16.9	46.3	43.5	278	297.6
$100 \frac{range}{Ave}$	109	56	74	43	21	39	14	46	39
Approx in %*	59	33	44	25	12	23	8	27	23

"Colorimetric Determination of Non Metals", Baltz, Interscience Publishers (1958)

* range/1.69

b. Chloride (Cont'd)

The nephelometry procedures have one distinct advantage in that the blank values are quite consistent and low (4 nephelos units).

The colorimetric Mercuric Thiocyanate procedure (B and W 1047-T-59) has been found to be reasonably accurate in the range of .5 to 5 ppm, and to have a linear extinction coefficient of about 0.045 $cm^2/\mu gm$. However, the blank value has been observed to be the equivalent of 0.5 ppm on both a spectrophotometric (463 $m\mu$) and on a filter photometer (470 $m\mu$ Filter).

The limitations of the procedures discussed above in the range below 0.1 ppm indicate that a concentration procedure must be used for the laboratory checks on the loop procedure. The use of a small Dowex 1 resin column has been found to be quite satisfactory to concentrate samples up to a factor of 100.

c. pH

The pH of the loop water has been about 7.4 (8.3 first day) as determined by colorimetric procedures and checked with a Beckman Model GS pH instrument. The accuracy of the data is questionable, however, since measurements in high purity water can vary considerably according to the measurement technique used.

d. Total Solids

The total solids content has been reasonably low in that the water conductivity was below 3 μmho and the turbidity generally below 15 nephelos units. The dissolved solids would be less than 1.5 ppm, based on a factor of 0.5 ppm per μmho conductivity.

e. Dissolved Oxygen

The dissolved oxygen concentration shortly after start-up, according to the thallium column observations, was about 0.02 ppm, decreasing to below 0.01 ppm. The levels thereafter increased to about 0.03 ppm and remained close to 0.03 ppm for the remainder of the cycle. The make-up water used is both demineralized and deaerated but did contain as much as 0.24 ppm oxygen. By reducing the leak rate and thus minimizing the make-up rate, it is anticipated that the dissolved oxygen can be maintained below the 0.01 ppm concentration specification. Better control of the hydrogen gas concentration may also help keep the oxygen concentration low.

Improvements in the operation of the make-up system will be made to reduce the oxygen concentration in the make-up water, thus making the control of the dissolved oxygen in the loop easier.

f. Dissolved Hydrogen*

The gas leak rate so far has been excessive with the hydrogen gas concentration decreasing a factor of two roughly every 2 hours. As a result, the hydrogen gas concentration has been within specification for brief periods only. With the batch addition method of adding hydrogen, an addition at least every hour would be required to keep the loop operating within specification. However, transient peaks above specification would probably occur due to the delay involved reaching equilibrium gas concentration conditions in the pressurizer. The pressurizer at equilibrium holds the predominant proportion of the total gas in the loop.

As an example, after adding the equivalent of 500 cc H_2 at 1300 psi over a period of 1 hour by 20 successive batch additions on December 11, the gas concentration in the loop reached 7.5 ppm H_2 within 1 hour after addition. About 2 hours after concluding the addition, the gas apparently reached equilibrium with that in the pressurizer and the resulting concentration reduced to 2 ppm (reduction also partly due to leaks). Four hours after the addition, the concentration of gas in the loop had decreased to 0.6 ppm, principally due to leakage.

f. Dissolved Hydrogen* (Cont'd)

Efforts are being made to reduce the apparent leak rate. An attempt will be made to add hydrogen continuously but the only practicable method of addition may be by batch addition on an hourly basis.

g. Fission Product Release

A few samples were analyzed for I-131 and I-133 during the course of the irradiation in order to provide an estimate of surface contamination of the fuel and to check the integrity of the fuel cladding itself. A typical sample taken December 10 contained 214 disintegrations per minute Iodine-133 per milliliter (dpm/ml) coolant and 16.5 dpm/ml Iodine-131. These values correspond to a source in the order of 0.3 milligrams 4.6% enriched UO_2 which is somewhat higher than was expected. This activity is presumed to be entirely from surface contamination of the fuel and the natural uranium traces in the structural material.

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*Subsequent data commencing early in Cycle #20 indicate that the hydrogen concentration is held within the specified limits.

2. Heat Generation and Peak Heat Flux

Table III summarizes the calculations of heat generation and peak heat flux applicable to the Cycle #18 irradiation.

The peak heat flux as predicted from physics data is based on the particular core loading and rod bank position for GETR Cycle #18. The predicted 340,000 Btu/hr-ft^2 is higher than the nominal heat flux of 300,000 Btu/hr-ft^2 largely because of the flux skew peculiar to the fuel loading of Cycle #18, and to a lesser extent the correction for gamma heating in loop fuel from reactor core gammas. During initial startup the heat generation was monitored based on flow and temperature data so as not to exceed a maximum of 300,000 Btu/hr-ft^2.

The data obtained on startup December 2 initially showed the power generation to be considerably lower than predicted. The position of the facility tube with respect to the reactor pressure vessel was therefore checked during the shutdown late December 2 and found to be further away from the pressure vessel than believed. The facility tube was then moved closer to the reactor pressure vessel supposedly to the position at which the neutron flux calculations were made.

The subsequent resulting heat generation calculations based on temperature and flow measurements were still low compared to the physics values predicted (experimentally measured total heat generation only 121 KW compared to 167 KW based on physics calculations). This discrepancy has been resolved to a limited extent but efforts are continuing to obtain a more accurate determination of the heat generation.

Subsequent to GETR Cycle #18, the facility tube position was re-checked and found to be a slight distance further away from the core than assumed. The separation was such that the physics calculations could be expected to be about 8% high. Reducing the physics calculations by 8% gives a new physics prediction of the peak heat flux of 313,000 Btu/hr-ft^2. Correcting this value for the difference in rod bank positions for cycle 18 and cycle 20 reduces the predicted heat flux another 5.5%, or to 298,000 Btu/hr-ft^2.

2. Heat Generation and Peak Heat flux (Cont'd)

Current loop operation during GETR Cycle #20 (GETR Cycle #19 was a non-operating cycle) has shown that the main flow recorder used in the heat generation calculations gives a lower value of main flow than that obtained by a heat balance on the main heat exchanger. On January 12, 1961, the main flow as determined from the main flow recorder was 80% of the flow as determined from the main heat exchanger heat balance. Assuming the same ratio existed between the two methods of determination of the main flow during GETR Cycle #18, the peak heat flux recalculated using the flow from the main heat exchanger heat balance was 298,000 Btu/hr-ft^2.

Our best estimate at this time is that the actual peak heat flux was close to the latter value of 298,000 Btu/hr-ft^2. Efforts are continuing to obtain a more accurate determination of the heat flux. Alternate methods of flow measurement are being evaluated and new data obtained. An additional instrument has been installed for improved differential temperature measurement across the facility tube.

In each of the methods used to calculate the peak heat flux, the experimentally obtained peak to average factor of 2.31 was used. The peak to average factor is highest at the beginning of a reactor cycle and is dependent upon the position of the reactor control rods. Considering the control rod position at the start of GETR Cycle #18, the actual initial peak to average factor should be somewhat less than the value of 2.31 used in the calculations.

TABLE III

Heat Generation and Peak Heat Flux

NMSR #2 Fuel

Irradiation Dec. 2-Dec. 12, 1960

GETR Cycle #18

	Physics* Calculations	Measured Heat Generation: Flow from main flow recorder	Measured Heat Generation: Flow from main HX heat balance
Total Heating (loop fuel fission plus gamma from core)	167 KW	121 KW	152 KW
Gamma heating outside fuel from core gamma (15,540 gm)	14.0 KW @0.90 watts/gm	14.1 KW @0.91 watts/gm**	17.7 KW @1.14 watts/gm**
Total heeting in fuel	153 KW	106.9 KW	134.3 KW
Gamma heating in fuel from core gamma (9,770 gm)	8.8 KW	8.9 KW	11.1 KW
Fission heating in fuel	144.2 KW	98 KW	123.2 KW
Average heat flux (3.54 ft^2 surface)	147,000 BTU/hr-ft^2	103,000 BTU/hr-ft^2	129,000 BTU/hr-ft^2
Peak heat flux (based on physics flux run peak to average of 2.31)	340,000 BTU/hr-ft^2	238,000 BTU/hr-ft^2	298,000 BTU/hr-ft^2

*Calculations expected to be about 8% high since facility tube was apparently not as close to core as assumed (for Cycle #18 only). Based on 18 inches start of cycle rod bank. Actual cycle #20 initial rod bank was 20". This corresponds to a 5.5% reduction of the peak flux, which, combined with position error, results in a corrected peak heat flux of 298,000 BTU/hr-ft^2.

**Gamma heating based on main flow and temperature data obtained on 11/25 and 11/27 prior to fuel insertion.

END