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DEVELOPMENT REPORT

THE SECOND REFUELING OF CORE 1 OF THE SHIPPINGPORT ATOMIC POWER STATION

AUGUST 1962

CONTRACT AT-11-1-GEN-14

BETTIS ATOMIC POWER LABORATORY, PITTSBURGH, PA.,
OPERATED FOR THE U. S. ATOMIC ENERGY COMMISSION
BY WESTINGHOUSE ELECTRIC CORPORATION

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UC-80: Reactor Technology
TID-4500 (17th Ed.)

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C. E. Center, H. Feinroth, and J. E. Yingling

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FOREWORD

This report tells the lessons learned from the second refueling of the Shippingport Atomic Power Station. Report WAPD-233 of July 1960 reviews the first Shippingport refueling.

Second seed replacement commenced August 16, 1961; the plant was returned to full power operation with Seed 3 on October 24, 1961.

The refueling was performed by Duquesne Light Company, operator of the station, under the general supervision of the Atomic Energy Commission. Personnel from the Bettis Atomic Power Laboratory, designers of the reactor plant, served as technical consultants to the Commission. Throughout the refueling, personnel and plant safety considerations were overriding factors; no person received radiation exposure greater than that permissible.

We believe the experience gained will result in future refuelings at Shippingport taking less time and will also prove of value to all civilian reactors. We hope this report will contribute to further advances in refueling operations and in refueling equipment design.

H. G. Rickover
United States Atomic Energy Commission

Phillip A. Fleger
Chairman of the Board and President
Duquesne Light Company, Pittsburgh, Pa.

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CONTENTS

CONTENTS (Continued)

THE SECOND REFUELING OF CORE 1 OF THE SHIPPINGPORT ATOMIC POWER STATION

C. E. Center, H. Feinroth, and J. E. Yingling

I. SUMMARY

The second seed refueling of the Shippingport Atomic Power Station was begun on August 16 and completed on October 24, 1961; the station was then returned to full power operation with Seed 3 installed. Sixty-three working days were required to complete the refueling and return the plant to full power operation, whereas 156 working days were required for the first refueling* of the Shippingport Atomic Power Station.

The purpose of this report is to describe the planning and the methods used to implement the second refueling operation and the lessons learned during this refueling. This report supplements the report of the first refueling; it compares the two refueling operations, indicating the results achieved from the lessons learned during the first refueling, and identifies areas where further improvements can be made.

The refueling operation was planned and carried out by personnel of the Duquesne Light Company. All procedures were approved by the Atomic Energy Commission and the Bettis Atomic Power Laboratory. Representatives of the Bettis Atomic Power Laboratory, operated for the Government by the Westinghouse Electric Corporation, designed the refueling equipment and served as technical consultants to the Commission throughout the refueling operation.

Personnel safety was the overriding consideration throughout the refueling; no personnel received a radiation exposure greater than the permissible dosage.

Since a major purpose of the Shippingport project is to advance reactor technology, many operations were performed during the refueling shutdown period because of the experimental nature of the plant. Some examples are the operations involving removal and installation of experimental core instrumentation, the extensive calibration given this instrumentation, and the several experimental physics and plant tests performed prior to and subsequent to initial core criticality. If this experimental work were eliminated, as would be the case

* The first Shippingport refueling and the lessons which were learned at that time are described in WAPD-233, "The First Refueling of the Shippingport Atomic Power Station," dated July 1960.

for a nonexperimental plant, it is estimated that the second refueling would have taken slightly more than 1 month, a period approaching the down time for a normal overhaul of a modern central station, high temperature, high pressure boiler.

A. Scope of Refueling Operations

The principal objective of the second refueling was to replace the 32 depleted seed assemblies with new seed assemblies. The original natural uranium blanket region, which produces approximately half of the power generated by the reactor, was left essentially intact, except for the replacement of two assemblies for inspection and destructive examination. The seed replacement was accomplished underwater through penetrations in the reactor vessel head. Control rod drive mechanisms and instrumentation were removed to gain access to the fuel and then replaced after the new fuel was installed. The plant was then pressurized to test all welds made on the reactor and the plant during refueling. This was followed by a period of calibrating special instrumentation, safety checks required for criticality, and several reactor physics tests to determine the reactivity of the core. The plant was then heated up and returned to power operation.

The above steps were the same for this refueling as they were for the first refueling, except that a 20-day modification to reactor vessel head instrumentation supports was performed during the first refueling.

Figure 1 presents a breakdown of effort devoted to the various phases of the refueling and compares this with the first refueling. This figure illustrates three significant features of the refueling operation:

1. Less than 10 percent of the refueling shutdown was spent replacing the 32 seed assemblies.
2. About 50 percent of the time was devoted to removing and installing control rod drive mechanisms and instrumentation.
3. About 40 percent of the refueling shutdown period was devoted to completion of scheduled plant maintenance and to checkout and testing of the entire plant after the last seal weld was completed on the reactor vessel head.

Based on the above observations, future effort should be directed towards (1) minimizing the amount of maintenance and plant checkout required after the last seal weld is completed, (2) improving the planning and preparation for the latter phases of the refueling shutdown, and (3) minimizing the amount of core structurals, mechanisms, and instrumentation which must be removed to gain access to the fuel in future reactor designs.

B. Preparations for Refueling

As a result of the lessons learned during the first refueling, increased emphasis was placed on the training of both supervisory and nonsupervisory personnel, the establishment of clearly defined refueling responsibilities, and the day to day planning and scheduling of refueling operations.

1. Site Organization and Training

Training of supervisory and key nonsupervisory personnel was begun shortly after the beginning of Seed 2 operation and was continued throughout Seed 2 lifetime. The supervisory personnel who were to be in direct charge of the refueling crews were given intensive training in the refueling operations. The increased understanding of the refueling program gained by the refueling supervisors during this training program enabled them to direct the refueling operations more effectively and, in particular, to take immediate corrective action to minimize work stoppage whenever technical problems arose in the reactor pit.

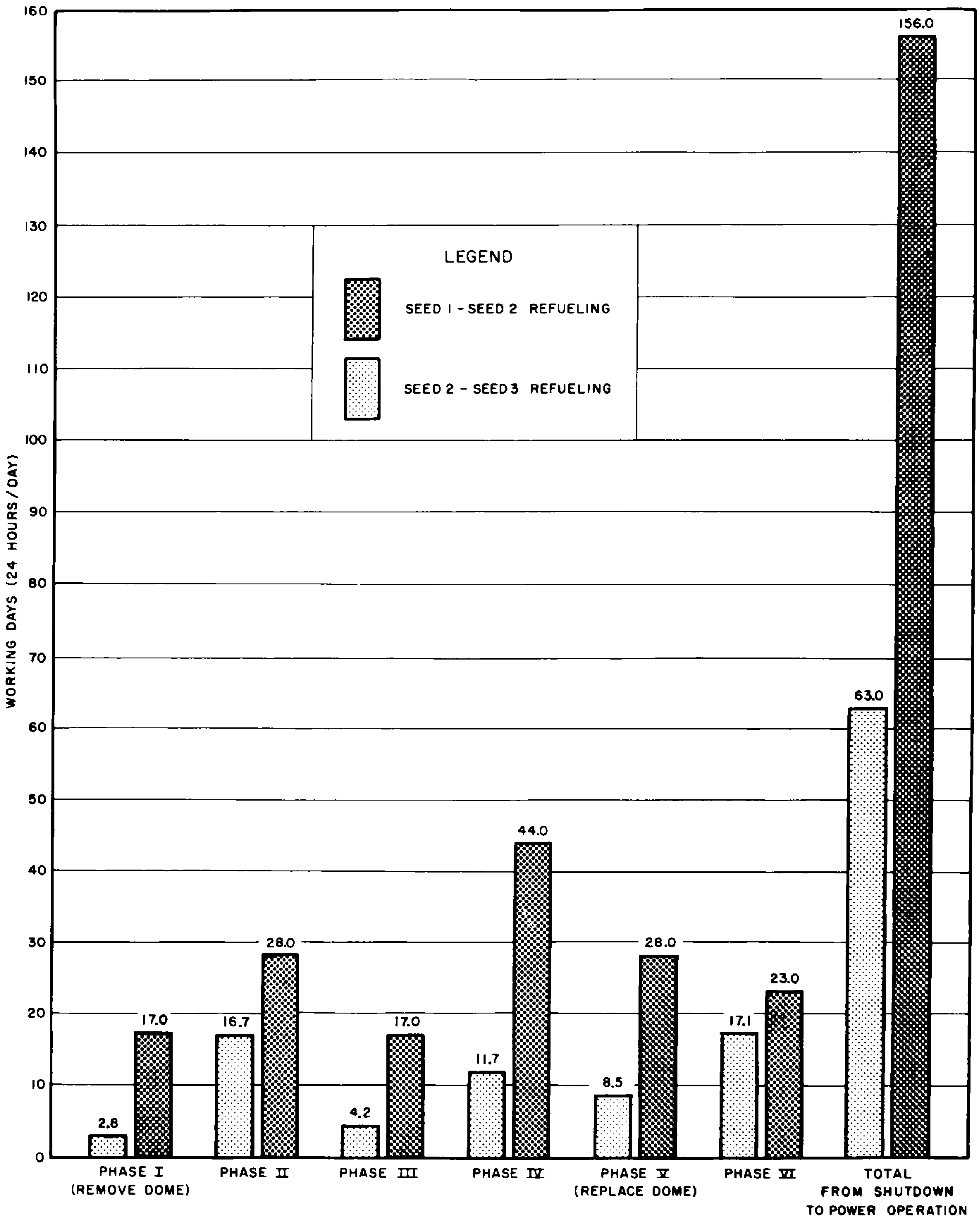


Figure 1. Comparison of Seed 2 and 3 Refueling Times.

Additional features of the organization and training for refueling which resulted from lessons learned during the first refueling and led to improvements in refueling operations are:

a. Refueling responsibilities were clearly defined early in the planning stages; one individual was given over-all responsibility for both the planning and the implementation of the refueling program.

b. Wherever possible workmen were selected who had experience during the first refueling. They were given thorough training in the specific jobs for which they were responsible; motion picture films of the first refueling were used as an aid in this training.

c. The training program for nonsupervisory personnel was initiated several months before the start of refueling, so that by the time refueling began personnel had become familiar with their assigned refueling tasks.

d. The refueling organization included a separate "technical planning" group comprised of engineers who had prepared the detailed procedures prior to refueling, and whose main responsibility during refueling was to plan ahead and take steps to improve the schedule wherever possible.

2. Planning

As during the first refueling, the basic refueling planning document was the refueling sequence, which outlined the steps and procedures to be used for refueling. However, more flexibility in sequencing the work was apparent during this refueling because of the "technical planning" group efforts and the increased understanding by the refueling supervisors of the refueling task. Thus, when problems arose in a particular operation, the supervisors immediately shifted the workmen to other operations which had been identified by the planning group as possible alternates.

3. Preparation of the Plant for Power Operations

The operations associated with plant checkout and test after completion of the last seal weld did not proceed as efficiently as the refueling operations on the reactor vessel head. There are several reasons for this, such as, (a) personnel supervising this phase of the operations were not as well prepared for the test operations, since the same personnel had been spending full time on refueling operations prior to initiation of this phase of operations, (b) refueling operations were completed ahead of schedule, whereas plant maintenance operations were not; thus, the plant maintenance had to be completed prior to checking out and testing the plant and reactor as a whole, (c) during the early part of these operations there was no single supervisor in charge of the expeditious return of the plant to power operation, (d) the need for instrumentation technicians during this phase of operations was greater than during previous phases and could not be met with available personnel, and (e) the operations did not have the benefit of an advanced technical planning group as did the refueling operations. More intensive planning and preparations should be made for this phase of operations to minimize the above difficulties during future refuelings.

C. Radioactivity Control

Based on the lessons learned during the first refueling, several changes were made to the procedures and techniques for minimizing the radiation exposure of refueling personnel. Because of these changes and the reduced total time, the total personnel exposure during the second refueling was less than during the first refueling.

1. During the first refueling, significant radiation levels (about 1R/hour) were encountered on the reactor vessel head as a result of

radioactive corrosion products (crud) entrapped in the reactor control rod drive mechanisms. During Seed 2 operation these radiation levels were carefully monitored and, to prevent further buildup of radiation levels, the control rod drive mechanisms were periodically exercised to wash off some of the crud from the mechanism surfaces. As a result of these steps, head area radiation levels were no higher than during the first refueling, despite the natural tendency for these levels to increase with operation.

2. Radiation exposure of personnel was reduced by more effective use of temporary lead shielding. The shielding was made in smaller segments and was easier to handle; it was, therefore, used more effectively and the personnel installing the shielding spent less time in high radiation fields.

3. During the first refueling, reactor structurals such as control rod shrouds and instrumentation support assemblies were removed from the reactor vessel under water in order to minimize personnel exposure; these components were handled dry during the second refueling. As it turned out, the radiation exposure was less during the dry handling operations because the rigging of the components required less time in radiation fields on the reactor vessel head.

4. Careful personnel control was maintained throughout the refueling, thus assuring that radioactive contamination was confined to the reactor pit and associated storage areas and that no personnel received greater than their permissible radiation dose throughout the refueling.

D. Reactor Components and Refueling Equipment

Improvements were made in the tools and equipment used for refueling as a result of lessons learned during the first refueling. For example, the omega seal cutting and welding machines were modified to avoid problems with galling and poor accessibility encountered during the first refueling. The omega seal weld cutting operations were performed in one-third the time required during the first refueling. Instrumentation on the reactor vessel head was more accessible for removal because of the large effort associated with modifying the instrumentation design during the first refueling.

Fuel exchange operations were performed under water in about four days with no difficulty. No personnel received a radiation exposure above background during these operations. The canal water pits afforded a convenient facility for examining and storing radioactive core components during the refueling. As during the first refueling the advantages of the 120-foot long fuel handling canal were again demonstrated.

All of the reactor components examined during and after refueling confirmed the conclusion that the first Shippingport core was capable of operating safely for several more seed lifetimes. In this regard, the original design lifetime specification for the reactor components was two seed lifetimes. Of particular significance is the performance of the blanket fuel which has now been irradiated to 14,700 MWD/ton peak and 4500 MWD/ton average. Laboratory examination of blanket fuel specimens removed during the refueling showed the fuel to be acceptable for continued operation. The examination revealed low hydrogen pickup of the Zircaloy cladding and no significant dimensional or structural changes, as compared to unirradiated fuel.

Two high power region blanket fuel assemblies gave indications of above normal activity during Core 1 operation, as determined by the Shippingport Failed Element Detection and Location (FEDAL) System. These assemblies were removed during refueling (after four calendar years of operation) and examination of the 840 fuel rods in each assembly revealed two small rod defects, one in each assembly. Detailed examination of one of the defects has been completed. The defect is about 0.003 inch in size and was probably in the reactor since initial startup in 1957; no significant dimensional changes and no visible loss of fuel occurred. It should be noted that the two blanket assemblies removed did not cause any detectable increase in gross primary coolant radioactivity during operation of Core 1. They were detected by means of the experimental FEDAL system and were removed primarily to confirm the accuracy and sensitivity of the FEDAL system. The detailed examination attests to the fact that the assemblies could have remained in the reactor without causing operational difficulties.

E. Lessons Learned

The principal lessons to be derived from the first two Shippingport refuelings are the importance of thorough training of personnel, checkout of equipment, and preparation of detailed plans prior to and during refueling. The following is a list of other significant lessons which were learned during the refuelings:

1. Preparations and Planning

a. An important element of successful refueling operations is an independent "technical planning group" responsible both for careful monitoring of refueling progress and for taking steps to improve the sequence of refueling operations where circumstances warrant.

b. If possible, power plant maintenance should be completed prior to completion of the last seal weld on the reactor vessel head in order to permit expeditious return of the plant to power operation. In any case an accurate schedule of parallel plant maintenance jobs should be prepared prior to the refueling shutdown. Maintenance progress should be continuously monitored and planning revised as necessary to ensure that plant maintenance is consistent with refueling progress.

c. Planning for the plant hydrostatic test, checkout of plant instruments and systems, and returning the plant to power operation should be on a level consistent with refueling planning. For example, supervisory personnel should be thoroughly trained in the sequence and details of operations necessary to return the plant to power operation after completion of the last seal weld; a technical planning group similar to the refueling planning group should be established for this phase of operations; the required number of personnel in key trades, such as instrument technicians, should be made available for the plant checkouts required. A single individual should be placed in charge of coordinating the return to power operation.

d. Although they did not cause significant delays in refueling, there were several difficulties during refueling which could have been avoided had more attention been given to following the detailed procedures. For example, during fuel port omega seal cutting the cutting machine was improperly mounted, resulting in an eccentric cut and requiring special corrective action during rewelding. Intensive training of working personnel should be emphasized to reduce these occurrences in the future.

2. Radioactivity Control

a. Periodic exercising of control rod drive mechanisms during plant operation is effective in reducing head area radiation levels during refueling.

b. Temporary lead shielding used for reactor maintenance operations is considerably more effective if it is designed in small segments so that it is easy to handle and presents a minimum of interference with the operations in progress.

c. Irradiated structural components can be handled safely and conveniently without a water shield, provided that adequate controls are instituted to prevent personnel from approaching the components during transit to underwater storage. Such handling significantly improves the efficiency of refueling operations and may reduce the total radiation exposure of personnel.

3. Reactor Components and Refueling Equipment

a. Careful design of refueling equipment to insure adequate access and proper fit of parts is an essential element of successful refueling operations. Improvements in instrumentation access and mechanism cutting machine performance demonstrated this important lesson.

b. Underwater refueling using a water-filled transfer and storage canal is an effective and safe means for replacing highly irradiated fuel in a power reactor.

c. The Shippingport Core 1 reactor components, including the natural uranium oxide blanket fuel, remain structurally adequate for continued operation of Core 1 with a third seed.

II. REACTOR COMPONENTS, REFUELING EQUIPMENT, AND REFUELING FACILITIES

For the benefit of those who are not familiar with the design of the PWR reactor, this section of the report describes the core and the parts of it which are affected by refueling, the replacement components that were installed during this refueling, the refueling facilities provided at Shippingport, and the special refueling tools used.*

A. Reactor Components

1. Description

The PWR core is contained in the reactor vessel (Figure 2) which is approximately 33 feet high with an inner diameter of 9 feet and a nominal wall thickness of 8-7/8 inches. The dome-shaped closure head containing the control rod drive mechanisms, fuel, and instrumentation ports completes the pressure vessel. Matching bolting flanges surround the top of the vessel shell and the lower part of the closure head. These bolting flanges are welded together by an omega seal which is not cut during seed replacement. The 32 control rod drive mechanism motor tubes and the instrumentation closure pieces are seal welded to their respective fuel-port housings; these welds must be cut to gain access to the reactor core.

The reactor vessel is in a pressuretight, spherical container with a dome which covers the part of the closure head that extends above the reactor chamber. To refuel PWR, the container dome is removed to expose the top of the reactor vessel head and the mechanisms, instrument leads, and associated wiring which must be removed before the seed and blanket fuel can be moved. After the container dome is removed and before the reactor pit can be flooded the closure head seal ring, a ring of 1-3/4 inch thick steel plate welded to the outer surface of the head at the top of the reactor container, is sealed by bolting a mating gasketed seal ring (refueling seal) to it to span the area between the closure head and the reactor container (Figure 3). This forms a water seal which prevents the entrance of pit water into the container.

The reactor has a seed and blanket core arrangement (Figure 4). The core is the active portion of the reactor and contains the fuel. Its main components are the core cage, which houses and supports all the other components; the fuel assemblies; the control rods; the sensing elements of the flow rate and temperature instrumentation; and, the connections for the Failed Element Detection and Location system. The fuel assemblies are supported within the core cage by a bottom support plate and a top grid, and are so attached that they may be moved individually.

* Refer to "The Shippingport Pressurized Water Reactor," Chaps. 4 and 6, Addison-Wesley Publishing Co., Inc., Reading, Mass., 1958.

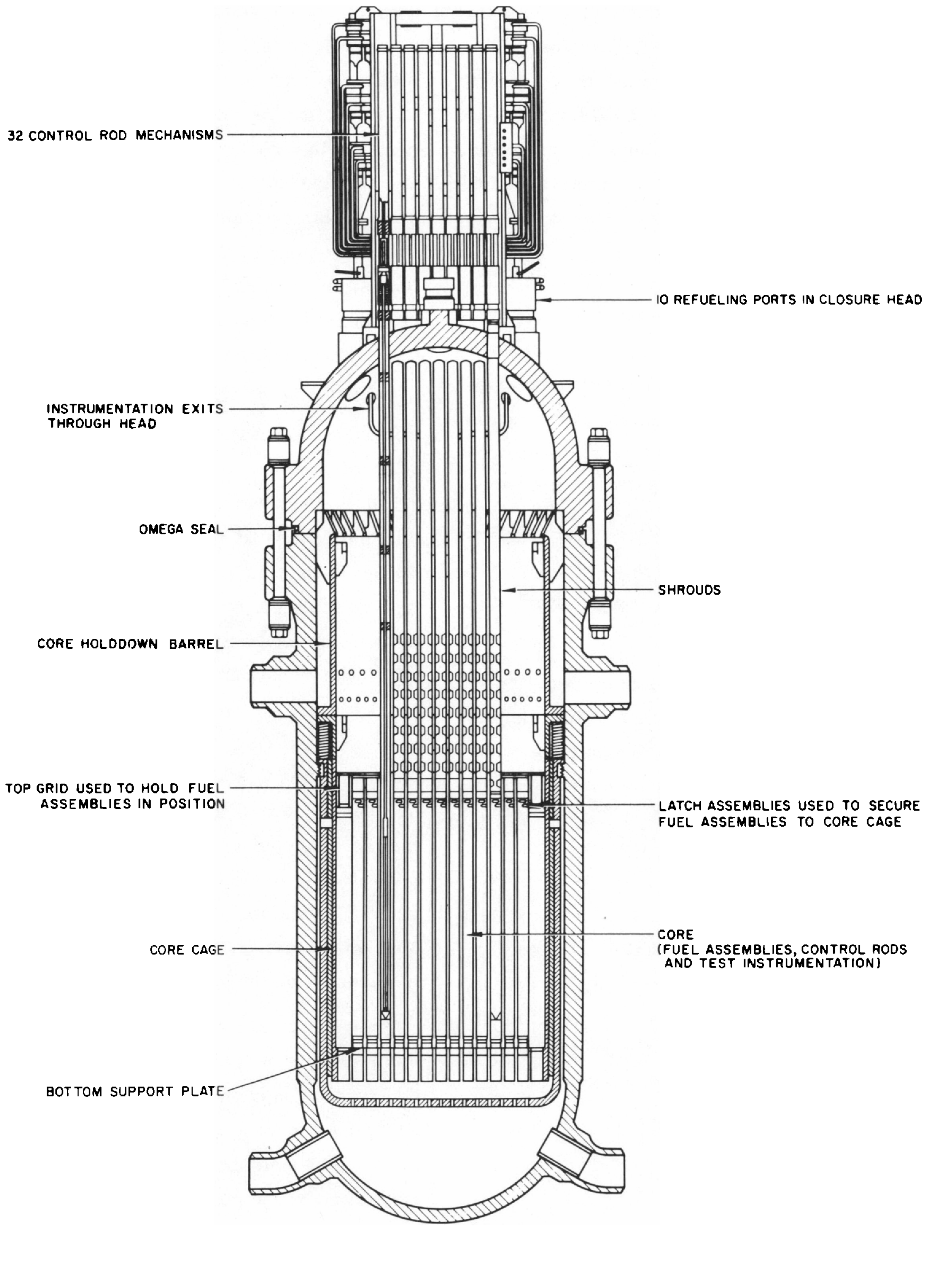


Figure 2. Reactor Vessel and Core 1 Cross Section.

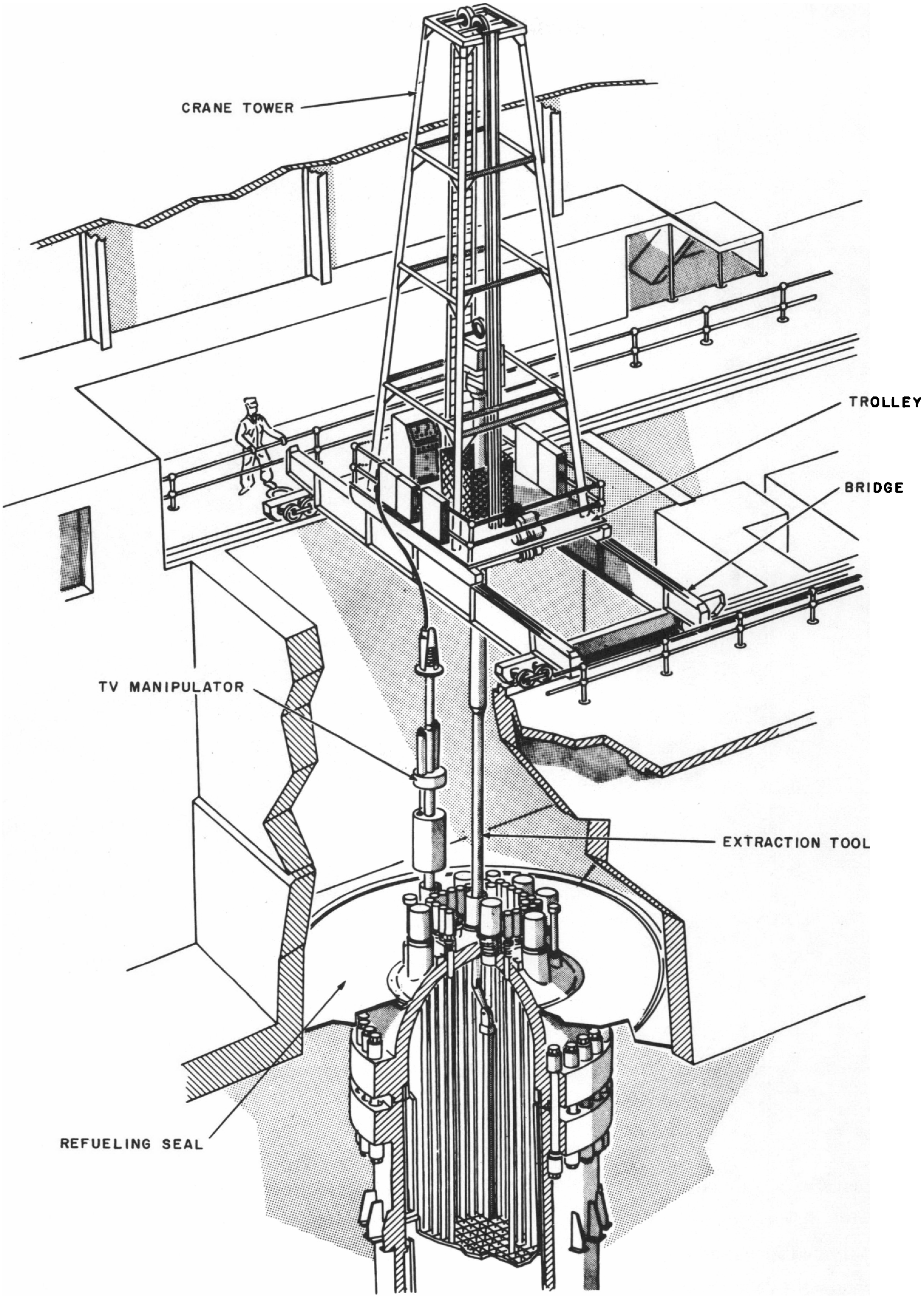


Figure 3. Reactor Pit with Refueling Seal Installed.

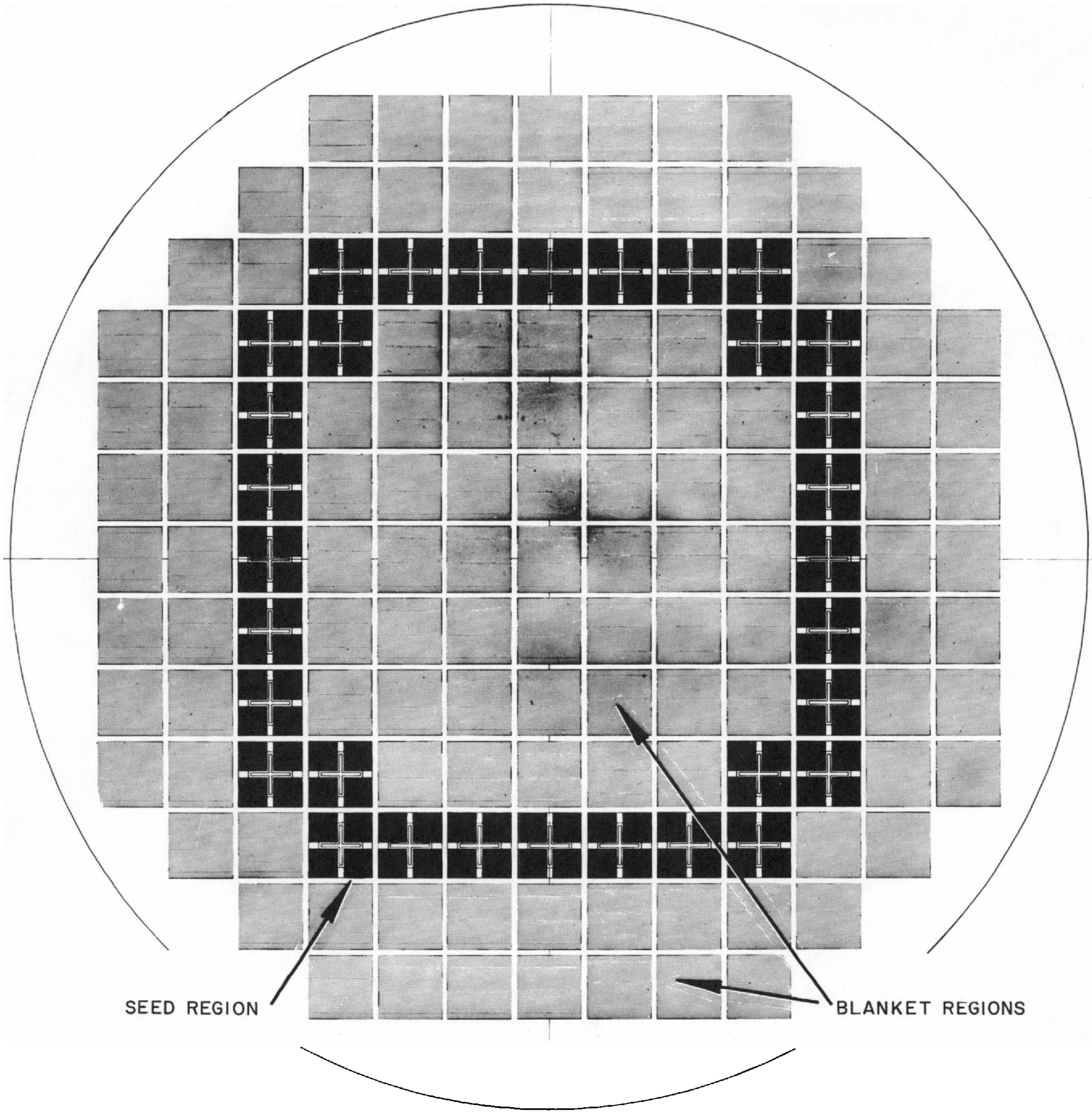


Figure 4. PWR Seed and Blanket Core Arrangement.

The seed region contains 32 highly enriched elements (clusters) arranged in a hollow square. The Core 1 seed elements are 5-1/2 inches square and contain four seed subassemblies welded together with half-inch spacers which form the cross-shaped channel through which the control rod travels (Figure 5). Only the seed requires control rods, since it is only in these assemblies that the concentration of U^{235} is high enough to exceed the fission rate necessary to sustain the chain reaction. When the core is refueled only the seed assemblies are replaced.

There are 113 blanket assemblies (Figure 6), 45 inside the seed square and the remainder surrounding the seed. The blanket assemblies are

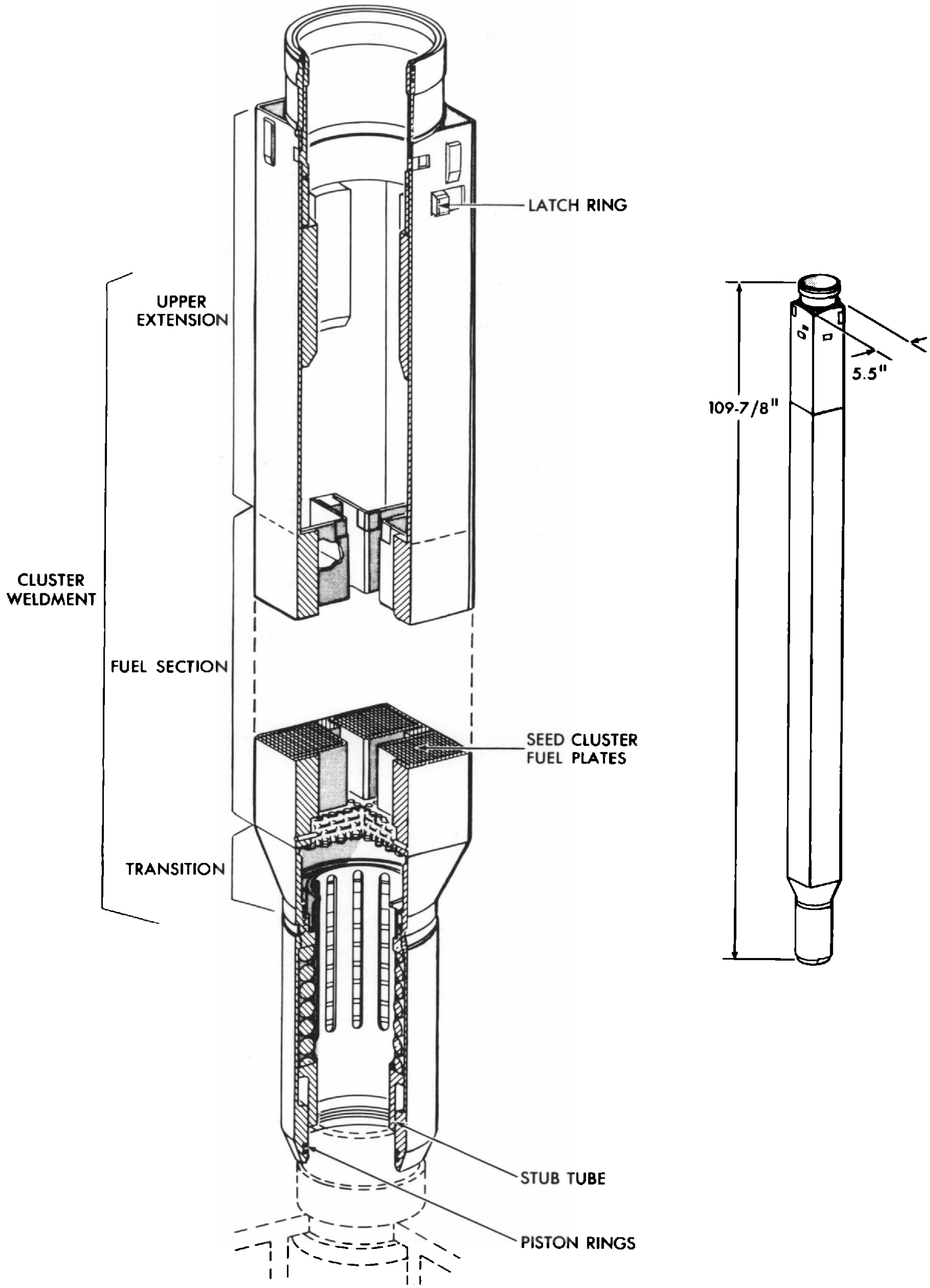


Figure 5. Seed Fuel Assembly.

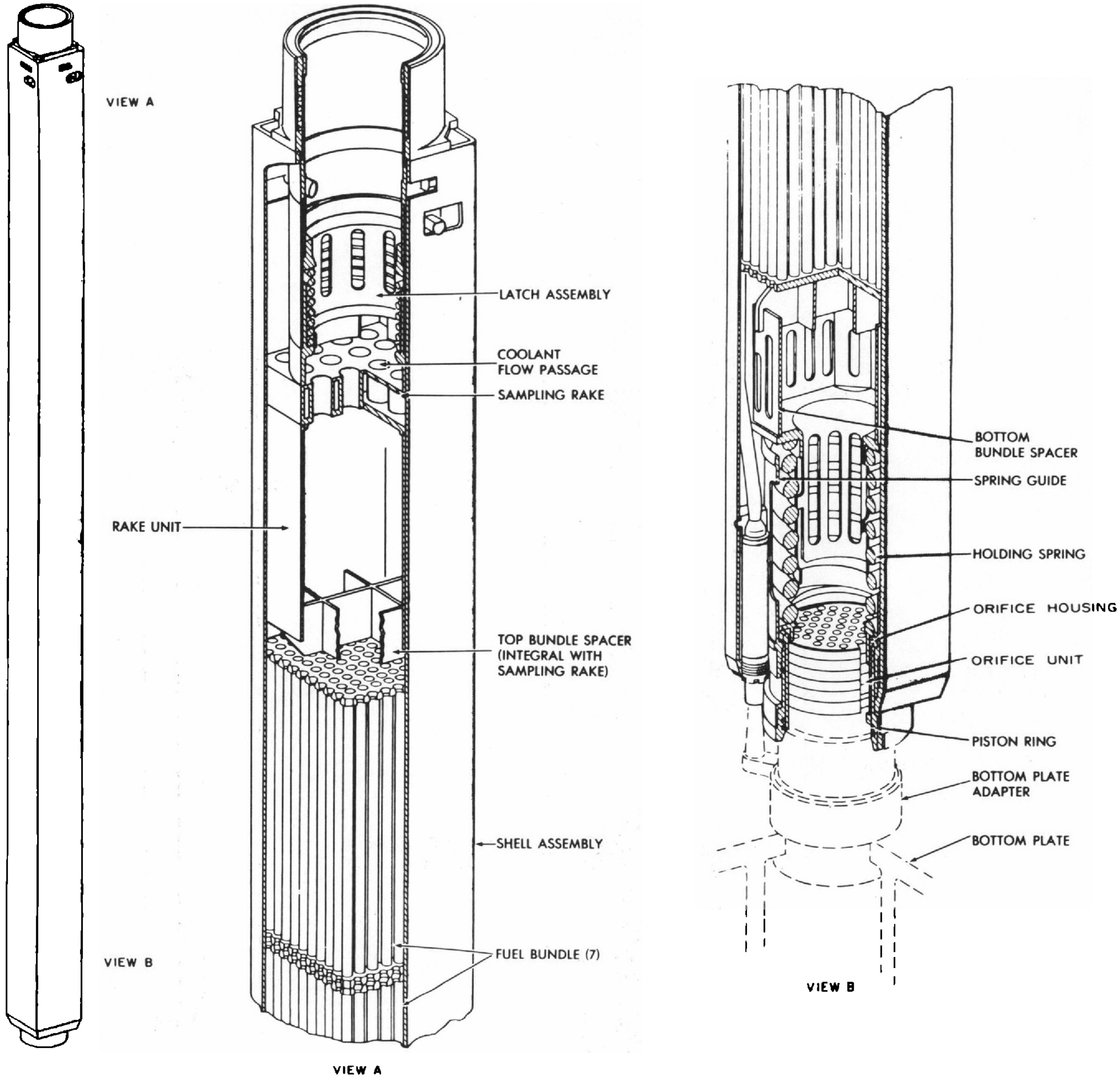


Figure 6. Core 1 Blanket Assembly.

divided into four regions according to the radial thermal neutron flux variation; thus, different flow requirements exist for each region. Restrictions to the flow, consisting of multihole orifices, are placed in the blanket assemblies to obtain the desired flow distribution among regions. The blanket fuel is natural uranium dioxide enclosed in rods assembled parallel to each other in groups called bundles. To make up a blanket assembly, seven bundles are stacked and enclosed in an open-ended Zircaloy shell.

Each of the 32 seed assemblies has a control rod, a control rod shaft and shroud assembly, and a rod drive system. Each control rod consists of two parts—the cruciform hafnium blade and an adapter at the upper end through which the rod is connected to the lead screw of the drive mechanism (Figure 7). The control rod shaft and shroud assembly consists of the shroud, a flange block and its accessories, support bearings, the drive shaft, and the tie rod. The shroud is about 17 feet long and 4 inches in diameter. The lower 90 inches of

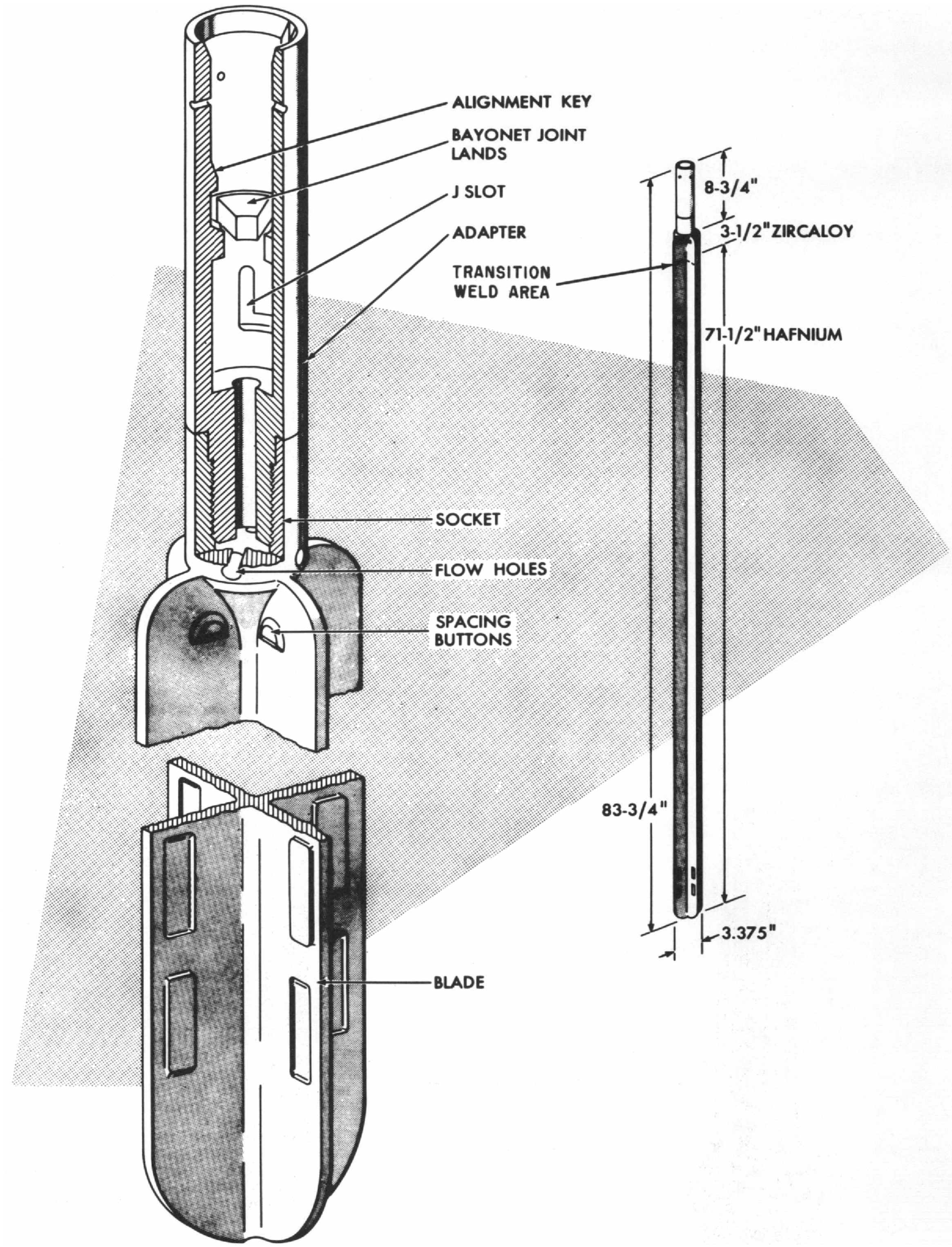


Figure 7. Control Rod and Adapter.

the shroud is fitted with internal flow baffles arranged to form a cross-shaped passage for the control rod. When the control rod is fully withdrawn from the seed assembly it is housed entirely within the lower tube. The upper part of the shroud tube houses the control rod drive (scram) shaft. The drive shaft is a hollow stainless steel (type 17-4 PH) tube through which the motion of the rod drive mechanism is transmitted from the mechanism lead screw to the control rod.

The control rod drive mechanism (Figure 8) is a rotating nut and translating screw device. The rotor of the motor drives transverse shafts on which are mounted two pairs of arms that pivot in a seesaw action. Ball-bearing mounted roller nuts are attached to the lower ends of these arms to drive the lead screw. The removal of any seed cluster requires disassembly of its associated control rod drive mechanism and accessories.

PWR is more highly instrumented (see Figure 9) than an ordinary operational plant to satisfy the experimental requirements of the reactor plant.

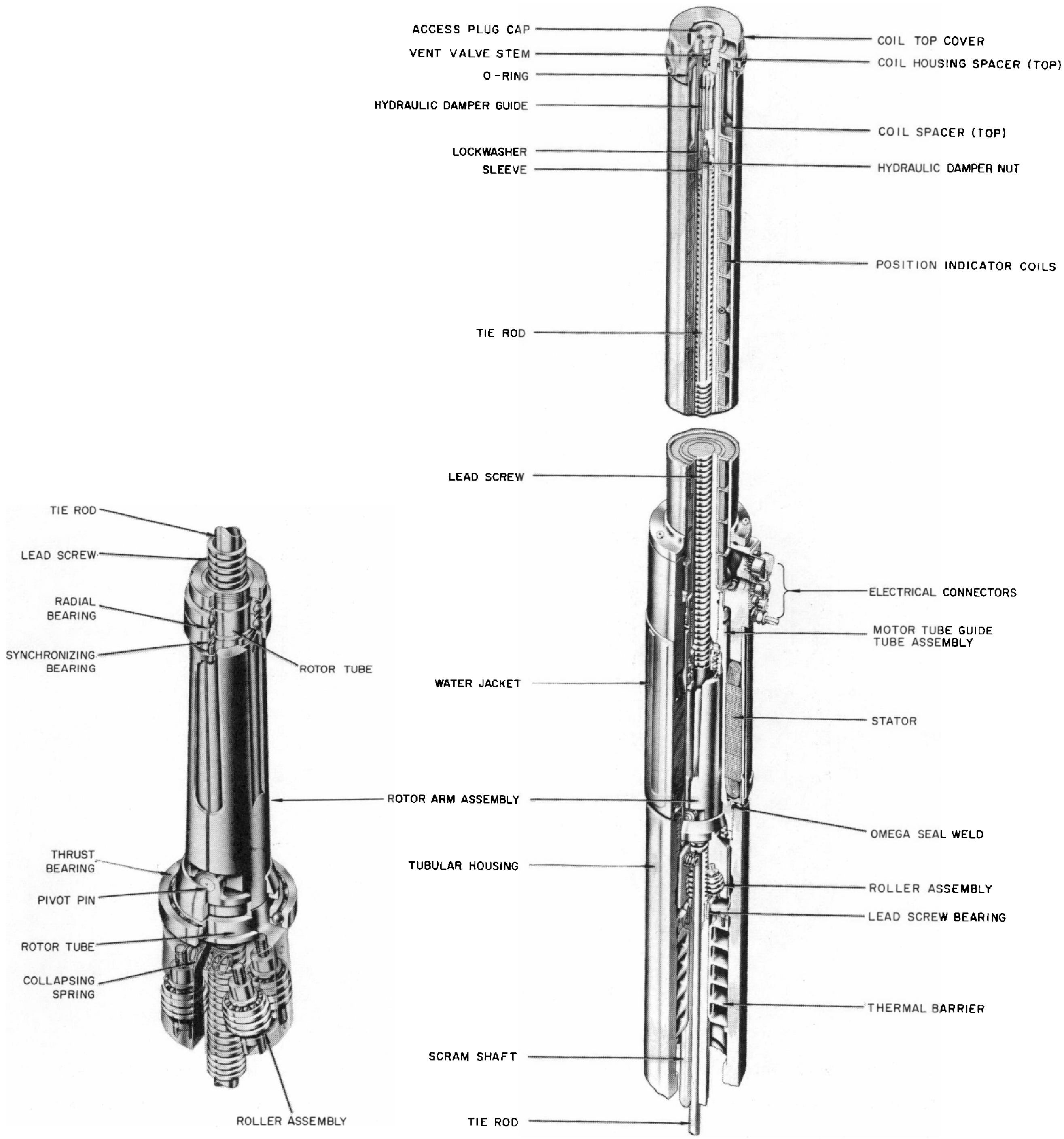
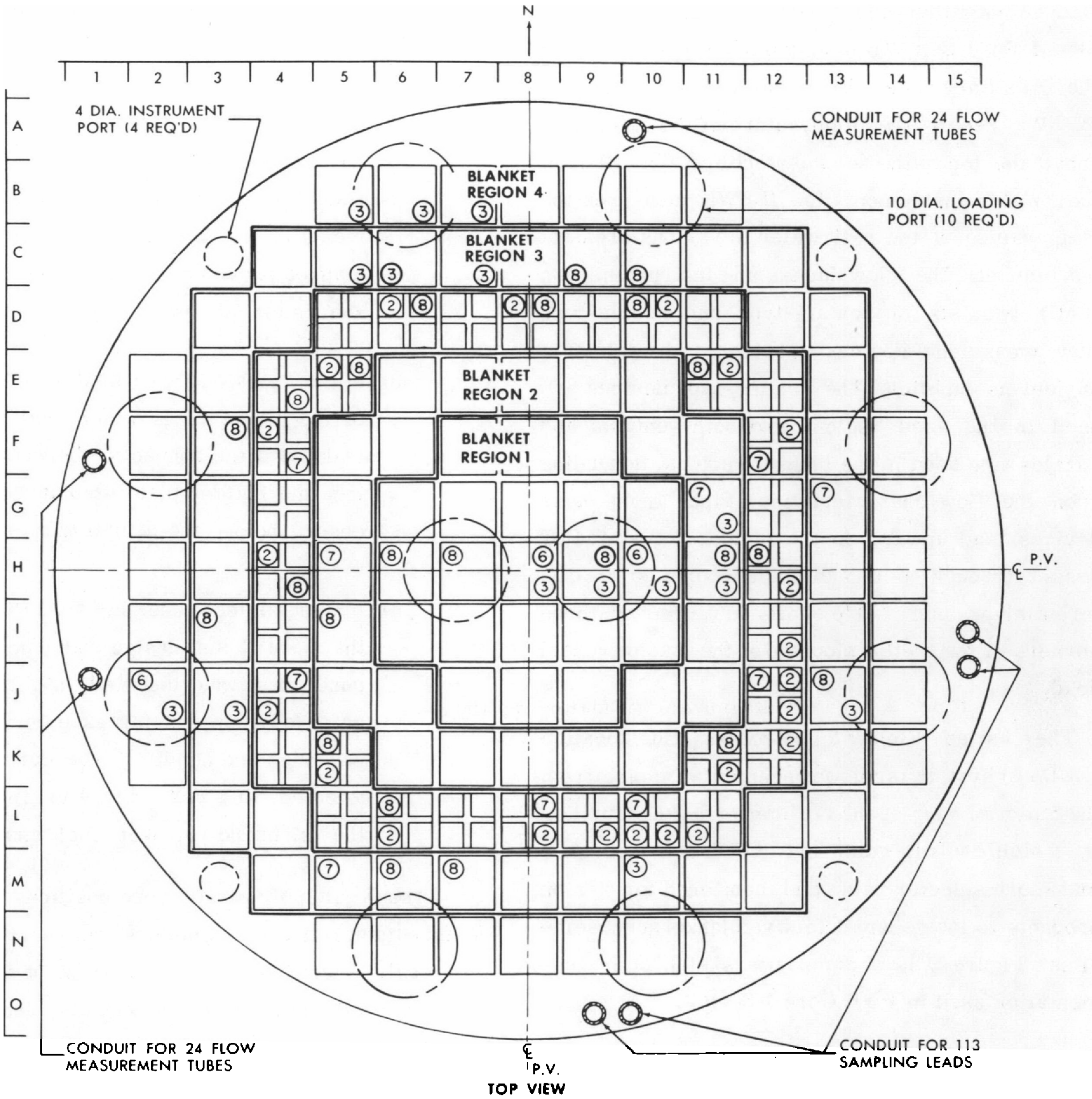


Figure 8. Control Rod Drive Mechanism.



LEGEND

(2) SEED EXIT WATER THERMOCOUPLE

(3) BLANKET ASSEMBLIES INSTRUMENTED FOR EXIT WATER TEMPERATURE

(6) FLOW NOZZLE WITH CALIBRATED D/P CELL

(7) VENTURI WITH CALIBRATED D/P CELL

(8) FLOW NOZZLES AND VENTURI WITH NONCALIBRATED D/P CELLS

NOTE:

ALL BLANKETS HAVE SAMPLING RAKES FOR FUEL ELEMENT FAILURE DETECTION SYSTEM.

Figure 9. Core 1 Cross Section Showing Instrumentation in Seed 3

During the Seed 3 refueling the installation of a Data Acquisition and Logging System was completed. Seed Exit Water Instrumentation (SEWI) and the Auxiliary Seed Exit Water Instrumentation (ASEWI) record the temperature of the exit water above the top of the seed assemblies. The Blanket Exit Water Instrumentation (BEWI) measures the temperature of the exit water in various areas of the blanket. The Flow Measuring Instrumentation (FMI) consists of venturi-type and nozzle-type flow measuring devices located at the inlets of 36 fuel assemblies. The venturi-type devices are used in the seed region, and both venturis and nozzles are used in the blanket regions, depending upon the flow characteristics. Pipes from these devices lead upward on the inside of the reactor vessel through 4-inch diameter ports to 36 differential-pressure (d/p) cells arranged on three corners of the trellis mounted on the reactor vessel head.

The Failed Element Detection and Location (FEDAL) system is an arrangement of piping from the outlet of each of the 113 blanket fuel assemblies by which coolant water may be drawn through a multiport selector valve and monitored for fission products to locate any defective blanket subassemblies. Figure 9 is a composite sketch of instrumentation used in PWR Core 1 Seed 3.

2. Changes in Reactor Requirements

The Seed 2 - Seed 3 refueling requirements and operations were similar to those for the Seed 1 - Seed 2 exchange which is described in Reference 1. The following section describes only those reactor requirements which differ from the first refueling.

a. Core Fuel Assemblies

1) Fuel plate thermocouples were not provided in Seed 3, since the Seed 2 thermocouples had furnished sufficient information to establish the accuracy of nuclear calculations on axial power distribution.

2) No artificial neutron sources were provided for Seed 3. Experience with Seed 2 indicated that the source-free quadrants were not irregular, and the approach to criticality could have been conducted on the sourceless channels alone. The reduction in artificial source requirements is possible because the irradiated blanket provides an increasing neutron source for startup.

3) Minor revisions were made in the design of the piston rings which seal against leakage at the connection between cluster and bottom plate stub tubes. This change helped the refueling operations.

4) A special test assembly (SOAP) to test the Core 2 fuel design under operating conditions was inserted into the H-8 core location. Insofar as external appearance and handling procedures were affected, this assembly was similar to the standard blanket fuel assembly.

Preselection of specific core positions for the as-manufactured Seed 3 clusters was required, as previously specified for the insertion of Seed 2. Measurements made following manufacture of the Seed 3 clusters indicated some variations among the clusters in the water-to-metal volume ratio and fuel boron content. Since these quantities affect the neutron flux distribution in the core, it was decided to select specific positions for each as-manufactured seed cluster in order to minimize any possible static power tilt. It was also decided to replace the blanket assemblies containing possible defects in the fuel elements, as determined by the FEDAL System (positions J-5 and K-8), and install the SOAP assembly in position H-8.

Several neutron diffusion theory calculations representing a horizontal section of the full core were performed by the Bettis Laboratory to evaluate the effects of asymmetries in core configuration necessitated by the above considerations. In these calculations, variations in the individual "as-built" seed compositions were represented, and the various seed clusters were placed in a configuration selected to minimize power asymmetries. Selection of the specific configuration also involved thermal as well as nuclear characteristics of the individual seed clusters because the more reactive clusters, which tend to be above nominal in power generation rate, are not necessarily those with highest thermal capability. However, in Seed 3 those clusters which are more reactive are, with few exceptions, also those with the best thermal capability. Even though the final seed configuration represented a compromise of nuclear and thermal considerations, the flexibility available in selective assembly was sufficient to result in a calculated power asymmetry of less than two percent.

Minimization of the static power tilt in the core by selective arrangement of the Seed 3 clusters also involved consideration of the influence of asymmetries in blanket assembly configuration. Calculations indicated that a power asymmetry of around 8 percent would occur in the seed if unirradiated blanket assemblies were inserted in core positions J-5 and K-8. This would occur because the unirradiated natural uranium blanket fuel, void of any plutonium isotopes, absorbs thermal neutrons at a lesser rate than the irradiated fuel with a resultant increase in the thermal neutron flux in the vicinity. This peturbation would extend into the adjacent seed cluster locations, causing a power asymmetry in the seed.

Results of calculations indicate that the magnitude of the perturbation effect is dependent on the amount of irradiation of the blanket cluster inserted. Use of a blanket unit whose irradiation history is less than that of the assembly being replaced results in a change in the adjacent seed cluster power output proportional to the change in thermal neutron absorption cross section. Consequently, the blanket configuration was arranged such that the assemblies replacing the defective J-5 and K-8 assemblies were those most nearly similar in irradiation history.

Based on the above considerations, the following blanket arrangement was used with PWR-1 Seed 3 (Figure 10). Blanket assemblies J-5 and K-8 were removed from the reactor completely. Assembly E-9 was moved to J-5, and assembly K-9 was moved to K-8. Assemblies J-2 and F-14 were moved to locations E-9 and K-9, respectively. New assemblies equipped with orifices suitable for region 4 were placed in outermost blanket locations J-2 and F-14 to minimize any seed power tilt which these might cause in the core. The relocation of assemblies from J-2 and F-14 required that new coolant flow adjusting orifice shims be inserted in those two units to adapt to the different flow requirements of the new blanket region.

b. Control Rods. A new set of control rods was provided for use in Seed 3. The original complement of 32 rods, with the exception of one rod removed during the first refueling for metallurgical examination, had been used for two seeds. It was decided to replace 30 of the rods during Seed 2 replacement and to continue use of two of the most highly depleted rods for Seed 3 to evaluate further the effect of irradiation on the hafnium rods. The corrosion behavior of the hafnium-to-Zircaloy welds in the original rods was the reason for control rod replacement. Selection of the two rods to remain for Seed 3 lifetime was based upon comparative visual examination of the welds in the rods from the inset corner locations, since these had had the highest hafnium depletion. Of the four rods in these locations, only that in K-11 had not seen two seed lives in an inset corner location. Therefore, the choice was to select two rods from the three in positions E-11, K-5, and E-5. The rod from E-5 was reused in that same position, and the one from E-11 was transferred to K-11. Section VI discusses the control rod examinations conducted during and after the second refueling.

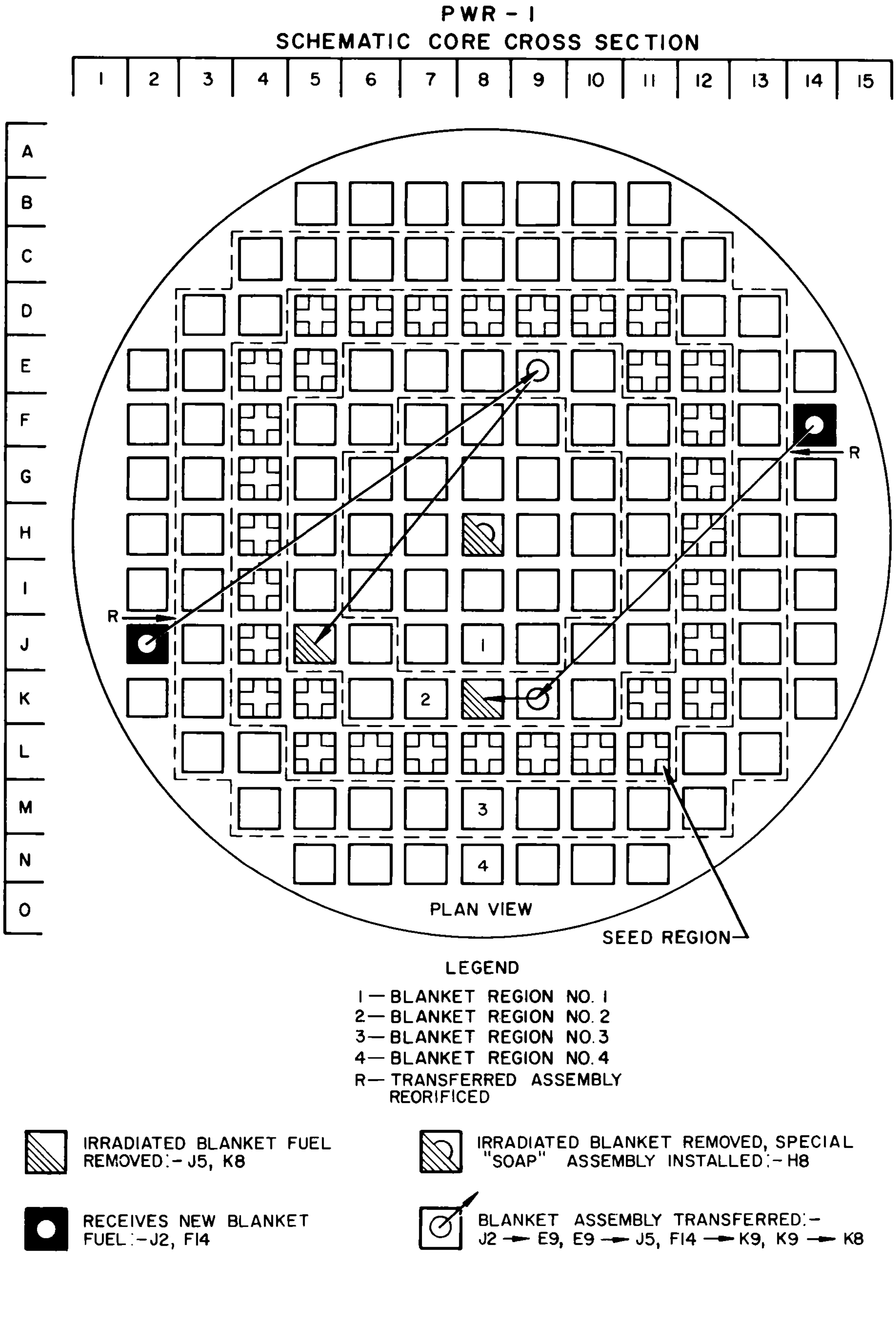


Figure 10. Blanket Fuel Changes during the Second Refueling.

c. Control Rod Drive Mechanisms. Requirements for the removal, inspection, and reinstallation of the rod drive mechanisms differed in several respects from those for the first refueling. At that time, an extensive program of inspection, overhaul, and testing of these devices was undertaken. This was done previously to provide motor tubes which were more corrosion resistant than the original motor tubes, and included magnetic-particle inspection of the mechanism housings and the reconditioning of motor rotors at the contractor's plant. During this refueling, these inspections, replacements, and overhauls were not required, and a simplified inspection was used to confirm that the rod drive mechanisms would be reusable with Seed 3. The previous program of overhauling motor parts had included the measurement of radial and thrust bearings to determine rates of wear as well as bearing condition. For the second refueling it was required only that comparative evaluations of wear be made between the mechanism bearings and standards of known bearing clearances. The results of this work are discussed in Section VI.

Only eight lead screw - scram shaft assemblies had been inspected during the first refueling. The additional exposure to reactor environmental conditions dictated that all 32 lead screws be inspected by a magnifying viewer while Seed 2 - Seed 3 fuel exchange was in progress. Two lead screw - scram shaft assemblies were scheduled to be removed for destructive examination and replaced at positions K-5 and J-4. As a result of a questionable surface indication noted during the above inspections, it was also decided to remove an assembly from location E-12 for further examination and replace it with a spare. The assemblies were removed for laboratory examination and evaluation of the changes in mechanical properties of the 17-4 PH stainless steel as a result of service with Seed 2 and for comparison with similar analyses made of assemblies removed from positions K-11 and L-6 during the first refueling. Section VI discusses these examinations conducted during and after the second refueling.

d. Core Instrumentation. As noted earlier, the reduction in the amount of core instrumentation work required at the second refueling involved, primarily, the number of thermocouples to be seal welded. About 20 days were saved because the installation of thermocouple housing extensions in 22 locations during Seed 2 refueling did not have to be repeated. In fact, the housings were more accessible in this refueling and, thus, saved additional time. Seed Metal (SMI) thermocouples were not to be replaced, but the 1/2 inch pressure boundary penetrations had to be sealed by blind plugs. Also, the number of Blanket Exit Water Instrumentation (BEWI) assemblies to be inserted through fuel port locations was reduced from 6 to 4. All instrumentation plug welding was specified to utilize the tungsten-inert-gas (TIG) manual welding process, which was known to result in better quality welds in less time than the manual metal arc process previously used.

1) Seed-Metal and Seed-Exit-Water Instrumentation (SMI and SEWI). Thirty SMI thermocouples had been provided in Seed 2 fuel assemblies at six locations (F-4, D-8, E-11, J-12, L-8, and K-5) to measure fuel plate and inlet water temperature. The information was used to confirm calculated axial power distribution. No requirement for measurement of this parameter was specified for Seed 3, inasmuch as the information obtained during Seed 2 operation is applicable, with minor correction, to Seed 3. Thermocouple access holes in SMI upper tube extensions were, therefore, sealed with 30 blind plugs.

Although 15 of the 16 SEWI thermocouple assemblies were operating prior to refueling, it was decided to replace all thermocouples with 16 new assemblies to insure continued reliability of the instrumentation systems. During the installation two assemblies were damaged and were replaced with spares.

The relative locations of the six SMI and sixteen SEWI lead wire penetrations and terminal box assemblies are diagrammed in Figure 11.

2) Blanket Exit Water and Auxiliary Seed Exit Water Instrumentation (BEWI and ASEWI). BEWI assemblies used in Seed 2 were of two types as a result of modifications made at the first refueling: an assembly containing individually replaceable thermocouples inserted at fuel port location H-9, and assemblies whose thermocouples were not individually replaceable at fuel ports B-10, F-14, H-7, J-2, and N-6. The two ASEWI assemblies at locations N-10 (Figure 12) and J-14 and the BEWI at H-9 were specified to be reinstalled during the replacement of Seed 2 with Seed 3. Of the other five BEWI assemblies, only those at H-7 and J-2 were to be reinstalled and then only if tests just prior to

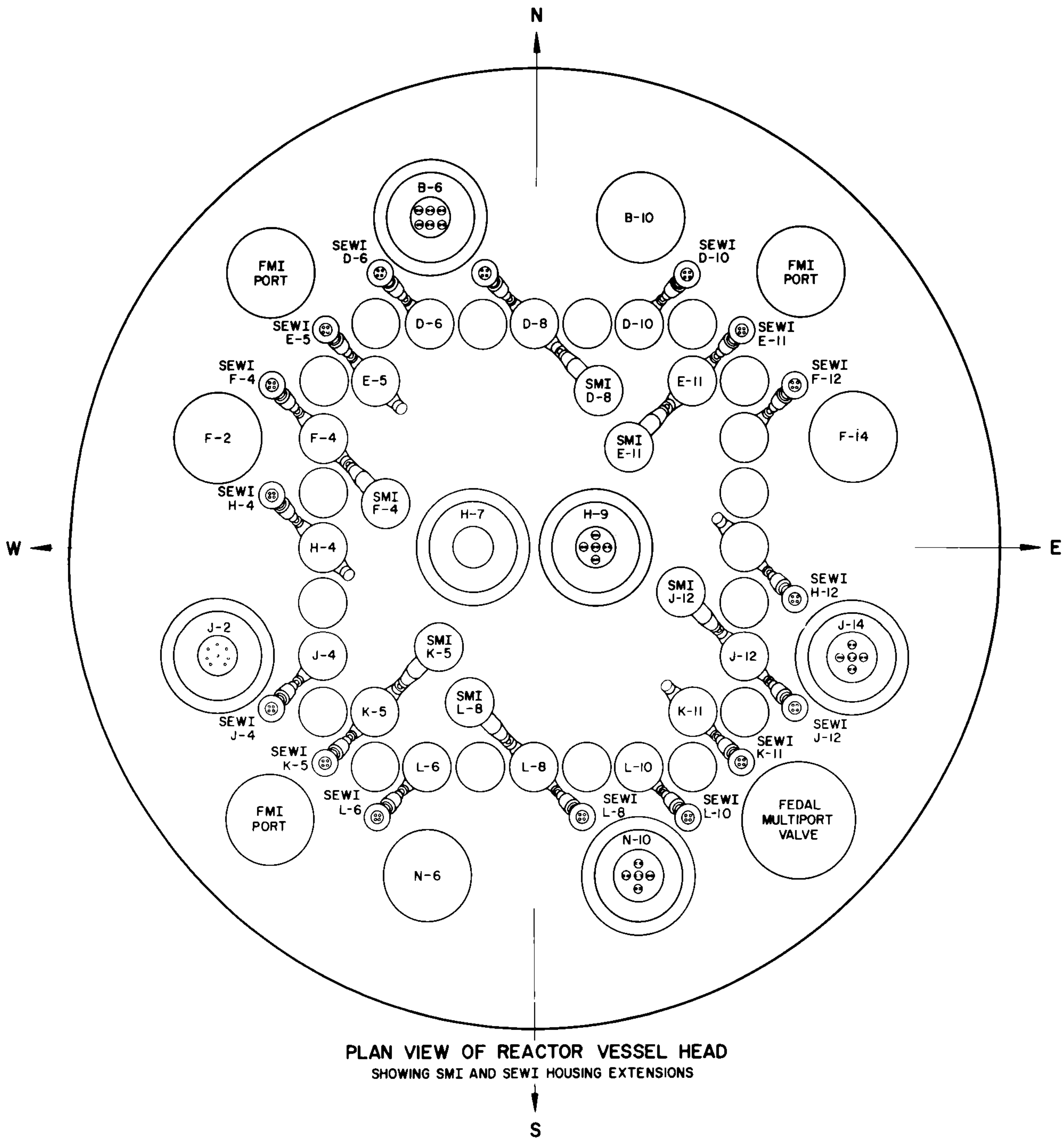


Figure 11. SMI and SEWI Location Diagram.

shutdown showed at least one thermocouple in each assembly was operable. A majority of thermocouples in the assemblies at B-10, F-14, and N-6 were known to have failed and could not be replaced. A new BEWI assembly with replaceable thermocouples was to be installed at fuel port location B-6 which previously had been a noninstrumented port. This new BEWI monitors the exit water temperatures of blanket assemblies in the northwest quadrant of the core locations B-5, B-6, B-7, C-5, and C-7. The arrangement of the BEWI and ASEWI for Seed 3 operations is shown in Figure 13.

Less refueling time was allocated to BEWI and ASEWI installation than in the first refueling. The irradiated assemblies to be reinstalled could

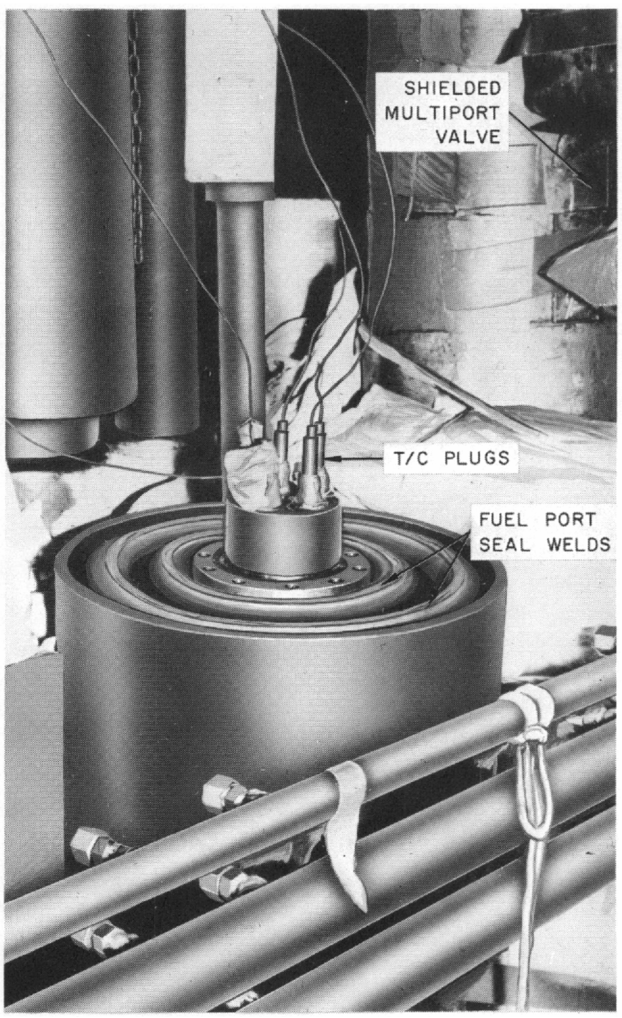


Figure 12. ASEWI Thermocouple Plugs in Location N-10 after the Welds Were Cut.

be handled in air because radiation measurements taken during inspection of these components indicated that the radiation levels would not be excessive.

3) Flow Measurement Instrumentation (FMI). FMI work required the recalibration of 13 preselected differential pressure (d/p) cells, most of which had been recalibrated during the first refueling. Six d/p cells for seed assembly locations and seven d/p cells for blanket assembly locations were recalibrated. The fuel locations served by these cells are diagrammed in Figure 14. In addition, the new experimental fuel assembly (SOAP) at H-8 required that a replacement d/p cell be provided.

Recalibration required over 14 days in the first refueling, although part of this time was spent in correcting malfunctions. In the second refueling six days were scheduled for recalibration, but it was actually accomplished in less than half that time. A significant time saving factor was the use of the newly installed Data Acquisition and Logging equipment which printed out the values for each calibration point, thereby eliminating the reading of recorder charts.

4) Failed Element Detection and Location (FEDAL) System. A FEDAL verification (gas bubble) test was run during the refueling to confirm the identification of blanket assemblies J-5 and K-8 which had been indicated as containing defects. This was to be done with the FEDAL multiport valve positioned to monitor blanket location J-5, with the FEDAL system pressurized with nitrogen. Bubbles would emanate from the J-5 assembly into the water in the reactor vessel if the position of the valve were correct. By viewing the vessel internals through the adjacent mechanism port J-4 the source of the bubbles could be seen and the FEDAL system indication confirmed. The test was repeated, viewing through location L-8 for blanket assembly K-8. The bubble test was performed and confirmed that the FEDAL position indication was correct.

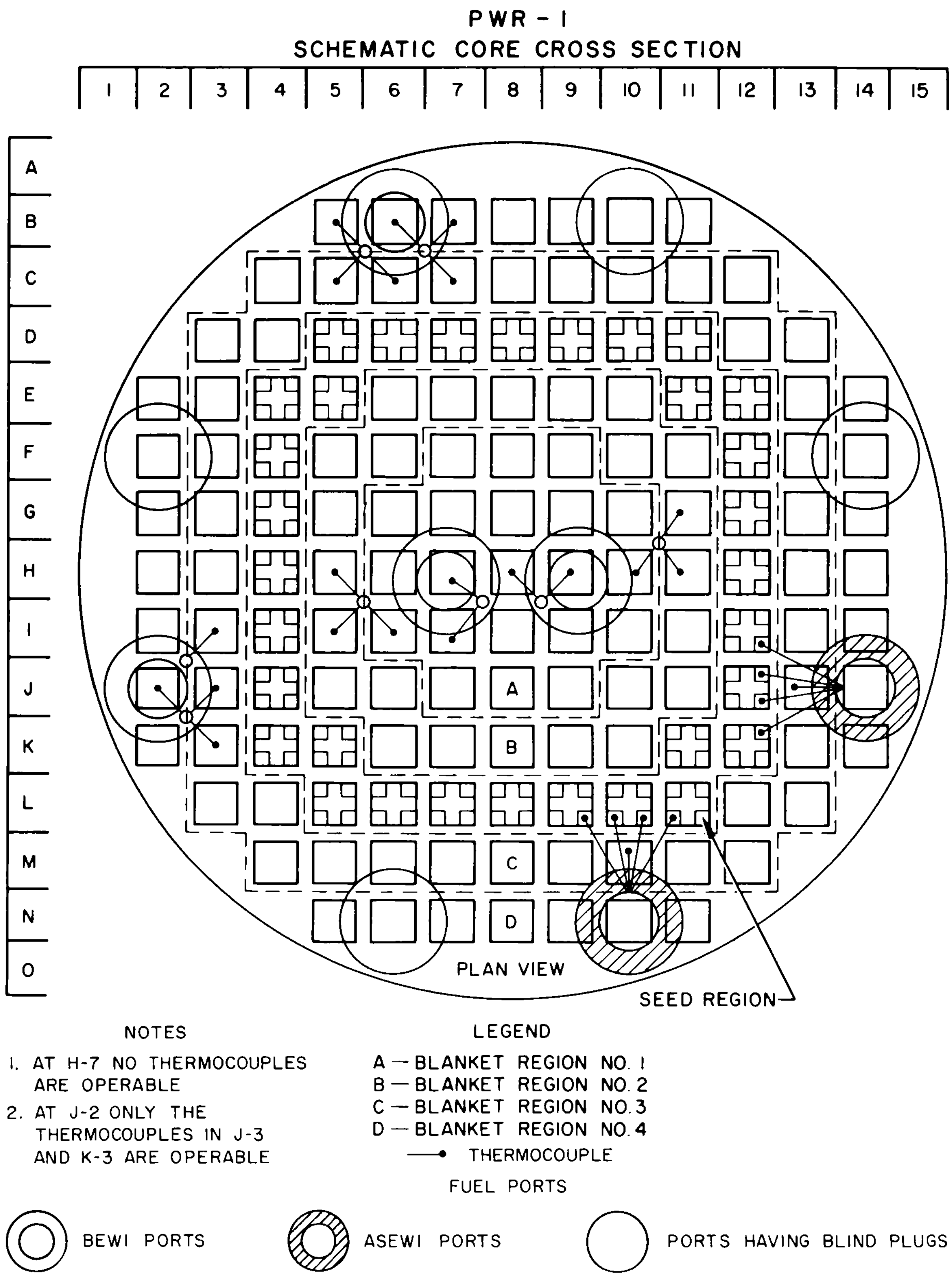


Figure 13. BEWI and ASEWI Location Diagram - Seed 3.

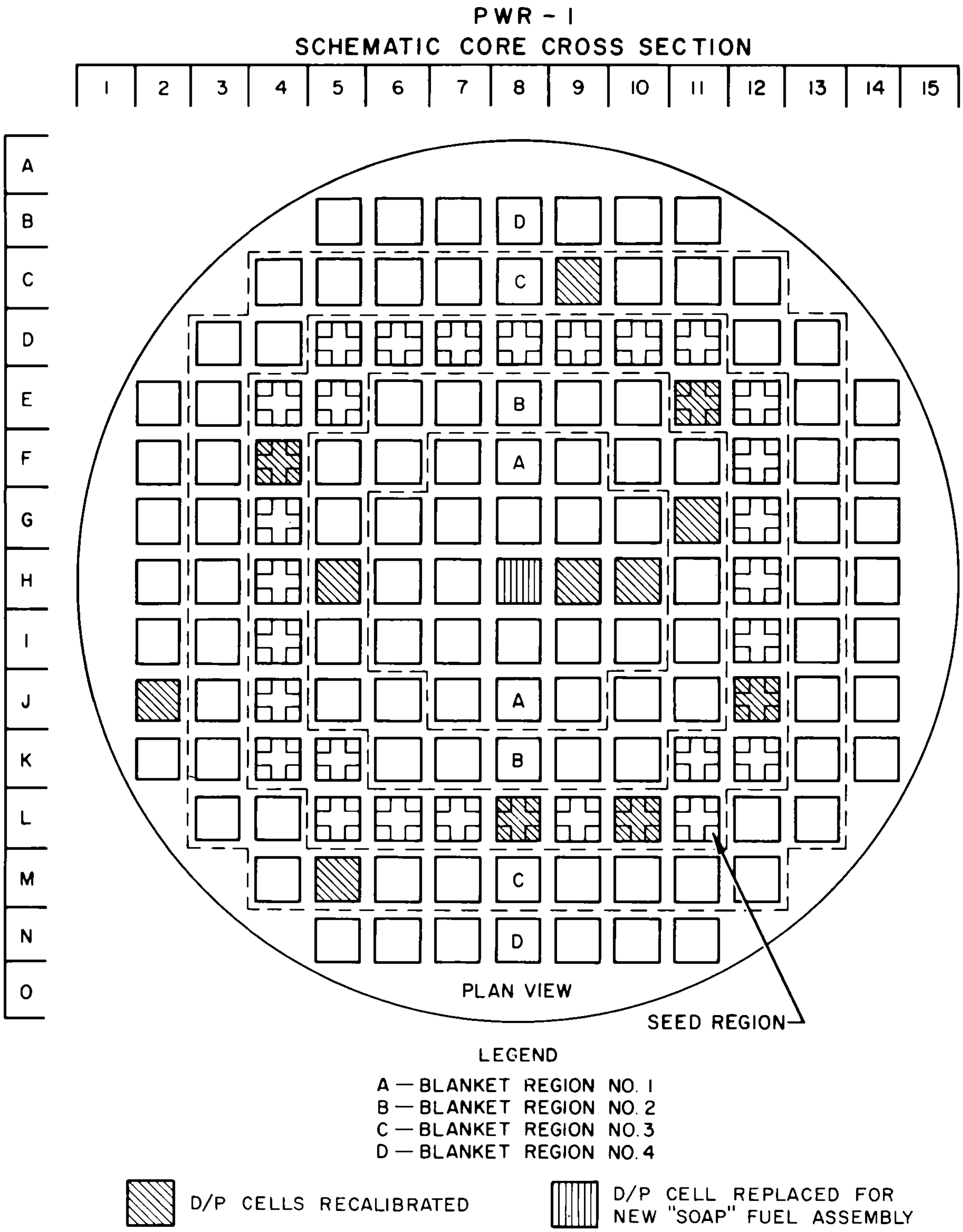


Figure 14. Locations Served by d/p Cells.

The FEDAL verification test required that the core top grid be viewed through mechanism ports J-4 and L-8. In order to see through these ports, it was necessary first to remove from each location the control rod shaft and shroud assemblies, which extend from the top of the fuel assembly up to the vessel head penetration. Calculations and the experience of the previous refueling indicated that the radioactive source strength of the shrouds was sufficiently low that radiation levels in all manned areas should not be excessive if shrouds were removed in air. Therefore, this method of removal was used for shrouds J-4 and L-8.

B. Refueling Equipment

Each of the special tools and pieces of equipment required for the refueling operation was evaluated in the light of its suitability, performance, and application in the first refueling; modifications were made accordingly. The more significant equipment and procedural modifications are described here. Refueling equipment is described in References 1 and 2.

1. Seal Weld Cutting Machines

The cutting machines are used for cutting the mechanism housing seal welds and the inner and outer diameter seal welds on the fuel ports, then for dressing the underside of the weld lips on mechanism housings and fuel port housings and fuel port closure pieces. Because the seal weld cutting operations were the most troublesome during the first refueling, design modifications were made to the two omega seal-weld cutting machines. The chief difficulties with the machines, when used during the first refueling for cutting mechanism motor tubes, were associated with:

a. Interference with the instrument trellis, the lead shielding, and with certain other motor tubes

b. Accessibility for setting up and operating the machine

c. The number of operations and proportion of total cycle time needed for installing and changing cutting tools

d. Galling between the motor tube and pilot tube of the cutting machine because of the very small diametral clearance between the motor tube and because the carbon steel pilot tube was not chrome plated

e. The inability of the parting tool to cut deep enough, in some instances, to separate the omega weld completely.

The cutting machine shown in Figure 15 is essentially a vertical lathe, except that the tool moves instead of the work. The design modifications were concentrated on the concentric tubes below the gearbox. To eliminate galling and alignment difficulties encountered earlier, a new system of bearings was designed for installation between the pilot tube and the mechanism motor guide tube. The pilot tube contains a nylon thrust bushing which bears on a tapered land and an upper nylon radial bushing which positions on the motor guide tube. Thus, the machine has been provided with positive alignment surfaces.

For the cutting operation, instead of three separate setups for grooving, slotting, and parting, only one initial setup and one tool change are required for all three cuts. Two V-groove cutters and a slotter are mounted in a single tool holder body as shown in Figure 16. Thus, the grooving and slotting are performed in one operation; the slotter is adjusted to cut 0.030 inch deeper than the groove cutters. When the slot is cut nearly through the weld the grooving tools are removed and the slotting tool bit is replaced by the parting tool. In actual practice the parting tool did not work satisfactorily, and the slotting tool was left in the cutter until the metal was broken through.

The original tool modification had provided a positive torque restraint and antichatter system of torque bars and an adjustable framework which bridged the trellis structure. However, when initially used for the Seed 2 - Seed 3 refueling, it proved to be too cumbersome and time consuming to set up. A long bar arrangement was substituted. Torque restraint to prevent the machine body from rotating was provided by a long bar extending horizontally from the body (Figure 15). The bar was long enough to extend through the row of motor

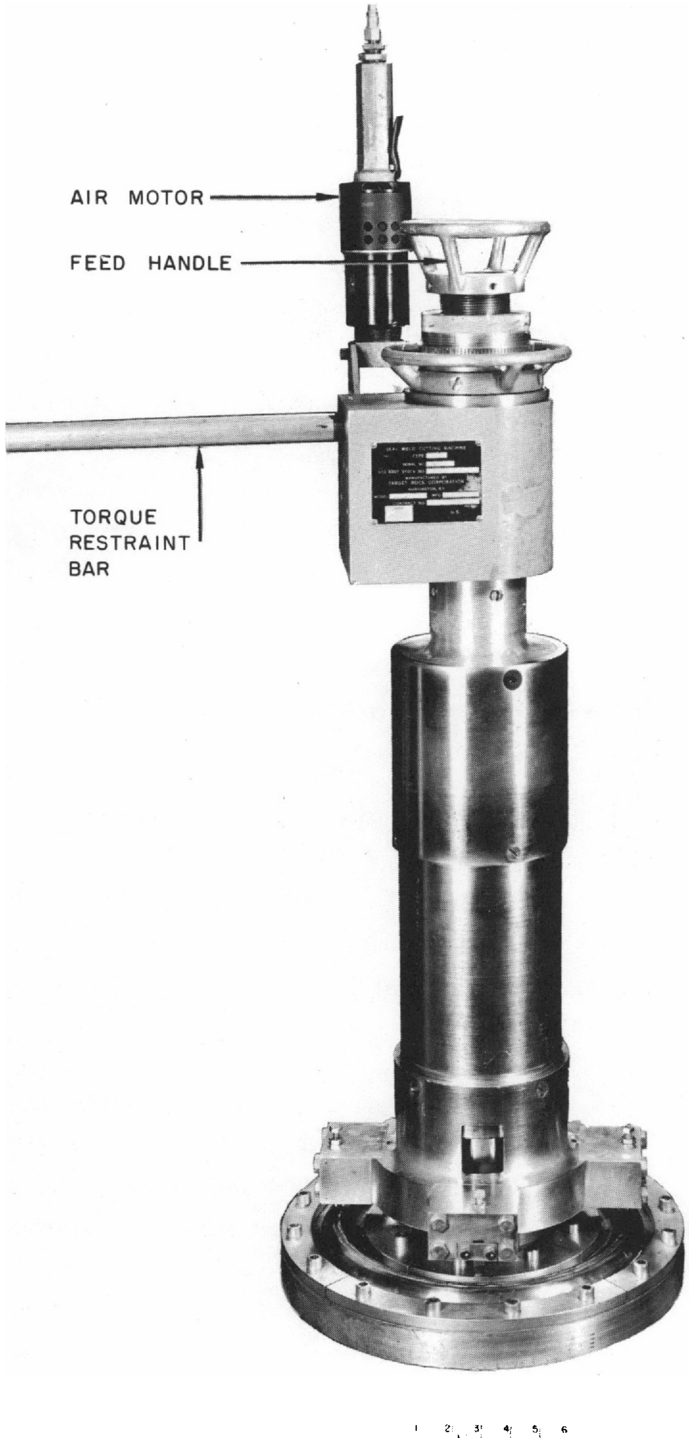


Figure 15a. Cutting Fuel Port Weld.

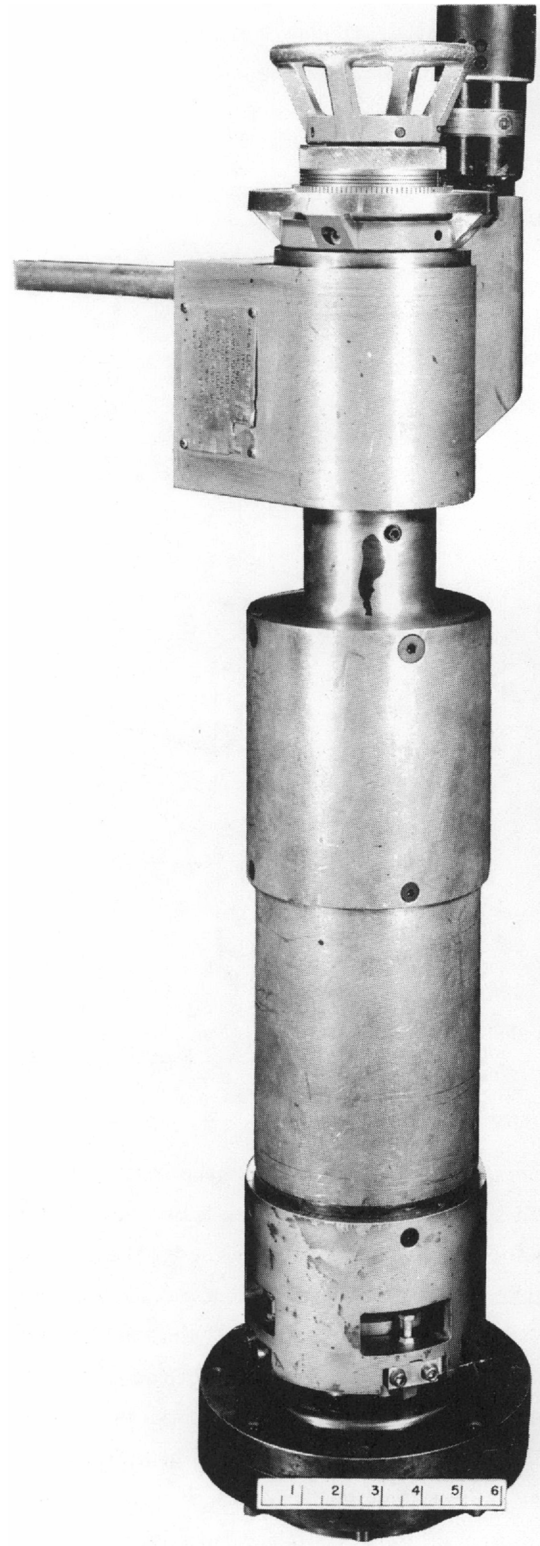


Figure 15b. Cutting Control Rod Mechanism Port Weld.

Figure 15. Omega-Seal Weld Cutting Machine.

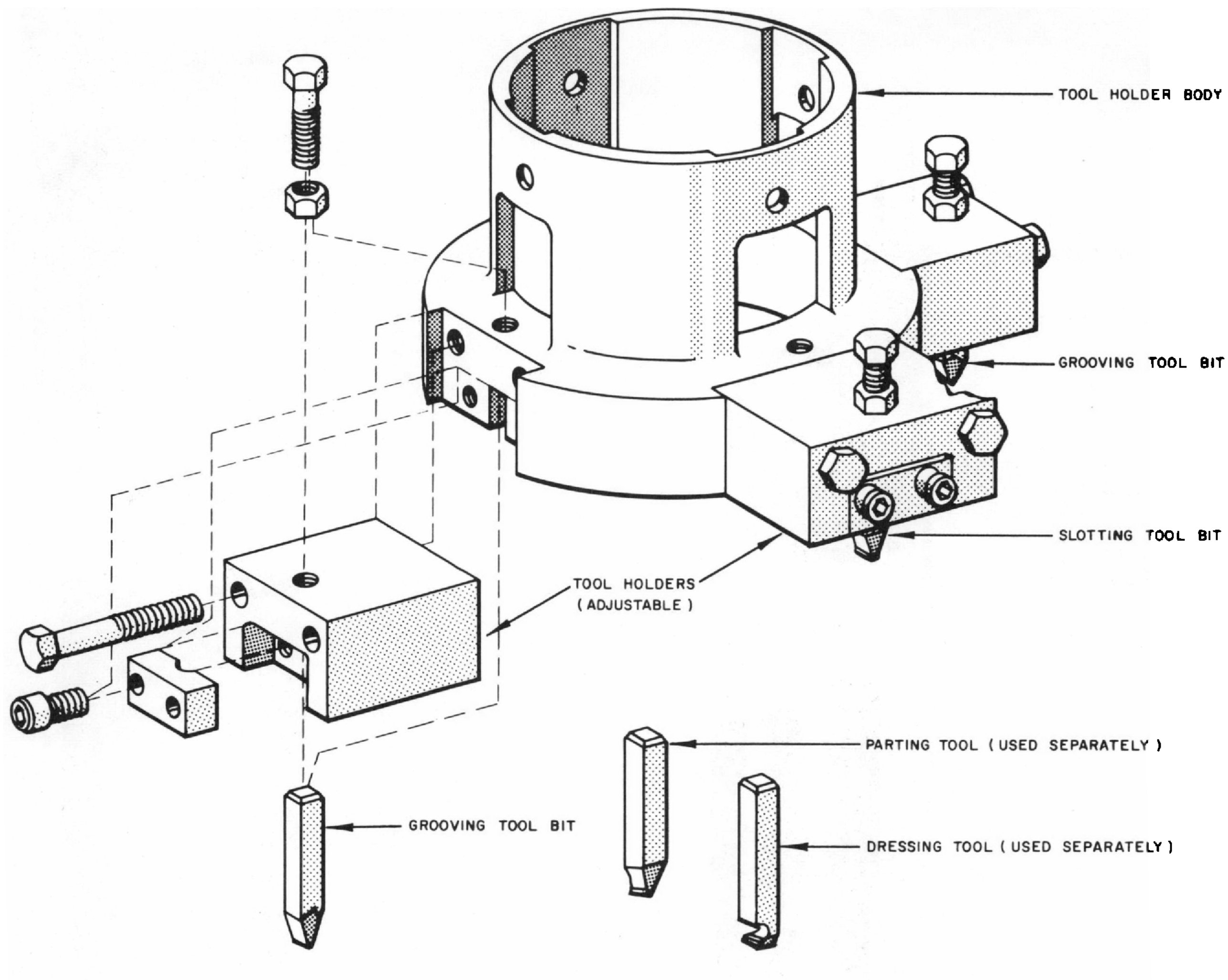


Figure 16. Seal Weld Cutting Machine Tool Holder.

tubes in the array opposite the cutting location, but did not use these tubes as a bearing surface. The outer end of the bar was lashed to the trellis structure with nylon rope to minimize backlash and chatter and was so tied off that the bar would not distort the vertical position of the cutting machine as it travels downward during the cut. When cutting the fuel ports located outside the mechanism array, the same long bar projected radially away from the array, and its outer end was lashed to the trellis structure. This arrangement was very successful.

2. Omega - Seal Welding Machine

For rewelding the mechanisms and fuel port closures after fuel exchange, an automatic welding machine head rotated around a motor tube (or tool post on the fuel ports) to make the circular welds. It used the tungsten-inert-gas (TIG) process with automatic wire feed. Considerable trouble had been originally encountered in obtaining acceptable welds on the mock-ups. A development program, begun immediately after the first refueling, concentrated on improving the constancy of rotational speed and wire feed speed. Among other detailed changes, the wire supply spool was moved to the welding head assembly from the power supply to reduce friction, and the wire feed mechanism was rebuilt. Before starting each weld, with the head stationary and no welding voltage applied, the wire feed time was measured at least twice with a stop watch to be sure that it would be constant and would feed the correct length of wire for the weld. New parameters permitted welding of a larger root gap than previously. The machine is pictured in Figure 17.

3. Thermocouple Seal Weld Cutting Machines

An additional thermocouple seal weld cutting machine was built so that two machines could be used to cut the welds on the thermocouple instrumentation seal plugs. The new machine provided a vertical travel of 5-1/2 inches, as compared to a 2-inch cutter travel for the old. Some of the thermocouples installed with Seed 2 have longer weld plugs than those in Seed 1. To cut the welds of these new plugs with the old machine, it was necessary to cut off a length of plug first. Then, in order to know when the cut had been completed, a wire was soldered to the top of the cut plug. Extending up through the machine's hollow shaft, this wire was bent at the top and was continually watched for rotation to indicate breaking through the weld by the cutting machine. The new machine has adapters which can accommodate all existing types of weld plugs and, unless a thermocouple wire is broken accidentally, this wire is used as a breakthrough indicator. Different cutters and alignment gages are required for most of the varieties of plugs.

4. Manual Welding

In both refuelings, the thermocouple plugs and mechanism guide tube vent plugs were welded manually. The small size of the thermocouple plugs and clearance and accessibility problems preclude the successful use of machine welding. Originally, these welds were made by the manual metal-arc process. For the second refueling, however, a new welding specification was prepared which required the use of the manual TIG process, using a preplaced filler wire for the root pass. Although either metal arc or TIG was permissible for subsequent passes, the latter was decided upon for consistency and savings in time and apparatus. This process produced a better quality weld which required less grinding prior to dye-checking for inspection. TIG welding was also used for at least the root pass of other joints in the primary plant where the weld could be in contact with the primary coolant.

5. Blanket Disassembly Equipment

The equipment for disassembly of irradiated blanket fuel assemblies was located in the deep pit section of the canal and was furnished for the underwater work necessary to inspect parts and replace fuel rod bundles in blanket assemblies. The blanket disassembly stand, used to position a fuel assembly for the various operations, contained hydraulically operated clamping and holding, rotating, and ram devices. During the first refueling the hydraulic control for the positioner (holder rotating) device was unsatisfactory because it did not provide reliable or accurate operation. This portion of the disassembly stand was subsequently rebuilt to operate by direct-connected cranks, shafts, and gears. The hydraulic ram, used to push a mandrel into the assembly for removal of parts, was not changed. The hydraulic tool, used to remove a flow orifice from an irradiated blanket fuel assembly, depressed a shim against a spring and rotated the orifice housing keyed with the shim to unlock these parts from the shell. A new orifice removal tool was built which accomplished these operations by direct mechanical means; the spring compressive force was from the weight of the tool alone. This equipment was used during the second refueling for two irradiated blanket assemblies for recrificing.

6. Inspection Stand

A special stand was built for the microscopic inspection of all control rod mechanism lead screws. The stand was located at the south edge of the fuel storage pit where the lead screw could be traversed vertically past the viewer by means of a chain hoist. It incorporated a cycloptic microscope having lenses permitting magnifications in the range of 3.5 to 12.5. The microscope had provision for attaching a camera. The inspection process is pictured in Figure 18.

Figure 17a. Welding the Outer Omega Seal on Fuel Port N-10.

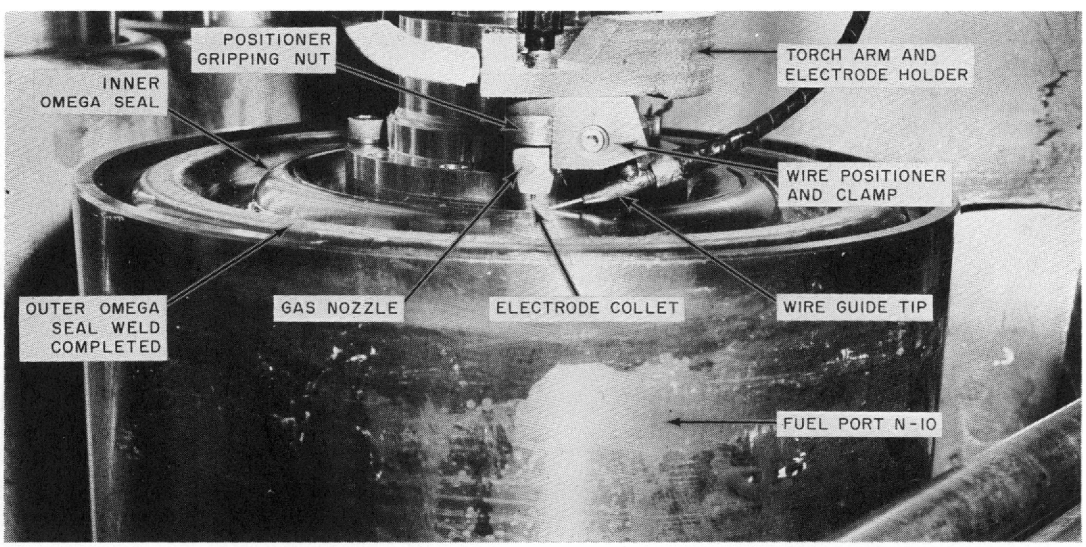


Figure 17b. Welding the Inner Omega Seal on Fuel Port N-10.

Figure 17. Seal Welding Machine.

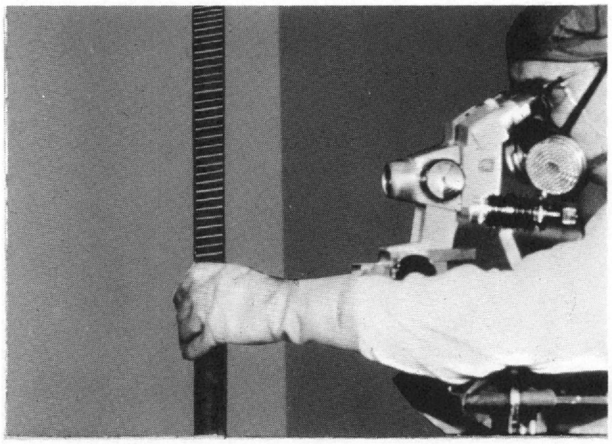


Figure 18. Lead Screw Inspection Using the Cycloptic Microscope.

7. Fuel Handling Cart

The fuel handling cart is used for transporting new fuel assemblies from the receiving area to the fuel storage vault, for inspection of fuel assemblies, and for the exchange of shipping rods for unirradiated hafnium rods in new seed fuel clusters. A minor modification was made to this cart for the receipt of Seed 3. The seed assemblies were lifted in a horizontal position from the shipping box by a two-legged sling equipped with straps. The straps were attached to the fuel assembly near each end. The assembly was deposited in the fuel handling cart whose cradle was rotated to a horizontal position. To allow removal of that lifting strap which was attached near the lower end of the assembly a notch was cut in the rubber lining of the cradle. The upper end of the fuel cluster projected beyond the end of the cradle, so that the strap there could be removed easily.

8. Stator-Coil and P.I. Assembly Storage

An item which saved considerable labor was the device in which the control rod mechanism stator-position-indicator coil assemblies were stored after their removal from the reactor vessel until needed for reinstallation. Racks had been built previously for the storage and protection of the large bolts which attach the reactor vessel head to the vessel body (Figure 19). It was found that the mechanism stator P.I. coil assemblies were about the same size as those bolts, and that, with only minor modifications to the bolt racks, the assemblies would fit into the rack cradles very well. Modifications included padding the cradles to protect the assemblies and changing one rack to accommodate 18 assemblies instead of 14.

9. Other Items

In the first refueling some of the special, but essentially simple, hand tools gave trouble. These were replaced or modified where necessary. For example, new tools were obtained for crimping or decrimping lockwashers used on mechanism stators, mechanism damper nut cup washers, and thermal barriers. A new damper guide spanner wrench was made for use on the mechanism assemblies.

C. Refueling Facilities

The fuel handling canal (Figure 20) is 109 feet long and 22 feet wide, with a capacity of 419,000 gallons, and is divided into four areas. The fuel storage pit located at the south end of the canal is 32 feet deep and has underwater storage racks for 36 spent seed assemblies, storage racks for 36 control rods, and a damaged-fuel storage tank. The deep pit is located in the middle section of the canal and was used as a transfer area during movement of fuel between the storage pit and the reactor pit. The south section of the deep pit is 32 feet deep and contains special fixtures and tools for the underwater disassembly of blanket fuel. Also located in the middle section of the canal is a dry pit for maintenance of the extraction tool and a shroud pit for storing irradiated shrouds and instrumentation under water. At the north end of the canal is the 26 foot deep reactor pit which provides access to the reactor during refueling.

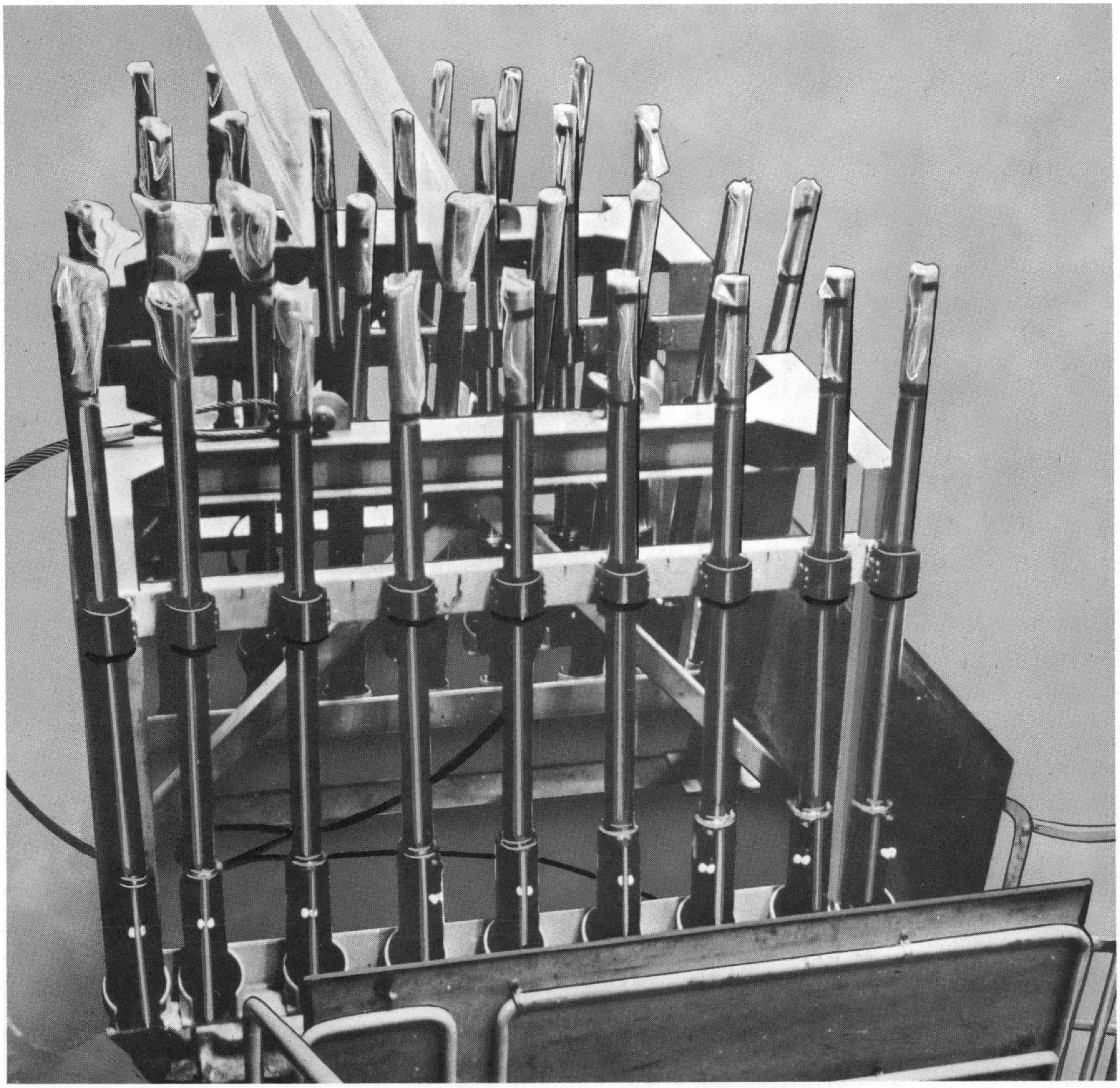


Figure 19. Mechanism Stator and P.I. Coil Assemblies in Storage Racks.

Piping and pumps are provided to permit filling or draining any of the canal sections individually with the canal gates in place and which recirculate pit water through ion exchangers to minimize waterborne radioactivity.

The Fuel Handling Building also contains support facilities, including a decontamination room, locker rooms, a core assembly area, and a core vault for storing new fuel. It is served by the main overhead bridge crane with 125-ton and 25-ton hooks, two traveling work bridges spanning the width of the canal, the special purpose fuel extraction crane (refueling tool), and jib cranes supported from building columns near the north and south ends of the canal.

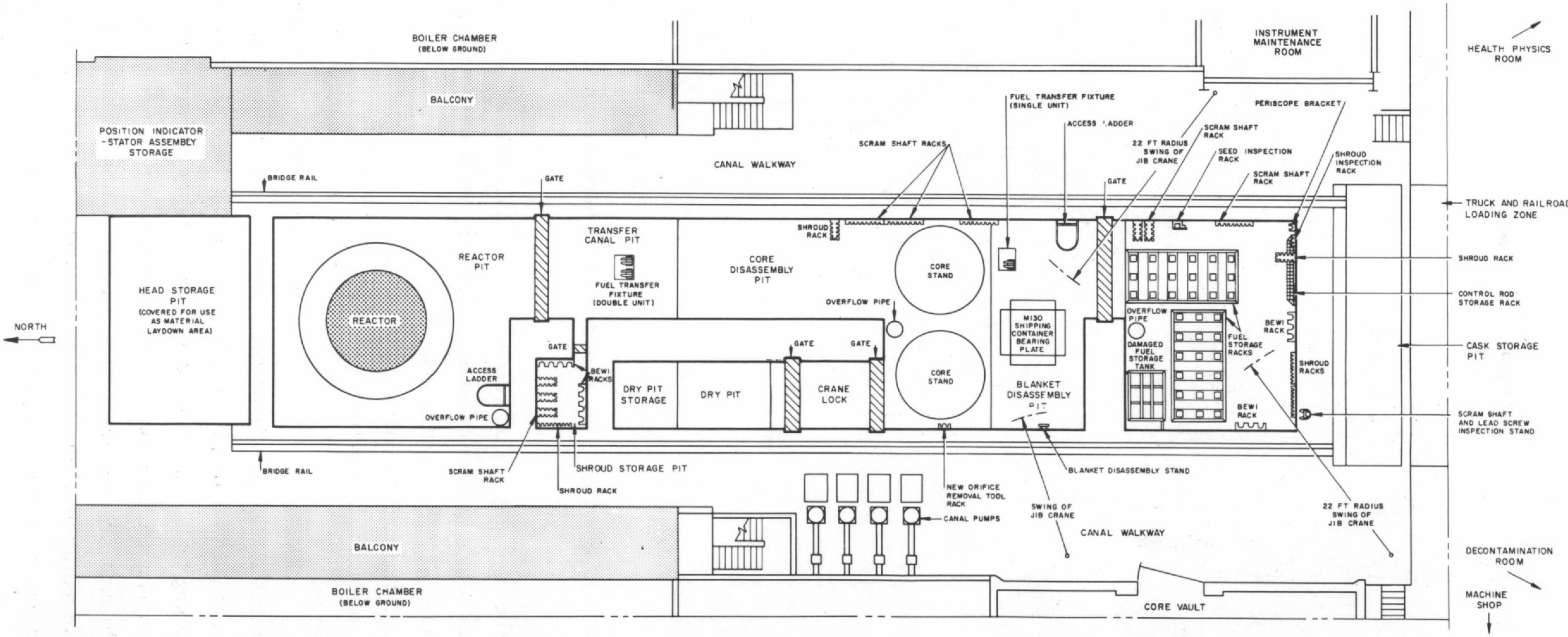


Figure 20a. Fuel Handling Canal (plan view).

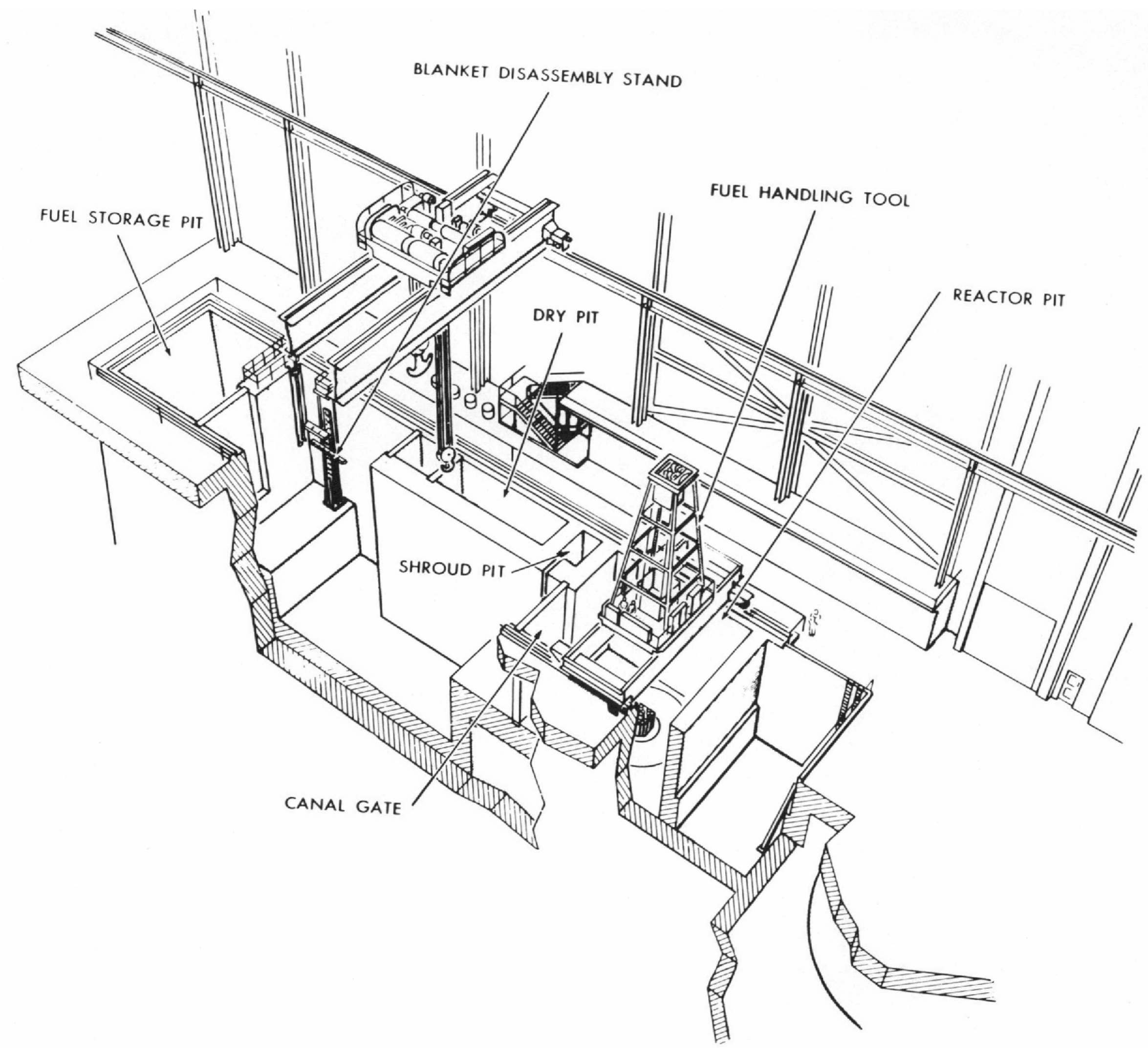


Figure 20b. Fuel Handling Canal (pictorial view).

The north end of the Fuel Handling Building contains a storage area used for the temporary storage of large devices, such as the refueling seal ring and the cable support ring, while they are being cleaned, painted, repaired, or awaiting installation into the reactor pit.

A cover built for the reactor vessel head storage pit provides a laydown area used for the placement of work packages containing tools, materials, and components to be used during the current step or the next few steps of the refueling operation. The forward planning and preparation of these work packages and their location immediately adjacent to the reactor pit where the items would be used made available to the working crews complete complements of materials required for each step.

The Decontamination Room contains both ultrasonic and chemical apparatus, together with baths and wash tanks for cleaning mechanism rotors, thermal barriers, and other small parts. The ultrasonic decontamination equipment is used to the maximum extent when practicable in order to conserve manpower and minimize exposure of personnel. Extremely long or large reactor components or inspection tools and equipment are partially cleaned in one of the flooded pits or are hand scrubbed with chemicals in one of the laydown areas because of the limited available space and tank size in the Decontamination Room.

The Laydown Building is connected to the Fuel Handling Building at the north end of the west balcony directly above one of the reactor plant primary loop chambers. This is for the storage of clean or slightly contaminated components, tools, and equipment. This is also used for a staging area for the preparation of packages of tools, parts, and components used for a particular series of refueling operations.

1. Changes Made to Refueling Facilities after First Refueling

Changes were made to the Fuel Handling Building and its facilities on the basis of the experience of the first refueling and requirements of the second refueling. The principal modifications were (1) relocation of the fuel assembly transfer fixtures and certain storage racks in the canal, thereby simplifying underwater movements of components; (2) the addition of a jib crane; (3) improvement in reactor pit lighting and installation of an air conditioner in the pit; (4) the expansion and rearrangement of the locker rooms to provide faster and more easily controllable access to the refueling area; (5) the rearrangement of materials handling, inspection, and storage facilities; and (6) erection of a machine shop.

The two-unit fuel transfer fixture (Figure 21), formerly in the fuel storage pit, was moved to the narrow section of the transfer canal area just south of the reactor pit, which is about 75 feet farther north than the original location. Movement of this fixture allowed fuel assembly inspection and orifice exchange in the blanket disassembly area to be done in parallel with fuel exchange in the reactor vessel. With the transfer fixture located in the fuel storage pit, travel of the fuel extraction tool crane would interfere with operations in the blanket disassembly area. An incidental time savings of about 4 hours was realized due to the decrease in the distance the extraction tool had to travel between the reactor pit and transfer fixture. The Seed 2 - Seed 3 fuel transfer steps are diagrammed in Figure 22. For comparison, the operations for the previous Seed 1 - Seed 2 fuel transfer are diagrammed in Figure 22.

The second transfer fixture (single unit), originally located in the blanket disassembly area of the canal, was moved to a position about four feet away from the east wall in the main canal channel of travel in order to make room for the type M-130 spent fuel shipping container. This single unit fixture was used in the Seed 2 - Seed 3 fuel transfer during insertion of irradiated control rods into Seed 3 fuel assemblies K-11 and E-5.

A new orifice removal tool rack was placed along the west wall in the core disassembly area, and changes were made in the orifice removal tool and blanket disassembly stand. Along the east wall of the canal in the fuel storage area some rearrangements of shroud and scram shaft storage racks were made and a new seed inspection rack was installed. The seed assembly storage racks in the fuel storage pit were so relocated within the fuel storage area as to make available a space about six feet wide adjacent to the south end wall so that the work bridge could be placed out of the way of main crane fuel handling operations. It also enabled the work bridge and two jib cranes to handle shrouds, control rods, scram shafts, and BEWI for inspection. Movement between racks at the south end of the canal was more efficient and the main crane was not needed as frequently. This work bridge is equipped with an A-frame and chain hoist boom, whereas the other work bridge normally used at the reactor pit is not so equipped.

Added to the walkway at the south end, where the eyepiece for an underwater periscope is located, was a new stand for scram shaft and lead screw inspection.

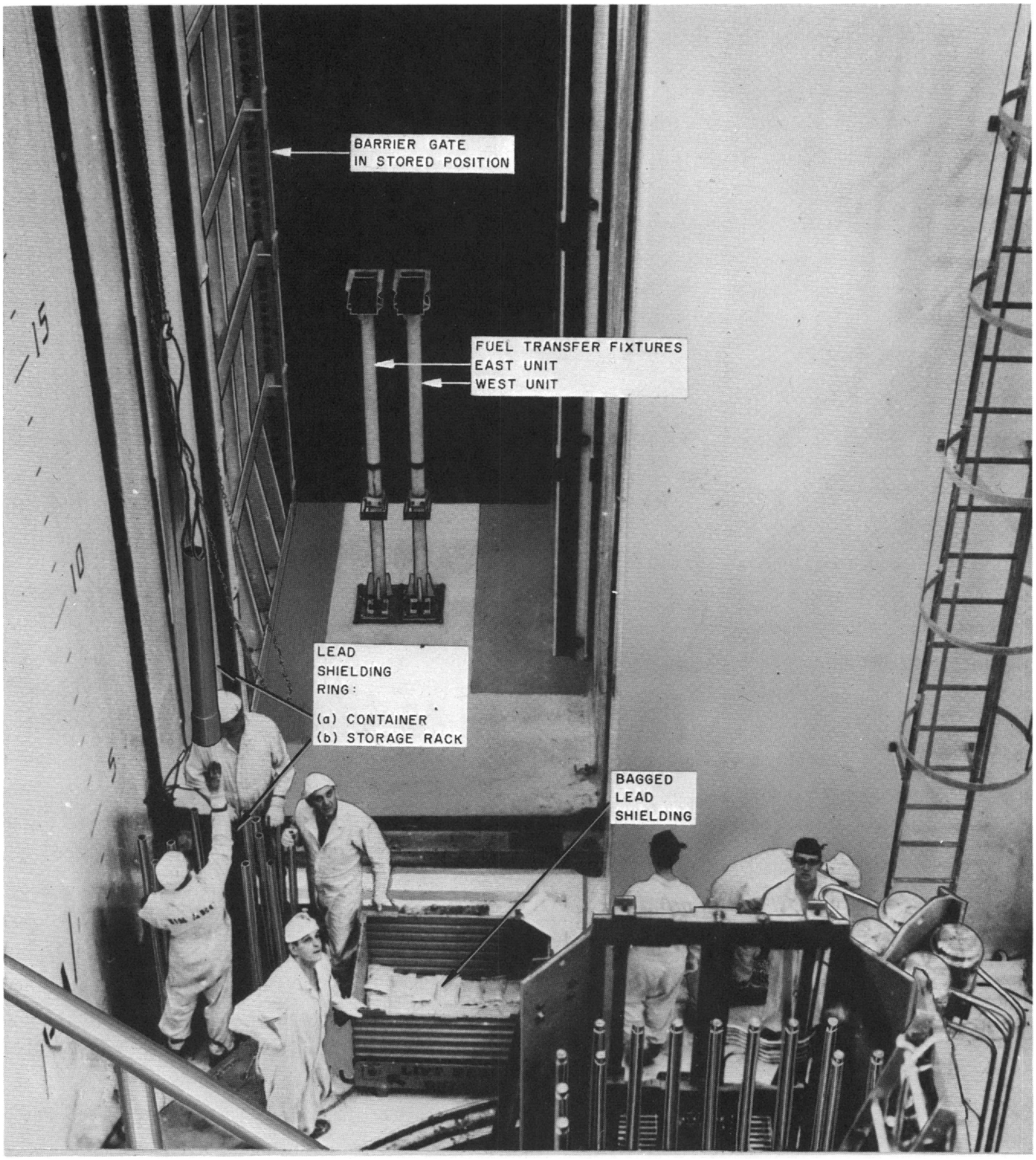
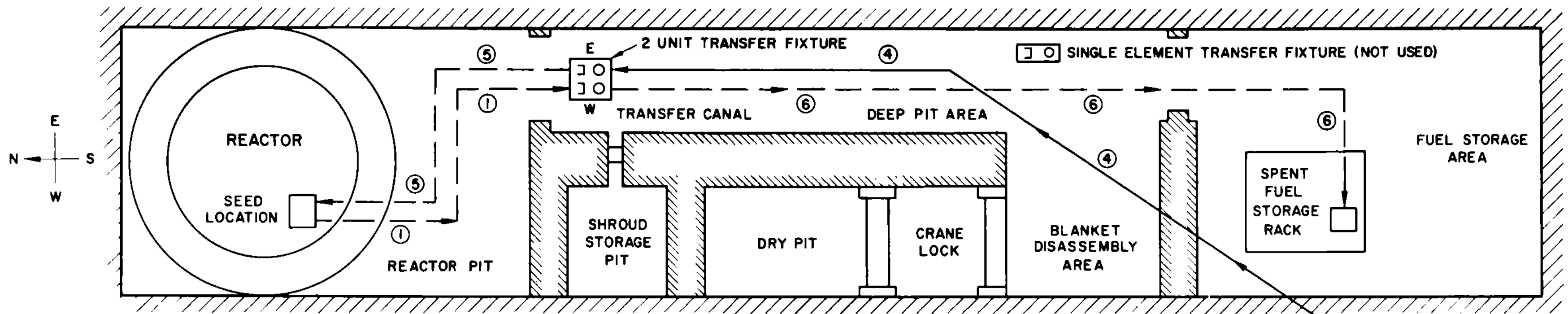


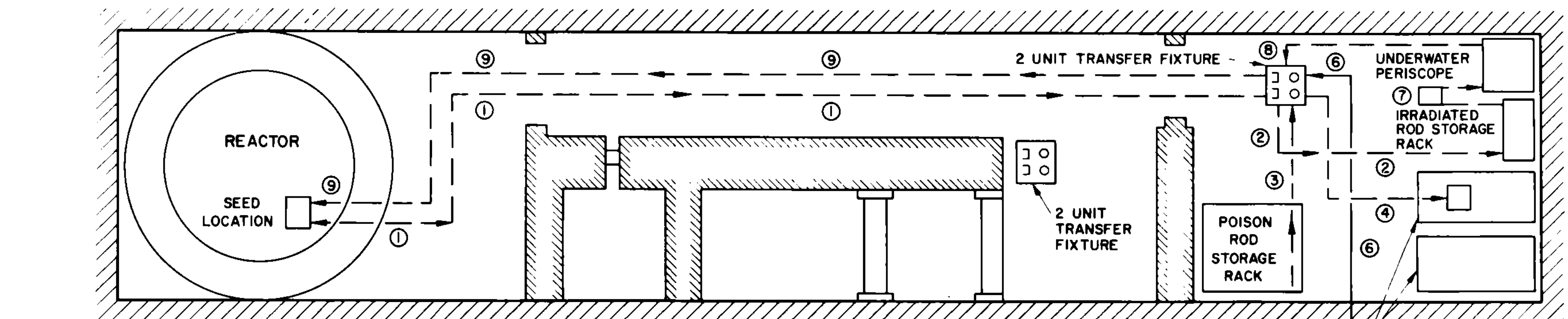
Figure 21. Fuel Transfer Fixture in (Dry) Transfer Canal.

For the first refueling a 1-ton jib crane with a 22-foot boom had been mounted on a building column near the southeast corner of the canal. This relieved the main crane of handling some components in this area. For the second refueling a duplicate jib crane was installed near the southwest corner of the canal. The two cranes thus provided almost complete coverage of the fuel storage pit and the area at the end of this pit.



① REMOVE SPENT SEED 2 ASSEMBLY FROM REACTOR TO WEST TRANSFER FIXTURE VIA FUEL EXTRACTION TOOL
② MOVE NEW SEED ASSEMBLY ON CART FROM VAULT TO CANAL WALKWAY
③ REMOVE NEW Hf CONTROL ROD, INSPECT AND IDENTIFY CLUSTER AND ROD; REINSTALL ROD
④ MOVE NEW SEED 3 ASSEMBLY TO EAST TRANSFER FIXTURE VIA MAIN CRANE
⑤ MOVE NEW SEED 3 ASSEMBLY TO REACTOR AND INSTALL VIA FUEL EXTRACTION TOOL
⑥ TRANSFER SPENT SEED 2 ASSEMBLY UNDERWATER TO SPENT FUEL STORAGE RACK VIA MAIN CRANE

Seed 3.



LEGEND
——— MOVEMENT IN AIR
— — — MOVEMENT IN WATER

① REMOVE SPENT SEED ASSEMBLY VIA FUEL EXTRACTION CRANE TO TRANSFER FIXTURE
② REMOVE IRRADIATED CONTROL ROD FROM CLUSTER; DEPOSIT IN ROD STORAGE RACK
③ TRANSFER POISON ROD FROM POISON ROD RACK AND INSERT IN SPENT CLUSTER
④ TRANSFER SPENT ASSEMBLY WITH POISON ROD TO SPENT FUEL STORAGE RACK
⑤ TRANSFER NEW SEED 2 ASSEMBLY FROM VAULT (5a) REMOVE SHIPPING ROD
⑥ TRANSFER UNRODDED SEED 2 CLUSTER TO TRANSFER FIXTURE
⑦ TRANSFER SEED 1 CONTROL ROD TO UNDERWATER PERISCOPE, INSPECT, VERIFY
⑧ TRANSFER INSPECTED SEED 1 ROD AND INSERT IN NEW SEED 2 CLUSTER IN TRANSFER FIXTURE
⑨ MOVE NEW SEED 2 ASSEMBLY CONTAINING SEED 1 ROD TO REACTOR AND INSTALL VIA FUEL EXTRACTION TOOL

Seed 2.

Figure 22. Fuel Transfer Sequence.

The core storage vault is located west of the fuel storage pit. This is a locked vault used for the storage of new seed clusters, control rods, shipping rods, and new blanket assemblies. For this refueling, the room was rearranged to allow the storage of motor guide tubes, fuel port blind plugs, closure pieces, closure nuts, thermal barriers, and other parts which were awaiting repair or reinstallation. A metal cage containing a door with a suitable lock was built around the area in which fuel assemblies were stored to separate fuel storage area from other component storage area within the vault. The fuel handling cart, into which seed assemblies are inserted for inspection and transporting from the vault storage racks to the edge of the canal, is also kept in this vault.

A machine shop adjacent to the fuel handling area was built as a completely new facility prior to the second refueling. Construction of this shop made it possible to fabricate needed new parts or repair contaminated parts within the controlled security area at the Site instead of having to arrange for such work to be done in outside shops. It is a 6000 square foot westward addition to the south end of the Fuel Handling Building adjacent to the decontamination room and the rail and truck loading dock. The new shop is equipped with machine tools, a welding area, and equipment for the manufacture of simple equipment and for the repair of many of the nonfuel bearing reactor components. Included in the addition are degreasing and wash tanks and a cage for storage of contaminated parts.

2. Evaluation of Facilities

The relocation of the fuel fixtures in the canal greatly improved the efficiency of the fuel transfer operations. Also, the relocation of equipment formerly in the fuel storage pit and on the south wall of the refueling canal increased efficiency. Further improvement could be made if the work bridge rails were extended north beyond the end of the reactor pit to allow the bridge to be moved completely out of the way.

Mechanism motor rotors awaiting decontamination were stored in concrete shielded casks, or drums, near the south end of the canal alongside the railroad unloading dock. While convenient to the decontamination room, this arrangement was awkward because the placement and removal of rotors in the casks required the main crane when it could have been used more efficiently for other refueling operations. The jib cranes could not handle these casks nor could they reach all parts of the area.

A sound-powered phone for communication between the reactor pit bottom and the canal walkway proved very useful, but the need for a signaling system was apparent. Therefore, it is suggested that a 3-station system be incorporated, with suitable signaling terminals located in the pit at the tool control barrier, in the shift supervisor's office and at the canal walkway.

The new shop facilities proved to be advantageous during the second refueling, mainly for making welding mock-ups and for dressing seal weld lips which, otherwise, would have been shipped to other facilities for machining.

III. DESCRIPTION AND EVALUATION OF REFUELING PLANNING, ORGANIZATION, AND TRAINING.

A. Planning

Immediately after the first refueling, the Duquesne Light Company assigned a small group of engineers to the Seed 2 - Seed 3 fuel change on a full-time basis. This key group was assisted by other engineers on a part-time basis where special areas of knowledge or experience would contribute to the planning. As the plans progressed, more full-time engineers participated. A Joint Refueling Committee was formed within three months of the termination of the first refueling. It was composed of representatives of the Duquesne Light Co., the Shippingport Branch Office (SBO) of the AEC, and the Bettis Atomic Power Laboratory. Starting in July 1960 the committee met monthly to identify and make plans for work item responsibilities, priorities and schedules, and to review the status of each major aspect of the forthcoming operation. Concurrently, the Bettis design engineers were modifying certain of the refueling equipment, and those materials whose deliveries required long lead times were ordered. The general scope of the required refueling operations was known immediately following the first refueling, although the specific sequence was not firmly established until shortly before the second refueling began. The early preparations by Duquesne also involved the identification of procedures in which no changes were anticipated. Each sequence step followed during the Seed 2 refueling was reviewed to determine whether it would be needed at all and, if so, in what order. Each existing procedure was reviewed to determine the adequacy of its scope, accuracy, and completeness. As a result some procedures were eliminated, several new ones were written, and the remainder were brought up to date.

Approximately 90 procedures were utilized. They were similar to those prepared for the first refueling, but the major advantage was that now they had been proved in service.

As more information became available concerning the components to be handled, the procedures, equipment and tools for doing the work, and the selection and training of the personnel, various time estimates for the refueling were made; these fluctuated between 50 and 60 working days. The scheduled and actual working time for the individual phases of the refueling work and subsequent plant startup operations is presented in Figure 23.

1. Procurement

The refueling progress was not delayed at any time for lack of working materials. Early in the planning stage a materials supervisor was assigned by Duquesne Light Company, and soon thereafter he was assisted by three engineers. In May 1960, immediately after the first refueling, an inventory was made of all existing equipment, supplies, and spare parts, together with their locations. Damaged items were identified. Bettis identified equipment for which it was responsible, including that on order or in need of repair; periodic status reports were issued. The Duquesne Light Company's materials personnel were assisted by a Bettis representative at Shippingport. In parallel with the work of Duquesne's Materials Group, the Industrial Hygiene Group made sure that all of its supplies and equipment would be available in the proper quantities when needed.

So that all the components, tools, and supplies would be immediately at hand when required for a particular sequence step, all the procedures and steps were analyzed and every item of material required was identified. Groups of materials needed for the performance of each related group of refueling sequence steps were arranged in packages. Each of the 18 package numbers was identified in the master refueling sequence index according to when it would be needed. Laydown areas adjacent to the reactor pit and in the fuel assembly vault were used for the spotting of these packages. After utilization, such tools and equipment as would be needed for a later package were immediately cleaned, repaired if necessary, and repackaged; others were stored for later overhaul and preservation for the next refueling.

2. Scheduling

During refueling, three 8-hour shifts worked six days a week and were rotated every two weeks instead of the normal shift rotation of working 10 days and off four. Shifts relieved each other on the job to exchange information at the work point. During each shift the men working in the reactor pit remained on continuous duty for two hours, after which a new crew was placed in service. This arrangement worked extremely well and contributed greatly to the reduced refueling time. During the

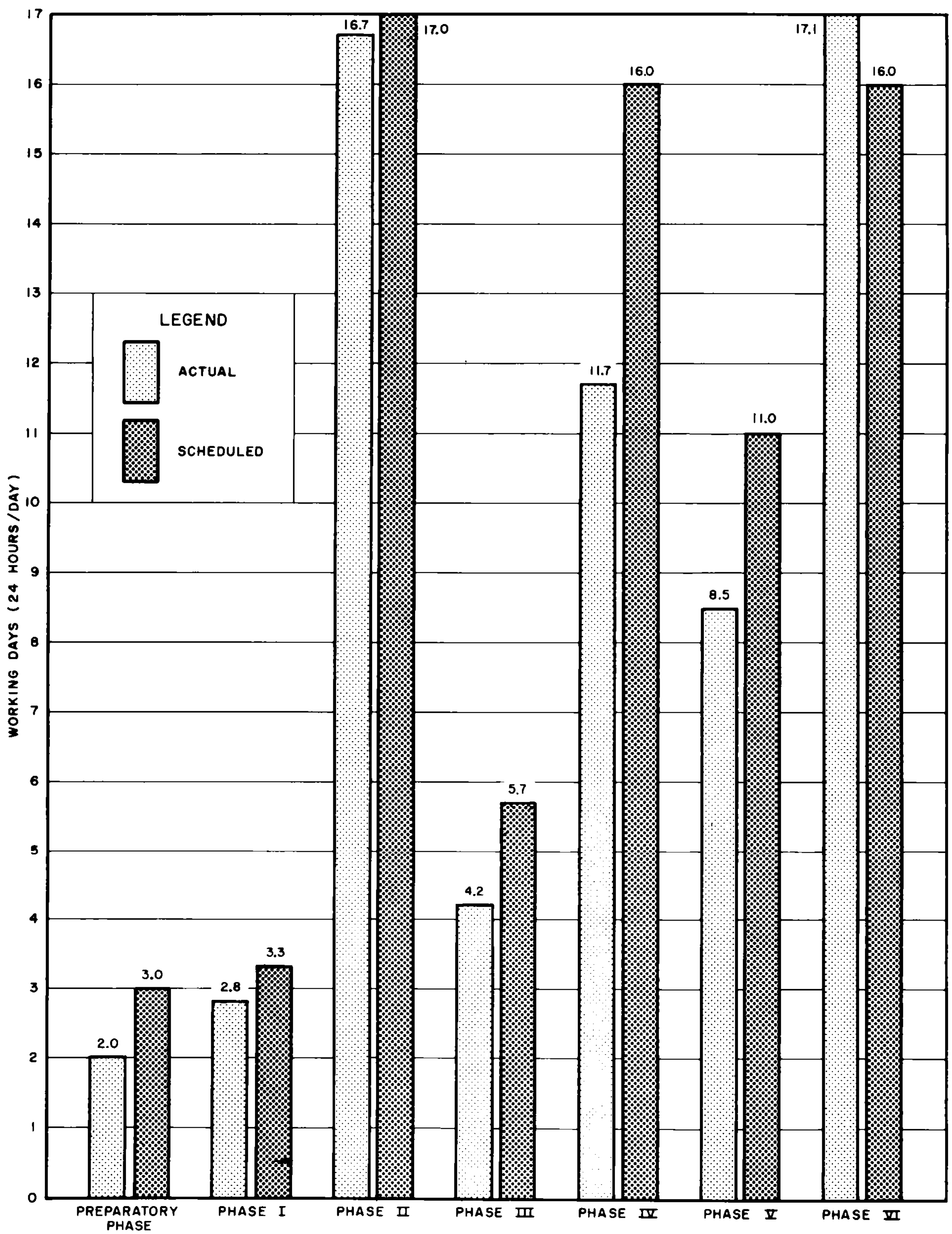


Figure 23. Comparison of Second Refueling Schedules and Actual Time from Shutdown.

fuel exchange, no shift changes were made until an exchange cycle was completed. During the seal welding the welders worked 12-hour shifts, and during the final stages of placing the plant in operation a 7-day week was worked.

During the refueling operation the Procedures and Technical Assistance supervisor maintained a schedule board in the work area which portrayed all schedule steps planned and all parallel tasks required for the next 48 hours. In addition, status charts were displayed on specialized tasks, such as seal weld cutting and dressing. Thus, a current picture of planned and actual events, which pointed up quickly those areas where trouble was occurring or was likely to occur, was always available to the supervisors and workmen. Prompt knowledge of where bottlenecks were developing allowed the necessary decisions and action to be taken to overcome a specific difficulty before significant delays occurred. Study of the progress charts at the end of Section IV will show numerous instances where work was performed at times other than scheduled in order to keep the job moving.

The radiation exposure estimate system, which has been used by the Duquesne Industrial Hygiene Group during the first refueling, was again applied to guide the over-all manpower deployment. The calculated doses, based on dose rates recorded during the first refueling, accurately predicted the time allotted to the craftsmen at their particular assignment. This permitted advanced scheduling of work rotation and prevented any delays which could have been caused by exceeding the maximum weekly dose.

An important means of anticipating potential difficulty was the practice, during refueling, of holding a Schedule Meeting once every week. This was attended by the Duquesne Light Company's supervisors of refueling canal operations, scheduling (Planning and Technical Assistance), and materials, and also by local representatives of the AEC and Bettis. At each meeting the work planned for the forthcoming six days was discussed and action was assigned to overcome any current or potential problem in any area, whether technical or one affecting labor, materials supply, safety, working conditions, or other applicable subjects.

While the direct refueling operations were in progress, power station maintenance, construction, and testing activities were being carried out in accordance with established plans.

As of August 1, the three-month Activity Schedule had identified initial criticality for the refueled plant as October 18 and the return to full power operation about November 1. The August 1 three-month Activity Schedule identified some 110 tasks whose duration ranged from one day to several weeks which were to be completed during the refueling shutdown. About 42 percent of the 87 maintenance and construction jobs identified required a week or longer to complete. However, as the work on the reactor progressed in early September, it became evident that, barring unforeseen circumstances, the refueling would be completed ahead of schedule and that the progress on the other plant work would control the date for return of the station to power. On September 25, refueling progress was 10 days ahead of schedule.

In view of these facts, work on the over-all power plant was intensified and workmen were reassigned from refueling to the station activities as rapidly as practicable. The scope of work to be accomplished was reviewed, so that on September 28 an over-all schedule was issued which identified approximately 75 major tasks requiring completion in the 10 days prior to October 8, the date then predicted for criticality. It was also planned at this time that full power operation could occur, if all went well, as early as October 20.

"Plan-of-the-day" schedules were issued daily after September 28 to identify all items to be performed within the next five to nine shifts (40 to 72 hours). New items were included each day as they came to light. Preparation of these near-term schedules was vital in order to keep all jobs moving and to identify such problems as personnel deployment, component deficiencies, and special system hook-ups.

As it turned out, initial criticality was achieved on October 7 and full power output was accomplished early on October 24. The work from the completion of the seal welding in the reactor pit to full power operation was accomplished in 26 working days. However, in order to maintain this schedule, a 7-day week was worked in all vital areas of the plant.

B. Organization

1. Description

The refueling required rearranging responsibilities within the Duquesne Light Co. Shippingport organization and adding a large number of personnel for the short period the reactor was shut down. It was desirable to have as many people as possible who had gained experience from the first refueling. In assigning experienced people to the refueling work and at the same time providing a good balance of crafts for the work to be done concurrently in other parts of the station, 66 of the 110 men needed were selected from those already employed at the Shippingport Station; fifty-five of these 66 Shippingport men had had previous refueling experience. The remaining 44 men were recruited from other Duquesne Light Company power stations, some of whom had had previous refueling experience. Sixteen of this latter group were given special training in specific refueling operations.

Of the 40 supervisory and technical personnel, 15 had had the same responsibilities during the first refueling and 9 had held other refueling positions. Sixteen men had no previous experience; of these, seven men came from test or operational assignments at Shippingport.

The refueling organization, as shown in Table I-A, followed generally the form used during the first refueling, except that the Industrial Hygiene Group retained its normal relationship to the overall station organization instead of reporting to the Refueling Supervisor; emphasis and responsibilities of the former Engineering and Planning Group were changed so that the successor Procedures and Technical Assistance group had charge of preparing plans, procedures, and schedules. The Canal Operations Group still had undivided responsibility for carrying out the work. The duties and responsibilities of each group are tabulated in Table I-B. In addition to these responsibilities assigned to the refueling organization, the station Test Group was responsible for performing certain tests associated with the refueling as well as their normal duties in other areas of the plant.

The Quality Control group was staffed with one inspector and two assistant inspectors for each shift. In addition to their responsibilities in the reactor pit for making measurements, performing inspections, and taking photographs as called for in the procedures, the assistants were also required to perform similar duties in other portions of the refueling area.

2. Evaluation

The refueling organization functioned smoothly. However, it became evident as the work progressed that there were not enough inspectors to cover adequately the simultaneous jobs in the reactor pit and in all the other areas where their presence might be needed. Also, deviations reported by the inspectors were not always corrected immediately, since a clearly defined procedure for reporting and correcting deficiencies detected by the inspectors was not available.

It is considered that improvements should be made in the inspection organization for the next refueling. A definite means should be arranged for the inspectors to report immediately any deviation

TABLE I-A.

SHIPPINGPORT ATOMIC POWER STATION SEED 2 - SEED 3 REFUELING ORGANIZATION

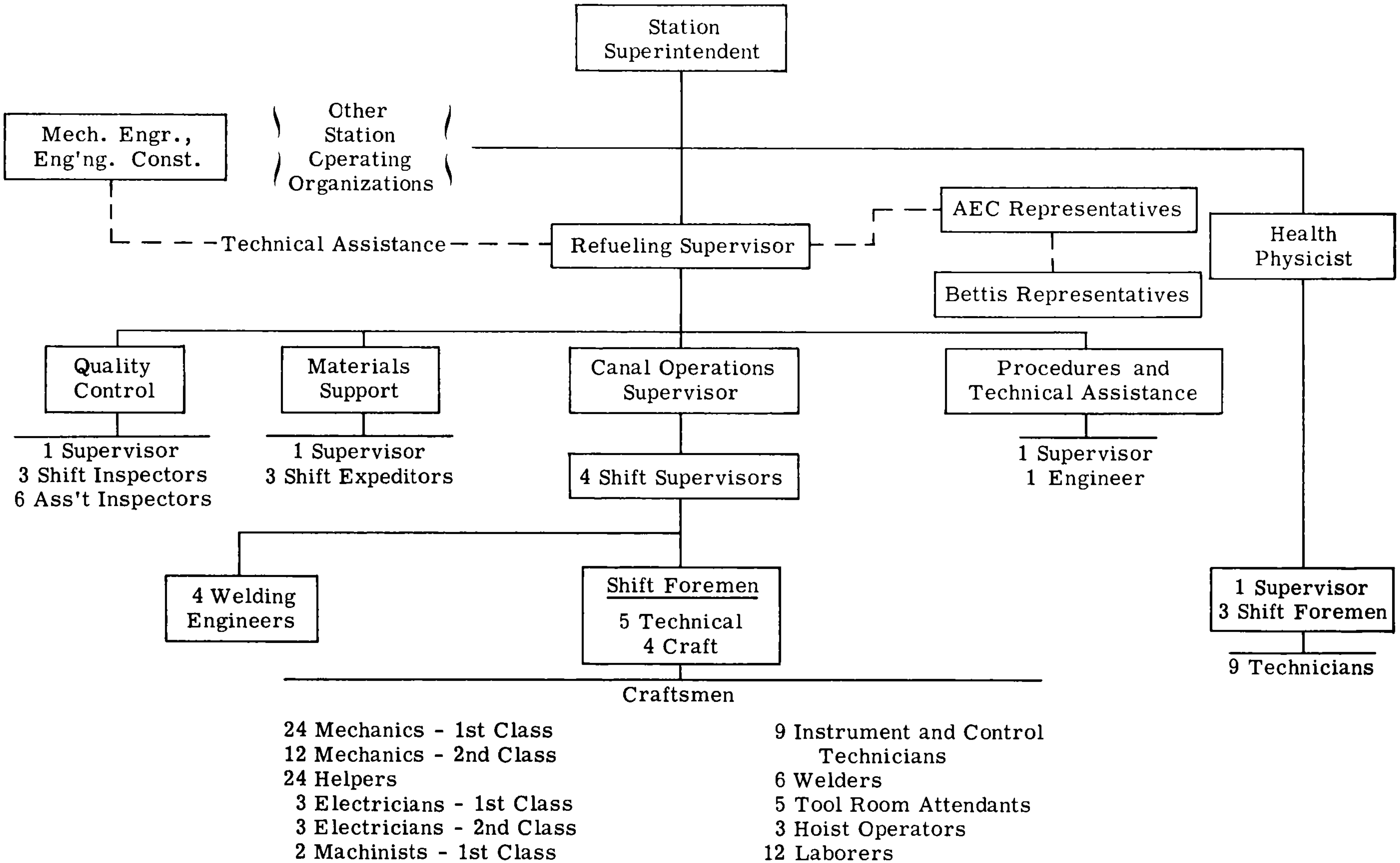


TABLE I-B.

SHIPPINGPORT ATOMIC POWER STATION SEED 2 - SEED 3 REFUELING ORGANIZATION RESPONSIBILITIES

A. Quality Control Group

1. Prepare inspection worksheets.
2. Perform inspections specified in refueling procedures.
3. Maintain all refueling logs and records.
4. Take inspection and general photographs as specified in procedures.
5. Prepare deficiency lists as necessary for components needing repair or replacement.
6. Cognizance of equipment tests as affecting safety of refueling.

B. Materials Support Group

To assure no delays due to lack of critical items:

1. Procure all tools, material, supplies obtained by DLCo except Health Physics supplies.
2. Cognizance of same obtained by Bettis.
3. Prepare all package lists of tools, components, equipment, supplies, including spare parts.
4. Assemble all packages for refueling steps in proper place at time required for use.
5. Expedite all items required during refueling.
6. Operate decontamination and cleaning facilities.
7. Store, inventory, condition all equipment at end of refueling.
8. Back up Refueling Group for all work done outside fuel handling building.

C. Procedures and Technical Assistance Group

1. Prepare detailed refueling plans and procedures.
2. Prepare time schedules.
3. Cognizance of technical details of special tools and equipment designed and ordered by Bettis.
4. Prepare and conduct training program.
5. Coordinate activities between DLCo and Bettis.
6. Reactor safety during refueling.
7. Prepare all reports and information documents required at end of refueling.
8. Technical assistance to all groups prior to and during refueling.
9. Participate in refueling operations which require additional technical supervision.

D. Industrial Hygiene Group

All aspects of Health Physics, including:

1. Monitor radiation levels.
2. Conduct periodic lectures to refueling personnel on health physics precautions.
3. Establish, identify, and control areas of Exclusion, Control, and Clean.
4. Conduct surveys as specified.
5. Maintain personnel exposure records; record all personnel dosimeter and film badge readings.
6. Maintain radiation instruments.
7. Provide clothing and supplies for personnel.
8. Supervise operation of laundry for contaminated clothing.
9. Supervise refueling area housekeeping.
10. Responsible to Operations Shift Supervisor to maintain control over personnel, equipment, working areas.

E. Canal Operations Group

(Conducts all refueling operations in fuel handling building. Assists Materials Group as required.)

in procedures or measurements to appropriate supervision so that corrective action may be initiated and efficiently followed through to completion. One means of implementing this arrangement might be to provide each inspector with "Deviation Notice Tags"; these are to be filled out for each piece of equipment not meeting requirements. One copy would be attached to the piece of equipment until its deficiency was corrected, and other copies would be forwarded to appropriate supervision for initiation of corrective action and for maintaining a log of all outstanding equipment deficiencies.

With regard to the over-all station organization, available forces had been deployed as efficiently as possible to carry on all station activities so that the entire plant would be ready for startup on schedule. Not enough experienced men were available to keep both refueling and maintenance crews up to the strength required because of the large proportion of the station's supervisors and engineers who had been diverted to refueling. Thus, when personnel were reassigned to their normal duties, they had to reorient themselves as to the current status and problems of the jobs in progress. Efforts will be made to minimize this problem in subsequent refuelings.

C. Training

1. Description

The training program was divided into these categories: (1) the indoctrination given to all refueling personnel, (2) supervisor training, (3) the operations training given to the craftsmen in the Canal Operations group, and (4) special training. Various training aids and techniques were used. These included lectures and demonstrations, the movie film of the first refueling, practice operations with mock-ups, and study of refueling procedures.

All refueling personnel received the following indoctrination:

a. A general review of the refueling program.

b. Health Physics lectures, given prior to the start of refueling and at frequent intervals throughout the operation.

c. Lectures on personal safety given just prior to the start of the refueling operation and frequently during its progress.

d. Lectures on maintaining reactor safety during the refueling operation given just prior to cutting the mechanism motor tube seal welds.

e. Lectures covering area security and personnel control, tool control, and plans for emergencies during refueling were given to all refueling personnel just prior to the start of refueling and at intervals during the refueling operations. These lectures included particular emphasis on the necessity of avoiding errors in refueling the reactor, the importance of component integrity, and the requirement of following explicitly the approved refueling procedures and instructions.

The Canal Operations supervisors and the Procedures engineers attended 26 two-hour classroom sessions to study all procedures and applicable drawings. The shift foremen and two engineers from the Procedures and Technical Assistance group were trained in the special mechanical skills required to perform the various refueling operations. Two additional engineers were trained in manual or automatic machine welding.

Fifty-one of the 110 craftsmen assigned to the Canal Operations group were selected to receive detailed training in the skills and techniques required for specific operations. Personnel who already qualified to do the work assigned to them received only the general indoctrination. The detailed training included:

a. Familiarization with all of the critical phases of the refueling operation.

b. Practice operation with actual refueling tools and equipment using mock-ups of core components and the reactor vessel to simulate actual working conditions. During refueling, mock-ups of especially difficult welds were made and used for practice.

The Materials Group gained familiarity with all refueling tools, equipment, and components during their preparation of materials package lists and the actual assembly of packages prior to and during refueling. A majority of the Industrial Hygiene Group was given additional training in the Health Physics refueling procedures. Temporary personnel added to this group received a thorough indoctrination in Health Physics techniques and requirements during a one-month period of on-the-job training prior to refueling.

2. Evaluation

The training effort was successful. Delays in the actual performance of the operations and in performing seal weld mock-ups were minimized during refueling. The men were more familiar with the radiation hazard and were neither careless nor overcautious.

IV. DESCRIPTION AND EVALUATION OF REFUELING OPERATIONS

The description which follows includes those tasks performed as specific steps in the refueling work together with other operations performed in preparing the plant for testing, initial criticality, and resumption of power operation. The effects of performing certain activities in a particular manner and the difficulties encountered are discussed as the events occurred.

A daily chronology of events and progress bar charts is given at the end of this section.

The phases of this refueling were:

Preparatory Phase	Terminate station power production; perform shutdown checkouts and tests; cool down plant, conduct base radiation surveys; prepare primary systems for wet layout and drain secondary systems; drain reactor pit.
Phase I	Remove reactor chamber dome and install refueling seal; remove components external to reactor vessel head preparatory to cutting primary system seal welds. Refueling Sequence Steps 1 - 18.
Phase II	Installation of water level recording instrumentation. Cut control rod mechanism, fuel port, and instrumentation seal welds, dress weld lips in preparation for rewelding, and remove mechanisms, shrouds, and instrumentation. Refueling Sequence Steps 19 - 88.
Phase III	Replace all seed-fuel assemblies, three blanket assemblies, and relocate four blanket assemblies under water. Refueling Sequence Steps (revised) 89 - 250.
Phase IV	Perform dry installation of irradiated components, mechanism components, new instrumentation, and seal welding in preparation for hydrostatic test of reactor vessel. Refueling Sequence Steps (revised) 280 - 315.
Phase V	Install external components, calibrate core d/p cells, remove refueling equipment from reactor

	pit, including refueling seal. Check out electrical cables and systems, hydrostatic test the reactor vessel and primary plant, repair any leaks, install reactor chamber dome, and flood the reactor pit.
Phase VI	Complete necessary plant tests and maintenance, and attain initial criticality with new fuel. Return station to full power operation.

A. Preparatory Phase - Station Shutdown

The load was removed from the turbine and the reactor was shut down in accordance with normal procedures on August 14, 1961. Control rods were then exercised to reduce the crud accumulation in the mechanisms, a radiation survey was made of the reactor vessel head, and the primary plant leak rate was determined. In addition, primary plant self-actuated relief valve leak rates were determined and selected valves noted for repairs; those not to be repaired were then caused to actuate. Following this, reactor plant cooldown procedures were executed and the shielding water pumped out of the canal reactor pit. By the afternoon of August 16 the reactor was cool enough to begin the physical work of disassembling reactor vessel head components. Concurrently, work was begun by the Duquesne Light Company and contractor crews on the many overhaul, modification, and testing tasks in other areas of the station planned for this shutdown. Chief among these tasks were the installation of new volute and new main coolant pumps in the 1D loops, the connection of existing steam and feedwater system to the new heat sink under construction for Core 2 operation, the installation and testing of a Data Acquisition and Logging System, the installation of work platforms in the Auxiliary and Boiler Chambers, and the modification of the Reactor Plant Container Air Cooling System.

B. Phase I - Removal of External Components

The pit was drained and, after the residual water was cleaned from the pit floor, the reactor container dome was removed with no difficulty. Figure 24 shows the head and cover after the dome was removed. Electrical cables (a total of 138) leading from the mechanism motors and position indicator coils and instrumentation terminal boxes to the ring bus were removed, tagged, and stored in polyethylene bags for later electrical testing. Lead shot shielding was placed on the reactor vessel head. The procedure specification that temperature of the head be below 100 F to guard against possible melting of the polyethylene cover over the bags containing the lead shot was revised to about 200 F, because the slow cooling rate of the vessel metal and water would have delayed operations and the bags could withstand the higher temperature. The revised specification value was acceptable and proved to be adequate.

After the rod mechanism holddown structure was removed from the top of the trellis, four scaffold support pins were welded to the top and bottom of each face of the trellis and a prefabricated scaffold was erected.

Cooling water headers and lines to the control rod drive mechanisms and multiport valve were drained and blown out without incident. When removing the 66 cooling water lines, difficulty was encountered in three instances where the fittings were very tight. Threads on the outlet header connector from mechanism location D-9 were damaged and a thread die was used later to dress the threads. The outlet connection adapter and nut threads at the stator, location E-12, were galled sufficiently to prevent removal with a wrench. The connector was heated, the stator water jacket was protected by asbestos cloth, and removal was accomplished. The stator connection threads were damaged and required subsequent repair. Eight

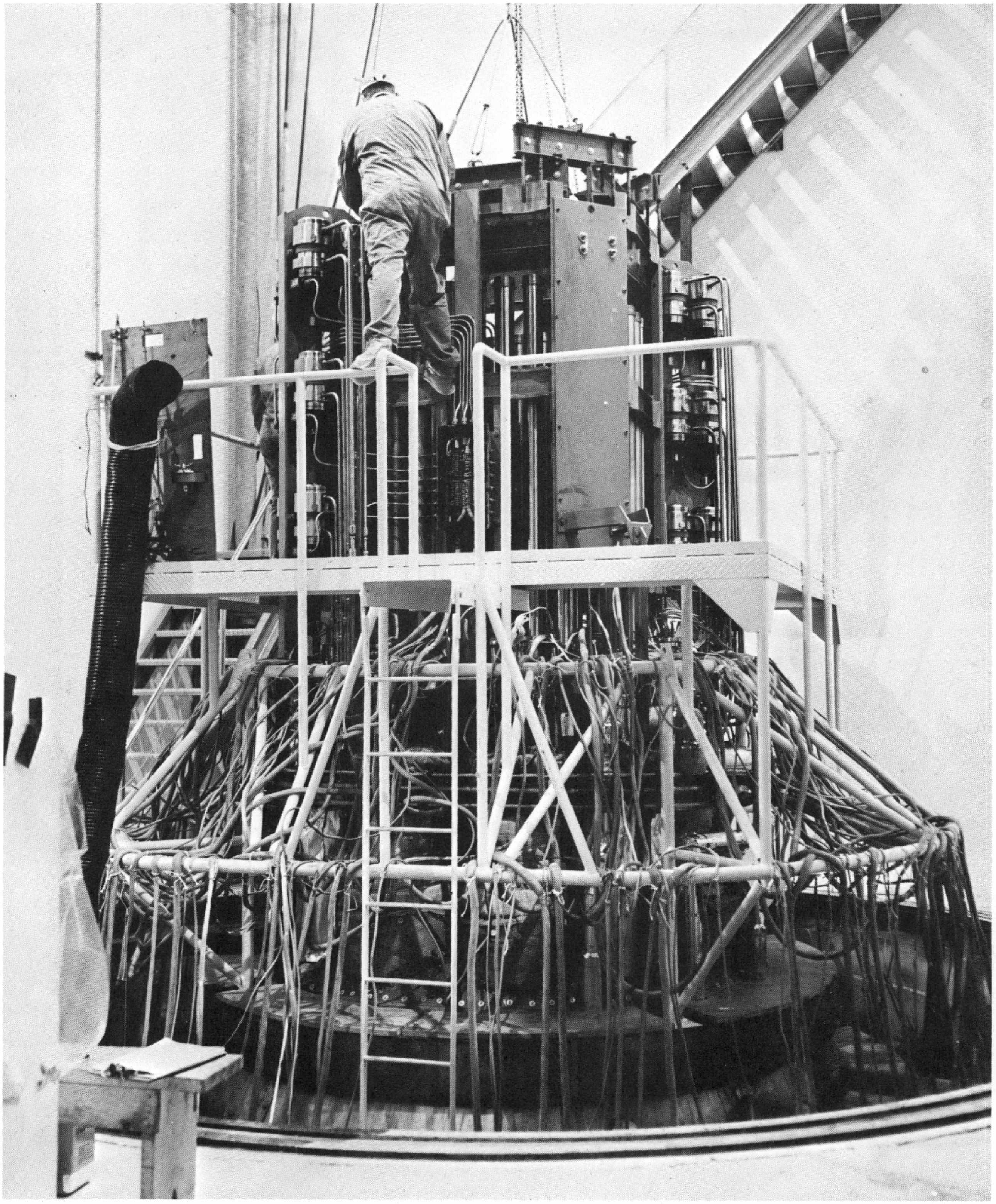


Figure 24. Reactor Vessel Head after Chamber Dome Removal.

pairs of the cooling water inlet and outlet lines did not have identification numbers etched on them; this was corrected before reinstallation. At one location (H-4) the mechanism position-indicator-water jacket assembly was seated loosely on the housing. None of these conditions delayed progress.

The installation of the refueling seal in the reactor pit proceeded smoothly. Bolts were tightened to 2000 ft lbs torque, and the resulting fits between the seal outer ring and the reactor chamber flange were well within the 0.233 inch specified. It was not necessary to leak test the seal.

The position-indicator coil and mechanism stator assemblies were removed (Figure 25) and stored in the modified head-bolt storage racks. The filled racks were hoisted to a balcony floor where, concurrently with other work, the electrical testing and mechanical inspection of the assemblies was performed. Removal of mechanism position-indicator coil covers and housing spacers revealed slight deposits of greenish corrosion product on some of the housing spacers, indicating galvanic corrosion between the brass covers and the stainless steel spacers and tubes. Also, slight corrosion products, asserted to be chromate, were found on three of the mechanism water jackets.

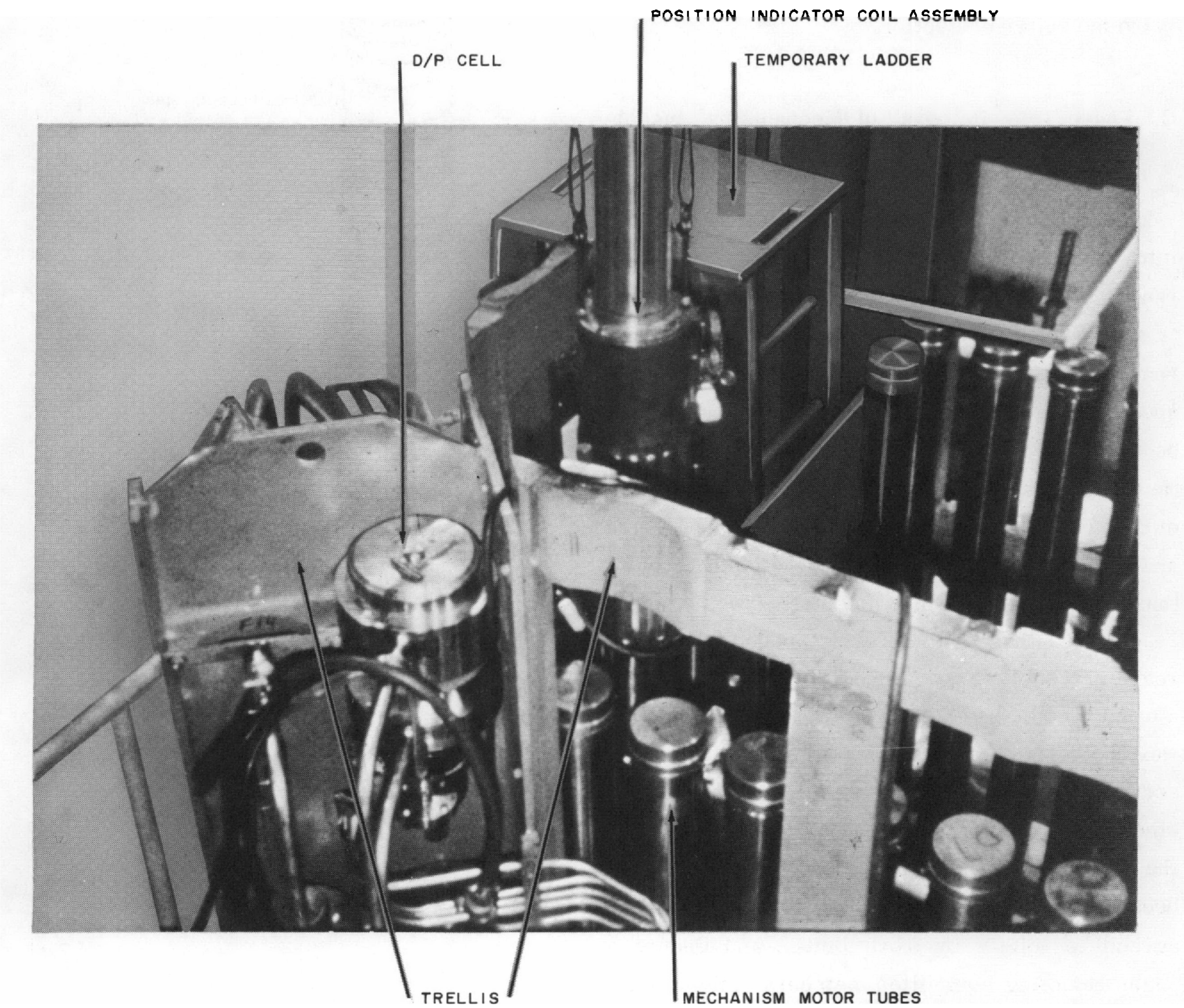


Figure 25. Control Rod Drive Mechanism Stator-Position Indicator Coil Assembly Removal.

Neither of these was considered significant; they were attributed to the accumulation of moisture during shutdown periods.

The new motor tube lead shielding rings and segments (Figure 26) were installed.

Phase I of the operations presented no significant equipment or procedural problems and was completed without delay.

The formed lead shielding on the multiport valve and mechanism rotor tubes was a considerable improvement over the lead sheet shielding used in the first refueling as it was easier to handle and provided better protection.

C. Phase II - Removal of Mechanisms, Instrumentation, Shrouds

The first major step was to calibrate and to put into service the reactor plant pressurizer wide range water level recording instrumentation. This is one of three systems required for indicating the reactor vessel water level, which must be at a specific level before a direct open connection can be made between the reactor vessel internals and the atmosphere by cutting a vent fitting. Calibration of the pressurizer wide range d/p cell was delayed and it was realized in lining up the piping system that the water level instrumentation would not be ready when the refueling crew needed to vent the reactor vessel. Preparations were made to install a backup water level indicator while this calibration was being done and, in the meantime, the refueling sequence was changed to permit performance of other work. About 19 hours were required to place the water level instrumentation in service, but because of the sequence changes proposed and agreed to jointly by AEC, Bettis, and Duquesne Light refueling committee members, only a small fraction of refueling time was lost. The following steps, which were not affected by reactor water level, were performed.

Step 30: Sections of piping leading to d/p cells which could interfere with later operations were

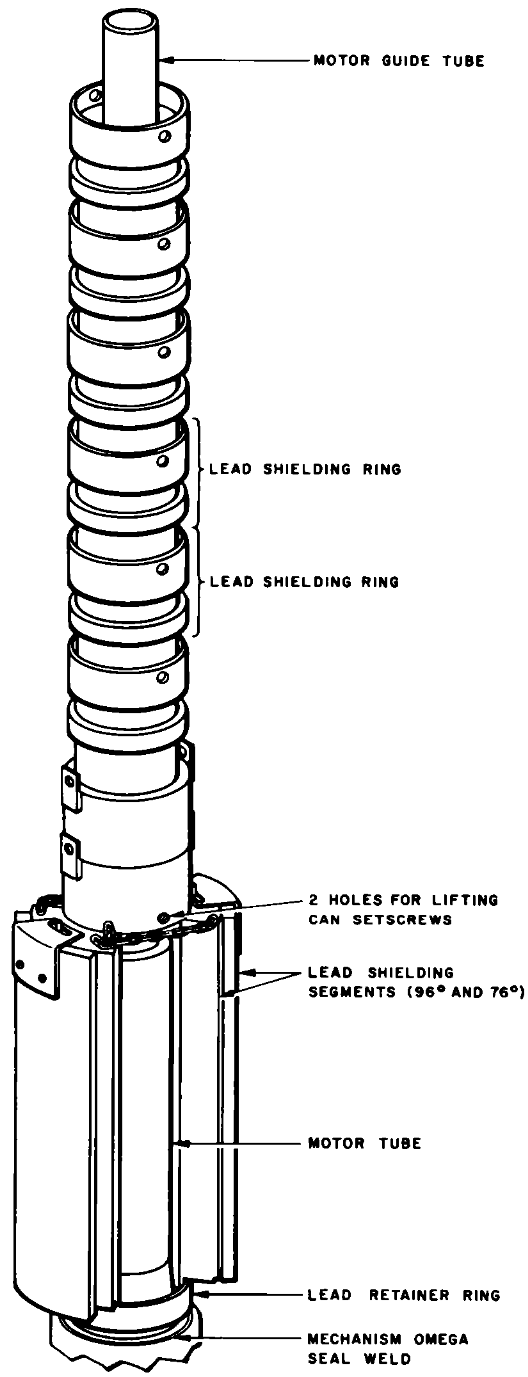


Figure 26. Mechanism Guide Tube Lead Shielding.

checked, and preparations made for removal of specified d/p cells.

Step 43: Seal weld cutting machines were set up on fuel ports F-2 and B-6 in preparation for cutting and removing their blind plugs so that fuel extraction crane bench marks could be checked.

Steps 20, 33, 37, 38: The SMI upper tube assemblies were removed. Thermocouple wires were cut 20 inches above the weld plugs, as specified. Similarly, upper tube assemblies were removed from two ASEWI and terminal boxes were removed from sixteen SEWI locations. Some connectors were saved for later inspection at Bettis.

The water level instrumentation was put in operation and the water was lowered to the required level in the reactor vessel. Then these steps were taken:

Step 19A: The vent plug seal weld on mechanism E-11 was cut to vent the reactor vessel.

Step 43: The fuel port plugs at F-2 and B-6 were cut and removed.

Step 45: The fuel extraction crane was moved into the reactor pit to check the bench marks which had been established during the first refueling. This consisted of starting from an established position marked on a scale at the craneway rails and moving the crane a known distance west, then moving the extraction tool north on the crane bridge to another established scale position adjacent to fuel port F-2, and then east with the tool to a coordinate location which, if no settling of the building had occurred or if nothing had disturbed the settings of the scales or telescopic sighting devices since the previous checkout, would permit lowering the tool into the center of the fuel port to within 0.015-inch clearance all around between the tool post OD and the fuel port cap ID. The operation was repeated to check fuel port B-6. Measurements taken at these two ports indicated a slight displacement of the canal rails relative to the reactor vessel head position during the initial refueling. Corrections to bench marks for the affected fuel ports were determined and previously prepared index cards giving directions for locating the fuel extraction crane were corrected. The maximum correction was 0.070 inch. The coordinates for fuel port H-9 and H-7 required calculation, inasmuch as these ports had not been entered by the extraction tool during the first refueling.

1. Cutting Mechanism and Fuel Port Omega Seal Welds

The cutting of the 32 mechanism omega seal welds was performed with relatively few mechanical equipment difficulties. Access and interference problems between the machine and adjacent structures were at a minimum. The cutting machines were installed and removed without trouble. The torque restraint framework arrangement proved to be too cumbersome and was replaced by a simple restraint bar. The chief difficulty in the cutting operation was with the parting tools.

The first four mechanism welds cut were the inside corner locations, E-5, K-11, E-11, K-5. These had to be cut first to eliminate interference with the cutting machines when cutting the mechanisms on the faces of the mechanism array. Then, starting in the diagonally opposite corners L-5 and D-11, the remaining 28 seal welds were cut, progressing generally in counterclockwise direction around the array. Two units, D-6 and I-4, were particularly troublesome. Cutting and parting the weld in D-6 was performed in parallel with the scheduled operations and occupied the major part of three work shifts. When the cutting machine was being installed, the grooving tool bit had not been tightened in the holder; the holder was rotated, resulting in a 0.010 inch deep groove being cut in the housing lip outside the weld area for about 90 degrees. The slotting tool was used to complete the break through. The lip required later repair. Tungsten inclusions during previous welding caused hard spots in the weld and two slotting tool bits broke. A piece of one became jammed in the slot. Finally, one side of the weld was cut through, but on the diametrically opposite side the weld was apparently thicker. Feeding the machine at 0.0005 inch per revolution and using hand parting tools, both of which frequently broke, the weld was finally parted. The resulting groove was irregular and the root gap varied from 0.045 inch to 0.061 inch in width.

Weld I-4, the last one to be cut, was nearly as difficult. Due to machine vibration filister head screws holding the feed nut loosened. The machine had to be removed, reassembled, and the tool bits reset. Hard spots in the weld broke four slotting tool bits. A special chisel type bit was also used. The machine was tight on the guide tube threads. In attempting to dress the guide tube threads and reinstall the machine, the threads became jammed and the machine had to be removed again. The second cutting machine was then used with the tool holder from the first machine, and the grooving and slotting operations were repeated to greater depth; then hand parting tools were used to complete the cut.

The double seal welds at the eight remaining fuel ports were cut after the mechanism welds were cut (Figure 27). Each of these ports contained thermocouple leads extending through welded plugs in a central closure piece. Between the closure piece and fuel port housing is an annular nut, threaded on the outside diameter. The seal welds join the nut to the outside housing, and the central closure piece to the annular nut. The closure nuts and pieces were removed, the housing threads inspected, and the housing weld lips dressed. Little difficulty was encountered in the cutting operation on any of these welds, except for thick welds and some which had irregular configuration, requiring the use of a planer tool on one and of hacksaw blades to part three welds. The nuts and closure pieces were removed satisfactorily, except the BEWI closure nut and closure piece from fuel port location J-2 (Step 48). The omega seal weld cutting machine had cut and parted the outer (11 inches nominal diameter) and inner (7 inches nominal diameter) fuel port seal welds without apparent incident. However, when the closure nut was turned for removal, the nut interfered with both the housing in the vessel head and the closure piece. When the nut was turned, the closure piece also rotated, tending to twist the thermocouple wires which project through it into the BEWI terminal box. The nut was turned down again, thereby carrying chips into the pilot between the nut and housing and into the fit between the nut and closure piece, thus, further preventing removal. Investigation disclosed an error in setting up the seal weld cutting machine. When attaching the tool post to the closure, dowel pins were not installed in the two locating holes. Without these pins there was enough clearance between the tool post holddown bolts and their holes to allow this tool post to be tightened down 0.020 inch off center. Thus, when cutting the welds, both omega seals were cut concentric with each other, but 0.020 inch eccentric with respect to the true centerline, or 0.040 TIR, causing the nut to bind between the central closure piece and the stationary housing.

The solution was to recenter the tool post and recut the outer omega seal to a diameter larger than normal by 0.040 inch. The inner weld was cut to a 1/2 inch wide V-gap so that the interference occurring between the nut and closure piece could be cleaned of chips. The nut could then be unscrewed. This cutting made both the closure piece and nut unusable, and also enlarged the diameter of the stationary housing seal lip. This cutting back of the housing lip decreased its height by about 0.010 inch, and the gap to be welded later, using a new nut, would be larger than nominal. Therefore, in preparation for later reinstallation,

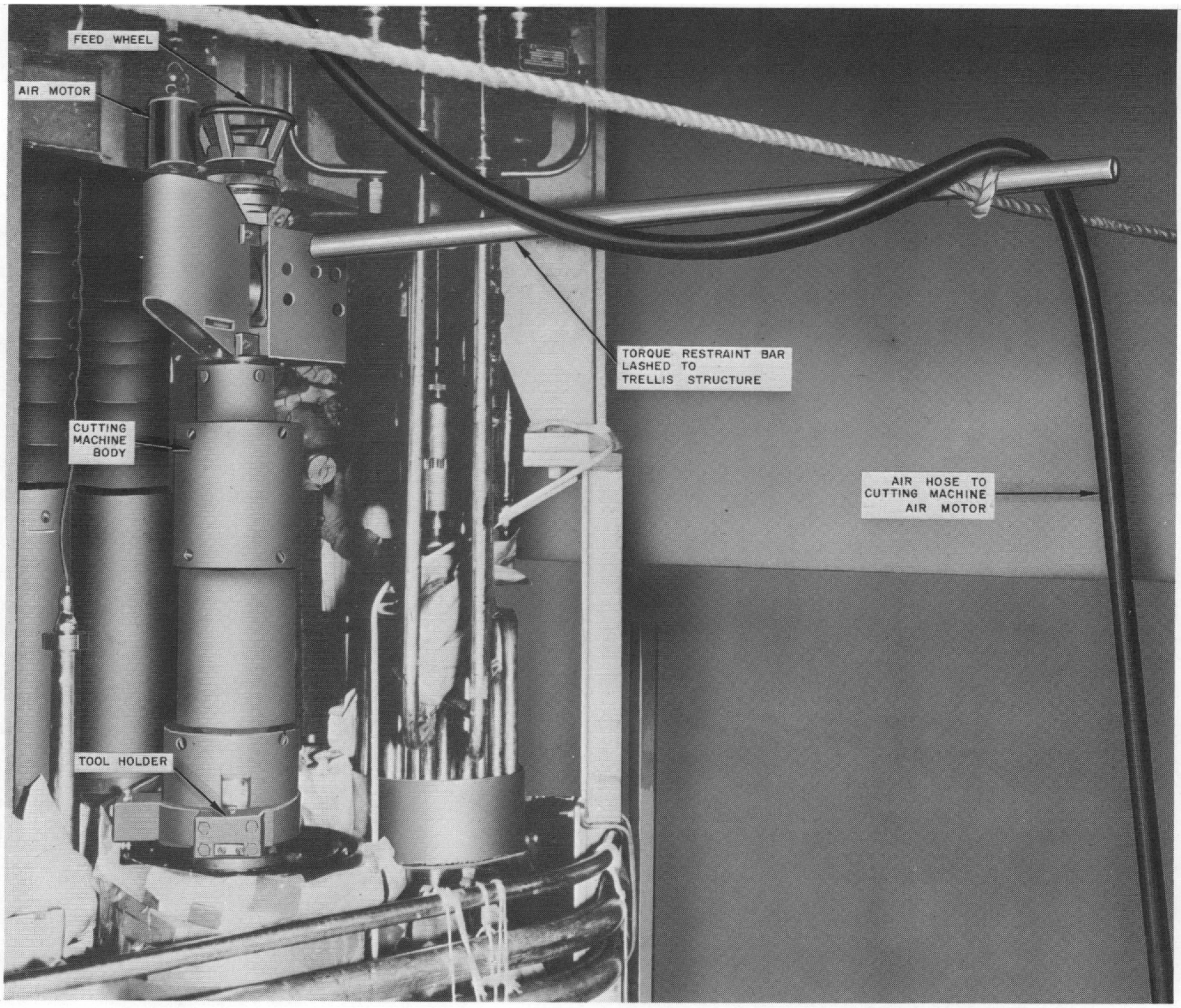


Figure 27. Seal Weld Cutting Machine on Fuel Port B-6.

a spare closure nut was prepared having an oversize lip diameter made to match the housing lip so that the root gap of the weld preparation would be within the range of the automatic welding machine. A replacement closure piece was also obtained from stock spares.

The mechanism and fuel port seal weld cutting difficulties were caused mainly by excessive underbead reinforcement and tungsten inclusions in the welds which had been made during the first refueling. These made the final parting cuts troublesome.

2. Instrumentation Plug Weld Cutting

The cutting of the thermocouple weld plugs at 6 SMI, 16 SEWI, 6 BEWI, and 2 ASEWI fuel ports was, in general, performed concurrently with mechanism seal weld cutting. The 16 SEWI plugs caused little trouble. Difficulties centered around the 30 plugs in the six SMI locations and BEWI plugs in the fuel port closures. Thick welds caused difficulty with tool travel in several instances; tool stops had to be reset. Many cutting machine shear pins were broken due to cutter jamming, implying too forceful feeding; also, several cutters were broken. On one SMI plug the machine adapter required modification because the length of the adapter would not allow the cutter to reach the bottom of the weld penetration. One machine required replacement of broken needle bearings and repair of a scored shaft.

3. Removal of Mechanisms

After completion of seal weld cutting and during thermocouple plug cutting, the control rod mechanism motor tubes, damper nuts, damper guides, tie rod sleeves, and motor rotors were removed. Concurrently, rod position indicating coils, motor stators, and electrical cables were being checked and all seal weld lips were being machine-dressed for rewelding. Decontamination of mechanism parts was begun immediately upon their removal.

Dressing of the weld lips on the motor guide tubes was performed in a lathe by a machinist rather than with the portable cutting machines. Thus, the operation was performed faster, more accurately, and required two less men. Following the dressing of the weld lips, the thermal barriers were removed and the internal surfaces of the 32 mechanism housings inspected.

4. Shroud Removal and Gas Bubble Test

Control rod shrouds J-4 and L-8 were removed dry in order to perform the FEDAL System Gas Bubble Test without a separate flooding and draining of the reactor pit.

Shroud locking rings, holddown nuts, and nut retainers were removed from the two mechanism locations at J-4 and L-8. Control rod shroud lifting bails and shaft lifting bails were then installed at each of these locations preparatory to the removal of these shroud-scram shaft assemblies. At this point, the importance of precautions to prevent criticality and the reasons for adopting explicit procedures were reviewed with all crew members, because this was to be the first time shroud assemblies had been removed from the reactor other than underwater. After the proper rigging and attachments to the 25-ton hook of the main crane had been made, the assembly from position L-8 was lifted slowly out of the mechanism port a short distance to be sure everything was free. The canal area was cleared of all personnel except two workmen and a radiation control technician on the workbridge and the operator in the cab on the main crane. The assembly was then lifted to a level such that the bottom of the shroud was just out of the mechanism port. This lifting operation was monitored continuously by the radiation control technician. The assembly was then raised until the bottom end was high enough to clear the reactor pit gate and it was transported south to the transfer canal. There it was lowered into the water and carried to the shroud storage racks along the east wall of the deep pit area. Because radiation levels were well within safe limits, the same process was repeated with the shroud and scram shaft assembly from location J-4. With this shroud at its highest elevation, the radiation dose about 20 feet from the lower end of the shroud was 100 mr/hr. The operation was safe and the decision was made to remove the other 30 shroud-scram shaft assemblies in the same manner. The lead screw was removed from the L-8 assembly and transported to the lead screw inspection stand at the south end of the canal for microscopic examination.

After these two shroud-scram shaft assemblies were removed, the Gas Bubble Test was performed on the FEDAL System to verify that the valve position was correct. With the FEDAL multiport valve positioned to monitor blanket location J-5 and with the FEDAL system pressurized with nitrogen, bubbles would emanate from the J-5 assembly into the water in the reactor vessel if the position of the valve were correct. By viewing the vessel interior through the adjacent mechanism port J-4 the source of the bubbles could be seen and the correctness of the FEDAL system indication confirmed. The test was repeated viewing through location L-8 for blanket assembly K-8. The test was completed satisfactorily; gas bubbles were observed coming from locations J-5 and K-8, and the settings of the FEDAL system multiport valve confirmed that the system had correctly identified locations J-5 and K-8.

Preparations had already been made for the underwater removal of the remaining 30 shroud and scram shaft assemblies. When agreement was reached to remove these assemblies without flooding the pit, the procedures were changed. The shrouds could be taken directly to the fuel storage pit in dry removal, whereas in wet removal they had to be taken to the shroud storage pit first, and then moved to the fuel storage pit later because they interfered with the fuel extraction tool.

All lead screws from these shroud-scram shaft assemblies were visually inspected in the fuel storage pit where they were lifted up past the viewing device of the microscopic lead screw inspection stand. Selected assemblies from locations E-12, H-4, H-12, J-4, J-12, L-8, and K-5 were chosen for disassembly and for internal and external inspection of the shroud or the scram shaft or both. K-5 was to receive a new scram shaft for Seed 3 operation. Shrouds to be inspected were from locations E-12, H-4, H-12, and J-4. The scram shaft assemblies from locations E-12, H-4, H-12, J-12, and L-8 were inspected. External visual inspection was made of the control rod shrouds from locations H-4, E-12, H-12, and I-12. The shrouds from H-4, E-12, H-12, J-4, and K-5 were disassembled for internal inspection. The K-5 and J-4 scram shafts were removed by placing each shroud in its storage rack and pulling the shaft out and up into the air, instead of using the scram shaft disassembly stand for underwater removal as orginally specified. Operators were working in a radiation field of about 20 mr/hr during this removal in air. The Seed 1 shroud removed during the first refueling from location L-8 was fitted with a new lead screw and scram shaft (Figure 28) and reinserted into location J-4 for Seed 3 operation. New scram shaft-lead screw assemblies were also installed into the shrouds for K-5 and E-12. The latter lead screw was selected from units which had been precipitation hardened at 1100 F.

All control rod mechanism lead screws were inspected by Bettis Laboratory engineers at the fuel storage pit by lifting them out of the water and viewing with the binocular provided with the lead screw inspection stand. Scram shaft assemblies from locations K-5, J-4, and E-12 were sent to Bettis for destructive examination and were replaced by new assemblies.

5. BEWI and ASEWI Assembly Removal

The last major operation before flooding the reactor pit was to rig the BEWI and ASEWI assemblies for underwater removal. These steps consisted essentially of "nesting" the BEWI sliding tubes for each assembly so that the bent feet at the bottom of the assemblies would be in the same vertical plane for clearance through the fuel port when lifted for removal. These operations for the BEWI's were performed satisfactorily. The ASEWI's at locations N-10 and J-14 were lifted and rested on the support frame thread protectors without difficulty.

Fuel port caps were then installed in their open position on the ten fuel port openings until after the instrumentation removal. Then the caps were closed to prevent foreign objects from falling into the reactor. Each fuel port cap had previously been scribed with the grid location of the fuel port on which it was to be installed, and was matchmarked for proper orientation on the fuel port in order to assure that it could be opened and closed under water without interference. The installation of these hinged covers occurred without difficulty, but it later developed during fuel transfer that the cover on location F-14 was incorrectly oriented. The cover would not close because the lid hit the d/p cell piping. At that time, a temporary cover

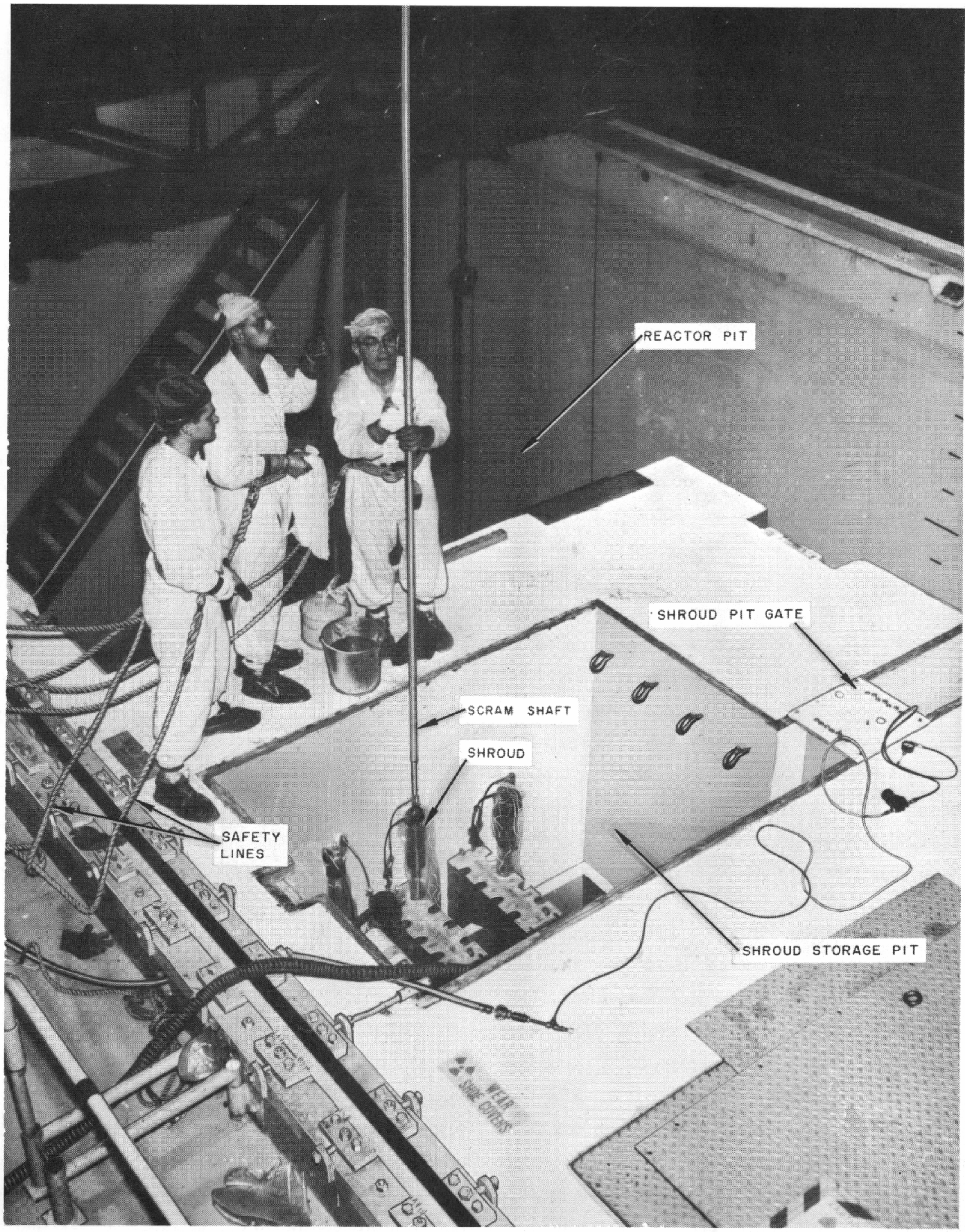


Figure 28. Installing a New Scram Shaft in Shroud Removed from Location L-8.

was made and lowered into place under water and, at a later time, the cover was rotated to permit it to close without interfering with the piping.

Floor gratings, barriers, tool cabinets, and all other working tools, equipment, and materials in the reactor pit were removed; the pit floor was cleaned; and, the reactor vessel head was cleaned and vacuumed in preparation for flooding the pit.

Water level in the reactor vessel was raised five feet to bring it up to a level just below the fuel port openings. During the flooding of the reactor pit, which required about 4 hours, the refueling seal was monitored for any leaks into the reactor chamber. No leaks occurred. At the completion of flooding, the reactor pit gate was removed and stored along the east wall of the canal transfer area. The two work bridges were rearranged in location so that disassembly of shroud and scram shaft assemblies could begin.

The BEWI and ASEWI are positioned in the reactor through eight of the ten fuel ports, and must be removed before fuel exchange can be effected through these ports. Because it was felt that these assemblies could be highly radioactive, they were removed under water.

6. Evaluation of Phase II Operations (Cut Omega Seal Welds and Remove Nonfuel Components)

Reactor vessel water level instrumentation difficulties caused rearrangement of the schedule. This problem was caused by inadequate integration of the power plant operations with refueling operations and should be corrected during future refuelings.

Omega seal cutting was accomplished during 4 working days, whereas in the first refueling this operation spanned 12 working days. The cutting operation on 9 fuel ports, 8 of which had 2 seal welds to be cut, had taken 9 working shifts during the first refueling; in the second refueling it required 6 shifts to cut 10 fuel ports.

The machines had been modified for the second refueling, and the operators had received considerable training and practice in cutting weld mock-ups. The mechanism weld cutting operation, together with removal of motor rotors and installation of protective covers on the mechanism housings, was scheduled to take 12 consecutive working shifts. The actual weld cutting operations were done over a period of five calendar days, but the cutting was performed in only 11 of the 15 working shifts included in that interval. In actual operation the seal weld cutting machines worked reasonably well as a result of modifications. Such difficulties as did arise are identified mainly with (a) the parting operation which was still unsatisfactory, mainly because of the large variations in the depth and hardness of the welds; (b) screws that came loose because of vibration; (c) overfeeding that caused increased torque on the small tool bits which, combined with hard spots in welds, broke parting tools and cutters. It is considered that the feed control can be improved by being made finer and by intensifying training in the operation of these machines.

The two cutting machines could not be used simultaneously during mechanism cutting because visibility of and access to the cutting head from outside the mechanism array were limited by other components; therefore, access to the cutting head for insuring its proper operation for cleaning the chips and access to the machine body for controlling its operation requires the operator to be inside the array where there is not enough room for two operators and two machines with the torque restraint bar.

The chief difficulty with the thermocouple seal weld cutting machines was in the breakage of shear pins. The machines should be modified to eliminate this problem.

The experimental removal of two shrouds in air for the FEDAL bubble test demonstrated the safety of this technique and, therefore, was applied to all shroud removal and reinstallation. Because operations had progressed into the preparation for underwater shroud removal, the time saved during this refueling was minimal. In future transfers the time savings should be somewhat greater, because no prior preparations will be made for underwater removal.

D. Phase III - Replacement of Fuel Assemblies

The prevention of a critical condition during fuel exchange is absolutely necessary; no control of criticality by motion of the control rod is available during this operation. Therefore, control of the refueling operations is accomplished with exacting attention to detail during every step. The order in which seed assemblies were replaced started with location L-11 and moved clockwise around the seed region. The blanket assemblies were rearranged and replaced at the same time the seed clusters were exchanged. Figure 29 shows the underwater removal of a typical irradiated seed cluster from the reactor using the fuel extraction tool.

1. New Fuel Handling

When the reactor was ready for refueling, the appropriate seed assembly was removed from the rack to the fuel handling truck and transported to

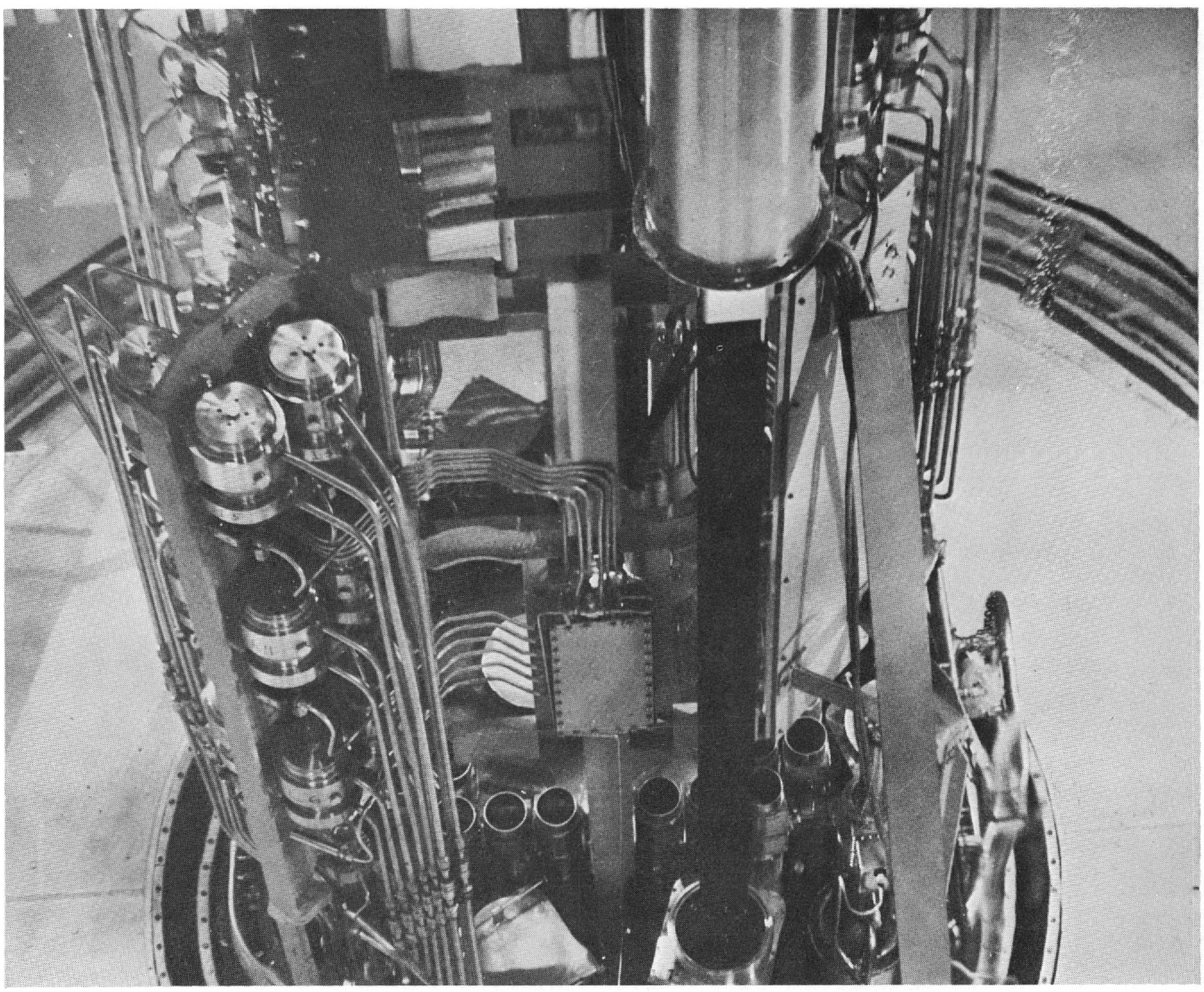


Figure 29. Removal of Seed Fuel Assembly from Location L-9.

the canal west walkway. The poison rods inserted in each seed during shipment to Shippingport had been replaced with a new Seed 3 hafnium control rod during inspection operations. The hafnium rod was temporarily removed and the assembly was inspected for cleanliness and to insure that no plastic from the shipping rod had been scraped off and adhered to the cluster. Spring scales were used on the hoists in all rod lifting operations, both to verify the weight of the rod and to indicate any hindrance to the passage of the rod into or out of the fuel cluster. The assembly was inspected again to insure that the rod was oriented properly within the cluster. A safety provision during all handling operations was that no more than two seed clusters, either new or spent, which did not contain either a poison rod or a control rod, could be present in the fuel handling building at one time.

The control rod installation was performed in air for the 30 new Seed 3 clusters which were equipped with new control rods, in contrast to the first refueling in which 31 irradiated rods had been installed underwater and only one new rod had been installed in air.

After the new Seed 3 cluster was inspected, the rodded cluster was then transported by the main crane to the transfer fixture in the canal where it was ready for insertion into the core by the fuel extraction tool. A safety provision of this transfer fixture was that it would hold only two fuel assemblies to prevent underwater concentration of fissionable material. Each assembly which was to be installed into the reactor was inserted only into the east section of the fixture, and each irradiated assembly upon removal from the reactor was inserted only into the west section, thus, assuring that no mixup of new and spent fuel would occur. All fuel assembly lifting operations used special engagement tools for that particular type of assembly; the spring scales on the hoists indicated whether the tool was or was not engaged and whether the fuel unit was moving freely. During handlings by the fuel extraction crane, strain gages on the tool head gave similar information. The fact that the new cluster contained a control rod was again verified at the extraction tool prior to insertion in the reactor by checking the load indicator on the tool console to be sure that the load on the tool was equal to the combined weight of fuel assembly and control rod.

2. Spent Fuel

Transfer of irradiated fuel units from the canal transfer fixture to either the spent fuel storage rack or the damaged fuel tank was performed under water, using the main crane 25-ton hoist, movable work bridge, and fuel handling tool. For personnel protection a minimum shielding water depth of 7 feet was required to be maintained over the top end of a spent fuel assembly. This was accomplished by using a protective spacer sling of fixed length between the hoist and the fuel grapple tool. The lifting arrangement was such that at the maximum lift the 25-ton hoist block would be at the upper travel limit and the attached chain hoist (for vernier control) at its upper limit. Then the length of the protective sling, the springscale, and the fuel handling (grapple) tool assured that there would be at least 7 feet of shielding water over the fuel unit.

The spent fuel storage rack in the fuel storage pit was used for irradiated seed assemblies and arranged so that proximity between assemblies could not be less than 24 inches. The damaged-fuel storage tank was arranged to provide maximum separation of the blanket assemblies stored in it.

3. Safeguards during Fuel Exchange

A review of criticality precautions and procedures was held with working personnel prior to a shift concerned with work involving possible movement of fuel or control rods.

The procedural provisions included the following requirements. Specific approval was required from the refueling shift supervisor before initiating any action involving movement of fuel or control rods in any fashion which could lead to a change in reactivity of the core. Since either seed or blanket fuel assemblies could be removed through a given fuel port, no blanket assembly position in the core, which could be reached with the refueling tool through a particular port, was permitted to be vacant while seed fuel assemblies were replaced through that port; this requirement insured that a new seed fuel assembly would not be inadvertently loaded into a blanket location in the core. An additional precaution was observed during the insertion of Seed 3 in that no blanket position adjacent to the seed was allowed to be vacant in the same quadrant in which seed material was being refueled. In order to avoid possible complications during shift changes of personnel, the new hafnium control rod must have been verified and the new Seed 3 cluster moved out of the transfer stand and inserted into the reactor by the same personnel on a given shift.

Nuclear instrumentation was used to monitor any change in reactivity conditions during the removal and insertion of seed assemblies. Four channels of source range instrumentation were in use to monitor BF_3 neutron detectors facing each of the four quadrants of the core. Four recorders were available in the power plant control room and two additional recorders were located in the console of the fuel extraction crane. Neutron count rate data from all four source range detectors were monitored on the control room recorders, while the repeater recorders on the refueling crane console displayed the count rate from the detectors closest to the refueling location in the core. Each recorder was equipped with visual and audible alarms with adjustable setpoints. Phone communication was available between the control room and refueling crane console so that the refueling supervisor and control room operator could confer in case of malfunctions.

The base count rate was determined at two different times. For the operations involving fuel exchange of the first 16 seed positions, the base count rate was determined immediately prior to the removal of the first depleted Seed 2 cluster from the core. The values were obtained from plateaus on discriminator curves for each instrument channel. A second base count rate, applicable for the remaining sixteen cluster locations of the core, was determined from discriminator curves obtained after completion of fuel exchange in the first 16 seed positions.

By changing the base count rate halfway through the refueling, an allowance was made for a normal increase in the count rate due to enhanced neutron multiplication as more new seeds were installed. The automatic warning system would have been activated if successive clusters were more reactive than anticipated and exceeded the setpoint of three times the base count rate.

In addition to the above safety precautions which were similar to those of the first refueling, new safeguard requirements were instituted to prevent the accidental withdrawal of a control rod from the reactor during operations involving removal of Seed 2 control rod drive mechanism parts and installation of Seed 3 parts. For disengaging a tie rod and scram shaft from a control rod, a mechanical stop was installed on the chain hoist in addition to a spring scale so that travel of the hoist hook was limited to 6 inches when it was engaged with the locked tie rod. Thus, if the tie rod were not actually disengaged from the control rod when it was thought to be free for lifting, the rod could not be lifted more than 6 inches. If the spring scale indicated more than a 95-pound load it was assumed that the tie rod and control rod were not disengaged and investigation was required. This condition did not occur at any time during the refueling. Subsequently, as a part of the physics test program after initial criticality of Seed 3, it was demonstrated that, even with the new seed clusters installed, the reactor would remain shut down with the control rod fully withdrawn from the most reactive cluster position.

4. Fuel Exchange

The following paragraphs describe the fuel replacement operations as they actually progressed.

The fuel extraction crane was moved into the reactor pit to location F-14, the nuclear instrumentation chart recorders were checked out and found to be operating properly, and blanket assembly F-14 was removed. It was noted that the compressive force required to latch this assembly with the extraction tool was 50 pounds higher than the normal 950-pound force. The assembly was removed without difficulty and, after having been deposited in the west fixture of the transfer stand, was further transferred to the blanket disassembly stand where it was later reorificed for reinsertion into the core at location K-9. Blanket assembly J-2 was then removed from the core with no difficulty and deposited in the west transfer fixture from where it was then moved for temporary storage in the damaged-fuel container.

Exchange of seed fuel assemblies through the outside fuel ports was initiated. There was some difficulty in closing the fuel port cover at location N-10, and opening the cover at N-6, because of poor visibility caused by turbulence of the reactor pit water. The underwater viewer was put into service to improve the visibility. Some difficulty was encountered in engaging the refueling tool with the new seed assembly for location L-6 in the east transfer fixture because the latch ring was not in the correct position. This was remedied; the fuel assembly was latched and inserted into the reactor with no further trouble.

For the removal of blanket assembly H-8 through fuel port H-7 it was necessary that the extraction tool enter this area by passing through the trellis structure. When removing blanket assembly H-8 and traversing the row of mechanism ports at the trellis east gate, the extraction tool strain gages registered an interference. It was found that the bottom of the H-8 fuel assembly had hit the mechanism port cap adapter at location H-12. The extraction crane was stopped immediately, and calculation of elevations indicated that there was about a 4-inch vertical interference between the fuel assembly and the mechanism port cap extension. It was found that due to a procedure deficiency these extensions had not been removed prior to the start of the fuel exchange operation. Therefore, the blanket assembly H-8 was returned to its position in the reactor, and the reactor pit drained about 6 inches below the opening of the mechanism port. Workmen were then lowered in a cage from the main crane, and the three mechanism port cap adapters were removed from locations G-12, H-12, and I-12. Three flat aluminum caps were placed over these mechanism ports. Removal of these extended adapters then provided adequate passageway through the east gate for the assemblies serviced by the fuel port H-7. While the water was lowered in the reactor pit, the occasion was taken to rotate the fuel port cover at F-14 so that it would operate without interfering with the d/p cell piping.

During the inspection of Seed 3 cluster for coordinate location D-9, preparatory to placing it into the canal transfer fixture, it was noted that a dummy neutron source holder had not been installed in that cluster. This was one of five seed clusters having cavities for neutron sources which were to be filled with dummy holders. Through oversight this had not been done. Four of the seed assemblies had already been installed in core locations I-8, I-6, D-6, and F-5 before the absence of these dummy source holders was noted. It was decided to place a dummy source holder in the D-9 Seed 3 cluster only, and not to remove the other four clusters for dummy holder insertion because it was determined that operation of the reactor would not be affected by these omissions.

5. Evaluation of Phase III (Fuel Exchange)

The fuel transfer operations progressed very smoothly. The time devoted to the fuel transfer was reduced from more than 10 days to 3-2/3 days. This is primarily the result of the relocation of the transfer fixture and the use of 30 new control rods which allowed the rod installation to be done in air. The only significant delays were due to (1) the necessity of draining the reactor pit in order to remove the long mechanism port cap adapters which interfered with the removal of the center blanket assembly, and (2) the short time required to arrive at a decision concerning the insertion of seed assemblies without dummy source holders. There was no trouble with the fuel extraction tool and crane.

Some of the long-handled manually-operated underwater manipulating tools gave trouble, such as the orifice tool and the rod removal tool. These tools employ latching and locking devices which must fit into precisely dimensioned mating devices of small areas, without lubrication, and with very little allowable play or backlash. The operation of the tools often depends upon feel rather than close visual observation. These tools had received comparatively hard usage during the training and check-out programs so that by the time they were placed in actual use for refueling some wear and strain on lugs and latches had occurred.

New blanket assemblies for locations J-2 and F-14 were equipped with orifices appropriate for the flow conditions of blanket region IV. In transferring the old blanket assembly from location F-14 to K-9, a new orifice was required for flow conditions in blanket region II. There was some minor difficulty with the lifting lugs being worn on the orifice removal tool, which required repair in the machine shop. This same tool gave trouble in the reorificing of the blanket assembly removed from location J-2, when a lug on the outer tube broke off due to an excessive load applied during orifice removal. Machine shop repairs were again required.

After removal of spent seed assembly K-5, the control rod was removed and placed in the control rod storage rack. A poison rod was then inserted into the spent seed. Later, when picking up the irradiated control rod, the rod fell off the underwater handling tool to the floor of the fuel storage pit. It was retrieved with a snare operated from the long-handled general purpose tool. To determine the reason for this accident, later tests were conducted using the hafnium rod handling tool. Numerous trials were conducted. On two occasions the tool was obtensibly locked to the adapter, but the control rod fell off the tool when the tool was shaken a little. A tool redesign will be made to eliminate the combination of clearance and tolerances condition which caused this problem from occurring again.

Tools worn during training operations should be repaired after training is complete so that the tools are ready if required during the refueling operation.

E. Phase IV - Replacement of Reactor Nonfuel Components and Welding of Omega Seals

After the refueling crane was removed from the reactor pit, the pit gate installed, and the reactor pit drained and cleaned, such working equipment as the work level grating, the lead shot and mechanism tube shielding, and tool control barriers were reinstalled in the pit. Then, before any components were reinstalled, reactor internals were inspected through the ten fuel ports, using an underwater light and binoculars. All visible surfaces and joints were examined carefully for wear, cracks, corrosion, crud deposits, foreign material, and general condition. No defects were observed and the color and condition of parts appeared excellent. Very slight wear marks were visible on the core grid at the intersection of locations I-4 and J-4 where the BEWI assembly engages the grid.

The ASEWI's at J-14 and N-10 were installed without incident. The BEWI's at J-2, H-7, and H-9 were installed and seated, but at two of these locations the No. 2 sliding tubes stuck and required force to free them for movement to their proper positions. Measurements taken to the top of the BEWI to the fuel port weld lip were within the specified 0.008 inch of the reference 5.569-inch dimension; therefore, the devices were considered seated.

Shroud installation produced several minor difficulties. At J-4, after about four feet of entry, the shroud did not move freely. By manual manipulation the shroud entered a little further and then its weight caused it to continue more easily. A slight bulge in the shroud tube may have been present. On installation of shroud D-6, the flange was found to have been nicked in two places so that the burr would not allow the shroud to seat. Upon removal and further inspection, the lifting bail was found bent at the lugs which engage the shroud. Repairs were made to the damaged parts and this shroud was installed satisfactorily. During the investigation for cause of the difficulty, the flanges on one end of one of the canal storage racks were found bent upward. The damage to the shroud rack indicated that the shroud had been lifted vertically before moving it out of the rack in a horizontal direction. These shroud assemblies were installed dry, similar to the removal method. All 32 shrouds and scram shafts were installed concurrently with the installation of some of the BEWI instrumentation.

The thermal barriers, bushing retainers, lockwashers, thermal barrier lockplates, holddown nut retainers, holddown nuts, and the lock rings for each of the 32 control rod mechanisms had been inspected and all were found suitable for continued use. Blind plugs were made available for insertion at all six SMI locations. At this time, the new BEWI containing six new replacement thermocouples was installed at location B-6. Its tubes were lowered into place and seated.

As a result of the reinstallation of the previously removed BEWI and ASEWI assemblies, plus the installation of the new BEWI at fuel port B-6, the arrangement of exit water instrumentation for Seed 3 operation resulted in BEWI's being installed at fuel ports B-6, J-2, H-7, and H-9, and ASEWI's at locations J-14 and N-10. This left fuel ports F-2, B-10, F-14, and N-6 to be equipped with blind plugs. The relative arrangements of the BEWI and ASEWI installations for Seed 3 and Seed 2 operations are shown in Appendix B. The four blind plugs were installed without difficulty. In the instrumented fuel ports, closure pieces and closure nuts were installed. At location N-10, the closure piece would not fit completely, because the ASEWI was not completely seated. The ASEWI assembly was lifted and re-engaged in the keyway, reseated, and then the closure piece was installed properly.

Control rod mechanism parts, including shroud locking rings, holddown nuts and retainers, plus thermal barriers were installed. Those thermal barriers for locations containing SEWI and SMI contain holes in the fins through which the thermocouple tubes pass. For the six locations which also formerly contained SMI, the original thermal barrier inserts containing SMI T/C tube holes were installed. Thus, 16 of the 32 thermal barriers installed were of the instrumented type and 16 were plain type barriers. Some difficulty was encountered at the instrumented location J-4, because the shroud locking device had not been inserted far enough into the mechanism housing for the barrier to seat properly. Once this condition was corrected, the thermal barrier was installed without further difficulty. All instrumented barriers were verified as they were installed to ensure that the thermocouple cables would pass through the tubes in the barriers satisfactorily.

The 16 SEWI thermocouple assemblies, containing two thermocouples each, were installed at the 16 locations. The thermocouple at location K-11 stuck about 3 feet above the seated position. Upon investigation, it was found that the spring sheath was damaged. The thermocouple was placed in storage, and a spare thermocouple assembly installed. The unit at location E-11 likewise could not be seated. After a third unsuccessful attempt, a thorough investigation of drawings and dimensions revealed that the new thermocouple for this location was longer than the Seed 2 thermocouple; it was replaced with a spare.

Installation of shroud locking devices was completed with few difficulties. At D-10, F-4, and L-9, the locking ring and holddown nut did not seat properly at the first trial. It was necessary to raise the shrouds in order to locate the difficulties. At D-10 a large burr was found on the shroud key; similarly, small burrs were found on the key at F-4. The burrs were removed, both shrouds were reinstalled, and locking devices properly seated. At L-9 there was no visible damage to the seating components, and a little extra pressure successfully seated the parts.

Mechanism rotor assemblies were reinstalled, all tie rods locked into place and verified, hydraulic damper guides installed, and the washers crimped. During the inspection of damper guide nuts, the one from location J-4 was noted to have slight corrosion marks on one edge; the nut was replaced. Several other damper guide nuts showed evidence of corrosion but were not replaced, since the corrosion was considered to be very minor. New tie rod sleeves were installed at L-6 and D-5 because they did not fit onto the lead screws as specified.

All 32 control rod mechanism motor guide tubes were installed with no difficulty. A new guide tube was installed at location D-6. During this operation, the trellis east gate was installed and the trellis support structure removed.

It was found that certain thermocouple blind plugs for the six SMI locations could not be inserted to the proper depth in the housings. The housings have a nominal diameter of 0.520 inch for a depth of about 1 inch and then decrease to a diameter of 0.501 inch in the lower portion. The plugs are stepped to fit these bores, but the lower plug diameter was too large to fit into the 0.501 inch portion of the housing bore. It was determined that the bore was slightly undersize, probably due to slight shrinkage after previous weldings; the outside diameter of the plug was at the maximum of its tolerance, hence the interference. The procedure adopted for fitting the plugs in the ten SMI holes at locations K-5 and L-8 was to cut off the plugs so that they would fit entirely within the 0.520 inch upper portion of the bore, and still leave enough of the plug projection above the surface of the housing so that the welds could be made and the thermocouple seal weld cutting machine properly installed, if at some future time the welds are required to be cut.

1. Omega Seal Welding

Seal welding of the fuel port omega seals together with the necessary repairs and inspection was accomplished in about 44 working hours over a span of 5 days. The seal welding machine and power supply are shown in Figure 30. The seal welding of the fuel port omega seals started with the welding machine being set up on fuel port J-14, immediately after the welders had qualified by welding three mock-ups satisfactorily. The inner diameter weld on J-14 was started, but it was found that the wire feed was set too near the inner lip when the machine arc was supposed to be applied to the outer lip. An uneven surface was being produced. The machine was stopped, readjusted, and the weld completed. A small hole was caused by the stopping and restarting of the welding machine; it was easily hand repaired. The outer diameter joint was welded with no difficulty.

The machine was set up on blind plug F-14 and at the end of the weld a blow hole occurred. It was determined that moisture in the blind plug of F-14 caused the blow hole. In order to reduce the possibility of vapor affecting the welding, the reactor water level was lowered another 6 inches. In order to repair F-14, strip heaters were installed around

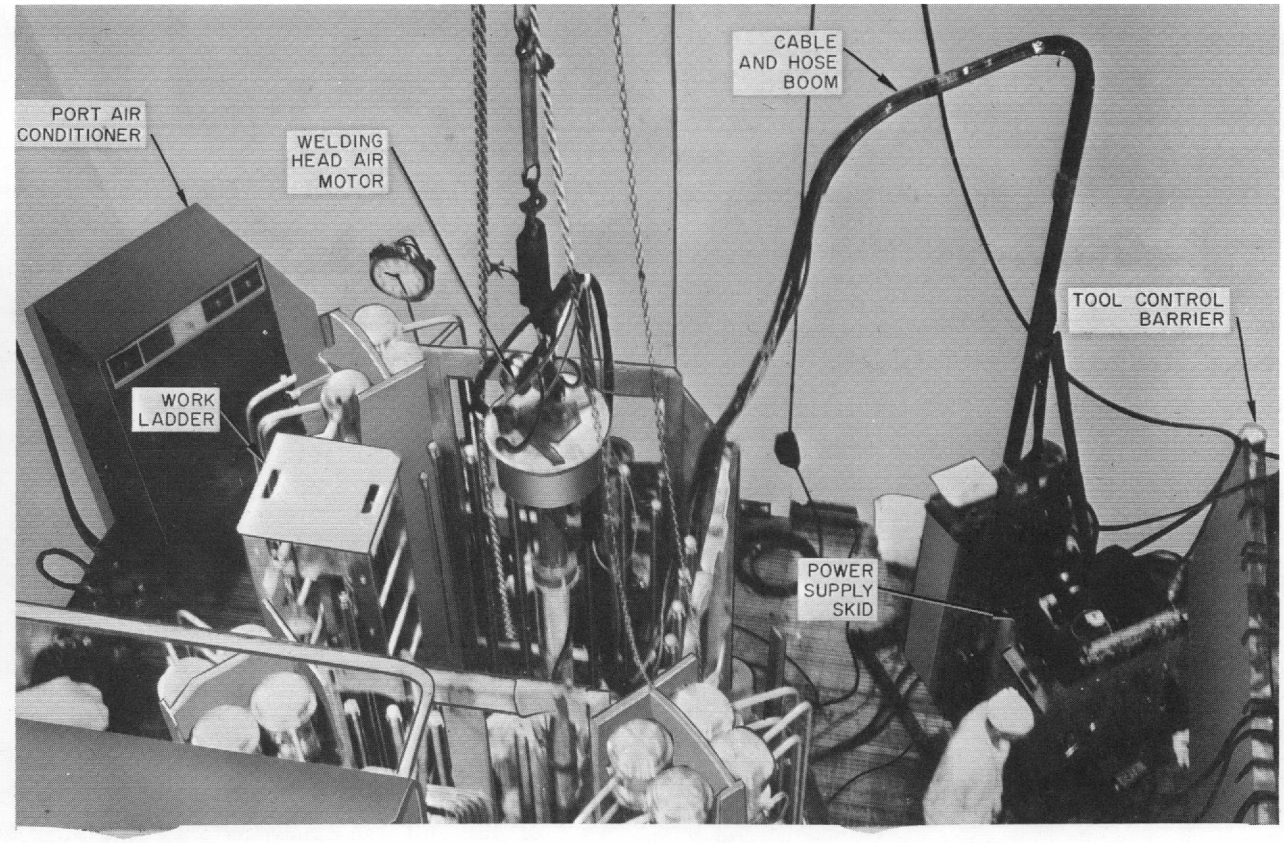


Figure 30. Omega Seal Welding Machine in Operation.

the fuel port housing to dry out the excess moisture on the underside of the blind plug. Other fuel port plugs were then removed and dried.

Of the sixteen omega seal welds made on the ten fuel ports, hand repairs were required for five. The welder qualified by performing a mock-up weld which had a defect similar to the actual one. The welding of these fuel ports demonstrated that the welding machine performed satisfactorily. Most of the defects were caused by the presence of moisture in the weld area.

Just prior to welding the mechanism omega seals, several mock-ups were welded. Immediately after two consecutive good welds were obtained on mock-ups, actual omega seal welding was begun. Of the 32 welds, 6 required manual repairs with either filler metal or grinding of spots, rewelding, or in some cases several dye-checks. Four of the weld preparations had wide root gaps.

At location D-6, which had given so much trouble during the cutting operation, the root gap was about 0.058 inch. Before welding on this particular unit, a mock-up was welded which had a root gap of 0.035 inch. On the D-6 weld, using the welding machine, a length of about 3 inches on the circumference of the outside seal weld lip did not completely fuse at the surface. The weld was manually repaired. After dye-checking, some small defects were still visible. These were ground out, repaired again, and the weld was finally acceptable. Root gaps on joints at E-5, F-4, and J-4 averaged from 0.067 inch to 0.070 inch. These three joints were prepared for welding by using a copper bar and gently hammering the lips inward to decrease the gaps to a maximum of 0.043 inch. E-5 and F-4 were welded satisfactorily but repairs were required in one area of J-4. The mechanism tube for location I-12 had a very thin weld lip in one area due to poor penetration at the first refueling;

therefore, a special mock-up was made of this joint, and the actual welding was performed satisfactorily. At location L-7, inspection showed a skip in one area and hand weld filling, repair welding, and redye-checking were required. At location L-19, a high bead in one quadrant was ground and repaired.

At the beginning of the welding operations on both the omega seal welds and the thermocouple plug welds, it was noted that drafts from the air conditioner, located in one corner of the reactor pit, could cause arc instability; therefore, the air conditioner was turned off when welding was being performed in those areas where air currents could interfere.

2. Thermocouple Plug Welding

The thermocouple blind plugs were welded concurrently with the mechanism seal welding using the manual Heliarc (TIG) process. The fitting of the SMI blind plugs has been discussed above. In preparing to weld these plugs, it was decided to work alternately between plugs at several locations, welding only one at a time in a given place, so that the heat from the welding would have a chance to dissipate in the thermocouple housing. Thus, by alternating the welding between locations, no time was lost while waiting for the housing to cool sufficiently to prevent damage to the thermocouple insulation. Also in order to save time, it was decided not to dye-check at a given location until all plugs in that location had been welded.

The 16 SEWI plugs were welded by making root passes and dye-checking, then final weld passes and dye-checking, and, if necessary, any grinding and repetition of dye-checking. In only one case was a root pass required to be ground and dye-checked a second time. However, only two of the sixteen welds had their final passes completed and dye-checks accepted without reworking. Nine of the welds required one grinding operation on spots which had shown dye indication. Three of the welds had to be ground twice, one three times, and one four times.

In the welding of the BEWI thermocouple plugs, all fifteen root passes were made and accepted on a first dye-check with the exception of the "A" plug at location H-9, where the thermocouple sheath became pinched during the seating of the adapter and closure piece. This broke the thermocouple wire and required the removal of the assembly. A complete new thermocouple was obtained from Bettis Laboratory and seated. The new thermocouple plug was welded without difficulty. In BEWI location H-9, thermocouple "E" required grinding of a small welding defect and, upon second dye-check, was accepted.

During the welding of the ten ASEWI plugs the root passes at location J-14 were accomplished satisfactorily, but four of the five plugs at location N-10 required grinding out of dye-check indications in the root pass. In welding the final pass on thermocouple "K", the thermocouple wire was inadvertently burned through, destroying the usefulness of the unit and requiring its removal. The parting of the just-welded plug weld was difficult, even though the cutting machine undercut 3/16 inch. A spare was obtained for installation by using an SMI-type thermocouple modified to fit this ASEWI location.

The thirty plug welds required at the six SMI locations were made concurrently with the other instrumentation plug welds. Root passes required grinding of defects on the plug welds at locations K-5 and F-4. On the final weld passes, only E-11 was able to have all five of its plugs acceptable on the first dye-check. At J-12 all five plugs required some grinding of defects and were accepted on the second dye check. At three of the remaining locations two plugs at each location required repair welding, and on the other five locations (K-5) three of the plugs required repair welding.

3. Evaluation of Phase IV (Reinstall Internal Nonfuel Components)

All operations in Phase IV were completed expeditiously. The dry installation of shrouds, BEWI, and SEWI was unquestionably faster than if performed under water. The machine welding of mechanism and fuel port omega seals was not delayed because of any difficulty with the machines. Such troubles as did occur were mostly with the condition of the joints themselves, wide root gaps, and the presence of moisture in the weld areas. Very little time was taken, as in the first refueling, for welding numerous mock-ups immediately prior to making individual welds. The welder training program had been extensive, and practice welding on mock-ups had been done concurrently with other work in prior stages of the refueling. Thus, only a few successful mock-up welds were required immediately prior to initiating the mechanism and fuel port welding. In two cases special mock-ups were made because of the condition of the weld lips on one mechanism seal and on one fuel port.

For the seal welding machine, the redesigned boom and the rearrangement of the cable bundling and its entry to the welding head and drive motor assembly accomplished these improvements:

a. New parameters permitted welding of a larger root gap than previously.

b. Permitted the power supply to remain in one place for all omega seal welds on the reactor.

c. Freed the main crane for other work.

d. Reduced the welder set-up time considerably.

However, the wire feed, although its constant speed was verified before starting each weld, tended to position the wire in a path not always equidistant from the two weld lips. Therefore, the consistency of the weld varied somewhat, producing areas which were thinner or thicker than the objective, and requiring manual repair welds in some cases. The machine should be modified to eliminate the wire feed mechanism or at least revise it to prevent recurrence of the above problems. The TIG welding process produced better quality welds which required less grinding and dye-checking.

F. Phases V and VI - Reinstallation of External Components, Hydrostatic Test, Initial Criticality, and Return to Power Operation

After completion of seal welding, over-all control of refueling and plant operations was shifted to the operations group in the control room, since many jobs and plant operations had to be completed in the reactor primary and secondary systems in order to fill, vent, and hydrostatically test the plant. Since refueling progress was 10 days ahead of schedule, several of the power plant work items were still in progress. These were, chiefly, the installation of a main coolant pump casing (volute) in the 1D loop, the retubing of a feedwater heater, the connection of new steam and feedwater piping for the heat dissipation system being constructed for use with Core 2, the installation of repaired main steam manual stop valves, the testing and checkout of the newly installed Data Acquisition System, and the performance of numerous periodic test programs. Work on the over-all power plant was intensified and personnel were reassigned from refueling to the station activities effort as rapidly as practicable. The personnel remaining in the refueling group were able to complete installation of external reactor components during these plant evolutions with no delay in achieving initial criticality.

1. Completion of Plant Maintenance, Venting, and Hydrostatic Testing

After completion of the seal welding on the reactor vessel head, some work on primary and secondary systems remained before leak testing could be done. This included completion of welding and weld repairs on main steam stop valves and heat dissipation system tie-in connections in the secondary system, line up of primary water charging and discharging systems, and completion of the sealing of the 1D main coolant pump which had just been installed. As noted in 3 below, when it became apparent that the 1D loop could not be sealed up in time to prevent over-all schedular delays, it was decided to perform the hydrostatic test on the other three loops first, and to test the 1D loop after initial criticality.

When the plant was ready for leak test, the water level in the reactor vessel was raised to the top of the mechanism rotor tubes, vent valves on rotor tube E-11 and the FEDAL sampling line were closed, and the entire primary system was vented. With the reactor vessel under primary water storage tank head pressure of about 35 psi a visual inspection was made for leaks at all of the seal weld areas. All fuel port, mechanism, and thermocouple plug welds were inspected and no leaks were detected. A small leak was found at one of the d/p cell vent valve plugs which was tightened to stop the leak.

The primary system was pressurized to about 500 psi, sufficient to operate the main coolant pumps, and the primary coolant was circulated for at least an hour to obtain representative samples from the purification system for gas analysis. At a pressure of 450 psi, and at a temperature of 172 F, no leaks were found in any of the reactor vessel head and reactor plant welds. As the pressure increased to 1200 psi, the welds were again inspected and found without leaks.

A pressurizer relief valve lifted at 1465 psig due to improper adjustment and was gagged. The pressurizer was isolated at 2000 psig.

Another check was made at a system pressure of 1800 psi, where a very slight leak was noted through two thermocouple sheaths at BEWI location H-7. These sheaths were sealed with welded plugs after the test.

At about 2650 psig, leaks were found around the bolting ring of the FEDAL pump, the flange of the 1C main coolant pump, and at a BEWI thermocouple plug at location H-7. The FEDAL pump required a gasket replacement and subsequent leak test. The 1C main coolant pump required that the flange bolts be retightened. The primary 1C coolant loop was subsequently leak tested with no leaks occurring. The final hydrostatic test pressure of 2750 psi was attained with no other leaks being found.

Concurrent with the above leak tests the plant pressure instrumentation was calibrated at various pressure levels. Also, the secondary (steam) sides of the steam generators and the main steam lines, up to the manual main steam stop valves, were given a hydrostatic test at 1150 psig as the stop valves had been removed for repair and rewelded. Several retests were necessary before minor leakage problems were corrected.

Leak rates of the four primary plant relief valves were measured both prior to and following initial criticality. As a result of these tests it was determined that the repair made on these valves was satisfactory. Leak tests on the remaining two relief valves had been performed prior to the refueling cooldown. A similar leak rate test was performed on 3-way selector valves in the valve-operating system. Although the leak rate tests were successful they had to be repeated several times before final leak rates could be established.

After the above leak tests were completed the reactor chamber dome was installed, the control rod drive mechanisms were tested to insure proper operation, the protection and control systems were checked and the reactor was then brought critical with Seed 3 installed on October 9, 1961.

2. Parallel Operations on the Reactor Vessel Head

While the plant was being prepared for leak testing, the mechanism motor stator-position-indicator coil assemblies were installed on the 32 mechanism guide tubes. During the assembly it was found that in five locations the motor guide tubes had not been welded in the same circular orientation as they had been prior to their removal. Thus the match marks did not line up and these stators were about 2 degrees out of circular alignment. Because of the horizontal spacing between stator water jackets, and the nesting required to maintain clearance between them, these five stator water jackets and P.I. coil assemblies had to be removed and disassembled. Then new lockwashers were installed and crimped in new positions such that the stators could be reinstalled in their proper nested positions.

The floor grating and all portable equipment were removed from the reactor pit and the pit was cleaned. The lead shielding was removed from the reactor vessel head and the refueling seal bolts were loosened. In this process six bolts were found exceptionally tight and, on removal, the threads on the bolts and on the flange were damaged. The flange threads were retapped.

The electrical cable support ring and mechanism cooling water lines were installed. Part of the thermal insulation was placed in position on the reactor vessel head.

After the reactor plant leak test was completed, the vent plug on motor tube E-11 was then welded into place. The electrical cables for the d/p cells, mechanisms, thermocouples, and FEDAL valve were installed. The reactor chamber dome was readied for installation. The reactor was ready for the d/p cell calibrations a week ahead of schedule.

During the calibration operation, electrical checks were made on all of the connections between the control room and the mechanisms, P.I. coils, and instrumentation. The stator of mechanism motor D-5 was found grounded, and three P.I. coils in the assemblies at K-4 and G-4 locations were found open circuited. These findings required the removal of the mechanism holddown structure, removal of the defective assemblies, and replacement with spare stator and P.I. coil assemblies. Also at this time, the operable thermocouple at location H-7 was broken. The BEWI instrumentation is operable for Seed 3 at locations J-2, H-9, and B-6.

The mechanism and multiport valve cooling water lines had been installed and leak tested at 165 psi, but retest was required after the above electrical trouble. On retest, one of the lines leaked at the component connector. It was found that the connection to stator K-4 had not been tightened. The southwest quadrant of the reactor vessel head had been sprayed with water but subsequent resistance readings taken of all stator windings disclosed that none of them had been wetted. Some of the thermal insulation had been soaked, requiring removal and drying. Heat lamps were set up to dry the general area.

The reactor chamber dome was reinstalled on October 6th.

3. Main Coolant Pump 1D Installation

The installation of the 1D loop main coolant pump was one of the major plant work items. Welding of the casing into the main coolant piping by the contractor was a difficult process in very restricted quarters. The 1D loop could not be included in the hydrostatic test of primary systems until this welding had been completed and the motor-impeller assembly installed and bolted tight. Because of this the remaining three primary coolant loops and the reactor vessel had to be given a separate hydrostatic test prior to initial criticality.

The 1D pump and loop were not given a hydrostatic test until the day after criticality, since several extra days had been required to radiograph the welds and replace several of the power supply cables. Upon hydrostatic test (2750 psig), leakage was discovered at the motor-volute flange of the 1D main coolant pump. This required draining the piping, unbolting and lifting the motor, and inspecting the flanged surface. Inspection of the pump flange and gasket surfaces revealed that (1) superficial radial scratches of varying lengths existed across these surfaces and (2) that a three to four mil raised edge, approximately 1/16 inch wide, existed on the inner edge of the volute flange. As a result of these findings it was concluded that the leakage was due to the marginal condition of the flanged surfaces; to correct these conditions, the motor-volute flanged surfaces were stoned, a new flange gasket was installed, and the pump was reassembled.

A subsequent hydrostatic test also gave a slight indication of flange leakage at a test pressure of 2750 psig. To eliminate this minor leakage the pump bolts were tightened further.

Another 2750 psig hydrostatic test gave indications of air-water mixture bubbling at the gasketed joint. The bubbling initially covered an arc of approximately 40 degrees. However, by the end of the hydrostatic test period the bubbling had contracted to only one small bubble. The bubbling was attributed to entrapped air located between the pump labyrinth seals and the gasketed flange joint. In order to vent the air the pump was placed under a pressure of approximately 2000 psig for eight hours. The pump was then subjected to a pressure of 2750 psig for one hour and finally hydrostatically tested at this pressure for 1/2 hour with the loop temperature at 185 F.

The final hydrostatic test was considered satisfactory, since no evidence of leakage existed at the gasketed joint. These difficulties, however, prevented the 1D loop from being ready for service until October 19.

4. Checkout of Reactor Protection and Control Systems

Tests of the Reactor Protection System were conducted in order to assure proper and acceptable time response characteristics of all the shutdown and rod insertion circuits. Although these tests are run periodically, these particular tests were used to determine the possible effects of circuit changes which were made in the nuclear instrument system during the refueling and they were also used to assure proper operation of all circuits at the new setpoints which were established for Seed 3 operation.

During the checkout and testing of the Reactor Protection System, it was found that some of the bistable magnetic amplifiers were firing at a level lower than permitted by the current setpoints. The location and correction of the difficulty and the rechecking of response times required considerable time and effort.

The Reactor Protection System tests also revealed an excessive time response on the loop status shutdown circuit. The function of this protection circuit is to avoid core thermal damage by scramming control rods in a predetermined minimum time upon loss or reduction of main coolant flow. Additional tests were prepared and run in an effort to resolve this problem and the additional testing showed that the pump power relays were reclosing for short periods of time under certain conditions of bus loading. The problem was resolved by temporarily reducing the intentional time delay of the pump timing relay.

During the course of the last stages of system checkout, on the day criticality was achieved, the scram breaker would not close properly, although it had operated satisfactorily previously. Trouble in the scram circuit breaker was traced to a defective "X" relay. A spare breaker was substituted so that criticality was not delayed. The spare breaker was found to have improper auxiliary contacts to provide cold water accident protection for the reactor; this was corrected promptly. The normal scram breaker was repaired and replaced three days later when plant conditions permitted.

Nuclear instrumentation gave periodic trouble both during refueling and during the stages leading to initial criticality and to power operation. Three distinct defects were responsible for delays on numerous occasions: (1) deterioration of the nuclear instrument cables, (2) defective source range BF_3 detectors (3) defective electrical connectors to these detectors.

The nuclear instrument cables were not well shielded and spurious signals were caused by electrical pickup from entirely dissociated devices. For example, when a certain alarm bell circuit was closed, a transient was introduced into nuclear instrumentation circuitry and appeared on the recording charts as an increase in neutron count rate. Such erroneous information proved misleading to the operators. The isolation of this difficulty was demanding on both manpower and time.

There are four BF_3 detectors; several of these required replacement on more than one occasion. The connector troubles were mostly in the vicinity of the reactor vessel neutron shield tank, and were judged to be the result of the combination of high radiation field, high temperature, and mechanical failures. Initial criticality was delayed while troubleshooting was performed on these detectors and cables. During later testing of the core negative temperature coefficient, rod worth, and excess reactivity, the test program was delayed for 12 hours while trouble in Channel B was traced to a source range detector, and the detector replaced. Still later, as the plant was being brought up to full power, shutdown was required and there was a delay of 16 hours to find the reason for erratic indications of power in the nuclear instrumentation power-range Channel C circuitry. Two replacements of compensated ion-chambers (CIC) were necessary.

The control rod drive circuitry developed difficulties which resulted in rod groups being withdrawn in unequal increments under automatic programming control. It was found that errors had been made in reconnection of wiring during the shutdown. Nuclear physics testing was interrupted for about eight hours while this problem was investigated and corrected.

5. Test Programs

In spite of the difficulties enumerated above, the tests of plant performance were conducted carefully and each test proved that the plant would operate with reliability and safety as predicted. The major tests at initial criticality were:

a. Initial Approach to Criticality.—This test was conducted in order to bring the cold reactor core to a condition of criticality in such a manner as to avoid nuclear hazards. In this test the control rod heights required for criticality were determined and nuclear instrument response characteristics were obtained.

b. Control Rod Drive Mechanisms.—Because disassembly and reassembly work had been performed on the rod drive system, it was necessary to assure that the direction of motion, latching and unlatching, position indication, and other operating characteristics of the mechanisms and their instrumentation were according to expectation immediately prior to reactor startup.

c. Calibration and Intercomparison of Control Rods.—This test, also conducted during the preoperational period at ambient temperatures, measured the reactivity symmetry of the core, the rod worths, and the critical heights for individual rods. In this test the core reactivity was measured with each rod individually in the fully withdrawn position. It was demonstrated that the reactor remains shut down in this condition.

d. Control Rod Positions for Criticality.—This operation determined the critical rod bank height and the reactivity worth for several rod configurations at both ambient and operating temperatures. From this information, the excess reactivity and temperature defect of reactivity of the core were determined.

e. Coefficients of Reactivity.—The temperature and pressure coefficients of reactivity were measured at zero power in order to compare these Seed 3 characteristics with calculations and with similar data for Seeds 1 and 2.

Other tests performed were:

Calibrate Core Flow Instrumentation

Calibrate Primary Plant Differential Pressure Instrumentation

Calibrate Primary Plant Pressure Instrumentation

Calibrate Primary Plant Flow Instrumentation

The rod control and nuclear instrumentation systems were checked out.

A newly installed console, located in the data acquisition center, provided in one place all of the special indicating and recording instruments required for collecting the data needed for the interpretation of the nuclear physics tests. This equipment saved time which otherwise would have been required for connecting separate instruments for each test, and kept to a minimum the number of people required in the control room. The new data logging and acquisition system was also helpful in reducing the time and manpower otherwise needed for manually logging the large quantities of data acquired.

The plant integrity was verified, the refueling and maintenance program completed, and full power output was achieved on October 24, 1961.

6. Evaluation of Phase V and VI Operations

Some of the more significant lessons which were learned during Phase V and VI operations which should be factored into the planning for future refueling shutdowns are as follows:

a. The fact that major plant maintenance was not completed when the reactor vessel was ready for hydrostatic test delayed the refueling. In the future, effort should be devoted to completing major maintenance before completion of the last reactor vessel head seal weld in preparation for hydrostatic test.

b. The malfunctions on nuclear and protective instrumentation caused delays in the refueling program. The basic problem was that these systems were not checked out soon enough in the schedule, so that when difficulties arose they caused delays in the entire test program. It is suggested that in the future all nuclear instruments and protective circuits be tested to the maximum extent possible prior to completion of the last seal weld (about 35 days after the start of refueling).

c. Specific steps have been or will be taken to minimize the difficulties with the nuclear instrumentation systems:

   1) Nuclear Instrument Cables. A short length of replaceable cable has now been provided between the shield tank connectors and the terminal block. These cables should be checked for insulation resistance and noise, and replaced if necessary immediately after refueling begins (source range cable only) and also shortly before the last reactor seal weld is completed (all cables).

   2) Source Range BF_3 Detectors. Before installing the three spare BF_3 elements in each detector at the beginning of refueling, these spare elements should be tested for proper insulation resistance, sensitivity, and plateau level.

   3) Cable Connector. The latest available cable connectors, which have been designed to eliminate radiation-induced deterioration, should be installed at all locations.

d. One leaking weld in a d/p cell was detected subsequent to the refueling during Seed 3 operation and caused instrumentation difficulties during Seed 3 operation. Before completion of the next refueling, instrument terminal boxes should be checked for the presence of water.

e. Extreme care should be exercised when performing operations on the reactor vessel head during phases V and VI to prevent damage to electrical cables and the relatively fragile thermocouple cables.

f. Faulty electrical connections were found on two control rod drive mechanisms after installation, requiring their removal and repair. Since mechanism electrical connections were checked electrically just prior to installation on the reactor, the above failures were probably caused during reinstallation. Personnel will be cautioned in subsequent refuelings to exercise care in reinstalling cables and P.I. coils to prevent damage.

G. Chronology of Refueling Operations

Table II presents the chronology of major events during refueling. A more detailed chronology of events is illustrated on Figure 31 which follows.

TABLE II
CHRONOLOGY OF MAJOR EVENTS

Date (1961)	Refueling Sequence Event	Parallel Operations
	Preparatory Phase - Station Shut Down	
August 14 (Monday)	Radiation survey of reactor vessel head. Reactor operated at power for crew training then shut down for refueling. Exercised control rods for radiation test data.	Installation of loop 1D coolant pump, heat dissipation system (future use), data acquisition and logging system, and plant overhaul work in progress. (continued throughout refueling).
August 15	Plant cooldown started. Flow distribution across core determined at 300 and 200 F. Drained reactor pit.	Layup of station systems for refueling.
	Elapsed Time - 2-2/3 days	
	Phase I - From Removal of Dome, Reactor Head Components to Venting Reactor Vessel	
	Refueling Sequence Steps 1 - 18	
August 16 (4:00 p.m.)	Refueling pit operations began. Water cleaned from pit floor. Reactor chamber dome removed.	Drained steam systems. Station construction, overhaul, and major maintenance programs in progress.
August 17	Electrical cabling reactor vessel head thermal insulation, holddown structure, cable support ring removed; mechanism cooling water lines drained.	Seed inspection and measurement stand installed in fuel storage pit.
August 18	Refueling seal installed. Lead shielding installed. Mechanism position indicator coils, coil housings, stator water jackets removed as units.	Reactor pit stairway installed. Trellis support structure installed and trellis east gate removed.
	Elapsed Time Comparison: Second Refueling - 2.8 days First Refueling - 17 days	
	Phase II - Seal Weld Cutting and Removal of Control Rod Drive Mechanisms, Instrumentation, and Shrouds.	
	Refueling Sequence Steps 19 - 82	
August 19 (Saturday)	Installed reactor pit work level grating. Established primary coolant level in reactor. Seal weld cutting machines in place on two fuel ports. Removed SMI upper tube assemblies.	Established tool control barrier.

TABLE II (Cont)

CHRONOLOGY OF MAJOR EVENTS

Date (1961)	Refueling Sequence Event	Parallel Operations
August 21 (Monday)	Vented reactor by cutting mechanism E-11 vent plug seal weld. Cut and removed fuel port blind plugs F-2, B-6, Established bench marks for fuel extraction tool. Began cutting mechanism seal welds and T/C weld plugs.	First three Seed 3 fuel assemblies received. Installed test rig for instrument terminal box air pressurization test. Cleaned deep pit.
August 22	Seal weld cutting of mechanisms and thermocouple plugs continued; removed two motor tubes and one rotor.	Deep pit filled. Decontaminating two fuel port blind plugs (F-2, B-6). Nuclear instrumentation being checked out.
August 23	Mechanism seal weld cutting continued. SEWI and BEWI thermocouple weld cutting suspended to concentrate on mechanisms.	Recirculating water in deep pit. Fuel handling truck modified to improve use for receiving seed clusters. Decontamination of 4 mechanism rotors started.
August 24	Mechanism seal weld cutting continued.	Jib cranes at south and north ends of canal tested. Four mechanism motor tubes decontaminated. Filled fuel storage pit. Source inspection stand installed in dry pit.
August 25	Resumed cutting thermocouple welds 1st shift. Mechanism seal weld cutting completed 11:30 p.m.	Lead screw microscope inspection stand installed; rigidity to be improved. Nuclear instrumentation check out continued. Resistance testing of electrical power and T/C cables, mechanism stators and position indicator (P. I.) coils in progress.
August 26 (Saturday)	Cutting thermocouple plug welds continued. Started disassembly and removal of mechanisms (guide tubes, damper nuts and sleeves, rotors).	Completed resistance checks of mechanism stators, P. I. coils. Six Seed 3 fuel assemblies received and inspected.
August 28 (Monday)	Cutting fuel port seal welds and thermocouple plugs welds continued. Completed removal of mechanisms from reactor.	Six Seed 3 fuel assemblies received and inspected.
August 29	Cutting fuel port and thermocouple welds continued. Dry removal of ASEWI thermocouples from assemblies in fuel ports J-14, N-10.	Completed resistance testing of electrical cables. Decontamination of mechanism motor guide tubes completed and of mechanism parts (damper nuts, sleeves, etc.) started. Fuel extraction crane strain gauge checked; crane moved to dry pit for inspection of tool. Installed underwater

TABLE II (Cont)

CHRONOLOGY OF MAJOR EVENTS

Date (1961)	Refueling Sequence Event	Parallel Operations
August 29 Continued		periscope at fuel storage pit. Six Seed 3 fuel assemblies received, inspected.
August 30	Cutting thermocouple plug seal welds completed. Dry removal of SEWI and SMI thermocouples and of replaceable BEWI T/C's at H-9. Dressing of mechanism and fuel port seal weld lips in progress.	Welders being qualified for seal welding.
August 31	Cutting of fuel port seal welds completed. Air lines connected from header to thermocouple terminal boxes for air leak test; test started.	Reactor vessel head vacuum cleaned. Six Seed 3 fuel assemblies received.
September 1	Dressing of fuel port seal weld lips completed; continued dressing mechanism lips. Cleaned reactor pit. Removed all thermal barriers from mechanism housings. Internal inspection of mechanism housings started.	Fuel extraction crane head tool reinspected; crane moved to deep pit ready for final operator training. FEDAL bubble test equipment preparation completed.
September 2 (Saturday)	Completed internal inspection of mechanism housings. Dressing mechanism seal weld lips continued. Air pressure testing of d/p cells, etc., for leaks in progress.	Cleaned reactor vessel head. Lead screw inspection stand prepared for service.
September 5 (Tuesday)	Shroud locking pieces removed; lifting bails installed. Two shrouds removed dry from reactor; radiation levels such as to permit continuation of dry removal; six more shroud and scram shaft assemblies removed.	FEDAL system gas bubble test performed to confirm location of blanket assemblies J-5, K-8 having failed elements. Start inspecting lead screws.
September 6	Shroud and scram shaft assembly removal completed. BEWI assemblies nested for removal under water. Lead shielding removed from reactor vessel head.	Decontamination and inspection of mechanism rotors and other parts in progress. Three Seed 3 fuel assemblies received.
September 7	Air pressure testing completed. Fuel port caps installed. Lifting cables attached to 6 BEWI and ASEWI assemblies. Reactor internals inspected. Removed tool control barrier, tool cabinets, work level grating from reactor pit; cleaned pit and vessel head. Reactor water level raised to fuel port nozzles ready for pit flooding.	Operators checked out on fuel extraction crane. Two Seed 3 fuel assemblies received. Mechanism thermal barrier decontamination completed. Nuclear instrumentation discriminator plateaus being plotted.
September 8	End of Phase II operations 3:00 a.m.	

Elapsed Time Comparison:
Second Refueling - 16-2/3 days
First Refueling - 28 days

TABLE II (Cont)

CHRONOLOGY OF MAJOR EVENTS

Date (1961)	Refueling Sequence Event	Parallel Operations
	Phase III - Replacement of Fuel Assemblies	
	Refueling Sequence Steps (revised) 83 - 250	
September 8	Reactor pit flooded and seal checked for leaks. BEWI and ASEWI assemblies removed from reactor. Began removal and insertion of fuel assemblies at 4:00 p.m. using fuel extraction crane.	Pit gate removed. Lead screw inspection continued. SOAP fuel assembly received.
September 9 (Saturday)	Fuel assembly exchange continued: Installed 8 seed assemblies. Removed blanket assembly H-8.	Lead screw inspection continued. Blanket assembly F-14 reorificed. Nuclear instrumentation-base count rates established for first 16 Seed 3 clusters.
September 11 (Monday)	Fuel exchange continued: The SOAP assembly was installed in reactor; blanket assembly K-8 removed and blanket assembly K-9 relocated to position K-8. Reorificed blanket (old) F-14 installed at location K-9, and new blanket installed at J-2. Seven seed assemblies installed.	Dressing fuel port blind plugs. Blanket assembly (old) J-2 was reorificed. ASEWI's being inspected and radiation surveys taken. Rotor decontamination in progress. New base count established for nuclear instrumentation.
September 12	Fuel exchange continued: Blanket assembly E-9 relocated to position J-5. Reorificed blanket (old) J-2 installed at E-9 and new blanket installed at F-14. Eight seed assemblies installed.	Three BEWI's inspected dry. Fuel port closures and nuts dressed. Shroud and scram shaft inspection continued.
September 13	Fuel exchange completed 10 a.m. with installation of nine Seed 3 assemblies. Reactor pit drained and cleaned. Reactor vessel head shielding reinstalled.	Shroud, scram shaft, ASEWI inspection continued; Fuel port closures and nuts dressed. Mechanism P. I. coil and stator mechanical inspection.
	Elapsed Time Comparison: Second Refueling - 4-1/6 days First Refueling - 17 days	
	Phase IV - Reinstall Nonfuel Components, Reweld Seals	
September 14	Shroud and scram shaft assemblies installed dry. Reactor internals inspected. Two ASEWI and three BEWI assemblies installed dry.	Mechanism internal parts inspection continued. Remote control pushbuttons communication and cable added to seal welding machine. Spent blanket assembly J-5 loaded into shipping container.
September 15	Shroud installation completed. Fuel port closure reinstallation.	Reinstallation lead shielding on reactor vessel started. Shipped irradiated lead screw E-12 to Bettis Lab.

TABLE II (Cont)

CHRONOLOGY OF MAJOR EVENTS

Date (1961)	Refueling Sequence Event	Parallel Operations
September 16 (Saturday)	New BEWI assembly B-6 installed. Fuel port closure and blind plug installation completed. Shroud locking devices installed. Mechanism thermal barriers being installed.	Spent blanket J-5 shipped to ECF.
September 18 (Monday)	Mechanism rotors installed. Thermal barrier installation completed. SEWI T/C installation begun. Motor guide tubes installed.	Trellis east gate installed. Trellis support structure removed. Welding equipment prepared for use.
September 19	SEWI T/C's reinstalled, plug welding started. Fuel port seal welding started. Blind plugs for former SMI locations welded.	Motor tubes lead-shielded and omega seal weld gaps taped to keep clean. Welders working 12-hour shifts. Stator cooling water lines repaired. Reactor dome bolts cleaned, cable support ring painted. Cleanup and repairs of tools and components.
September 20	Fuel port seal welding completed except for repairs required. BEWI T/C plug welding in progress. SEWI T/C plug welding continued. Mechanism seal welding started.	Core instrumentation cable checkout from ring bus to data acquisition center in progress.
September 21	Repair seal welds on 5 fuel ports. Mechanism and SEWI seal welding continued. T/C's inserted in ASEWI and BEWI B-6 assemblies.	Reactor plant primary system being prepared for hydrostatic test.
September 22	Mechanism port seal welding, SEWI and SMI plug welding, and fuel port repair welding continued.	Position-indicator coil resistances rechecked on two assemblies K-11, J-4.
September 23 (Saturday)	Mechanism ports being welded, dye checked, and repaired as necessary. SMI, SEWI, BEWI, and ASEWI plug welding, dye checking, repairs in progress.	Plant preparations for hydro test continued. Cable connector reinspection.
September 25 (Monday)	Mechanism port weld dye checking completed. Instrumentation plug welding and checking continued. Replaceable BEWI T/C inserted in H-9 and plugs welded. New ASEWI T/C inserted in N-10 and welded.	Welding of d/p cell in progress. Reactor chamber dome being readied for installation. Thermal insulation for reactor vessel head prepared.
September 26	Another BEWI T/C inserted in H-9 and plug welded. Reactor vented through E-11 valve at head tank pressure. Components inspected for leaks. Primary plant systems filling and venting started.	Motor guide tube shielding removed. Plant preparations for hydro test continued. Reactor water level indicator disconnected. Cable connector repairs. Primary loop 1D cleaned for installation of main coolant pump assembly.

TABLE II (Cont)

CHRONOLOGY OF MAJOR EVENTS

Date (1961)	Refueling Sequence Event	Parallel Operations
September 27	Continued checking instrumentation weld plugs, cable connectors, weld inspection for leaks; preparation of mechanism stator water jacket and P. I. coil assemblies for installation, etc.	Primary systems filling and venting for hydrostatic test in progress. Filled secondary (steam) sides of steam generators A, B, C. Nine plant testing programs and seven major plant construction and maintenance programs in progress, prerequisite to plant startup.
	End of Phase IV Operations Elapsed Time Comparison: Second Refueling - 11-2/3 days First Refueling - 44 days	
	Phase V - Replace External Components, Hydrostatic Test, Prepare for Initial Criticality	
September 28	Mechanism stator and P. I. coil assemblies and housings reinstalled. T/C resistances checked. Refueling seal removed. Reactor vented again. Reactor vessel head insulation partially installed. Twenty-one mechanism cooling water lines installed.	Work level grating removed from reactor pit. Cable support structure readied for installation. Lead shot shielding removal. Miscellaneous fixtures removed from reactor pit. Operations crew continuing system pressurizing and venting procedure. Coolant temperature 142 F at 11:30 p.m. Installed main coolant pump motor assembly in primary loop 1D and torqued bolts.
September 29	Plant being brought up to pressure and temperature required for hydro test. Vessel head welds had no leaks at 450 psi and 1200 psi. Cable support ring installed.	Removed P. I. coil housing cover caps to inspect old vent plug welds for leakage during hydro test. Nuclear instrumentation discriminator plateaus checked.
September 30 (Saturday)	In raising pressure for hydro test two T/C tubes leaked slightly at 1800 psi due to broken electrical insulation. No further leaks on reactor head welds at full 2750 psi. E-11 motor tube prepared for welding vent plug. Upper tube assemblies for BEWI's installed.	Repair items made for leaking T/C boss. FEDAL pump and main coolant pump 1C leaked at bolt flanges on hydro test. System pressure reduced to 2000 psi for pressure instrumentation calibration.
October 1 (Sunday)	Certain reactor protection system tests completed. Trouble on NIS magamps in investigation. Leak tests of secondary systems performed; leaks found. Primary system heatup started via main pumps 1A and 1B. Five test programs prerequisite to criticality in progress.	Pressure instrumentation calibrated per portions of test procedure. Diesel generator load tested. Head dissipation facility tie-ins to treated water and other systems in progress; discharge tunnel out of service for construction work.

TABLE II (Cont)

CHRONOLOGY OF MAJOR EVENTS

Date (1961)	Refueling Sequence Event	Parallel Operations
October 2 (Monday)	E-11 vent plug welded, dye checked. Installing and connecting electrical cables of T/C's and mechanism stators; also cooling water lines. Plant pressure instrumentation calibrated. Rod control system testing. Start primary system cooldown for physics tests. Reactor protection system testing.	Installed electrical cables for calibrating 13 d/p cells. Depleted seeds F-12 and H-12 inspected underwater. Multiport valve stator removed, reoriented, reinstalled. Feedwater heater 1B being retubed (since August). Fill secondary sides of steam generators 1A and 1B. Main steam system leak repairs completed. Reactor plant container air lock testing. Heat sink tie-in work continued. Valve operating system 3-way selector valves leak tested.
October 3	Motor tube holddown structure fitted and installed. Hydrostatic test of cooling water lines.	Calibration of d/p cells started.
	Ready for chamber dome installation pending completion of d/p cell calibration, when accidental damage to one mechanism and two P. I. coil assemblies occurred.	
	Removed ventilation duct, ladder, and holddown structure to gain access, then removed grounded D-5 stator jacket and P. I. assembly. Electrical cables in reactor pit installed and being tested. Primary system cooldown completed.	Checkout of station electrical and fluid systems in progress. Steam generator 1A maintenance. Hydrazine addition vessel discharge line leak tested. Reactor protection system magamp repairs. Two pressurizer steam relief valves repaired and adjusted.
October 4	Installed spare stator in D-5. Renewed grounded P. I. coils and reinstalled in G-4 and K-4. Reinstalled holddown structure. Investigated bad BEWI connector.	Two of 13 d/p cells calibrated. Loaded spent blanket assembly K-8 in shipping container. Checking stator thermocouples. Reactor plant container air lock testing completed. Reactor coolant piping and component external radiation level tests completed. FEDAL pump and main coolant pump 1C repairs in progress.
October 5	Reinstalled ventilation duct and ladder. Electrical cables checked out. Pre-criticality checkouts prior to rod latching tests started. Vented three main coolant pumps, steam generators; pressurizing and heatup for 1C main pump hydro test begun.	Leak tests conducted on certain station systems. Mechanism stators meggered. Leak tested VOS 3-way valves and other primary system components. Number 2 rod drive MG set bearing failed. Completed core d/p cell calibration.

TABLE II (Cont)

CHRONOLOGY OF MAJOR EVENTS

Date (1961)	Refueling Sequence Event	Parallel Operations
October 6	Reactor chamber dome installed 1st shift. Refueling operations in reactor pit completed. Station startup check-outs. Precriticality check completed. Rod drive mechanism tests performed.	Check operation of main coolant pump 1D. Completed BEWI H-9 T/C repair. Leak test pressurizer relief valve discharge piping.
October 7 (Saturday)	Initial approach to criticality tests conducted. Source range B trouble investigated and corrected. Criticality achieved 11:30 p.m. Started determination of radionuclides test.	Pressurizer steam relief valves tested satisfactorily. Reactor scram circuit breaker closing coil trouble. Activated spare breaker.
	End of Phase V Operations Elapsed Time Comparison: Second Refueling - 8-1/2 days First Refueling - 29 days	
	Phase VI - Conduct Physics Tests, Complete All Necessary Plant Work, Attain Full Power Operation of Station	
October 8 (Sunday)	Reactor shutdown. Pressurizer steam bubble formed. Reactor coolant loop 1D hydrostatic tested, leaks found. Control rod calibration and comparison test started.	Rod drive MG set no. 2 repaired. Leak test FEDAL valve welds and pump gaskets completed. Main steam system weld repairs; heat sink tie-in work continued. Automatic data logging system tests in progress.
October 9 (Monday)	Reactor critical as necessary for cold, sub-power physics tests, control rod tests. Primary system level and flow instrument calibration continues.	Radioactive waste disposal vent gas compressors 1B and 1C developed leaks. Faulty source range indication channel C corrected. Spent blanket fuel assembly K-8 shipped to ECF-Idaho.
October 10	Cold physics tests interrupted for 12 hours to correct NIS source range troubles in channel B; defective BF_3 detectors. Primary system level instrumentation calibration tests completed. Reactor coolant pump 1D motor removed to correct flange seal leaks.	Valve operating system alignment. Vent gas compressor 1C back in service; 1B on standby. Steam, feedwater pipe weld repairs continue. Scram circuit breaker and pump 1B breaker repairs completed.
October 11	Sub-power physics tests in progress; reactor critical as needed. NIS Channel B intermediate range difficulties. Control rod calibration and comparison tests completed. Control rod positions for criticality testing started. Start calibration test of pressurizer and loop 1D pressure instruments.	Heat sink tie-in to feedwater system weld repairs cause reinstallation of original feedwater piping.

TABLE II (Cont)

CHRONOLOGY OF MAJOR EVENTS

Date (1961)	Refueling Sequence Event	Parallel Operations
October 12	Primary loops 1A, 1B, and 1C heatup started for hot physics tests and attempt to go to power operation. Completed control rod positions for criticality tests. Coefficients of reactivity measurements tests started.	NIS channel B intermediate range erratic. Weld repairs on main steam system and its heat sink tie-in continuing with radiographs. Feedwater heater 1B retubing continuing.
October 13	Physics tests below power range in progress. Cool down for steam generator filling. Coefficients of reactivity tests interrupted.	Fill steam generators B and C for leak testing of boiler steam side and four steam lines. Radiation monitoring system (ORMS) Channel 8 modification in progress.
October 14 (Saturday)	Reactor shut down. Primary system pressure taken to 2000 psi to leak check reactor vessel d/p cell vents and BEWI H-7 weld plugs. Pressurizer system returned to service. Leak test steam generators A, B, and C steam side. On startup Group I rods programmed incorrectly; wiring error corrected.	Take primary system gas samples. Main coolant pump 1D leak tested 2750 psi. Feedwater original piping hookup completed. New compensated ion chamber (CIC) installed in NIS channel B.
October 15 (Sunday)	Reactor taken critical twice. Test of rod positions for criticality (hot) started. Temperature instrument calibration. Reactor coolant flow and pressure drop characteristics test started.	Checkout primary loop drain valves, core heat removal system valves, blind flanges. Check out turbine supervisory instruments.
October 16 (Monday)	Reactor scrammed due to incorrect low pressure protection setpoint. On restart, high startup rate safety insertion occurred due to incorrect rod withdrawal rate. Reactor caused high startup rate safety insertion; same reason as above. Scram occurred later from reactor protection system tests.	Trouble in Loop D 3-way selector for drain valve. Steam plant to heat sink tie-in modified to permit station operation. Weld radiographs completed. Completed leak rate tests of pressurizer relief valves.
October 17	Completed rod positions for criticality (hot) test; also coefficients of reactivity test. Reactor flow instrumentation calibration in progress. Primary coolant and purification self-actuated relief valve tests initiated.	Repair welds in main steam header and manual stop valve 1A. Completed leak tests on boiler feed system and secondary of steam generator 1A. ORMS channel 8 modification completed; channel in service. Type M-130 spent seed fuel shipping container being prepared for use.
October 18	Completed relief valve operational tests, hydrostatic test of loop 1D, main steam system header, and retest of steam generator 1A steam side. Operational checks performed on complete reactor plant.	Calibrated ORMS channels. NIS channel B magamp 3N3 checkout. Replace no. 1 FEDAL d/p cell. Feedwater heater 1B retubing completed.

TABLE II (Cont)

CHRONOLOGY OF MAJOR EVENTS

Date (1961)	Refueling Sequence Event	Parallel Operations
October 19	Reactor protection system tests performed. Completed hydro test steam generator 1D steam side. Started calibrating temperature sensing elements. Loop 1D placed in service. Checking time response of main pump shutdown relays.	Hydrogen and lithium-hydroxide added to primary system. Circulate turbine oil and exercise controls. Spent fuel container M-130 loaded.
October 20	Protection system tests disclosed response times considered excessive on coolant pump shutdown time-delay relays. Relay circuits adjusted for time constants as result of studies. Performed tests of flow distribution across the core and determination of coolant system pressure drops. Pre-criticality checkoff; started control rod drive criticality test but interrupted for 1-1/2 hours to correct reactor protection system magamp difficulties.	Leak test heat sink tie-in connections. Leak test feed-water heater 1B. Check out turbine speed changer, etc. Complete lagging of all piping. Leak test main air ejector modifications. Spent fuel container M-130 hydrogen buildup being investigated.
October 21 (Saturday)	Certain bistable magamps replaced in protection system and retested system operation. Completed high temperature calibration of temperature detectors. Control rod drive mechanism tests continue.	Added hydrogen to primary coolant system. Checkout FEDAL system. Manually cycled coolant pump time delay relays for time response. Weld surge line isolation valve cap. Clean reactor plant chambers.
October 22 (Sunday)	Completed rod drive mechanism tests. Checkout protecting system time response. Start FEDAL system check-out test. Precriticality checkoffs completed again.	Installed new timing relay in pump 1B circuit; then removed time delay from all pump relay circuits.
October 23 (Monday)	Reactor critical for approach to power operation. Turbine warmed up and overspeed trip checked. Station synchronized with power system; loaded to 8MW. Started test of station performance at steady-state loads when interrupted by defective compensated ion chamber (CIC) detector in channel 1C. Returned to power operation after 13-hour delay.	
October 24	Full power station output attained 6:05 a.m. in a step load buildup during which calorimetric determinations of steam system made. Reactivity lifetime testing started. Station performance at steady-state load test completed.	

End of Phase VI Operations

Elapsed Time Comparison:
Second Refueling - 16-1/4 days
First Refueling - 25 days

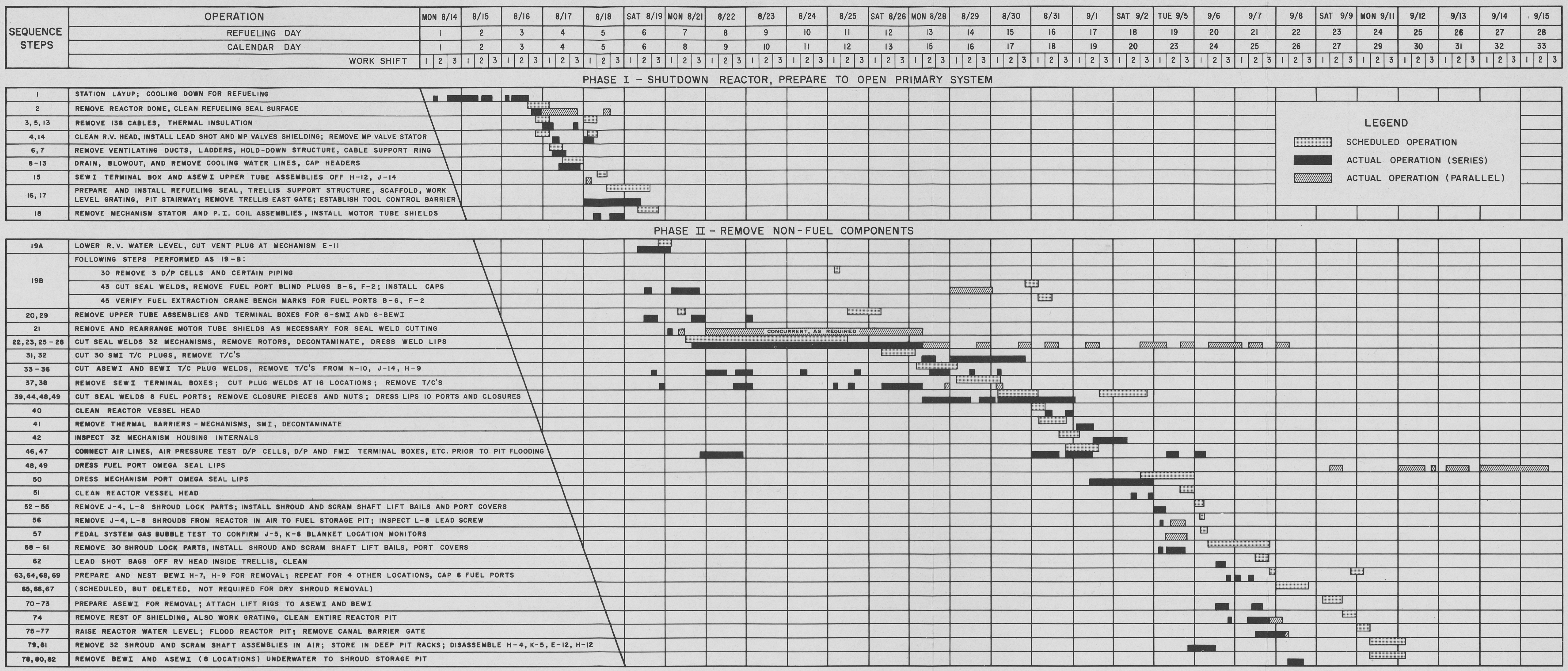


Figure 31. Bar Charts of Refueling Work Progress. (Sheet 1 of 4)

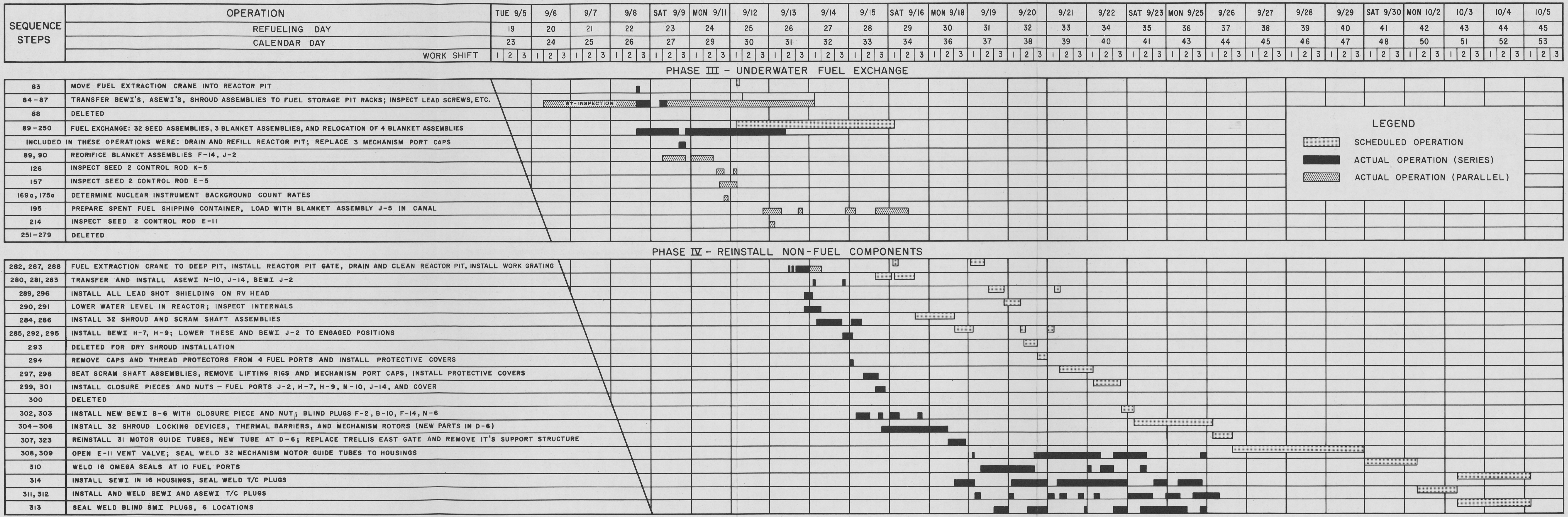


Figure 31. Bar Charts of Refueling Work Progress. (Sheet 2 of 4)

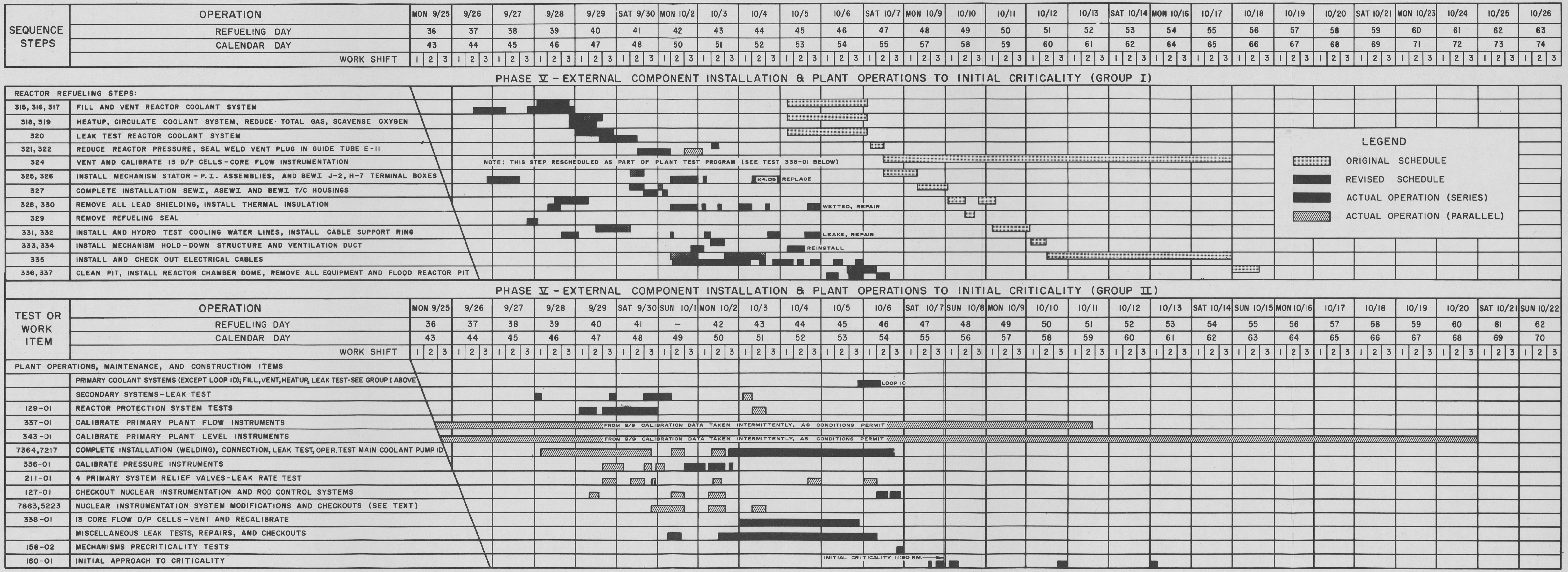


Figure 31. Bar Charts of Refueling Work Progress. (Sheet 3 of 4)

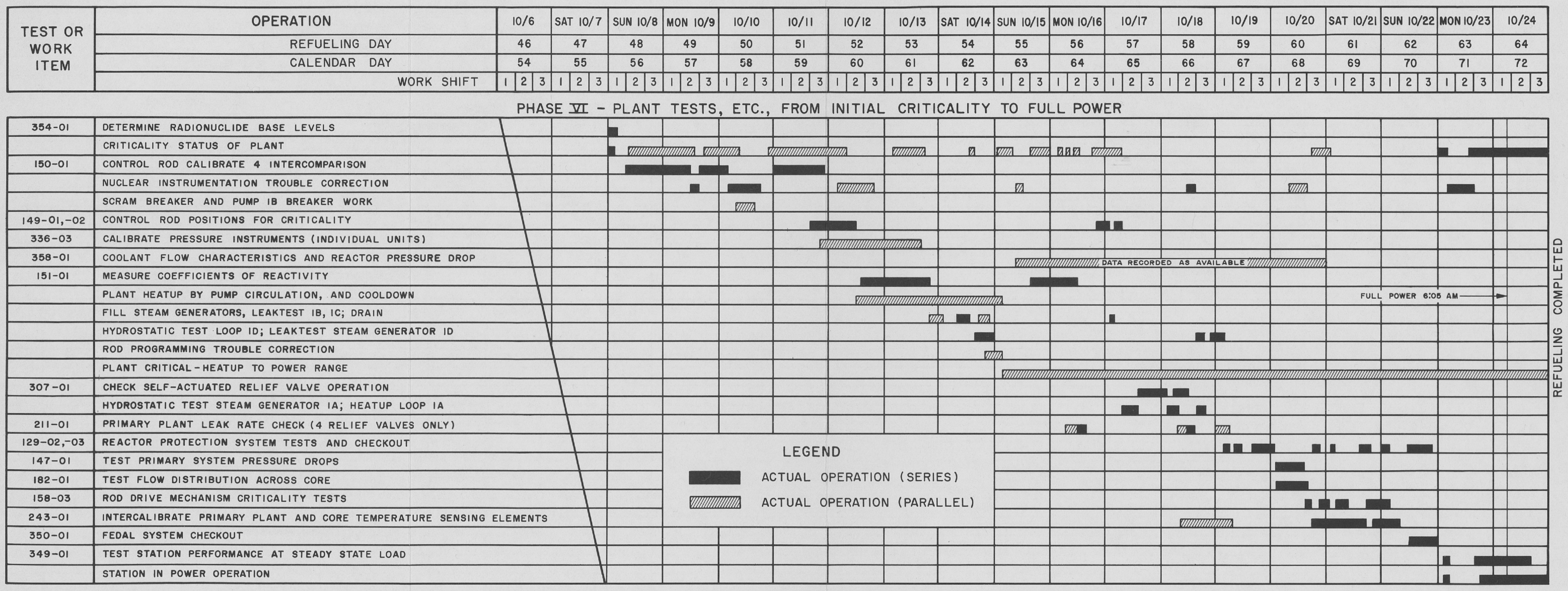


Figure 31. Bar Charts of Refueling Work Progress. (Sheet 4 of 4)

V. RADIOACTIVITY CONTROL

Radiation exposure control was accomplished efficiently and safely during the second refueling. Several important improvements were made in the methods of exposure control. For example, improved shielding techniques were utilized, and more efficient means for handling irradiated shrouds and BEWI assemblies were devised. As a result of these improvements and the reduction in total refueling time, total radiation absorbed by all refueling personnel during the second refueling (about 225 roentgens) was significantly less then the total exposure during the first refueling (about 280 roentgens). No single person received more than the allowable 3 roentgens during the refueling period.

A. Personnel Protection

The minimization of personnel exposure required familiarity with, and the conscientious application of, all radioactivity control procedures, and general knowledge of the sources of radioactivity and its control. The major sources of radioactivity during a refueling are neutron-induced gamma activity and fission products in the fuel-bearing and nonfuel-bearing core materials; corrosion and wear products deposited as crud on mechanism and instrumentation surfaces; and, loose contamination which could be spread by workmen to the building surfaces or into the air. The most important factors contributing to personnel exposure were the direct radiation fields and hot spot radiation levels emanating from the reactor head. Direct radiation is controlled by time, distance, shielding, and decontamination. Variations of these methods were used to restrict accumulated personnel radiation doses to the lowest practical limit and to distribute these doses most efficiently among the men as they performed their specialized jobs in the refueling process.

1. Time

A radiation exposure estimate system was worked out by the Duquesne Light Company Health Physics Department during the first refueling to guide the over-all manpower deployment by controlling the length of time spent in a high radiation field. For the second refueling the estimated dose was calculated from reactor pit radiation levels measured during early refueling and the expected decontamination factors of the components to be reinstalled. These calculations resulted in more efficient distribution of the dosages among the men in each trade group and insured that the men would be available for performing a particular forthcoming refueling step. Throughout the course of the refueling procedures, re-evaluations were made which compared the predicted dosages with the doses received and work was assigned accordingly. The system was also an aid in determining additional shielding requirements.

The exposure limit is 3 roentgens per running 13-week period. The weekly exposure was kept to less than 900 mrem as much as possible. The estimate system for controlling exposures was effective in that no personnel received more exposure than permissible and no time delays were caused by personnel achieving the allowable dosage and then being restricted from radioactive areas.

Intensive training conducted prior to actual performance was vital to increased efficiency and to reduced exposure time during the refueling procedure. Periodic lectures given to refueling personnel by Industrial Hygiene representatives helped maintain an awareness of the necessity for spending a minimum amount of time in a radiation field to prevent accumulating unnecessary doses.

2. Distance

Whenever particularly high radiation levels existed or were anticipated, as during shroud removal, concurrent refueling operations were discontinued to limit the number of men exposed to high radiation levels. To perform an operation causing high radiation levels, the reactor pit was evacuated, except for a radiation control technician and workmen essential to performing the particular procedure. Decontamination and component inspections were performed away from high fields whenever possible.

3. Shielding

Water in the refueling canal and minimum depth restrictions provided shielding which kept the dose rate to background levels during fuel transfer and inspection of irradiated components.

Various forms of lead shielding were used on the reactor vessel head to reduce the general levels of direct radiation when water was not in the pit. Supplementary shielding was added on hot spots until the point was reached where workmen would receive greater doses during the installation and removal of the shielding than would be prevented cumulatively by the additional shielding. Because of the complex configuration of the external working surface of the vessel head, bagged lead shot was used to protect against radioactivity from the crud on the interior surface of the head and on the mechanism parts generally below the motor, such as thermal barriers and shroud upper parts. Formed lead was used on the mechanism motor guide tubes for protection against deposited radioactivity on the mechanism rotor, roller nuts, and other operating parts above the vessel head. The lead shot was packed in mildew resistant, double bags, 10 pounds per bag. These bags were then sealed in yellow polyethylene covers. The shot bags were stacked on the reactor vessel head inside the mechanism boundary to just below the top of the center fuel ports. This area was then protected by 3/4 inch thick polyethylene-wrapped plywood covers, made in 4 sections, and laid flush with the fuel port tops. Outside the mechanism fence, between this boundary and the outer fuel ports and 12 vertical plywood retainers, bags were similarly stacked and covered. In all, about 10,000 pounds of lead shot were installed. The lead shot was contained in box-type skids set on the work grating in the reactor pit, which had been strengthened to bear the weight.

Formed lead shielding, designed on the basis of data gathered during the Seed 2 refueling and during Seed 2 operation, was used for the multiport valve and mechanism motor tubes because the sheets of lead used during the first refueling were awkward and inefficient.

The multiport valve is a crud trap and, therefore, a source of high radiation. Special lead segments were formed from sheet lead and cut to fit snugly in multiple layers, using a spare multiport valve as a form. The largest of the shielding segments weigh about 50 pounds each, and all were installed by hand. Following the removal of the multiport valve thermal insulation and stator water jacket assembly, the "cap-type" shield was installed over the motor tube as shown in Figure 32. The multiport valve bypass pipe and the two sample lines were shielded by wrapping them with 3/16 inch lead sheet until sufficient thickness was reached to attenuate radiation to a minimum practicable level. This shielding was effective and reduced the radiation levels at the multiport valve by a factor of about 4.

To take the place of lead sheet shielding of the mechanism motor tubes, several types of cast lead cylindrical segments, spacers, full cylinders, and covers were designed and manufactured (Figure 26). The large segments, weighing about 50 pounds each, were provided with short chains with snap locks for connecting pairs of segments together around the lower sections of each motor tube. These chains became twisted or jammed and some snap locks broke. An improved mechanical support is being devised for future refuelings. For the upper areas of the motor tubes, 20 pound cylindrical rings (donuts) were slipped down over the tops of the guide tubes. Up to seven such pieces could be stacked and nested on each motor tube. These were small enough to be handled manually, but a cylindrical handling container, which could hold a complete stack of rings, was also used to facilitate covering the motor tubes. To make room for installing the weld cutting machine on one motor

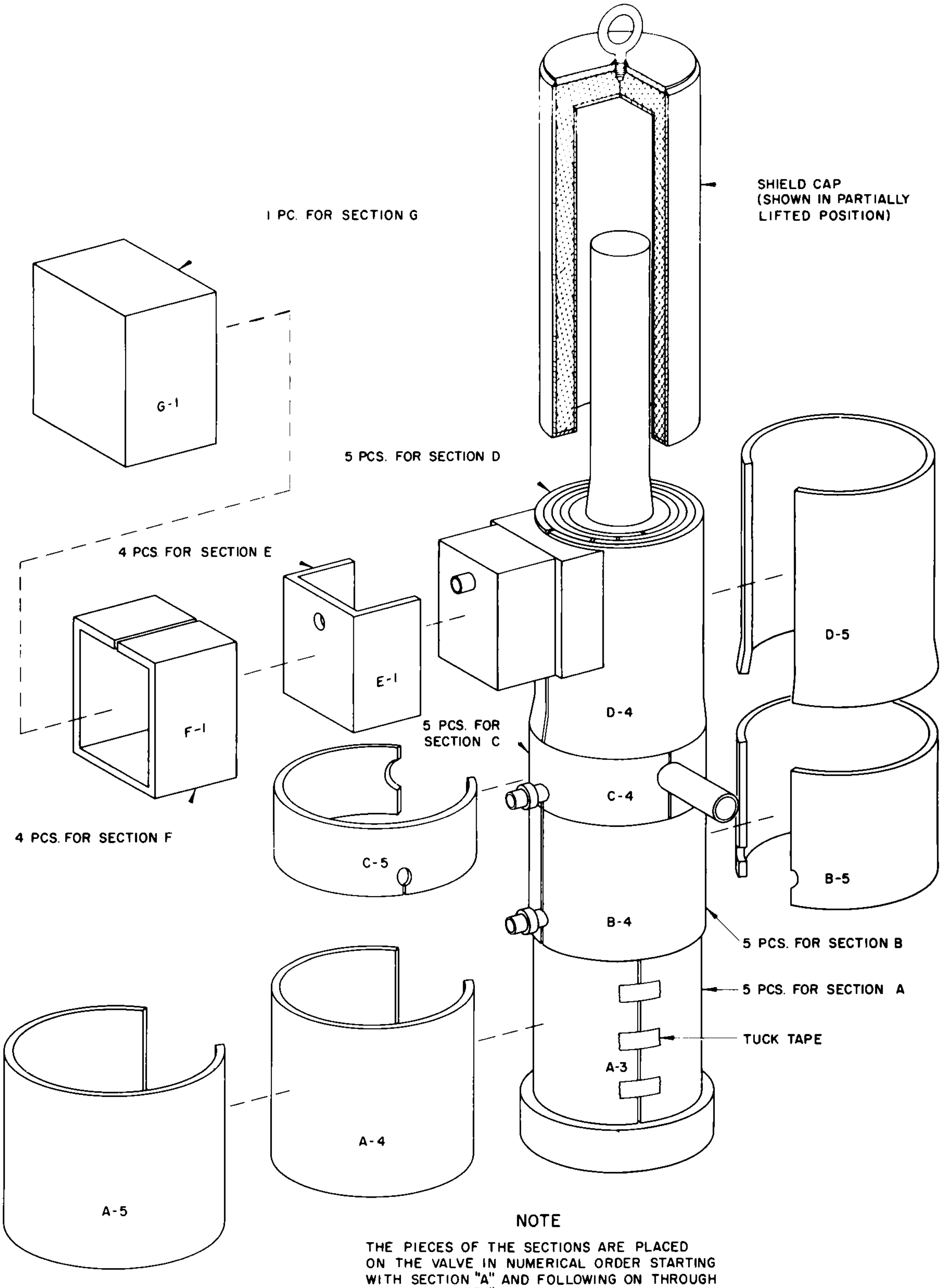


Figure 32. Multiport Valve Shielding.

tube, the lead cylinders and segments had to be removed from adjacent motor tubes. Then, as the machine was moved to the next tube, the lead donuts and segments were removed from the next tube ahead and replaced on the trailing tube. This mechanism shielding, when all shields were in place, reduced the radiation levels at these components by a factor of about 5.

4. Decontamination and Access Control

Personnel were required to wear anti-contamination coveralls (anti-C's), head covers, gloves, and boots when working in the refueling area. If there was airborne activity (during most of the operations over open ports on the reactor vessel head), respirators were also required. Decontamination of areas and equipment by vacuum cleaning, mopping, and wipedown controlled air activity to prevent an ingestion hazard. Decontamination of components by ultrasonic baths (Figure 33) and scrubbing reduced the radiation levels during their reinstallation. Other personnel protection measures included maintenance of clean areas, normally required control areas, and highly radioactive restricted areas, in addition to continuous area monitoring including the mounting of film badges at various locations in the reactor pit. The contaminated change room and showers provided a buffer area which effectively prevented the spreading of contamination to the clean locker rooms or

Figure 33. Ultrasonic Decontamination of Mechanism Rotor from Location D-10.

beyond. The practice of removing the extra set of coveralls and gloves upon leaving the reactor pit during periods of high contamination also helped in controlling contamination.

B. Area Access

The contaminated area access facilities are under the jurisdiction of the Industrial Hygiene Group. A new section was added to the south end of the Reactor Service Building to provide expanded health physics working areas. The expansion of the change room facilities provided improved clothes locker space and better contamination control. Adjacent to the refueling work area and change room are a laundry and the Industrial Hygiene work area where log books and records are prepared and maintained, monitoring instruments kept and calibrated, and samples counted. All other entrances to the refueling area were closed and sealed with lead seals.

Only one entrance from the outside was provided; during refueling a guard stationed here required that all persons entering have either a badge identifying them as having regular duties in the refueling area, or proper identification as visitors entitling them to enter that area. Film badges and dosimeters were assigned to all refueling personnel. On leaving the clean locker room area, all personnel were required to check themselves with the hand and foot counter, and pass through a portal monitor before reaching the guard station. Any handcarried materials to be brought out of the refueling area must have been passed by an Industrial Hygiene representative and registered with the guard. Personnel entering the refueling area through a single entrance from the clean locker room had to pass through the change room. Personnel leaving the contaminated area entered the change room where the anti-C clothing was left. The only other access made available during refueling was through the overhead door at the railroad/truck dock, which was appropriately guarded when being utilized. The single entrance facilities for personnel proved to be very effective in preventing the spread of any possible contamination.

C. Personnel Metering

Table III is the average weekly dose based on film badge dosimeter readings, developed weekly, for the 190 refueling personnel.

During the 8-week period from reactor shutdown until reinstallation of the reactor chamber dome, the average accumulated dose, calculated from a weighted weekly average, was 1.53 rem per person; the maximum accumulated dose received by one individual was 2.89 rem.

D. Radiation Surveys

Three types of radiation surveys were conducted in various locations by the radiation technicians—direct radiation level surveys, smear surveys, and air sample surveys—to monitor for excessive dose rates and control personnel exposure. These radiation surveys were taken for operational purposes; therefore, radiation monitoring procedures were not directed toward obtaining experimental data under controlled conditions.

Survey sheets representing the plan views of the reactor pit and the entire canal area were used by the radiation control technicians to record the measurement of radiation levels, smears, surveys, and air samples. A survey sheet of the reactor pit profile was used to record radiation levels at various elevations.

The control procedures required that a barrier be set up with radiation warning signs at the 5 mr/hr line of an exclusion area. A high radiation area was defined as the area inside a perimeter line at which the dose rate was greater than 100 mr/hr.

1. Direct Radiation Surveys

Dose rates in the reactor pit, presented in Table IV, were affected by the operation in progress and

TABLE III

WEEKLY SUMMARY OF RADIATION DOSAGES ACCUMULATED DURING SECOND REFUELING

Week Ending	No. of Men Considered* in Average Radiation Dosage	Number of Men Receiving Dosage in Ranges					Ave. Weekly Dosage mrem/man	Refueling Work in Progress
		> 400 mrem/wk	300-400 mrem/wk	200-300 mrem/wk	100-200 mrem/wk	<100 mrem/wk**		
8/18/61	140	1	10	23	36	70	114	Removed reactor chamber dome, electrical cables, and cooling water lines; installed refueling seal; removed mechanism external components.
8/25/61	157	34	37	18	16	52	232	Removed SMI upper tube assemblies and fuel port blind plugs; cut mechanism seal welds and thermocouple weld plugs for BEWI and SEWI assemblies; removed mechanism motor guide tubes and rotors.
9/1/61	171	20	46	30	15	60	218	Cut thermocouple plug seal welds and fuel port seal welds; disassembled mechanism motor guide tubes, damper nuts, and sleeves; removed mechanisms from reactor, ASEWI thermocouples from fuel ports, SEWI and SMI thermocouples, and replaceable BEWI thermocouples; dressed mechanism port and fuel port seal weld lips; removed mechanism thermal barriers; inspected mechanism housing internals.
9/8/61	158	6	40	34	20	58	188	Dressed mechanism seal weld lips; removed shrouds and scram shafts from reactor; performed FEDAL system gas bubble test; nested BEWI assemblies for underwater removal; installed fuel port caps; inspected reactor internals.
9/15/61	138	0	2	10	54	72	99	Flooded reactor pit; removed BEWI and ASEWI assemblies from reactor; replaced seed and blanket assemblies under water; dressed fuel port blind plugs; drained and cleaned reactor pit; dressed fuel port closures and nuts; installed shrouds and scram shafts; inspected reactor internals; installed ASEWI and BEWI assemblies; installed fuel port closures.
9/22/61	156	27	25	27	27	50	234	Installed new BEWI assembly, fuel port closures, and blind plugs; installed mechanism thermal barriers, rotors, motor guide tubes, and SEWI thermocouples; seal welded SEWI plugs, fuel ports, blind plugs at former SMI locations BEWI thermocouple plugs, and mechanism ports; repair welded fuel ports; installed ASEWI and new BEWI assemblies; prepared reactor plant loops for hydrostatic test.
9/29/61	130	24	21	32	32	21	262	Welded, dye checked, and repaired mechanism ports, SMI SEWI, BEWI, and ASEWI plugs; welded differential pressure cells; prepared for hydrostatic test; installed mechanism stators, position indicator coils, and housings; removed refueling seal; installed reactor vessel head insulation and cooling water lines; raised reactor coolant to temperature and pressure.
10/6/61	114	13	17	18	23	43	189	Conducted hydrostatic test; welded mechanism vent plugs, installed BEWI upper tube assemblies, electrical cables, and cooling water lines; removed and reinstalled multiport valve stator; installed mechanism motor guide tubes and hold-down structure; calibrated differential pressure cells; conducted precriticality tests; installed reactor chamber dome.

* Supervisory and refueling personnel from Duquesne Light Company receiving dosages above background.

** Not including 0 mrem/week.

TABLE IV

TYPICAL RADIATION LEVELS IN REACTOR PIT DURING SECOND REFUELING

Date	Shielding and Port Access Status	Radiation Levels (mr/hr) Minimum-Maximum	Locations Monitored*
8/16/61	Before removal of the reactor chamber dome	∿ 4	Outside reactor chamber dome
8/17/61	After removal of the reactor chamber dome; before installation of the lead-shot shielding inside trellis later in the day	300-350	Inner fuel ports H-7 & H-9
		130-400	Outer fuel port
		65-150	Inner seal ring
		20-55	Mid-radius of seal
		13-25	Near pit walls
		700	Multiport valve hot spot
		800	Flow measurement instruments
		350-400	Near mechanism array
8/18/61	After installation of lead-shot shielding inside trellis; during installation of lead-shot shielding outside trellis; removed position indicators and stator water jackets from mechanisms, before installation of preformed multiport valve shielding after installation of additional shielding on northwest FMI	-------	Inner fuel ports H-7 & H-9
		90-260	Outer fuel ports
		70-150	Inner seal ring
		30-70	Mid-radius of seal
		5-20	Near pit walls
		1000	Multiport valve hot spot
		65-170	Flow measurement instruments
		300-550	Near mechanism array
8/19/61	After installation of all lead shields and donuts on motor guide tubes; established coolant level in reactor; removed SMI upper tube assemblies	75-85	Inner fuel ports H-7 & H-9
		30-95	Outer fuel ports
		30-50	Inner seal ring
		20-30	Mid-radius of seal
		5-11	Near pit walls
		240	Multiport valve hot spot
		50-200	Flow measurement instruments
		60-70	Near mechanism array
8/21/61	Vented reactor	70-70	Inner fuel ports H-7 & H-9
		35-75	Outer fuel ports
		30-30	Inner seal ring
		19-40	Mid-radius of seal
		9-11	Near pit walls
		300	Multiport valve hot spot
		45-60	Flow measurement instruments
		-------	Near mechanism array
8/22/61	Removed motor guide tubes E-11 and K-5	150	Inner fuel ports H-7 & H-9
		45-90	Outer fuel ports
		10-45	Inner seal ring
		10-30	Mid-radius of seal
		7-13	Near pit walls
		220	Multiport valve hot spot
		85-225	Flow measurement instruments
		45-500	Near mechanism array
8/26/61	More reactor vessel head shielding, mechanism dust cover, and motor guide tube shielding had been added on 8/23/61; removed and disassembled mechanism components	110-130	Inner fuel ports H-7 & H-9
		30-120	Outer fuel ports
		10-40	Inner seal ring
		11-20	Mid-radius of seal
		8-12	Near pit walls
		180	Multiport valve hot spot
		60-90	Flow measurement instruments
		60-250	Near mechanism array

TABLE IV (Cont)

TYPICAL RADIATION LEVELS IN REACTOR PIT DURING SECOND REFUELING

Date	Shielding and Port Access Status	Radiation Levels (mr/hr) Minimum-Maximum	Locations Monitored*
8/28/61	Completed removal of mechanisms from reactor	60-75	Inner fuel ports H-7 & H-9
		40-80	Outer fuel ports
		30-60	Inner seal ring
		17-23	Mid-radius of seal
		4-12	Near pit walls
		160	Multiport valve hot spot
		11-120	Flow measurement instruments
		50-140	Near mechanism array
8/29/61	Removed ASEWI assemblies (dry)	50-60	Inner fuel ports H-7 & H-9
		35-70	Outer fuel ports
		30-50	Inner seal ring
		11-15	Mid-radius of seal
		4-10	Near pit walls
		220	Multiport valve hot spot
		20-190	Flow measurement instruments
		90-180	Near mechanism array
8/30/61	Removed SEWI, SMI, and replaceable BEWI assemblies (dry)	60-80	Inner fuel ports H-7 & H-9
		100-110	Outer fuel ports
		25-80	Inner seal ring
		16-25	Mid-radius of seal
		6-21	Near pit walls
		190	Multiport valve hot spot
		20-60	Flow measurement instruments
		70-150	Near mechanism array
8/31/61	Metal fuel port covers in place	60-100	Inner fuel ports H-7 & H-9
		50-100	Outer fuel ports
		20-30	Inner seal ring
		14-19	Mid-radius of seal
		4-8	Near pit walls
		180	Multiport valve hot spot
		60-160	Flow measurement instruments
		90-240	Near mechanism array
9/1/61	Cleaned reactor pit; removed thermal barriers	110-150	Inner fuel ports H-7 & H-9
		65-120	Outer fuel ports
		30-60	Inner seal ring
		16-22	Mid-radius of seal
		4-9	Near pit walls
		200	Multiport valve hot spot
		15-300	Flow measurement instruments
		90-250	Near mechanism array
9/5/61	Started shroud removal (dry); installed mechanism port caps	115-160	Inner fuel ports H-7 & H-9
		85-125	Outer fuel ports
		35-75	Inner seal ring
		21-35	Mid-radius of seal
		5-13	Near pit walls
		240	Multiport valve hot spot
		30-400	Flow measurement instruments
		120-300	Near mechanism array

TABLE IV (Cont)

TYPICAL RADIATION LEVELS IN REACTOR PIT DURING SECOND REFUELING

Date	Shielding and Port Access Status	Radiation Levels (mr/hr) Minimum-Maximum	Locations Monitored*
9/6/61	After installation of multiport valve shielding; completed shroud removal; removed lead-shot shielding from inside and outside trellis on reactor head	120-150	Inner fuel ports H-7 & H-9
		55-115	Outer fuel ports
		60-100	Inner seal ring
		30-55	Mid-radius of seal
		6-12	Near pit walls
		225	Multiport valve hot spot
		60-120	Flow measurement instruments
		90-130	Near mechanism array
9/7/61	After nesting BEWI tubes for removal; before installation of fuel port caps; raised water level in reactor in preparation for pit flooding	1000-1200	Inner fuel ports H-7 & H-9
		200-800	Outer fuel ports
		65-105	Inner seal ring
		40-60	Mid-radius of seal
		9-21	Near pit walls
		400	Multiport valve hot spot
		150-350	Flow measurement instruments
		400-500	Near mechanism array
9/8/61 through 9/13/61	No surveys were taken while the reactor pit was flooded for underwater fuel replacement.		
9/14/61	Drained and cleaned pit; installed lead-shot shielding inside and outside trellis; installed shroud and scram shaft assemblies (dry); after installation of lead sheets around multiport valve and FMI assembly; before installation of ASEWI and three BEWI assemblies (dry)	900-1000	Inner fuel ports H-7 & H-9
		75-350	Outer fuel ports
		30-70	Inner seal ring
		25-35	Mid-radius of seal
		7-16	Near pit walls
		120	Multiport valve hot spot
		110	Flow measurement instruments
		215-350	Near mechanism array
9/15/61	Removed fuel port caps and split-thread protectors; installed aluminum closures outside trellis; before installation of more lead-shot shielding on hot FMI in northwest corner	120-160	Inner fuel ports H-7 & H-9
		50-220	Outer fuel ports
		35-80	Inner seal ring
		13-40	Mid-radius of seal
		6-25	Near pit walls
		165	Multiport valve hot spot
		25-120	Flow measurement instruments
		80-180	Near mechanism array
9/16/61	After installation of shielding inside trellis and at corners outside trellis; installed BEWI assemblies; before installation of thermal barriers, blind plugs, and remaining fuel port closures	30-42	Inner fuel ports H-7 & H-9
		22-44	Outer fuel ports
		15-60	Inner seal ring
		14-20	Mid-radius of seal
		6-11	Near pit walls
		30	Multiport valve hot spot
		16-42	Flow measurement instruments
		50-250	Near mechanism array
9/18/61	Installed mechanism rotors and motor guide tubes; placed additional shielding around corner positions and inside trellis; before installation of additional shielding on motor guide tubes	80-80	Inner fuel ports H-7 & H-9
		50-100	Outer fuel ports
		18-50	Inner seal ring
		13-25	Mid-radius of seal
		6-18	Near pit walls
		130	Multiport valve hot spot
		30-110	Flow measurement instruments
		70-240	Near mechanism array

TABLE IV (Cont)

TYPICAL RADIATION LEVELS IN REACTOR PIT DURING SECOND REFUELING

Date	Shielding and Port Access Status	Radiation Levels (mr/hr) Minimum-Maximum	Locations Monitored*
9/19/61	Shielded motor guide tubes	30-45	Inner fuel ports H-7 & H-9
		19-45	Outer fuel ports
		20-50	Inner seal ring
		12-19	Mid-radius of seal
		6-12	Near pit walls
		50	Multiport valve hot spot
		12-40	Flow measurement instruments
		19-180	Near mechanism array
9/21/61	Installed ASEWI and BEWI assemblies; removed shielding from some motor guide tubes	50-50	Inner fuel ports H-7 & H-9
		40-65	Outer fuel ports
		20-35	Inner seal ring
		10-20	Mid-radius of seal
		5-10	Near pit walls
		110	Multiport valve hot spot
		26-130	Flow measurement instruments
		70-140	Near mechanism array
9/26/61	Removed motor guide tube donut shielding; installed position indicator coils and stator water jackets	40-50	Inner fuel ports H-7 & H-9
		20-60	Outer fuel ports
		15-27	Inner seal ring
		12-20	Mid-radius of seal
		3-10	Near pit walls
		80	Multiport valve hot spot
		16-25	Flow measurement instruments
		30-70	Near mechanism array
9/27/61	Removed shielding inside trellis after taking FMI and seal ring readings	140-150	Inner fuel ports H-7 & H-9
		27-80	Outer fuel ports
		19-50	Inner seal ring
		11-19	Mid-radius of seal
		5-12	Near pit walls
		150	Multiport valve hot spot
		16-25	Flow measurement instruments
		600-800	Near mechanism array
10/2/61	After removal of multiport valve shielding; removed and reinstalled multiport valve stator; removed shielding from reactor vessel head	100-300	Inner fuel ports H-7 & H-9
		40-170	Outer fuel ports
		35-140	Inner seal ring
		17-50	Mid-radius of seal
		5-15	Near pit walls
		600	Multiport valve hot spot
		80-150	Flow measurement instruments
		70-80	Near mechanism array
10/6/61	After installation of reactor chamber dome	3.5	Outside reactor chamber dome near multiport valve

*Waist level readings except for hot spot reading near multiport valve and northwest FMI.

the placement of shielding on the reactor head. The most significant operation was the removal of the control rod shroud assemblies with the reactor pit dry. The maximum dose rate recorded during the removal of the first shroud was 150 mr/hr 4 inches from the upper end of the shroud. The dose rate about 20 feet from the lower end of the second shroud removed was 100 mr/hr. Because the radiation levels were within the tolerable limits, the remainder of the shrouds were removed and reinstalled dry. Instrumentation was removed and reinstalled dry. A dose rate of 2000 mr/hr one foot from the L-8 SMI thermocouple was recorded. Personnel were evacuated from the reactor pit during the shroud and instrumentation removal and reinstallation. The BEWI and ASEWI were removed underwater; however, the upper portions of BEWI H-9 and ASEWI J-14 were lifted above water three days later and indicated 150 mr/hr at 3.5 feet and 400 mr/hr at 2 feet. These assemblies were reinstalled dry. The dose rate along the canal walkway during installation was 70 mr/hr.

2. Smear Surveys, Air Samples, and Water Samples

The radiation samples taken during refueling show the effectiveness of continuous vacuum cleaning, wipedown, and mopping in controlling transferable beta-gamma contamination and air activity in the reactor pit. Alpha contamination was negligible as has been the case throughout the life of Core 1.

During the course of refueling the radiation control technicians took smear samples at various locations in the canal area and on the reactor pit floor and walls. Between 10 and 20 smear samples were taken per shift in the reactor pit. Smear surveys indicated the areas to be controlled or decontaminated. The control limitations were defined as follows: Clean area - < 100 dpm/100 cm^2; Controlled area - 100 to 500 dpm/100 cm^2; and Contaminated area- > 500 dpm/100 cm^2.

E. Radiation Alarms

Difficulties were experienced with the nuclear instrumentation system source range equipment during refueling. The coaxial cables and connectors used with this equipment are known to deteriorate under prolonged irradiation. This deterioration is not apparent until the cables and connectors are disturbed, as during a refueling operation.

The faulty cables and connectors resulted in erratic meter indications and periodic alarms. This condition was not unlike that which existed during Seed 1 - Seed 2 refueling. During the first operation, however, the alarm circuitry was in the main control room only. For Seed 2 - Seed 3 refueling, an additional alarm was added at the pit. This alarm required immediate evacuation and did not allow for evaluation. The evaluation of these deficiencies and corrective action for future refuelings is discussed in Section IV. F.

On August 28 the nuclear instrumentation alarms were initiated three separate times. On each occasion the reactor pit was evacuated, but each alarm had been caused by malfunction in the instrumentation circuitry. On September 9 the radiation alarm in the fuel storage pit was activated during the transportation of shroud I-12 from the deep pit shroud storage rack to the shroud storage rack on the south wall of the fuel storage pit. The dose rate 25 feet from the shroud was 100 mr/hr.

During the dry removal of shroud assemblies from E-11, G-4, H-4, and L-10, the air monitor at the west wall was affected by the direct radiation from the shrouds, and it recorded levels 5 times higher than normal.

F. Decontamination of Control Rod Drive Mechanism Components

Decontamination of control rod drive mechanism components was performed entirely at the Shippingport site during the second refueling instead of shipping the mechanisms to a contractor for final decontamination, as was done during the first refueling.

A combination of hand scrubbing and ultrasonic baths was used to decontaminate mechanism components which contained small crevices where crud had built up. Other mechanism components with smooth surfaces were scrubbed in a solution of Alconox and water. Radiation measurements indicated that the position indicator coil top cover, coil spacer, coil housing spacer, and external assembly were approximately equivalent to the background beta-gamma radiation level of 0.05 mr/hour, and, therefore, did not require decontamination. To determine the effectiveness of the decontamination processes, certain components were monitored before and after decontamination.

Relatively large variations in radiation readings were noted among varying points on identical mechanism components. Fluctuations of this nature may be expected to occur because of the change in high spots with respect to the specified monitoring points, and because of the critical nature of monitoring methods at distances as small as 1/4 inch from the radiation source. However, the high radiation readings did not cause trouble in transporting and handling parts during refueling.

Eight survey points were selected along the length of each rotor assembly. Table V lists the Jordan survey meter readings from eight typical rotor assemblies. The average of the eight readings before ultrasonic decontamination of each rotor was divided by the average of the eight readings after decontamination to obtain the decontamination factor. The maximum (not spot) reading and the background level when the readings were taken are given. The top of the rotors was consistently the hottest point.

Table VI lists the Jordan survey meter reading on the damper guides of the same eight mechanisms for which the rotor readings were listed. The hot spot was most frequently located at point 3 on the bottom of the damper guide. The damper guides were decontaminated by scrubbing without the ultrasonic bath. With this exception the remainder of the data is similar to the data listed in Table V. Table VII lists the radiation levels on the damper guide cup washers, which were not decontaminated because they were replaced by new ones.

Table VIII summarizes the results of the Duquesne Light Company test requiring radiation survey of six thermal barriers.

Table IX summarizes the Jordan survey meter readings for the motor tube-guide tube assemblies of the eight mechanisms surveyed throughout this decontamination analysis.

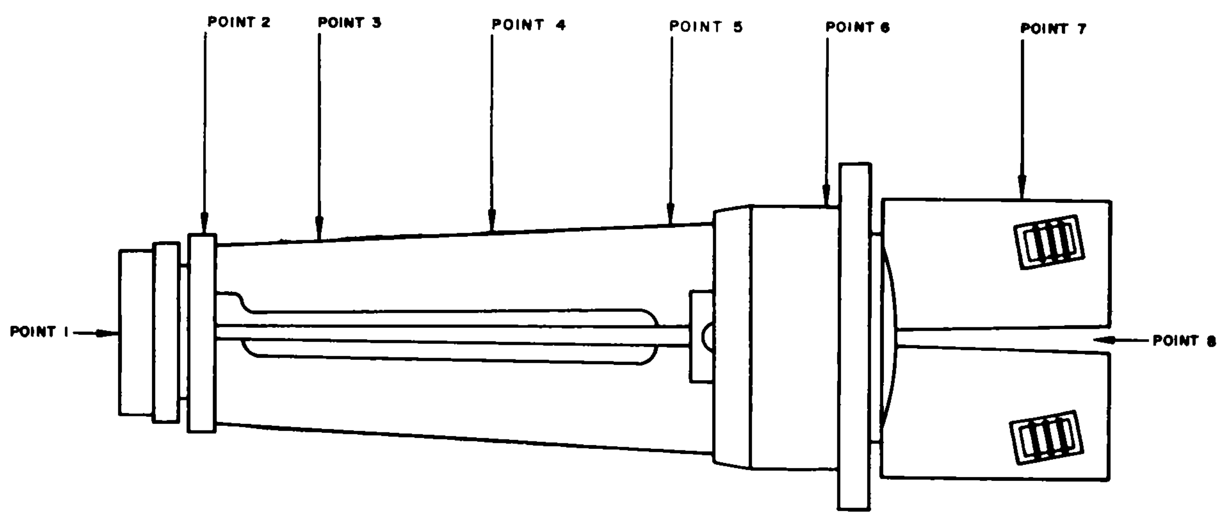


Figure 34. Rotor Assembly Radiation Survey Points.

TABLE V

ROTOR ASSEMBLY RADIATION SURVEY

A. Gamma Activity before Ultrasonic Decontamination (mr/hr)

Point on Figure 34	K-5	L-9	D-9	L-5	J-12	H-4	F-12	D-5
1	800	350	1500	3000	800	2000	1300	350
2	350	350	600	3300	800	1500	700	200
3	150	300	350	800	250	600	300	150
4	65	70	120	200	110	150	140	80
5	70	70	95	90	65	140	110	70
6	110	110	200	100	110	200	200	140
7	110	120	200	100	120	150	250	130
8	125	130	250	120	150	300	250	140
Average	225	190	415	960	300	630	410	160
Maximum	800	350	1500	3300	800	2000	1300	350
Background	2	2	2	2	2	2	2	2

B. Gamma Activity after Ultrasonic Decontamination (mr/hr)

	Number of Decontamination Baths (~ 2 hr)							
	five	two	two	two	two	five	two	one
1	350	100	300	350	225	600	200	400
2	200	60	250	200	150	450	150	400
3	40	25	55	40	50	90	60	100
4	15	20	30	15	25	40	40	50
5	15	20	35	12	30	40	45	60
6	20	35	45	15	60	90	70	100
7	30	35	60	15	70	85	80	100
8	60	40	100	20	80	125	150	100
Average	90	40	110	85	85	190	100	165
Maximum	350	100	300	350	225	600	200	400
Background	< 0.5	< 0.5	<0.5	<0.5	<0.5	<0.5	< 0.5	<0.5
Decontamination Factor of Averages	2.5	4.7	3.8	11.3	3.5	3.3	4.1	0.9

All readings are Jordan contact readings taken 1/8 in. to 1/4 in. from the surface of the assembly. Note that the hot spot is point 1 near the top of the rotor.

TABLE VI

DAMPER GUIDE RADIATION SURVEY

A. Gamma Activity before Decontamination (mr/hr)

Points on Figure 35	K-5	L-9	D-9	L-5	J-12	H-4	F-12	D-5
1	6	15	2000	30	7	50	450	95
2	6	20	100	30	10	150	115	60
3	8	45	400	30	20	1000	800	400
Average	7	27	833	30	12	400	455	185
Maximum	8	45	200	30	20	1000	800	400
Background	2	2	2	2	2	2	2	2

B. Gamma Activity after Decontamination (mr/hr)

Points on Figure 35	K-5	L-9	D-9	L-5	J-12	H-4	F-12	D-5
1	0.5	2.0	1.0	1.0	2.5	0.8	6.5	3.0
2	1.5	2.0	2.0	0.9	2.0	2.0	3.5	3.5
3	6.5	9.0	12.0	6.0	7.0	15.0	3.5	20.0
Average	2.8	4.3	5.0	2.6	3.8	5.9	4.5	8.8
Maximum	6.5	9.0	12.0	6.0	7.0	15.0	6.5	20.0
Background	< 0.5	< 0.5	< 0.5	< 0.5	<0.5	< 0.5	< 0.5	<0.5
Decontamination Factor of Averages	2.5	6.3	167.0	11.5	3.2	67.7	113.7	21.3

All readings are Jordan contact readings taken 1/8 in. to 1/4 in. from the surface of the assembly.

Decontamination was accomplished by hand scrubbing the damper guides in a solution of alconox and water.

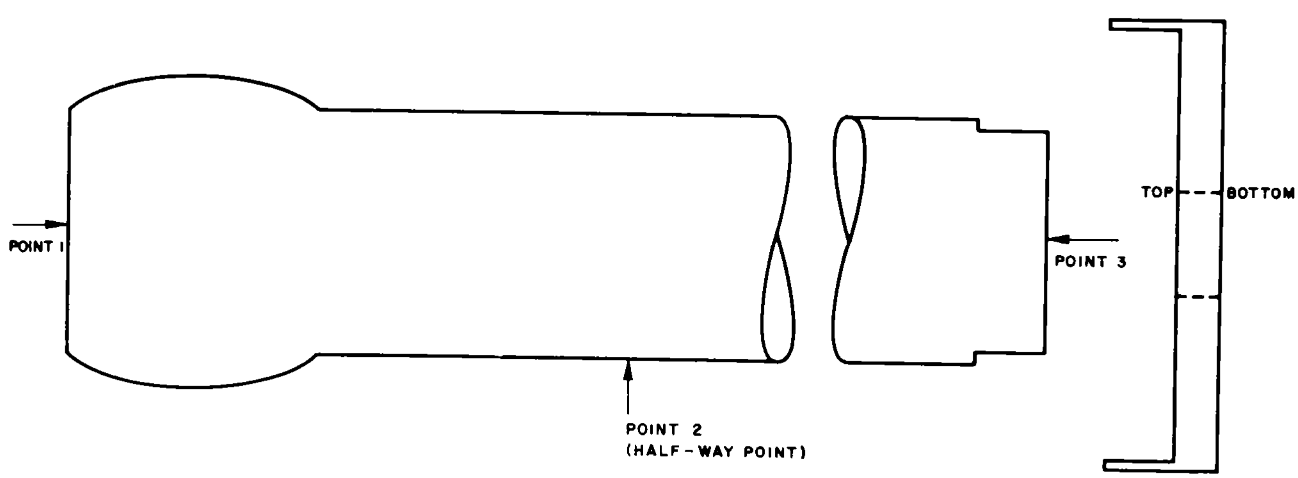


Figure 35. Damper Guide Radiation Survey Points.

TABLE VII

DAMPER GUIDE CAP WASHER SURVEY

Point on Figure 35	K-5	L-9	D-9	L-5	J-12	H-4	F-12	D-5
	Gamma Activity after Removal (mr/hr)							
Top	15	20	500	30	9	2500	550	550
Bottom	20	23	400	30	8	2000	500	550
Average	18	22	450	30	8	2250	530	550
Maximum	20	23	500	30	9	2500	550	550
Background	2	2	2	2	2	2	2	2

All readings are Jordan contact readings taken 1/8 in. to 1/4 in. from the surface washer.

The cut washers were not decontaminated; new ones were placed in the mechanisms.

TABLE VIII

THERMAL BARRIER SURVEY

Gamma Activity (mr/hr)

Core Location	Instrument* J - Jordan C - Cutie Pie	Bushing Retainers Before Decontamination Maximum	Bushing Retainers Before Decontamination Minimum	General Survey Before Decontamination Maximum †	General Survey Before Decontamination Minimum ‡	General Survey After Decontamination Maximum
G-4	J	300	125	350	125	
	C	350	300	850	300	120
K-4	J	150	100	350	300	
	C	300	200	900	700	120
D-7	J	200	200	300	300	
	C	300	300	1000	800	no data
D-8	J	200	110	5000	600	
	C	500	500	6500	1000	300
E-11	J	250	250	3000	550	
	C	400	400	7000	1000	155
F-12	J	125	100	400	200	
	C	500	400	900	500	185

* Survey instruments were held approximately 1 in. from thermal barrier component.

† Maximum readings occurred at bottom fin of barrier.

‡ Minimum readings occurred at top fin of barrier.

TABLE IX

MOTOR TUBE - GUIDE TUBE ASSEMBLY SURVEY

A. Gamma Activity before Decontamination (mr/hr)

Point on Figure 36	K-5	L-9	D-9	I-5	J-12	H-4	F-12	D-5
1	35	80	60	60	60	200	90	80
2	9	75	40	40	40	100	50	55
3	9	36	20	22	12	20	45	30
4	10	35	25	25	15	40	50	30
5	40	10	4	3	2.5	7	40	7
6	30	2	3	1.5	1.5	3	25	2
7	40	90	40	40	25	50	40	60
8	20	30	30	40	30	95	50	40
9	30	30	30	45	30	90	45	40
10	35	16	80	65	90	90	70	100
Average	25	40	33	34	31	70	50	44
Maximum	40	90	80	65	90	200	90	100
Background	2	2	2	2	2	2	2	2

B. Gamma Activity after Decontamination (mr/hr)

Point on Figure 36	K-5	L-9	D-9	I-5	J-12	H-4	F-12	D-5
1	7	11	22	7	14	24	15	20
2	6	5	23	6	15	18	15	15
3	2.5	8	12	15	12	15	25	9
4	4	15	12	12	11	20	20	15
5	0.9	55	2	2	20	6	1.5	5
6	<0.5	<0.5	0.9	<0.5	0.6	3	0.5	0.5
7	20	50	20	20	15	15	25	40
8	3.5	5	15	15	12	11	11	10
9	4.0	4	18	18	12	12	15	11
10	1.5	10	40	40	20	45	50	55
Average	5.0	11.4	16.5	13.5	13.2	18.9	17.7	18.1
Maximum	20	50	40	40	20	45	50	55
Background	< 0.5	<0.5	<0.5	<0.5	<0.5	<0.5	< 0.5	< 0.5
Decontamination Factor of Averages	5.0	3.5	2.9	2.6	2.3	3.7	2.8	2.4

All readings are Jordan contact readings taken 1/8 in. to 1/4 in. from the surface of the assembly.

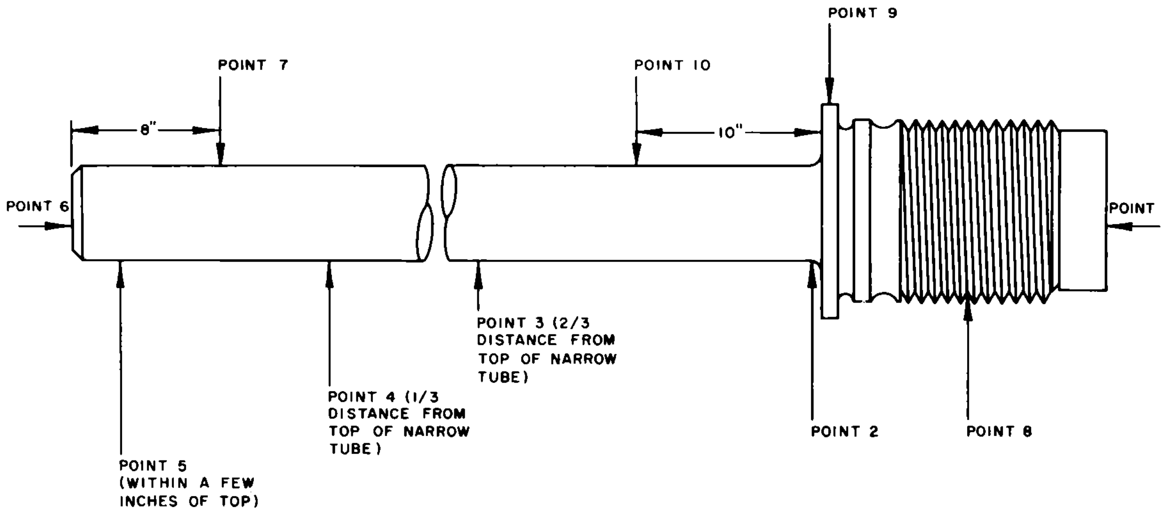


Figure 36. Mechanism Motor Tube — Guide Tube Assembly Radiation Survey Points.

VI. PERFORMANCE OF REACTOR COMPONENTS DURING TWO SEED LIVES OF CORE 1 OPERATION

Visual examination of a number of core components during refueling revealed no abnormalities or unusual conditions and it was determined that all reactor components remain structurally adequate for continued operation of Core 1 with further seeds. The results of these inspections, including control rod drive mechanism, shroud, control rod, and instrumentation assemblies, are discussed in the ensuing paragraphs.

In addition to the confirmatory examinations at Shippingport, additional destructive examinations are being performed on certain stainless steel shafting, and on the seed and blanket fuel materials to determine the effects of long-term reactor exposure on the properties of the PWR Core 1 structural and fuel materials. These examinations are still in progress and are described below only where significant results have already been obtained. Examinations not reported will be included in subsequent reports.

A. Examination of Nonfuel Components

This included visual examination of each of the 32 control rod drive mechanisms and lead screws; visual examinations of five shrouds, three BEWI's, two ASEWI's, and three control rods; and, destructive examination of three lead screw-scram shaft assemblies and a control rod.

1. Control Rod Drive Mechanisms

The mechanism rotors were inspected locally to assure that there were no gross flaws in the roller nuts or bearings and that they were adequate for use with Seed 3. This inspection was preceded by ultrasonic cleaning to remove loosely adhering material and to reduce the handling complications from radioactivity. The work was performed in the decontamination room under clean room conditions and required the use of protective clothing.

The inspection of the roller nut and ball bearing components consisted of careful visual examination of wearing surfaces and a comparison between bearings being inspected and a calibrated set of sample bearings. The radial play of the synchronizing ball bearing could not be checked without disassembly; however, operating experience has shown that wear on this bearing is negligible in comparison with that on the radial and thrust bearings.

In general, the results of the inspection were as predicted and all mechanisms were found to be adequate for reinstallation into the reactor. The ball bearing wear was within the normal range expected, and the bearings were found to have adequate wear life for at least Seed 3 operation. The chipping, pitting,and wear marks observed on the roller lands (Figure 37) are also normal for these parts; these marks were not deep and the affected area was small. The chipping which occurred on the lower edge (radius) of the roller lands has little effect on mechanism operation or wear life, since these surfaces are loaded only during latching. There were no apparent signs of ratcheting which could be attributed to Seed 2 operation.

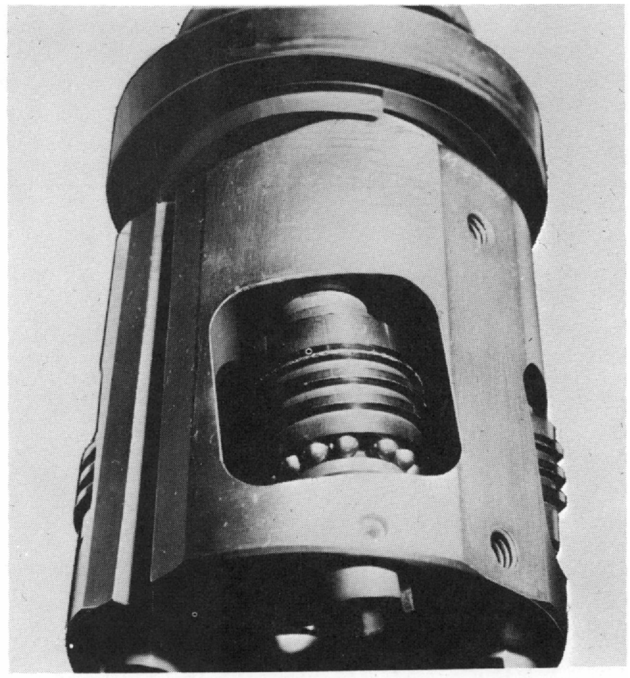


Figure 37. View of Mechanism Rotor Assembly Showing Chipping of Roller Lands.

2. Control Rod Shrouds

Five shrouds, located during Seed 2 operation at coordinates E-12, H-12, H-4, K-5, and J-4, were inspected during the refueling. An over-all inspection of the external surfaces for evidences of dimensional distortion, fretting, and corrosion was made with the underwater periscope. A similar inspection of the four internal bearing surfaces was made with the underwater boroscope. No significant defects were noted. Several minor scratches on some of the bearing surfaces, and mars and longitudinal scratches on the outer shroud surfaces appeared to be the result of the shrouds rubbing on portions of the mechanism housing ledges during their removal from the core. All 32 Seed 2 shrouds were reinstalled for used during the life of Seed 3, except the one at location J-4 which was retained for more detailed examination.

3. Lead Screw - Scram Shaft Assemblies

Five scram shafts, the lower ends and upper threads of three tie rods, and thirty lead screws were inspected at Shippingport during the Seed 3 refueling. The scram shafts, equipped with bails, could be partially withdrawn from their shrouds for inspection; the shrouds were hung in clips on the canal wall. The lead screws were inspected as the components were slowly hoisted past a Binocular Microscope. Magnifications in the range of 3.5 to 12.5 were obtained.

The topmost two feet above the latch area were clean, bright, and free of imperfections. Although the latch area (Figure 38) was marked because of repeated engagement (on startup) of the mechanism roller drive nuts, this area is not detrimental to operation. Below the latch area were smaller marks caused by ratcheting. It was interesting to note the absence of ratchet marks on the I-6 and K-11 lead screws, which were new at the start of Seed 2, indicating that no ratcheting occurred during Seed 2 operation.

In the course of the on-site visual examination of lead screws, an indication which appeared to be a crack was observed in a thread root of the E-12 lead screw. The E-12 tie rod, lead screw, and scram shaft were removed and returned to the Bettis Laboratory for nondestructive inspection. Also returned to Bettis were the lead screws, spline shafts, and tie rods from the K-5 and J-4

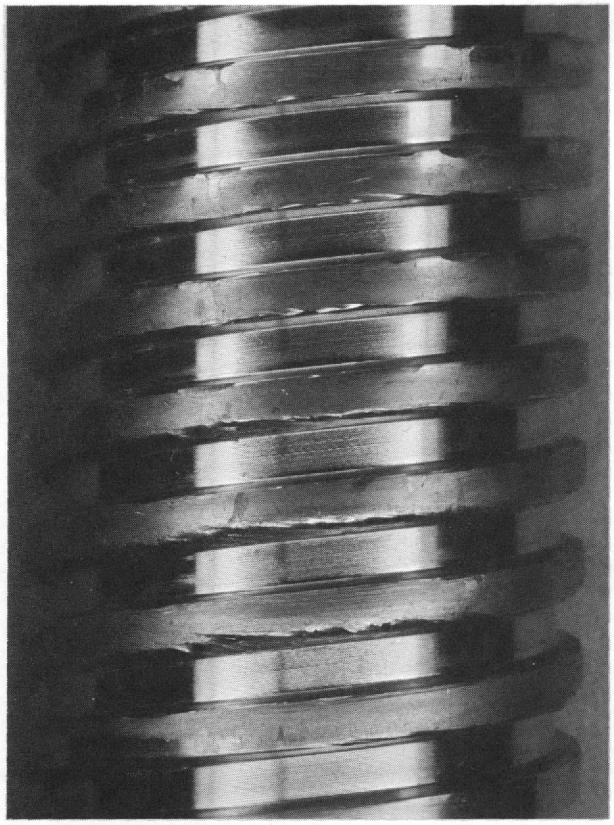


Figure 38. Latch Area of E-12 Lead Screw after Seed 2 Life; X8.

positions. These three assemblies were replaced with new unirradiated scram shaft assemblies for Seed 3 operation.

a. Nondestructive Tests—Fluorescent penetrant and fluorescent magnetic particle inspections were performed on the above scram shaft assembly components in the as-received condition after they had been subjected to dynamic stress equivalent to twice the maximum operating stress. No cracks were observed and all components appeared to be acceptable for further reactor use.

A slightly different decontamination procedure was used to prepare the E-12 lead screw for inspection. To prevent possible surface material smearing which might mask small imperfections, decontamination was performed using APAC, detergent, and bristle brushes. Because no abrasive cleaning methods were used on this part, many more fluorescent penetrant and fluorescent magnetic particle indications were observed in this component than in the others inspected. The indication noted during on-site inspection was barely discernible after decontamination and was not observed after either fluorescent penetrant or magnetic particle inspection. Visual examination at 60X showed it to be only a deep scratch from the grinding operation.

b. Mechanical Properties Tests. Mechanical properties tests were made on the K-5 scram shaft assembly to determine the effects of aging on 17-4 PH components during service in two seed lifetimes. The K-5 assembly was selected because, with respect to reactor centerlines, it was positioned symmetrical to the K-11 assembly and was, therefore, subjected to equivalent environmental conditions. The K-11 assembly had been evaluated after Seed 1 life. Test results are shown in Tables X and XI. The significant conclusions from these tests are:

1) Long time service at operating temperatures results in a slight increase in NDTT; however, the effect is not of sufficient magnitude to affect the ability of the parts to perform their prescribed functions. The low stresses on the lead screw will not cause brittle fracture to occur.
2) Long time service at 500-515 F has little or effect on the tensile properties of 17-4 PH aged at either 875 or 1100 F.
3) The K-5 scram shaft assembly components were free from microfissures as evidenced by the ductility in tension and the good impact properties of the material.
4) The maximum radiation to which the scram shaft assembly was exposed (3.1×10^{18} nvt) has no significant effect on the mechanical properties of 17-4 PH in the H1100 condition.
5) Minor variations in the H875 heat treatment and heat treatment response of 17-4 PH have

a greater effect on hardness and NDTT than long time service at 500 F.

In summary, the results of the mechanical proof tests indicate the adequacy of the 17-4 PH stainless steel components in the PWR Core 1 application during Seed 3 application. Scram shaft components from K-5 and J-4 core locations were dynamically loaded to twice operating stress after Seed 2 operation (components from K-11 and L-6 were similarly tested after Seed 1 operation). Repetitive tests at this condition were conducted; no adverse effects resulted.

4. BEWI Assemblies and ASEWI Assemblies

Of the six BEWI and two ASEWI assemblies which were removed with Seed 2, three BEWI (J-2, H-7, H-9) and the two (N-10, J-14) ASEWI assemblies were reinstalled with Seed 3. These irradiated assemblies were inspected while in the fuel storage pit. ASEWI N-10 was inspected under water, using the periscope. A slight gouge mark was noted on the housing assembly. A radiation survey was performed on ASEWI J-14 and BEWI H-9 to determine if subsequent examinations and reinstallation could safely be performed dry. The radiation levels were within tolerable limits and BEWI's J-2, H-7, H-9, and ASEWI J-14 were inspected dry. No defects were found and photographs were taken of the spring-loaded bearing pads and thermocouple fingers.

5. Control Rods

Thirty of the thirty-two control rods were replaced by new assemblies at the end of Seed 2 lifetime. The control rod positions which receive the greatest radiation are K-5, K-11, E-5, and E-11. K-11 was removed for examination after Seed 1 lifetime. After Seed 2 lifetime, three control rods, those from positions E-5, E-11 and K-5, were each examined under water with a periscope and underwater lighting. The rod from the K-5 position was shipped to the Expended Core Facility for destructive examination and testing, while the other two were reinstalled for additional irradiation.

The visual examinations revealed no unusual conditions of wear nor any dimensional distortions. Vertical wear markings were noted on the bearing pads, rubbings shoes, and, to a lesser extent, along the length of the rods near the blade tips. Similar observations were made during the first refueling, and the extent of wear was not considered to be excessive. This conclusion is reinforced by operating experience, since there have been no cases of control rod insertion or withdrawal difficulties. The connections between the control rod and the scram shaft assembly appeared to be clean and free from abnormal corrosion and galling; this indication was confirmed by the fact that this connection was disassembled in each case without difficulty.

The main reason for examining the control rods was to evaluate the extent of corrosion in the hafnium-Zircaloy transition weld areas and to check for any evidence of excessive corrosion in the high-flux areas occasioned by the buildup of tantalum via transmutation. No accelerated corrosion effect of tantalum was visually discernible, and the weld areas were found to exhibit corrosion to no greater extent than did some rods after Seed 1 operation (Figure 39).

It was concluded that these two control rods would safely operate throughout Seed 3 life because (1) examination of the weld joint prior to reinstallation revealed no gross defects in the weld joint and no significant increase in corrosion over that noted at the end of Seed 1 life, and (2) tests at Bettis have shown that the tensile strength of the weld joint material would remain high (about fourteen times higher than the maximum operating stress) even after coolant exposure equivalent to four seed lives. A detailed examination of the weld joint of one of the rods removed after Seed 2 operation has been completed and confirms the above conclusion. Specifically, the area of accelerated corrosion remains small (about 10 percent of the weld area), and the tensile strength remains about fourteen times higher than the maximum operating stress. No unusual condition was noted during the detailed examination of the control rod.

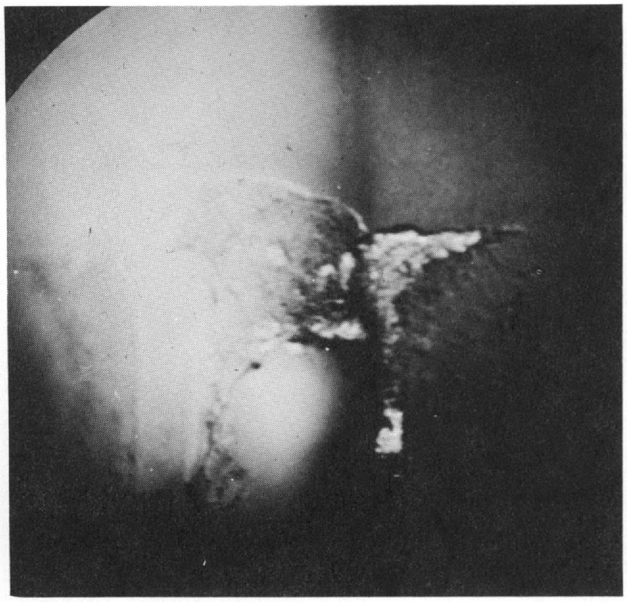


Figure 39. Closeup View of Accelerated Corrosion in the Transition Weld Area of the Rod Removed from Position E-5; 2.5X.

B. Examination of Seed Fuel

Two Seed 2 cluster assemblies representative of the highest and lowest burnup regions of the seed were chosen for prompt visual and dimensional examination at Shippingport. The results of the visual examinations of the seed assemblies revealed no unanticipated phenomena and no deterioration. The Zircaloy portions were covered with a normal black lustrous corrosion film unchanged since insertion. No significant crud deposits were observed on any parts open to visual examination. The top and bottom stainless steel extremities of the seed cluster evidenced a dark black film. All moving parts of the seed cluster (those used for positioning and latching the fuel) operated as designed without difficulty, indicating no adverse effects of corrosion, fretting, or crud deposits.

A total of nine Seed 2 clusters, including the two mentioned, were measured for length at Shippingport before shipment to ECF. The measurements were made to determine the seed length increase

TABLE X

IMPACT PROPERTIES OF 17-4 PH COMPONENTS FROM PWR K-5 SCRAM SHAFT ASSEMBLY

Component	Heat Treatment	Service Environment (F)	FATT ‡ 50% shear (F)	Estimated NDTT** (F)	Hardness (R_c)
Lead Screw*	H875	None	260	260	43.3
Lead Screw	H875	150-300	310	310	44.3
Lead Screw	H875	500-515	275	275	41.5
Tie Rod †	H1100	150-300	100	50	36.2
Tie Rod	H1100	150-300	50	50	36.0
Tie Rod †	H1100	500-515	160	110	37.3
Tie Rod	H1100	500-515	110	110	36.9
Spline Shaft	H1100	500-515	130	130	38.8

* Tests conducted in 1958 on a lead screw of the same heat and heat treatment.

† Standard Charpy V-notch specimens; all others are subsize (0.200-in. sq.) V-notch specimens.

‡ Fracture appearance transition temperature.

** NDTT estimated as corresponding to FATT for subsize and 50 F below FATT for standard Charpy V-notch impact tests, as discussed in text.

TABLE XI

TENSILE PROPERTIES OF 17-4 PH COMPONENTS FROM PWR K-5 SCRAM SHAFT ASSEMBLY*

Component	Heat Treatment	Service Environment (F)	Test Temperature (F)	Ultimate Tensile Strength (psi)	Yield Strength 0.2% Offset (psi)	Reduction in Area (%)	Elongation in 1 in. (%)
Lead Screw†	H875	None	75	216,150	213,880	47.7	7.6
Lead Screw	H875	150-300	75	208,350	202,000	43.5	8.4
Lead Screw	H875	150-300	300	189,250	181,200	40.2	6.9
Lead Screw	H875	150-300	500	177,150	163,600	33.5	7.0
Lead Screw	H875	500-515	75	193,000	184,550	54.9	10.2
Lead Screw	H875	500-515	300	172,250	160,600	64.0	10.2
Lead Screw	H875	500-515	500	166,800	149,600	53.2	9.5
Spline Shaft	H1100	500-515	75	179,600	175,650	62.0	7.0
Spline Shaft	H1100	500-515	300	163,350	160,000	59.2	7.4
Spline Shaft	H1100	500-515	500	155,650	151,300	44.7	4.8

* All test specimens were 0.125-in. diameter with 1-in. gauge length.

† Tests conducted in 1958 on a lead screw of the same heat and heat treatment.

experienced during Seed 2 operations as a result of fuel irradiation and to correlate the results with similar Seed 1 data previously obtained at the Expended Core Facility. The maximum recorded length increase measured was of the order of 1/16 inch. The measured growth on Seed 2 was slightly greater than Seed 1 which was to be expected because of the higher burnup.

C. Metallurgical Examination of Blanket Fuel Rod—Test Results

Blanket bundle number 0545 from assembly J-5 containing 120 Zircaloy-clad UO_2 fuel rods was subjected to metallurgical examination and burnup analysis. This bundle was exposed to in-pile conditions for both Seed 1 and Seed 2 lives for a total of approximately 13,700 EFPH and was located in the third position from the bottom of the assembly. It is considered to be the most highly depleted bundle in this region. Rod number ZDP-2087, located in the southwest corner of this bundle, is computed to be one of the most highly depleted rods in the blanket, with an average burnup of about 4×10^{20} fissions/cc at a maximum surface heat flux of 300,000 Btu/hr-ft^2 It was returned to the Bettis Hot Laboratory for destructive evaluation. Examination of this rod permits a direct comparison with a similar rod examined at the end of the first seed life. Additional rods from this bundle are being examined at ECF.

1. Visual Inspection of Cladding Surface

The rod was examined for evidence of corrosion using a stereo viewer and low-magnification (4 to 6.6X) photographs. No accelerated corrosion attack has taken place, and the rod retained its unirradiated black lustrous film over the entire surface (Figure 40). Some minor surface scratches were evident, but these were apparently due to the post-irradiation handling process.

2. Leak Testing of Sheath

Leak testing was conducted by immersing the rod in liquid nitrogen until temperature equilibrium was attained and then rapidly transferring the rod into an alcohol bath at room temperature. In this manner small defects in the sheath would become evident by a steady stream of gas bubbles emanating from the hole as the entrapped nitrogen gas boiled off. No leaks were detected in the rod sheath.

3. Diametral and Warpage Measurements

Dimensional measurements were taken on the diameter of the rod at half-inch stations along the length. To eliminate any effects of eccentricity the rod was then rotated 90 degrees and readings were again taken at the same station locations.

Measurements taken over the rod showed an average value of 0.4108 inch and a range of 0.4097

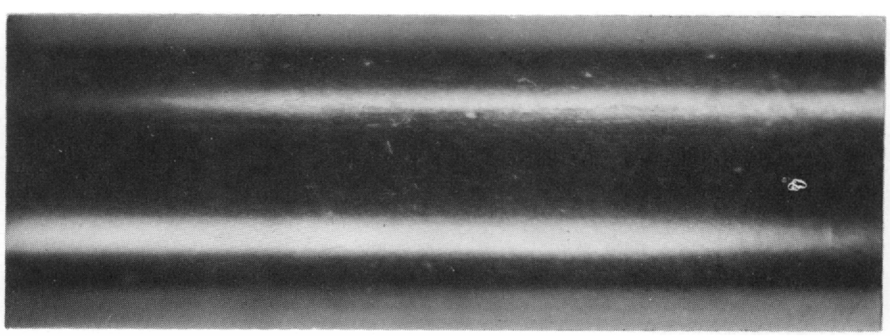


Figure 40. General View of Rod Surface; X4.3.

to 0.4116 inch. Since the in-bundle specification dimensions are 0.411 ±0.002 inch, these results show that the rod did not change in diameter as a result of the in-pile exposures.

4. Fission Gas Release

Upon completion of the measurements described above, the rod was prepared for checking fission gas release by placing it in a vacuum chamber and drilling through the 0.027-inch cladding at the interface between the end cap and fuel pellet with a 7/64-inch drill. The evolved fission gases were collected and then analyzed for stable fission gas atoms by gamma ray and mass spectrometry. A blank unirradiated rod was also punctured in order to determine background release.

The analysis of the total stable fission gases released from the UO_2 fuel indicates a release of 5.3×10^{18} stable fission gas atoms. Based on an estimated 4.1×10^{20} fissions/cc this figure represents about 0.29 percent of the total number of the noble gas atoms born in the fission process.

5. Metallographic Examination of Fuel and Cladding

Metallography samples were prepared by sectioning the fuel rod at the locations shown in Figure 41. This operation was accomplished by using a tubing cutter to section the cladding at pellet interfaces, thus keeping disturbance of the fuel to a minimum. A total of four fuel pellets were then pressure mounted using Hysol epoxy resin to penetrate each crack. The pellets were sectioned on a transverse cut and polished and etched to bring out the grain structure of the cladding or fuel. Longitudinal samples were prepared in a similar manner as shown in Figure 41. The Zircaloy cladding and end cap, designated as Met 4 and Met 5 in Figure 41, were examined in the longitudinal and transverse directions to examine the weld metal structure. The oxide film on the outside of the rod was examined under higher magnification, and average thickness values were obtained by direct measurement. Special mounting and polishing techniques were used so as not to disturb the film.

The cut faces of the sectioned fuel pellets displayed a radial crack pattern of the type shown in Figure 42. There was no fragmentation of fuel in the metallographic samples located in the body of the rod; however, the pellet located next to the end cap, Met 4A of Figure 41, fractured fairly extensively into large coherent pieces which became dislodged from the cladding. This fracture is attributed to the thermal stresses set up during the welding operation of the end cap. Microexamination of the transverse and longitudinal sections revealed fine uniform grains with no indication of grain growth or centerline melting. Typical grain size of the fuel at the pellet center is shown in Figure 43; that of UO_2 fuel from the most highly depleted rod at the end of the first seed life (1.77×10^{20}

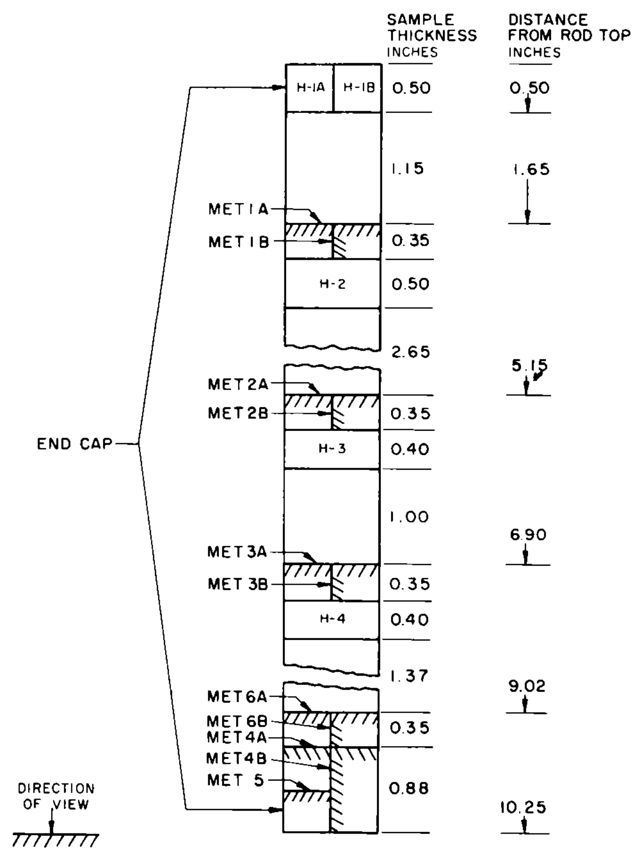


Figure 41. Sectioning of PWR Core 1 Blanket Fuel Rod for Hydrogen Analysis and Metallographic Examination.

Figure 42. Transverse View of Ground and Etched Oxide Fuel Pellet after Two Seed Lives of Operation; X6.6.

fissions/cc) is shown in Figure 44. Grain size at the clad surface is shown in Figure 45, and a typical microstructure of nonirradiated UO_2 is shown in Figure 46. These comparisons indicate that no significant structural changes occurred. The lack of structural or dimensional changes in the fuel after exposure is also shown in Figure 47 where the diametral clearances between fuel and cladding may be seen. The as-fabricated clearance was specified as 0.002 inch, which compares with the measured annulus of 0.0018 inch in Figure 47.

6. Hydrogen Analysis of Zircaloy Cladding

Cladding sections designated by "H" in Figure 41 were completely separated from fuel and carefully etched in boiling HNO_3 acid to remove any residual UO_2 fuel, then analyzed for hydrogen content by the hot vacuum extraction method. This method allowed a correlation between microstructurally observed zirconium hydride needles and the hydrogen content of the cladding.

The measured hydrogen contents of the cladding, using the hot extraction technique, are presented in Table XII. Assuming an average cladding surface temperature of 515 F over the entire period of pile exposure, and estimating an initial concentration of hydrogen of 40 ppm, the hydrogen pickup in the fuel rod cladding can be calculated as follows (Reference 3):

	H_2 (ppm)	Total H_2 (ppm)
Initial hydride after vacuum anneal	40	40
Corrosion test 3 days at 680 F	10	50
Pickup during Seed 1 (660 days)	15	65
Pickup during Seed 2 (462 days)*	6	71

* Although the blanket rods had about 132 percent more fuel depletion during Seed 2 lifetime than during Seed 1 (2.33×10^{20} fissions/cc), the total time exposed to hot water was less due to a greater reactor use factor.

TABLE XII. HYDRIDE ANALYSIS OF ROD ZDP-2087

Location	Hydrogen (ppm)*
H1A	97
H1B	133
H-2	†
H-3	68
H-4	79

*Analytical precision ±15 percent

†H-2 lost through temperature excursion of RF Generator

This accumulated hydrogen content value agrees very well with the average experimentally determined hydrogen content (73 ppm) of the two cladding samples H-3 and H-4 taken in the body of the rod. The end cap hydride analyses indicated concentrations that are 24 to 40 ppm higher than that observed in the cladding. Since the end caps act as heat sinks for the rod, and considering that the time in-pile is sufficiently long for thermal diffusion of hydrogen to take place, these results do not seem unreasonable (Reference 4). Figure 48 is a photomicrograph of a longitudinal section through the lower end cap and is assumed to be similar to a section through the upper end cap where the

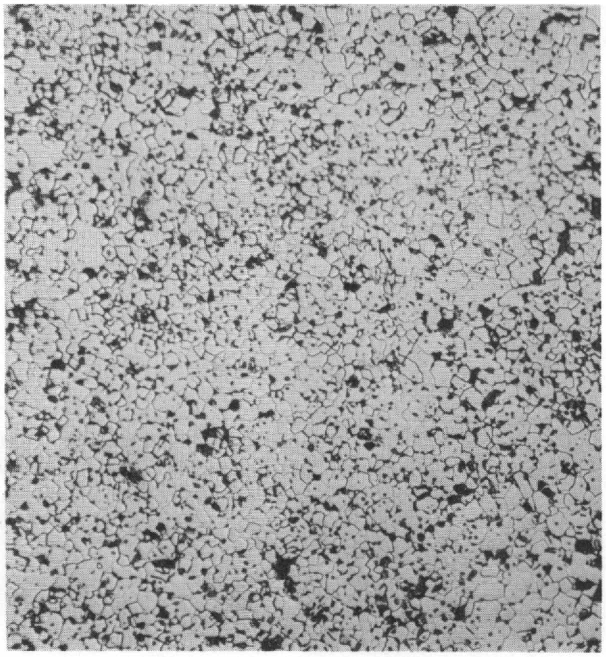


Figure 43. Typical of the Fine Grain Microstructure at Center of Pellet 2A at the End of the Second Seed Life; X250.

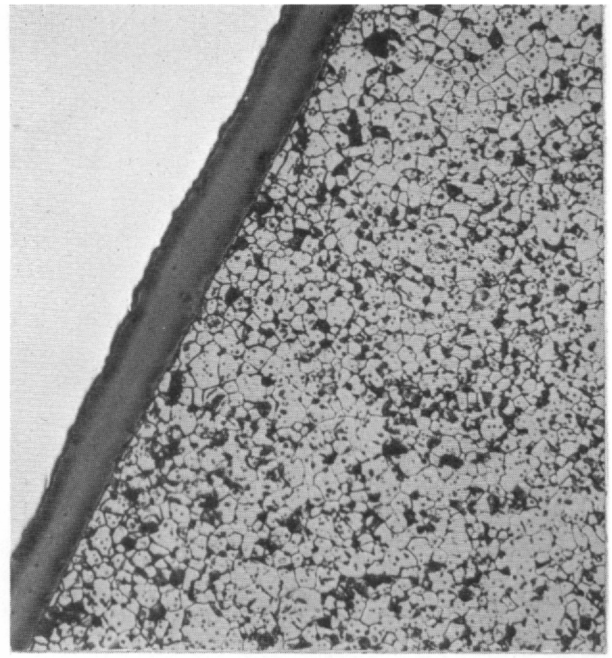


Figure 45. UO_2 Microstructure at Pellet Clad Interface of Pellet 2A; X250.

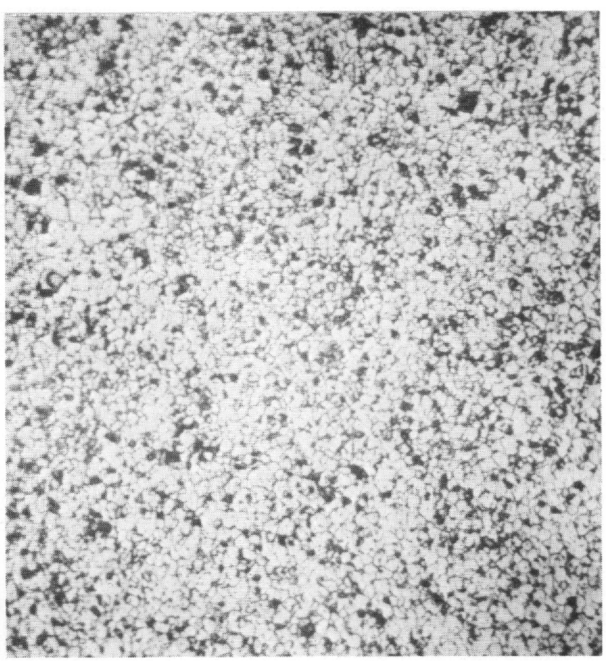


Figure 44. Fine Grain Microstructure of UO_2 at End of One Seed Life; X250. (No change in grain size in comparison with figure 43; micrographs are of same general rod area.)

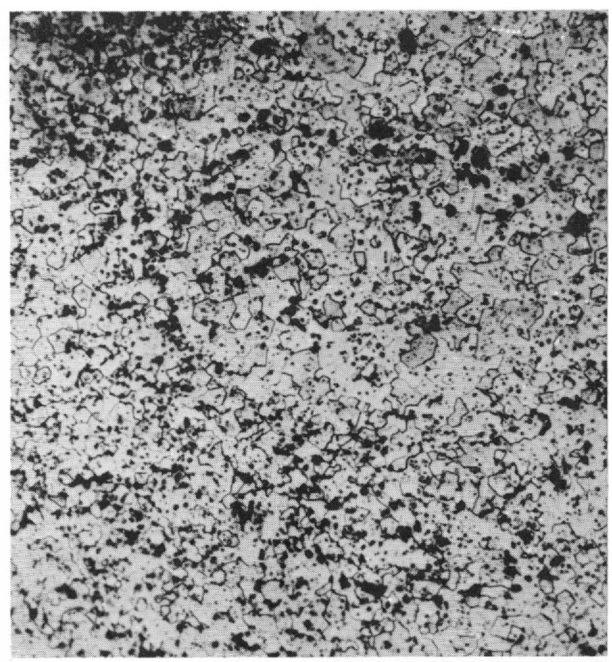


Figure 46. Typical Microstructure of Unirradiated UO_2; X250.

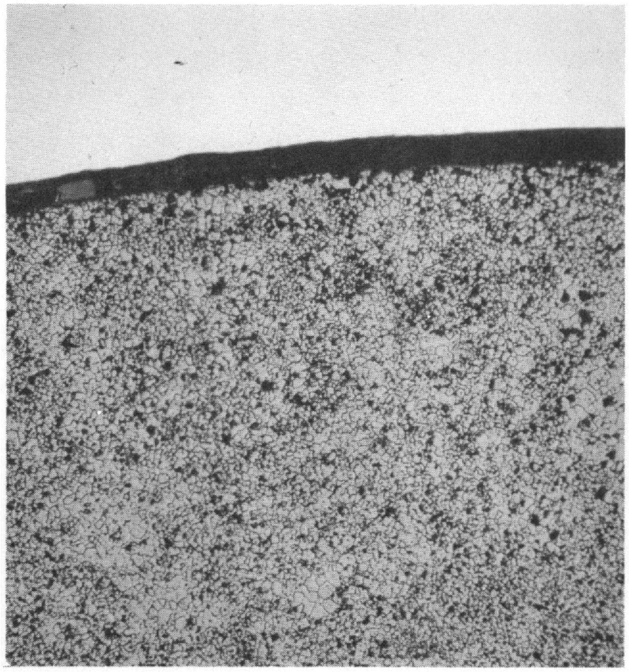


Figure 47. Fuel - Clad Interface of Typical Pellet Is Filled with Epoxy Resin and Retains the Diametral Clearance which Measures Approximately 0.0018 Inch; X100.

hydrogen analysis was made. Comparison of this structure to that of the cladding shown in Figure 49 (sample adjacent to H-3) confirms the low hydrogen content values obtained.

7. Measurement of Oxide Film Thickness

Measurements of the oxide film thickness made on the cladding are presented in Table XIII, and a representative photomicrograph is shown in Figure 50. These results showed an average oxide film thickness of 0.000056 inch. The theoretical oxide film buildup can be summarized as follows:

	Oxide Film (inches)	Total Buildup (inches)
Three days at 680 F	0.000024	0.000024
Buildup during Seed 1 (660 days)	0.000024	0.000048
Buildup during Seed 2 (462 days)	0.000011	0.000059

The theoretical value agrees very well with the average of the measured values.

TABLE XIII. ZrO_2 FILM THICKNESS OF ROD ZDP-2087

Measurement Location	Number of Readings	Film Thickness (inches)
Met 1-B	6	0.000070
2-B	6	0.000046
3-B	18	0.000055
4-A	51	0.000053

Average = 0.000056

Maximum noted = 0.000090 at Met 1-B

Minimum noted = 0.000035 at Met 4-A

8. Conclusions

Examination of a highly depleted rod in the PWR Core 1 blanket at the end of two seed lives and after an exposure of about 4 x 10^{20} fissions/cc

Figure 48. End Cap Weld; X150. (Note good weld penetration of longitudinal section.)

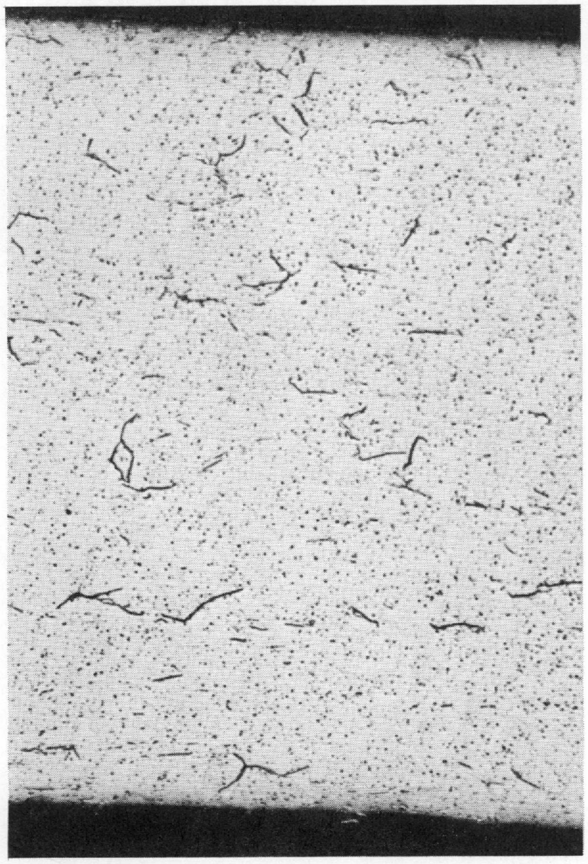


Figure 49. Hydride Distribution in Longitudinal Section of Cladding; Water Exposed Surface Is to the Left; X150.

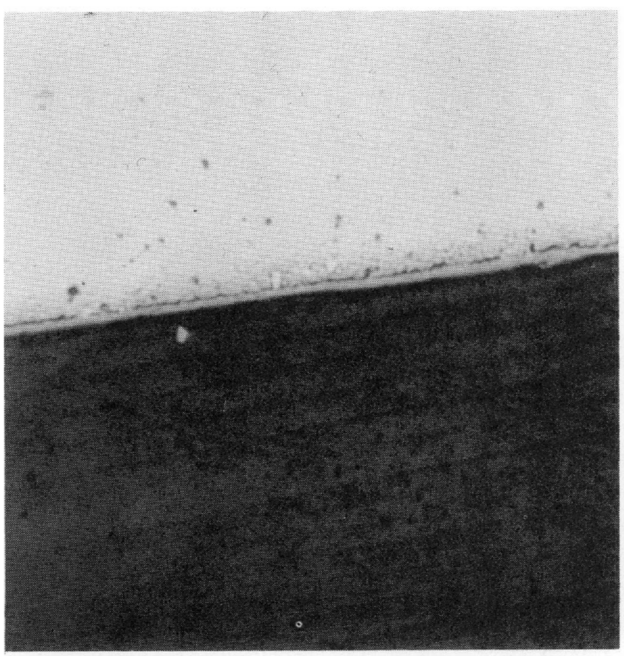


Figure 50. Representative ZrO_2 Film; X1600. (Note high conformity and continuity.)

(13,000 MWD/T) at a maximum surface heat flux of 300,000 Btu/hr-ft^2 has shown that no significant dimensional or microstructural changes have occurred in either the UO_2 fuel or the Zircaloy-2 cladding. The fission gas release values were extremely low and were consistent with results obtained in irradiation tests of similar elements subjected to the same exposure (Reference 5). The hydrogen pickup of the Zircaloy-2 cladding was consistent with theoretically calculated values expected under these operating conditions.

D. Examination of Defected Fuel Rod in Blanket Assembly J-5

1. Description

Complete cladding penetration was observed in rod IDJ 3269 in bundle 0435 of blanket assembly J-5 removed from the PWR Core 1 blanket at the end of Seed 2 life. The penetration, 0.004 inch diameter at the external clad surface and 4 5/8 inches from one end of the rod, apparently occurred shortly after startup of the Shippingport reactor. Operation of this rod for approximately 13,700 EFPH at a time averaged surface heat flux of 98,700 Btu/hr-ft^2 and a depletion of about 2.22×10^{20} fissions/cc was satisfactory in all respects and confirms in-pile loop data previously obtained on test rods. No dimensional changes of consequence or evidence of a waterlogging or hydriding type failure were observed despite the small size of the cladding penetration at the surface. No adverse effects on fuel appearance or structure due to operation in a steam environment were noted. The fuel-clad assembly gap was retained, indicating no appreciable fuel swelling. Also, the fission gas release from the adjacent nondefective irradiated rod IDJ 3340 was normal, 0.17 percent, and in agreement with the more highly depleted rod taken from the same shell assembly and bundle 0545 (Reference 6).

The maximum hydrogen content of the cladding of rod IDJ 3269 was about 200 ppm (about 150 ppm pick-up during operation) and decreased to 52 ppm at the rod end. The average hydrogen content was about 125 ppm, indicating that the rod could have operated for about 30,000 EFPH before this value would rise to 250 ppm, presently considered an upper limit for avoiding possible hydride-type failures. The distribution of hydride through the clad was normal in that a uniform gradient existed consistent with the known diffusion of hydrogen toward the colder external surface. Similarly, the thickness of corrosion oxide film at the external surface, 0.056 mil, was normal and consistent with that calculated from out-of-pile corrosion data for an equivalent exposure time and temperature. The thickness of the corrosion oxide film at the inside cladding surface and at the external surface in the flow pattern area downstream of the defect was 0.4 to 0.6 mil. Thickness values of this amount have been observed for defected in-pile loop tested elements and are attributed to the accelerated corrosion of Zircaloy contaminated by fission products when operating in a high temperature water environment. Thus, the behavior of the cladding, like the fuel, in the presence of the cladding penetration was normal and presented no unanticipated or unusual conditions.

The cladding penetration observed in defective rod IDJ 3269 was not caused by irradiation. The defect was initiated during fabrication as a transverse crack at the inside surface of the cladding. Cracks of this nature frequently occur in tube extrusion or drawing operations and are referred to as drawing cracks or checks. For example, other defects similar in nature but varying in size and depth were present in the same rod. Only one crack in rod IDJ 3269 apparently extended practically all the way through the clad thickness on initial insertion in the Shippingport core.

The nondestructive inspection techniques, in particular the Eddy Current Test (Probolog), used to detect tubing flaws were shown to be incapable of detecting the transverse type defects observed. Improved equipment now available (RADAC) or techniques such as die penetrant testing followed by boroscoping would probably reveal the defects.

Despite the presence of the defects in the tubing the mechanical strength of the tubing, as determined by pressure burst testing of both irradiated and nonirradiated rods, was not adversely affected. Burst pressures of 12,500-13,600 psi for the irradiated rods compare favorably with 11,000-11,400 psi for unirradiated rods. Thus, the known presence of defects which penetrated the cladding up to 88 percent of the original wall thickness did not significantly deteriorate burst pressure characteristics. The effect of the cladding defects on the cladding ductility was to increase local strain concentration and to reduce the average cladding ductility only in the case of cracks which penetrated the cladding 7 mils or more. The reduction in average cladding ductility, however, is quite small. In the worst case observed, involving unirradiated rod IDJ 3114 which had the largest total number of defects as well as the largest number of defects greater than 7 mils deep, a reduction of 11 percent in ductility was noted.

2. Conclusions

a. Rod IDJ 3269 satisfactorily operated in the PWR Core 1 Blanket, virtually since startup, for approximately 13,700 EFPH at a time averaged heat flux of 98,700 Btu/hr-ft^2, and achieved a fission depletion of about 2.22 x 10^{20} fissions/cc despite the presence of a defect which completely penetrated the cladding.

b. Based on the average hydrogen content of the cladding of about 125 ppm, rod IDJ 3269 could have continued in operation without restriction for at least 30,000 EFPH before the hydrogen content would have attained a level of about 250 ppm and resulted in possible gross failure.

c. The cladding defect in rod IDJ 3269 did not result from irradiation but originated during initial fabrication of the tubing. It was a transverse crack which started at the inside surface of the cladding at which point it was about 50 mils long and 60 mils wide. It decreased in size and changed in shape as it penetrated the tubing wall and was 4 mils in diameter at the exterior surface.

d. The presence of cladding defects does not significantly reduce the internal pressure required to cause rupture of the tubing in either irradiated or unirradiated rods. Room temperature burst pressures are consistently above 11,000 psi even when defects penetrate 88 percent of the clad thickness.

e. The presence of cladding defects causes a small reduction of average cladding ductility on pressure burst testing when they penetrate the cladding 7 mils or more. In the worst case of defects which penetrate up to 88 percent of the wall thickness the average cladding strain is reduced by only 11 percent.

REFERENCES

1. H. Feinroth and T. D. Sutter Jr., "The First Refueling of the Shippingport Atomic Power Station," WAPD-233, July 1960.
2. "The Shippingport Pressurized Water Reactor," Addison - Wesley Publishing Company, Inc., Reading, Mass., 1958.
3. B. Rubin, "Examination of PWR Core 1 Blanket Fuel Rods for Microstructure, Hydrogen Pickup and Burst Strength," WAPD-TM-264, February 1961, pp. 20-22.
4. J. M. Markowitz, "Thermal Diffusion of H in Zr," in "Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1958," Vol. 6, p. 235, United Nations, Geneva, 1958.
5. J. D. Eichenberg, R. M. Lieberman, and F. P. Mrazik, "Irradiation of UO_2 Fuel Rods - The X-1-L Experiment," WAPD-208, April 1960.
6. L. R. Lynam, "Metallurgical Examination of Fuel Rod from PWR-1 Blanket at the End of the Second Seed Life," WAPD-TM-308, December 1961.