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PIQUA NUCLEAR POWER FACILITY
REACTOR OPERATIONS ANALYSIS PROGRAM
SEMIANNUAL PROGRESS REPORT NO. 9
JULY 1 – DECEMBER 31, 1966

AEC Research and Development Report

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GENERAL, MISCELLANEOUS,
AND PROGRESS REPORTS

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I. REVIEW OF OPERATIONS

A. PREVIOUS OPERATING HISTORY SUMMARY

Initial criticality at the Piqua (Ohio) Nuclear Power Facility was achieved in June 1963, fuel loading was completed in July 1963, and power operation first took place in November of the same year. The first full-power operation was achieved on January 23, 1964. The detailed operating history summaries prior to the current reporting period are available in previously published documents.*

After accumulating over 10,000 Mwd integrated exposure, the reactor has been in a shutdown condition since January 13, 1966, shortly after which a sizeable deposit of carbonaceous material was discovered in the outer moderator region. Subsequently, a plan of action was developed which had the following objectives: (1) complete unloading of the core and inspection of all fuel elements, (2) general core and system cleanup of all carbonaceous deposits, (3) accomplish corrective system modifications, and (4) restoration of the plant to power operation. Actual fuel element unloading operations commenced on May 2, 1966, and were completed on June 6, 1966. Twenty of the 57 driver elements and 9 of the 13 control rod elements were found to have some "buckles" (folds, wrinkles, or creases) in the outer process tubes. A detailed description of the examination of the elements removed is presented in the preceding progress report.†

B. ON-SITE ACTIVITIES DURING REPORTING PERIOD

1. System Cleanup

The principal on-site activities during this period were directed toward core and system cleanup. Following the removal of all fuel elements, a series

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*PNPF Reactor Operations Analysis Program Semiannual Progress Reports:
No. 3, July 1-December 31, 1963
No. 4, January 1-June 30, 1964
No. 5, July 1-December 31, 1964
No. 6, January 1-June 30, 1965
No. 7, July 1, 1965-January 13, 1966
No. 8, January 1-June 30, 1966

†PNPF Reactor Operations Analysis Program Semiannual Progress Report:
No. 8, January 1-June 30, 1966

of carbonaceous samples were obtained from the outer moderator region through the use of a special coring tool which was lowered into the core positions normally occupied by the fuel elements. Several methods were considered for gross removal of the carbonaceous material from the reactor core. Included in these were pulverizing of the matter by the use of high-pressure water, direct mechanical contact with the material, remote contact, and any combination of these. Combustion of the material, or impact pulverization was discarded from safety considerations. A remote cleaning process was selected in which the lower guide grid, upper core grid plate, and inner support barrel were removed. With this method, the cleaning of the reactor was facilitated and the final inspection of the reactor made more reliably.

For shielding purposes during removal of the upper core at grid plate and core barrel, it was decided to replace the normal OMP terphenyl coolant with HB-40, thus permitting the maintenance of a liquid in the core at a lower temperature, (freezing point of OMP ~350°F, of HB-40 <32°F).* Accordingly, about 5000 gal of HB-40 at 340°F were added to the reactor vessel via the degasifier tank. Circulation through the vessel, pressurizer loop, and decay heat removal system was established with the vessel being cooled to approximately 110°F. At this point the vessel was opened and all of the support pins, which were installed after each fuel element removal, were removed and packaged for disposal off-site. HB-40 circulation was resumed when the temperature in the area adjacent to the lower guide grid rose to 130°F. This appears to have been due in part to a steam tracing circuit left in service on one of the vessel penetrations, and in part to the inadvertent addition of a quantity of 350°F OMP coolant to the HB-40. The vessel temperature went down to about 117°F, but would not go lower. As the situation seemed to be caused by leaking steam tracing circuits in the pressurizer system, all steam to the reactor building was shut off. The pressurizer pump was started, and the reactor vessel was cooled to 105°F.

Installation of television equipment for remote viewing of the removal of the inner barrel assembly was started in late July, and completed in early August. The barrel was raised to a height such that the upper core grid plate

*H. Mandel, "HWOCR Physical Properties of Some Polyphenyl Coolant," AI-CE-15.

was above the HB-40 level in the reactor vessel; radiation readings did not exceed 5 r/hr at the reactor floor level. At this point, a dummy fuel element was discovered suspended from the upper core grid plate. The barrel was lowered to its normal position, and the area near the suspended dummy was probed to determine its exact location. The dummy was located in Position F-5 which had been used recently in operational testing of the modified Mark II control rod. It was removed and placed in the spent fuel storage pool turret. The barrel was again raised until the core grid was above the HB-40 level. A television examination of the barrel assembly was made during its transfer across the 100-ft level to the storage pool. This examination is described in detail in Section I. B. 2. Following this transfer, the television monitoring equipment was relocated to permit surveillance of the reactor vessel during its cleanup.

A work platform was installed inside the outer core barrel, air-powered drilling equipment was lowered into the vessel; and, as part of the system modifications, three 3/4-in.-diameter vent holes were drilled in the outer barrel. The purpose of these vents is to eliminate possible trapped gas accumulation and provide local cooling flow. The former vent holes in the inner core barrel were sealed. At the completion of the drilling operation the work platform and equipment were removed, and the upper guide grid was reinstalled.

The plan developed for cleanup of the carbonaceous material consisted of the following principal steps.

1) Utilizing remote viewing techniques and a pool-type manipulator, grasp the large pieces of carbonaceous material and insert them into a remotely operated container. Insert plugs in the lower grid plate fuel element positions to minimize deposition of matter below the grid plate during the removal operations.

2) Verify the initial cleaning process by remote inspection of the vessel; employ a remotely controlled commercial vacuum system to sweep small particulates from the core region. Inspection of the area below the lower grid plate to be attempted by the use of a light pipe or borescope.

3) Perform a final comprehensive remote inspection of the reactor vessel to assure adequate removal of the carbonaceous material.

4) Replace the inner barrel and grid plate assembly, and the freshly charged in-vessel and degasifier filters; fill the system with HB-40. Raise the HB-40 temperature with pump heat then replace the HB-40 with OMP, and circulate at maximum flow. Continue the operation, recharging the filters as necessary, until satisfactory system purity is achieved.

5) Drain the heat transfer system, remove the in-vessel filters and drain the reactor vessel coolant to permit a visual inspection of the vessel.

6) Inspect all accessible areas inside the reactor vessel for carbonaceous material. The cleanliness criteria are based upon: (a) complete viewing of the core area, to ascertain that foreign material is not left, and (b) verification that each opening in the lower core grid plate is clean by inserting a plug gauge.

7) The surge tank, the pump inlet screens, the auxiliary building drain tank, and the reactor building drain tank will be opened and inspected. Any debris will be removed. Both upper and lower guide grid plates will be inspected at the 100-ft level.

8) After approval of the inspection by the plant superintendent, fill the system with purified organic coolant and maintain circulation and cleanliness.

Following the draining of the HB-40 from the vessel and P/D loop to T-1, in-vessel lights were installed for viewing of the cleanup procedure. The upper grid was also removed to permit inspection of the carbonaceous mass. At first, considerable organic fuming was encountered and visibility was limited. However, it was observed that approximately 50% of the coke mass was still in an upright position, with the remainder having broken off and fallen to the lower core grid plate. Other than this centrally located coke mass, the surface of the lower core grid plate appeared to be clean. Figures 1 and 2 illustrate the deposit as first observed. With improvement of visibility, it was discovered that lower grid plate plugs had not been installed in core positions K-7 and B-15 after the nuclear instrument thimbles had been removed from these locations; plugs were installed.

Removal of the carbonaceous material was initiated after the manipulator was installed and a 30-gal drum, used as a receptacle for coke fragments, was

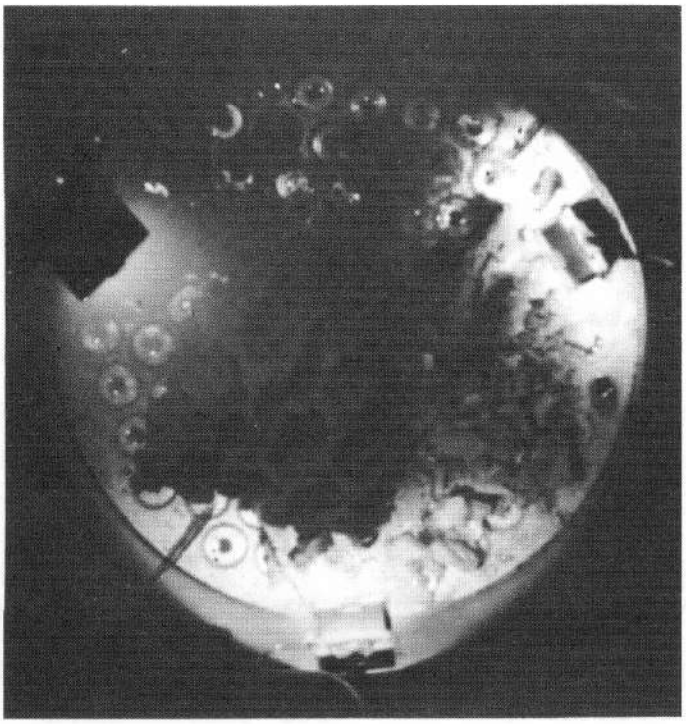

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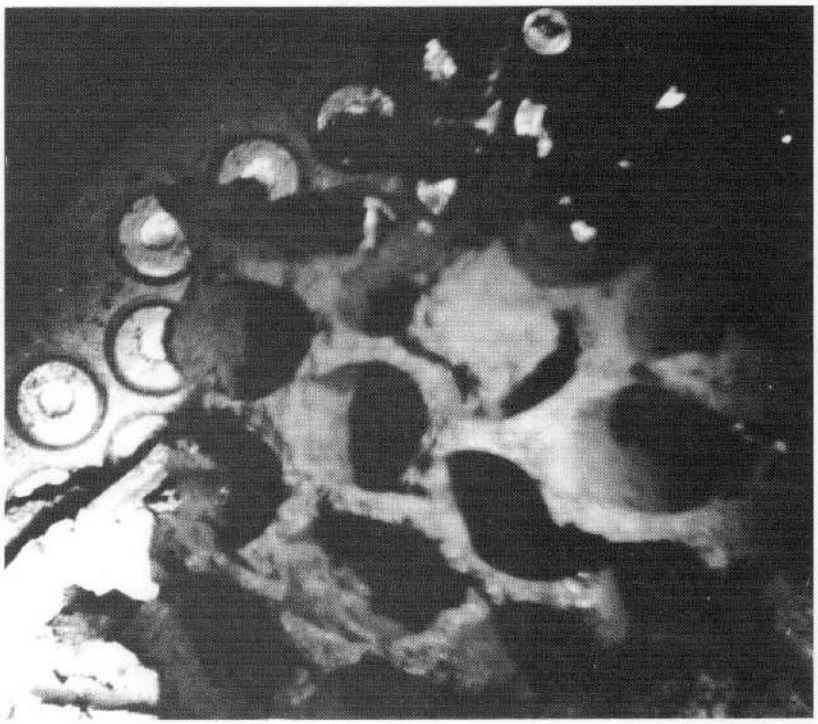
Figure 1. Coke Mass From Reactor Floor Level



6-23-67 UNC 7686-5107

Figure 2. Closeup View of Coke Mass

lowered into the vessel. The manipulator had a small wire brush attached which swept the smaller pieces of coke into a dust pan arrangement which, in turn, was dumped into the 30-gal drum. When this drum was being removed from the vessel, both hooks accidentally slipped out of the lifting eyes and the drum fell approximately 10 ft onto the remaining coke mass. However, few core positions remained upright, and identifiable samples (three pieces weighing a total of 27 lb) were loaded into another 30-gal drum. The largest intact piece recovered is shown in Figure 3. On completion of this phase of the cleanup, the fragments were transferred to new drums for shipment and analyses. Three barrels were designated for analyses, while the remainder of the material was sent for burial.

6-23-67 UNC 7710-2570

Figure 3. Largest Intact Piece of Outer Moderator Deposit Recovered From Core

A vacuum cleaner was used to remove the small granular material left on the lower core grid plate. Certain problems were encountered with this procedure, however. In the recess between the lower core grid plate and outer barrel, fragments were discovered that were too large to be picked up by the

cleaner; the pieces were crushed to facilitate removal. When semisolid material from grid plug recesses caused plugging of the vacuum cleaner hose and the rest of the grid plate was clean, the plugs were removed from the vessel one at a time and cleaned. A special tool was fabricated to remove small pieces of coke which had lodged between the lower grid plate and the support barrel. Inspection indicated that the grid plate and the lower grid-support barrel recess were free of coke except for fine, gritty pieces adhering to an organic film; this material was expected to be removed by the flushing operation to follow.

Inspection of the thermocouple (TC) support plate, begun in August, continued during the first half of September. For the initial examination two sections of the inspection station periscope were added to the in-vessel unit. Although this resulted in a reduction of the field of vision to one grid plate penetration, a sharper and more detailed image was provided. Examination of the support plate below the B-15 penetration clearly showed a deposit that completely covered one of the Failed Element Location System (FELS) flow tubes in the southeast quadrant of the position. This particular position did not have a lower core grid plate plug installed during the initial phase of the coke removal operation. Attempts to vacuum the coke sediment from the support plate below the B-15 penetration were not successful, so the vacuum cleaner assembly was removed from the vessel. Inspection of the D-11 core position showed that the support plate and FELS flow tubes were clean and shiny with no visible coke deposit or fragments.

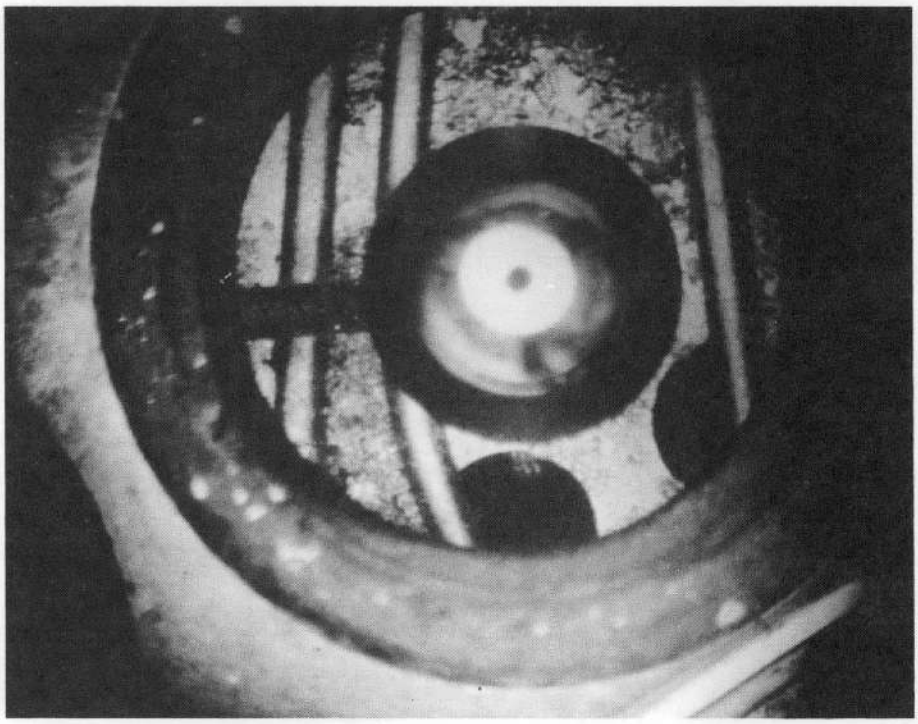
At this point, all lower core plate plugs were removed from the vessel and additional lamps were placed in the vessel to facilitate the inspection. Lighting was a limiting factor in use of the flexible boroscope for inspection. Even with a new objective section, visibility was severely limited at distances of more than 2 to 3 in.

On September 7, additional extension sections were added to the periscope to permit positioning of the objective lens about 2 ft above the lower core grid plate. A second 3-in. periscope was installed to provide an overall view of the plate and to position the in-vessel lights. With this arrangement, the TC support plate was inspected at all core positions.

In general, the inspection revealed a layer of sludge covering the support plate with the depth varying from about 1 to 2 in. at the core periphery to a thin

film near the center of the core. At some locations a layer of brown coolant could be seen. In 6 of the locations the surface of the coolant appeared to be pockmarked, as if droplets had hit the surface after the surface had become a semisolid. Small flakes or fragments of coke-like material could be observed on much of the surface, and ranged in size up to about 1/2 in. on a side. The few pieces larger than this were reduced by crushing to less than 1/2 in. on a side, to ensure their removal during flushing. From their locations on the support plate and between the TC support tube and wall of the grid plate opening, it is considered likely that most of the flakes were dislodged during fuel removal, prior to installation of the lower grid plate plugs.

A photograph of the support plate surface was taken (Figure 4), and samples of the carbonaceous material were obtained from the plate below the K-13 and B-11 positions. A cursory examination showed the material to be granular, brittle, and highly saturated with HB-40. As the samples were removed, it was noted that the surrounding material would immediately flow in and fill the vacated area.



6-23-67 UNC 7686-5127

Figure 4. Thermocouple Support Plate as Seen Through Lower Grid Plate Opening (prior to core flushing)

To confirm the mobility of the remaining material, a few selected core positions were flushed with about 5 gal of HB-40 with the result that material on the walls and beveled surfaces of the holes was removed. A second trial flush with HB-40 was made at Position D-11. After this flush the residual HB-40 was blown from the area, revealing the bare metal beneath. The material on the support plate appeared to be easily moved, but the FELS flow tubes on the plate surface prevented complete drainage with this 1 to 2 gpm flush. It was believed that a full-flow flush would effectively remove all residue from the plate.

All fuel element positions in the lower grid plate were cleaned with a new wire brush tool, and then checked with a plug gage. Ten of the holes still contained enough foreign material to prevent full insertion of the gage. They were rebrushed and, after the gage was polished to remove burrs, were determined to be satisfactory.

Following this operation, special tools and equipment were removed from the vessel, the in-vessel work platform was installed, and HB-40 was again added to the vessel to provide shielding for the drilling of the barrel support ring for installation of the new core pressure drop instrumentation. The hole was drilled and tapped, and the tubing installed; the tubing external to the vessel was plugged to prevent leakage during the flush.

After removal of all remaining equipment from the vessel, the inner barrel was transferred from the spent fuel storage pool to the vessel. During the transfer, a comprehensive visual examination was made of the lower side of the upper core grid and the external surface of the barrel. A thin, black oily film was present on the surfaces of both the barrel and grid plate, but the walls of the grid plate holes were clean and smooth. No carbonaceous deposits were in evidence nor was mechanical damage observed from a distance of 3 to 4 ft.

The inner barrel was bolted into place, and grid plugs were installed in 25 core locations of the lower guide grid, including the 13 control rod positions and alternate positions in the outside ring of holes. Plugs were also placed in the Charpy and source positions.

Circulation of HB-40 through the pressurizer loop was initiated to heat the vessel in preparation for filling it with OMP coolant. The HB-40 was drained

from the vessel on September 28 at a temperature of 332°F, and OMP was pumped in at 340°F. Circulation was established through the pressurization loop and F-2A filter.

The vessel was opened September 29 to: (1) install 60 in-vessel filter supports and filters in the remaining positions, and (2) remove drain piping to obtain measurements of the difference in elevation between the upper surface of the lower grid plate and the top of the TC housings. Measurements of this difference at 5 core positions indicated an average distance of 5 ± 0.03 in. The vessel lid was then reinstalled, and filling of the MHTS resumed.

Upon completion of the refilling operation, high-flow flush operations were initiated to remove residual carbonaceous material from the reactor vessel. The initial attempts to obtain high flow rates were hampered by an immediate increased pressure drop across the main pump suction screens. The high ΔP proved to be due to plugging of the screens with organic saturated coke. During the month of October these screens were removed and cleaned on three occasions. Quantities of material removed were as follows.

Cleaning Number	Screen Deposit (pints)	
	P – 1A	P – 1B
1	4*	4
2	1	4
3	1	1

*A small piece of wire screen and flat metal pieces also found on the screen appear to be from the prestartup period, since they show no signs of activation nor do they appear to match materials or tools used in the core cleanup process.

After the first cleaning of the screens, pressure gage assemblies were installed in the lines between both screens and pumps to provide more accurate information relative to subsequent screen plugging.

During the initial replacement of the suction screens, the FELS flow tubes were blown out with nitrogen. An increase in pressure drop across the F-2A

filter was coincident with this operation; there was also some evidence of plugging of the screen in the CV - 211A valve position. Although it could not be established whether the material was due to the nitrogen purge or was a consequence of the inclusion in the circuit of an external loop, the operation was repeated on the resumption of flow in an effort to agitate the contents of the lower portion of the reactor vessel to provide better flushing. There was no evidence, however, that additional material was deposited on either the pump screens or filters. This flushing procedure was repeated again during operation prior to the third cleaning of the suction screens in an effort to transport as much coke material as possible to the screens before their removal.

During shutdown for the third suction screen cleaning, the 75-micron in-vessel filter media were replaced with 25-micron media. Previously, the 75-micron filters had shown a ΔP increase from 3.9 to 7.1 psi, normalized to a flow rate of 14,000 gpm at a temperature of 575°F.

After the third pump screen cleanup and the resumption of circulation, excessive leakage developed in the suction screen flanges, so the pumps were shut down and the lines isolated while the flanges were retorqued. Excessive leakage from the P-1B flange necessitated its isolation, and circulation was restored with one pump. When a small amount of leakage developed at the P-1A flange, circulation in the MHTS was stopped to retorque it.

Coolant circulation through the main heat transfer system to effect system cleanup through continuous filtration continued during November, except for one short interruption. During the weekend of November 4 to 6, ten lower grid plate plugs were removed to achieve partial filter bypass, and to ease the filter pressure drop (structurally limited to 10 psi) so that full two-pump flow at 17,000 gpm could be established.

On this same weekend, the main pump suction screens were reinstalled and circulation through the vessel drain piping was established to improve flow conditions under the thermocouple support plate. The flushing procedure continued through November 22nd, with all system parameters remaining constant within the limits of reading and/or instrument accuracy. During the weekend of November 25 to 27, the reactor inlet block valve was cycled twice, causing flow variations from 16,800 to about 2,000 gpm. The 7-sec cycles resulted in no

detectable radiation increases at the pump suction screens, or in increased differential pressure readings. Coolant circulation was accomplished with one pump (P-1B) for the last two days of the month, while P-1A was repaired. The flush was terminated on December 1.

Upon completing the flushing operation, the reactor vessel was opened, and the in-vessel filters and filter support plugs were removed from the reactor. The filter media were removed from their housings. A few pea-size pieces of coke were found in the support plugs themselves.

After draining, the initial inspection of the reactor vessel indicated that the lower core grid plate was clean, and that other core internals were also clean; but it appeared that about 50 small pieces of coke lay on the T/C support plate. The basic inspection device was a Kollmorgan periscope. The Kollmorgan was also provided with an attachment for taking pictures.

Some of the pieces of coke on the T/C support plate reached a maximum size of about 2 in. on a side. The locations in which most of the coke appeared were the outer core positions, and generally to the south and east side of the vessel; that is, opposite the location of the main heat transfer system suction piping. This condition indicated that some short-circuiting of flow had taken place. Also, the layout of the piping on the T/C support plate appeared to be an influential factor; that is, T/C piping perpendicular to the flow path retained pieces of coke, whereas piping parallel to the flow permitted the escape of pieces. In some of the positions in which coke was initially thought to be identified, it was later proved to be either an accumulation of organic crystals or a possible lighting effect, because when attempts were made to break up the pieces of coke 1/2 in. or larger by probing, the suspected coke in some cases would simply disappear. The total volume of coke on the T/C support plate was estimated at about 13 in.^{3}

Of the 16 core positions initially reported to have coke pieces larger than 1/2 in. on a side, 6 were subsequently found not to have the reported coke. Two of the positions had a buildup of organic crystals which disintegrated when touched with a probe; two positions had a "stalagmite" of organic-presumably drippings from the grid plate; and two of the positions had only a discoloration of the T/C plate. All but two of the remaining pieces of coke larger than 1/2 in.

were crushed or broken up. In the two cases, the coke flipped out from under the probe, and could not be relocated. It is likely that the pieces were broken up in the process.

Two pieces of foreign material, a fitting elbow and a piece of shielded cable about 1/4 in. in diameter and 6 in. long were seen on the support plate and were retrieved. Visibility through the holes in the T/C plate was not uniformly good, but in a few locations it appeared that the coolant heel was clean, and no coke fragments were seen. Attempts to obtain samples from the T/C support plate resulted in nothing more than a black oily film being retrieved from the plate.

During inspection of the reactor vessel, the radiation exposures were about 450 mr/hr at the top rotating shield with the vessel empty and the upper grid plugs removed. Without the top rotating shield in place, a dose rate of 2.2 r/hr was measured at the upper guide grid with a grid plug removed. In general, a radiation field of 50 mr/hr existed at the periscope operator's work location.

2. Examination of Inner Core Barrel Assembly

The barrel assembly- the inner barrel, lower guide grid, and upper core grid assembly- was visually examined by means of closed circuit television while being moved from the reactor to the storage pool. The underside of the upper core grid was examined in detail with the results described below.

The underside of the grid plate was uniformly covered with a black, oily substance (also observed on the inside and outside surfaces of the inner barrel and encountered on the inside surface of the outer barrel during the barrel drilling operation). The black, oily substance could be compared to fine lamp-black. The uniform black coating was, in turn, covered with highly reflective droplets of material which appeared to be organic. Occasionally, droplets were observed to collect and drip off the grid plate. The highly reflective nature of the droplets interfered with the examination, and an attempt was made to remove the organic by a swab consisting of a bundle of rags on a long pole. The swab proved to be effective, and droplets were removed from approximately 80% of the underside of the grid plate. One reexamination, some of the droplets were observed to have been smeared, supporting the observation that this material consisted of droplets of organic. Reexamination of those areas cleared of the droplets revealed only an expanse of black, oily surface. With exception of

the black coating, surfaces appeared to be clean and free of foreign material. Edges or intersections of surfaces, as, for instance, the chamfers on the holes, were sharp and clean. By changing the angle viewed by the camera, the inside surfaces of the holes in the grid plate could be viewed obliquely and appeared to be clean.

Particular attention was given to the observation of possible cracks but none were observed. The apparent resolution of the viewing system was such that a hairline crack on an area observed could have been overlooked; however, any crack 0.030 in. or greater in size would have been observed in those areas examined. It is estimated that a minimum of 80% of the lower surface of the upper core grid was examined.

By focusing the television camera through the holes in the upper core grid, a cursory examination of portions of the inside surface of the barrel and underside of the lower guide grid was made. The interior surfaces of the barrel assembly so examined did not appear to be significantly different from the underside of the upper core grid.

The television equipment used for the inspection was mounted on a pan-tilt unit located in an instrument pit in the floor of the 100-ft level. The camera position was chosen to permit the detailed examination of the underside of the upper core grid, as the barrel assembly was suspended above the pit. Illumination was provided by lamps also located in the instrument pit and directed upward. Photographs were obtained of the display on the television monitor by means of a 4 x 5 camera equipped with a Polaroid pack.

3. Inspection of Miscellaneous Plant Subsystems/Components

The inspection ports on the main boiler B-1, reactor building drain tank T-1, surge tank T-2, and seal drain tank T-19 were opened and the vessels were inspected. There was no evidence of coke in either the inlet or outlet lines and other cavities of the boiler. The boiler piping had the typical uniform black film.

The reactor building drain tank was cleaned, and approximately 30 gal of material (composite of organic and coke) was removed. The material removed was measured for activity and showed 1.35×10^{-2} cu/gm. Radiation readings

in the T-1 tank were 2 to 3 mrad/hr. Not only was the material cleaned out of the tank, but the walls were wiped down, and prior to buttoning up, screens were installed over the tank drain line. The drain line and level instrument bridles were also cleaned.

The T-2 surge tank contained a very small amount of black residue (similar to that found in T-1) in the bottom of the compartment and in the flange neck (approximately 1 to 2 gal). The radiation in this region was reported between 0.1 and 0.6 mr/hr.

The heel of coolant and black material (about 2 in. in depth) found in the seal drain tank was similar in appearance to that found in T-1.

4. Other Activities

a. Pump Repair/Maintenance

A repair of the conoflow controllers for the pressurizer pump speed controller system was completed. The pressurizing pump motor (P-3A) was installed with a new spare armature, but proved to operate unsatisfactorily at high speed. The manufacturer was called in to assist in resolving this problem.

Both of the main coolant pumps, P-1A and P-1B, had developed bearing noise. A thrust-bearing replacement was completed on P-1A, but the results were not completely satisfactory, so a more complete disassembly and inspection were made. The problem associated with P-1A was found to be the result of deterioration of the Loc-Tite cement which was used to seal the inboard bearing race to the bearing housing. The cement became loose and the race was turning in the housing, causing the noise.

The thrust bearing which was first removed from P-1A was found to have ball bearings which were out-of-round. The inboard bearing was also replaced. The pump shaft and inboard bearing housing were both metal-sprayed and turned down to proper size.

Since the cause of the slight noise found in the P-1A pump (which was similar in P-1B) was of such a nature that it would probably not fail, P-1B was not reworked.

b. Core Vessel Nitrogen Purge Modification

Installation of a new nitrogen purge distribution ring was completed. It was installed inside the upper guide grid shroud and will provide a much more uniform distribution of nitrogen over the vessel coolant level, thus reducing the potential air-coolant contact area when the vessel lid is off. The distribution ring has 12 holes equally spaced around the ring. Two holes (1/2 in.) were drilled in the inner shield shroud to assure presence of the nitrogen blanket in the region of the control rod jumper lead holes at the periphery of the vessel.

c. Reactor Building Containment Leak Test

The reactor building containment leak test was initiated on December 30. The building was pressurized to 5 psig, using the instrument and utility air compressors. Preliminary results indicate there was essentially no leakage from the building proper, and only a very minimal (approximately 0.02% per day) leakage through the 30-in. ventilation system butterfly valves. The percent of leakage is based on the volume of air contained in the building at 5 psig. One percent per day is the allowable leakage rate.

C. SYSTEM MODIFICATIONS

1. Core Modifications

Core modifications during this report period include changes to the fuel elements, dummy elements, and instrumented fuel element. New components added to the core include downcomer elements, in-vessel nitrogen purge header, and core D/P cell instrumentation.

Modifications will be made to the fuel elements to (1) increase the outer moderator flow, and (2) increase the velocity in the inner process tube. These modifications will provide a velocity of 2.0 ft/sec (46 gpm) in the inner process tube. The velocity before the modification, with the drain valve removed, was approximately 0.4 ft/sec (9 gpm).

In the original Core I configuration, coolant flow into the outer moderator region was provided solely by three weep holes in the upper head extension of each fuel element, discharging below the upper grid plate. The calculated total flow was 317 gpm, providing an average velocity of 0.05 ft/sec. This velocity could have been as high as 0.5 ft/sec, generated by thermal gradients, but in the opposite direction. With the modifications, the average coolant velocity is calculated to be 4 ft/sec, and in the upward direction, taking advantage of the buoyancy forces created by internal heat generation in the outer moderator plenum.

b. Dummy Elements

There will be 20 dummy elements in the 41-element restart core. The dummy elements fill both the upper and lower grid plate holes, assist in maintaining the desired coolant velocity in the outer moderator region, and provide locations for additional fuel elements in the core. The dummy element has been redesigned to improve core neutron economy and to increase coolant velocity through the interior of the dummy to minimize film formation.

The dummy element top fitting is fabricated from Type 304 stainless steel, and the upper portion is the same as a fuel element. The top fitting contains three 0.133-in.-diameter side leakage holes at the upper grid plate identical with a fuel element. These holes provide a bypass flow of 4.0 gpm to assist in cooling the upper grid plate. The tube connecting the upper and lower end-fittings is fabricated from 6061-T6 aluminum, and is pinned to the end-fittings.

An 0.277-in.-diameter orifice hole in the top fitting provides a flow of 9.4 gpm through the dummy to minimize the formation of film inside the aluminum

tube. The lower fitting is fabricated from Type 304 stainless steel, and has a radial clearance in the lower grid plate of 0.010 in. (cold). This radial gap provides a bypass flow of 4.0 gpm to assist in lower grid plate cooling. The upper and lower end-fittings have been fabricated from Type 304 stainless steel to minimize the thermal expansion between the dissimilar metals of the aluminum tubing and the end-fittings. The envelope size of the new dummy is compatible with the present handling and storage facilities.

The relative reactivity worths of steel and aluminum dummy elements in the 41-element restart core were calculated, and show that there is a significant increase in core lifetime by changing the carbon steel tubes to aluminum in the dummy elements. This increase in lifetime for the reactivity-limited 41-element core is approximately 930 Mwd/MTU (3000 Mwd, or 5 months at 20 Mwt).

c. Downcomers

There are 24 downcomers in the 41-element restart core located at the periphery of the core. The downcomers are new components in the core. They are designed to collect the outer moderator coolant at the core periphery and discharge it downward into the lower plenum. They also promote the desired outer moderator coolant velocities to minimize film formation.

The downcomer has approximately the same envelope size as a fuel element and is compatible with the fuel-handling machine. There are 64 graduated diameter inlet holes arranged in a spiral pattern in the tube connecting the upper and lower fittings. The hole pattern starts just above the lower grid plate and extends almost to the bottom of the upper grid plate. The tube connecting the upper and lower fittings has twenty-four 1-in. holes in the lower section, twenty 1-1/2-in. holes in the center section, and twenty 2-in. holes in the top section, all arranged in a spiral pattern. The top fitting is similar to a fuel element. It blocks the flow of coolant from the plenum above the upper grid plate from going directly into the downcomer. The top fitting has three 0.113-in.-diameter radial holes that provide a bypass flow of 4 gpm for upper grid plate cooling. The lower fitting has an annular clearance of 0.010 in. (cold) with the lower grid plate. This clearance results in a bypass flow of 4 gpm to assist in lower grid plate cooling.

d. In-Vessel Nitrogen Purge Header

The in-vessel nitrogen purge header is designed to provide uniform gas blanketing over the reactor coolant during fuel handling, to minimize oxygen contamination of the coolant. It is a new component that has been added to the reactor vessel just below the upper guide grid. The new header will improve the control of oxygen (air) in-leakage by providing additional nitrogen cover gas over the reactor coolant during refueling and maintenance operations.

The in-vessel nitrogen purge header is fabricated in a 72-1/2-in.-diameter ring from 1-in. carbon steel pipe, and is of all-welded construction. The ring header is welded to support clips on the top inner shield guide plate skirt. Twelve equally spaced 1/16 in-in.-diameter holes have been drilled in the header to equalize gas flow distribution with the 8 to 10 scfm flow rate requirement. The maximum flow rate will also limit the amount of organic vapor to an acceptable level above the reactor during fuel handling and maintenance operations. Removable piping has been fabricated for connecting the header to the reactor drain line. Two 1/2-in.-diameter holes were added to the top inner shield guide plate skirt to allow nitrogen blanketing in the annulus between the skirt and the vessel wall. The new nitrogen purge header and associated piping were fabricated and installed by the City of Piqua during the recent reactor cleanup operation.

e. Grid Plate Support Barrels

During installation of the in-vessel nitrogen purge header, three 3/4-in.-diameter vent holes were drilled near the top of the lower grid plate support barrel. These holes will vent any possible gas that may generate in the moderator plenum during reactor operation. The holes will also permit the flow of organic between the inner and outer support barrels to eliminate stagnant coolant in this annulus.

f. Core D/P Cell Instrumentation

During the recent reactor cleanup operation, a new line for a differential pressure cell (D/P) was installed in the reactor vessel by the City of Piqua. The line was fabricated from 1/2-in.-diameter steel tubing. The line enters the reactor vessel through a Conax fitting in the "T" nozzle. A 1/2-in. hole was drilled and tapped through the lower core barrel support ring. The line is

connected to this tapped hole with a Swagelok fitting. The location of the line in the support ring will permit measurement of the coolant pressure before it enters the exit nozzles of the reactor vessel. The new location of this pressure tap will eliminate the pressure losses from the lengths of main cooling piping that were encountered in the old pressure tap location.

There are two other pressure taps in the reactor vessel located above and below the top guide plate. These taps measure the operating pressure drop through the in-vessel coolant filters. Using the lower leg of these pressure taps and the new line, the operating pressure differential across the core can be measured. An increase in differential pressure will indicate any increase in pressure drop across the fuel elements. Installation of the new line and the D/P cell and associated instrumentation will provide more reliable information of the coolant pressure drop across the core, and thus assist in the core surveillance program.

g. Instrumented Fuel Element

The 41-element restart core will include one instrumented fuel element (IFE). Instrumentation requirements were established from the operations and surveillance program experience with the original instrumented fuel elements and from the fuel element fouling surveillance program.

The modified IFE employs the same general arrangement as the original instrument fuel elements. An extension housing is located above the upper head of the fuel element, and extends to an elevation just below the upper shield grid plate. The fuel element together with the upper extension housing can be transported in the fuel-handling machine.

The extension housing provides the means of routing and protecting the instrumentation leads from the in-core section to an elevation where connection to the jumper leads can be accomplished. The upper end terminates in a thermocouple disconnect plug. The pin size, socket construction, and mounting provisions are patterned after the original IFE jumper design, but modified to prevent pin damage when making a connection. This pin and socket construction allows a maximum of 16 instrument leads (8 T/C's), due to availability of space within the IFE housing. Analysis of the information obtained from the original IFE's indicated that temperature information is more important than the data

that had been obtained from the flowmeter installation. Therefore, all of the available leads in the IFE/upper housing are to be used for thermocouples. A discussion of the thermocouple locations follows.

Eight thermocouples will be available on the IFE provided for the restart core. The preferred locations of these couples are shown in Figure 5. The following considerations went into the choice of these positions.

1) The method of installation dictates that in-fuel thermocouples can be located only in the outer fins of either the inner or outer fuel sections.

2) Locations above the gaps between fuel sections are more apt to detect the variations which contribute to hot channel factors because of extensive mixing across the gaps.

3) It is important to determine axial and radial temperature variations in the fuel.

4) Center channel temperatures will be closer to average values for a given elevation than will the corresponding outer channel values, due to cross-flow shear in the counter-spiraling fins.

5) A precise reference thermocouple is needed for use in indicating the IFE temperatures as a ΔT rise above inlet.

6) IFE exit thermocouples can be used as reference points to monitor the performance of the usual fuel exit thermocouples. These exit thermocouples can also be used for the transient analysis proposed to evaluate fouling of the fuel element.

The thermocouple hot junctions buried in the cladding fins are located above the unfinned gaps between sections of the fuel cylinders, so that the T/C readings will not be affected by an unknown degree of coolant mixing which occurs in the gaps. Iron-constantan thermocouples will be used to match the material of the existing leadout wires. Stainless steel pins and sockets will be used at the jumper disconnect. The thermocouple wire and the stainless pins and sockets will be connected by welding to provide a high quality structural and electrical joint.

The control of coolant flow through the modified IFE and the resultant flow split in the lower fitting is similar to a regular fuel element. There is no provision for an orifice in the IFE. The upper extension housing is adequately

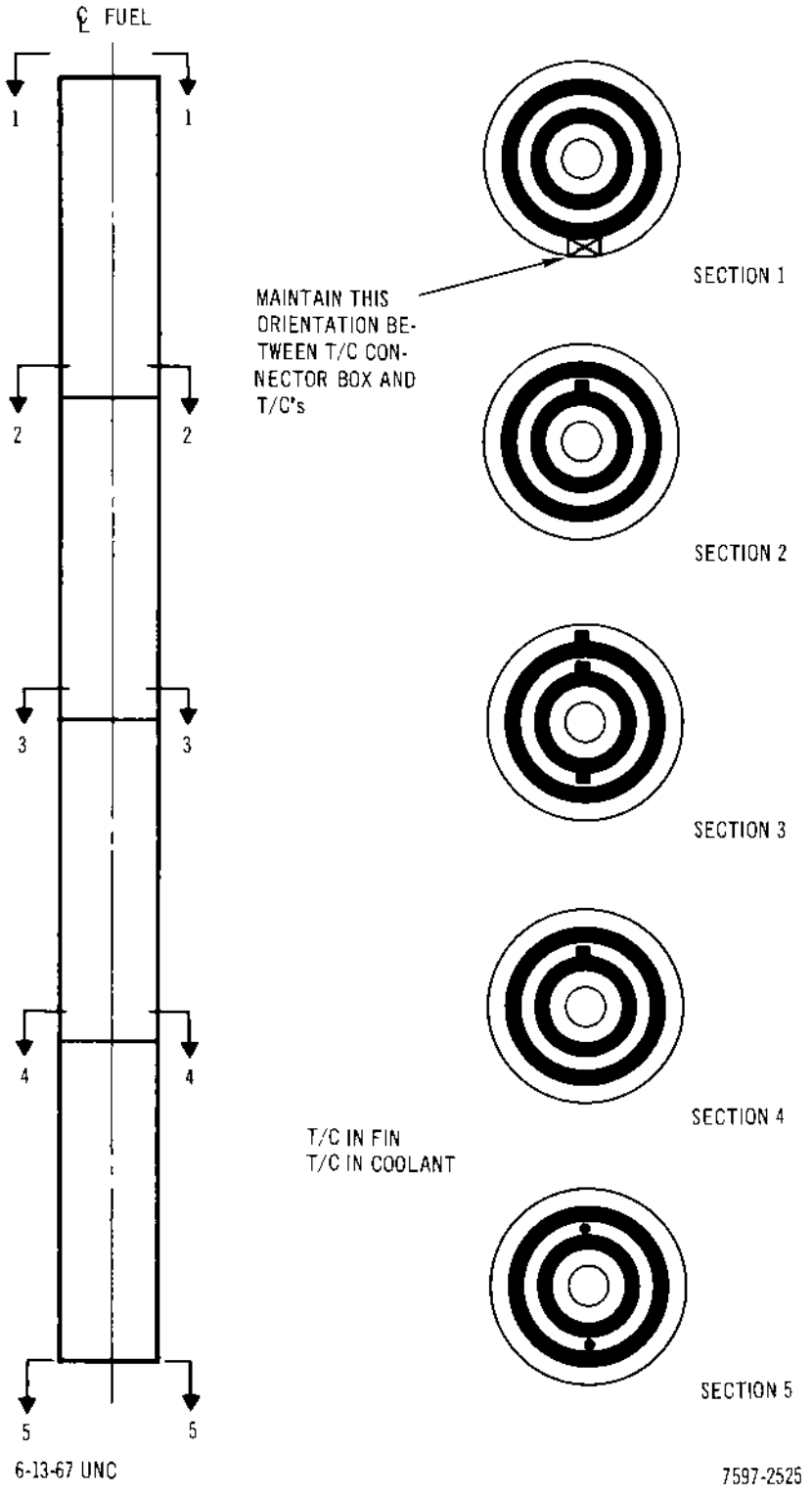


Figure 5. Thermocouple Locations in Instrumented Fuel Element

ported so that entrance losses will not restrict coolant flow into the fuel element. In addition, there is sufficient porting just above the fuel element upper head to allow coolant flow through the fuel element if the coolant is lowered to a level about 3 to 4 ft above the top of the upper core grid plate. The IFE extension is connected to the fuel element handling head with large shoulder bolts. The shank of these bolts has a slight radial clearance. These features allow for a small flexing of the IFE at this point, thus reducing the fabrication problems in respect to close tolerance for straightness over the length of the unit. The quality assurance requirements for the IFE will be identical to those required for the modified fuel element. The T/C's will be checked for continuity and grounding during and after fabrication, to assure high quality temperature measurements.

h. Measurement of Fouling by Inner Process Tube Devices

Two devices have been suggested for installation in the inner process tubes (IPT) of a number of fuel elements; both are intended to provide information on surface fouling in the PNPF. The first device has been used before at ORNL to determine velocity effects on corrosion rates. In the Piqua application, it would provide similar information on fouling rate. It also appears possible that it could be used to permit an on-line evaluation of the extent of fouling. The second device is concerned directly with measurement of the effects of a number of parameters which may influence fouling rates.

i. Summary

The proposed modifications are designed to meet the objectives of increasing moderator coolant velocities, limiting oxygen contamination of coolant, and improving in-core D/P instrumentation. All of these changes will further improve the performance of the reactor.

2. Control Rod Modifications

The original control rod drives were shipped from the site to AI in June of 1966. A cursory examination revealed problem areas not previously considered. A comprehensive evaluation of failures and failure modes was instituted and completed. Some of the failure modes were found to include features of the control system external to the reactor vessel as contributory factors. As a result of the investigation, the scope of the control rod drive modifications was altered

and enlarged to include portions of the control system external to the reactor vessel. Work on the modifications is currently in progress.

a. Failure Mode Investigation

Each type of failure was investigated to establish prime and contributory failure modes. The results of this investigation are presented in NAA-SR-Memo-12142. A summary of the failure modes associated with each type of failure follows.

(1) Coil Case Electrical Feed-Through Failure

Electrical failure was found to be caused by shorting through carbonaceous deposits between pins and between pins and shell. In addition, the feed-throughs had failed to maintain their seal integrity. Analysis, confirmed by test, established that the feed-through conductors operated at temperatures of 200 to 400°F above the environment when carrying normal operating currents. The power supply circuit protection was such that in the event of a shorted conductor, the current in the conductor increased to as much as twice the normal value before the protective devices would trip. The additional heating due to the increased current led to the rapid subsequent failure of the remaining circuits handled by the feed-through.

(2) Shorting of Conductor to Ground Due to Slippage of Overbraid Terminations

No examples of this type of failure were available for examination. The failure modes associated with this type of failure are thermal cycling and flexing, either of which is capable of moving the overbraid in relation to the conductor and insulation. Overbraid termination technique alone appears incapable of accommodating variations in lead length and number of bends. The insulator overbraid stops originally proposed were judged to be adequate.

(3) Shorting of Conductor to Ground Due to Carbon Buildup Between Conductor and Overbraid

This type of failure has been encountered only once, immediately after the shutdown during the summer of 1965. No examples were available for study at the time of the investigation. The failures occurred in 18 AWG conductors within the drive upper housing. This was the only location in which this size of conductor was used. At normal operating currents the 18-gauge wire was

calculated to operate up to 50°F above the temperature of its environment. This temperature would also increase in the event of a short at the power coil case feed-through with the type of circuit protection provided. No records were available to check for possible correlation between these lead failures and previous feed-through failures.

Another possible and unrelated failure mode consists of coolant conditions and cover gas control during the shutdown immediately preceding the failure occurrence. It is known that the oxygen content of the cover gas and coolant was high during certain periods of the shutdown, and that the coolant cover gas interface was at the level of the affected leads for long periods. Insufficient information was available to allow a complete evaluation of this possible failure mode.

(4) Shorting of Conductors to Ground Due to Mechanical Damage Occurring During Fuel Handling Operations

It was found that the original installation had inadequate provision for protecting the jumper leads during fuel handling. The leads were bundled in a wire wrap. At the junction to drive plug, the abrupt transition to flexible leads tended to concentrate flexing at that point. The leads could not be removed during fuel handling. They were disconnected from the drives and arranged along the periphery of the vessel to clear as many fuel positions as possible before installing the top rotating shield. If further lead movement was required to clear additional fuel element positions, the leads were moved with a bar operating through the top rotating shield. The wire wrap around the leads would provide no protection against a misplaced edge of the bar. The originally proposed modifications were judged to be adequate.

(5) Shorting of Conductors to Ground at Vessel Penetrations

This was found to have been corrected by the modifications accomplished during August 1965. The cause of the failures previous to that time was due to conductor heating on the air side of the Conax packing glands. This caused the Teflon insulation to decompose, shorting the lead to ground. The modification consisted of mounting the Conax packing glands on extension spool pieces to improve cooling geometry, and of increasing conductor size from 16 to 14 AWG, to reduce conductor heating.

(6) Power Coil Degradation

Upon disassembly of the power coil cases at AI, all were found to be filled with reactor coolant. Two coil sets were found to be shorted, and the majority of coils were found to have lower than normal loop resistance. Examination of the shorted coil sets revealed that one coil of each set was shorted to its bobbin. In both cases the short had occurred close to the start end of the coil winding. Several of the coils with low loop resistance were unwound. A great number of turn-to-turn and layer-to-layer shorts were found. In every case investigated, a flaw was found in the anodized insulation on the aluminum wire. The types of flaw included porosity, scratches, and abrasions. Flaws were apparently created during the anodizing process, during subsequent handling through coil manufacture, and possibly during thermal cycling of the coils during service. Concentrations of carbonaceous material were found at many of the short locations, but this was judged to be a secondary effect. The condition of the organic away from the shorted areas indicated that the organic had little if any direct effect on the insulation integrity of the anodized wire.

The coil construction was reviewed. It was found that there was only one source of anodized aluminum wire sufficiently flexible for coil winding. Adequate quality control procedures are not available to assure complete integrity of the anodized insulation as fabricated. The anodic coating is very thin and subject to damage in any handling operation. The coil wire-insulation system is very fragile in nature because of the thin relatively brittle insulation, the soft nature of the aluminum wire, especially at temperature, and, the high coefficient of thermal expansion characteristic of aluminum. For these reasons the anodized aluminum wire-insulation system employed in the original drives was judged incapable of providing the required degree of reliable operation.

(7) Coil Case Leakage

All disassembled drive coil cases were found to be partially or totally filled with the reactor coolant. Leakage paths traced to the electrical feed-throughs ranged from the obvious, where the glass feed-through insulator was cracked, to sound-appearing feed-throughs which were found to leak at reactor operating conditions. However, in at least two instances, the source of the leak could not be isolated to the feed-throughs. The leakage path was not found in either of these

instances, probably due to masking of the leakage path by the reactor coolant. Possible leakage paths not eliminated are: (1) defects in the welds; and (2) stringers in the material.

(8) Position Indicator Coils

Although the circuit in which the position indicator coils operate had been operating satisfactorily, examination revealed that potential failure was imminent. Forty-seven percent exhibited a significant loss of loop resistance, and 85% had an insulation resistance of less than 50,000 ohms at 50 volts. Twelve of the indicator coils were destructively examined. Reactor coolant was found in all twelve. All twelve of the opened coils exhibited varying degrees of the coil glass insulation material cracking, spalling, and pulling away from the wall of the coil case. Low-resistance paths to ground were found at the coil electrical feed-throughs, and on the surface of the coil insulation material. Two of the coils were unwound, and numerous turn-to-turn and layer-to-layer shorts were found. The glass encapsulant employed in the indicator coils is a proprietary product of the Turboglaze Corporation. It appeared to be incompatible with the reactor coolant.

(9) Long Position Coils

All of the long position coils were found to be in good condition electrically, with insulation and loop resistances very close to the original values. The conductor terminations showed some evidence of deterioration. The decontamination procedures had washed the Foster Seal potting compound away from the windings. This will necessitate rebuilding the long position coils.

b. Control System Modifications

(1) Control Rod Drive

As a result of the failure mode investigation carried out on the drives returned from the reactor site, the scope of drive modifications was greatly increased over that presented in the previous report.* The drive modification now consists of a major redesign of the drive incorporation new wire-insulation systems in the drive and indicator coils and mineral insulated cable leads from the coils to the upper drive plugs. The new drive configuration is shown in Figure 6.

*NAA-SR-Memo-12148

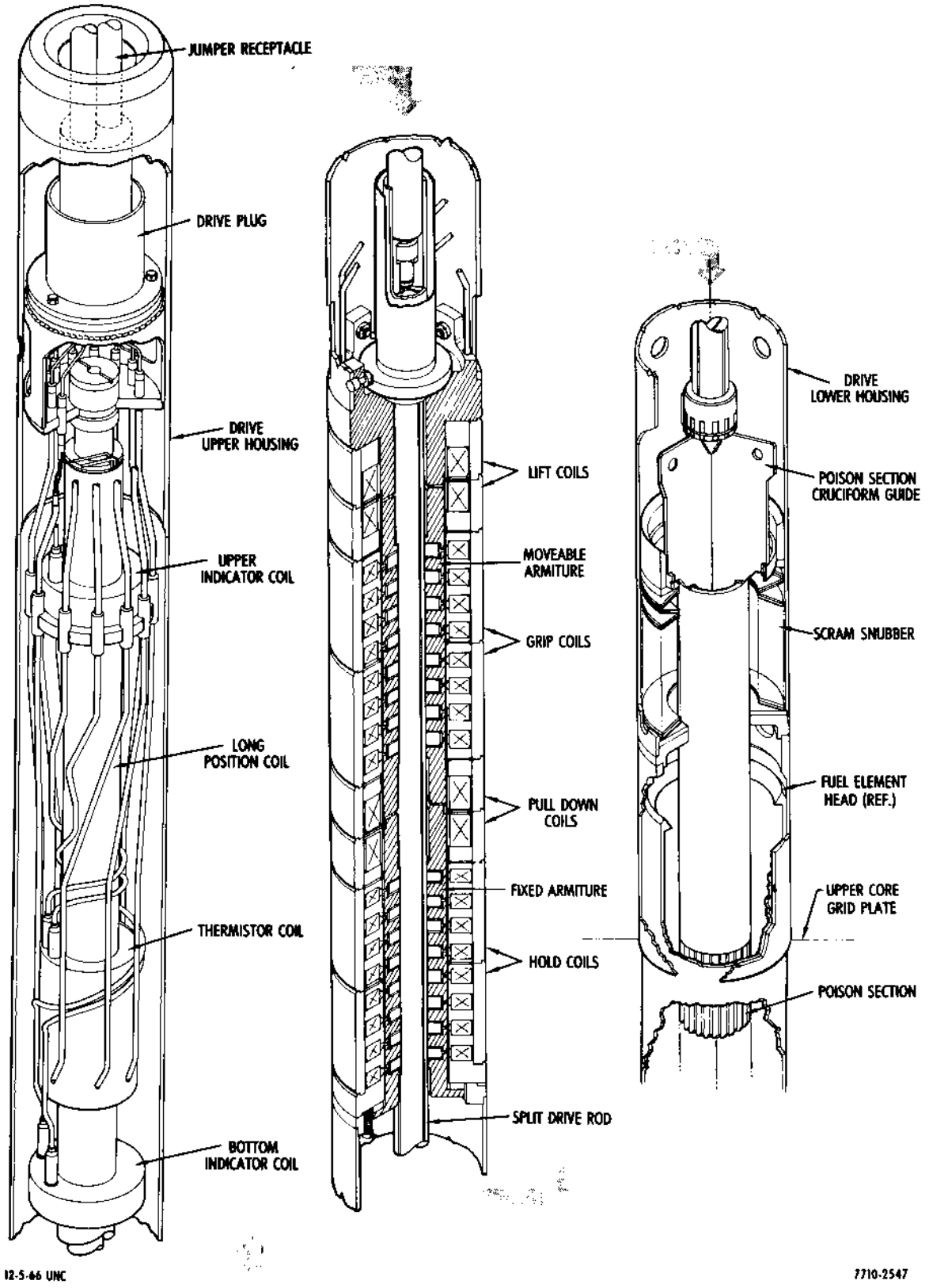


Figure 6. PNPF Control Rod Drive Assembly – Mark III

(a) Drive Coils

The drive coils are redesigned using a new wire-insulation system to prevent the coil degradation experienced with the original design. The coils are wound of copper wire double-served with fiberglas insulation and encapsulated with AI-1 high temperature encapsulant. The heat transfer properties of this wire-insulation system do not equal those of the original anodized aluminum system. In order to maintain the desired operating temperature, the coil pole pieces were redesigned to provide more coil volume. Bimetallic construction, using magnetic and not magnetic materials, is used to reduce magnetic flux leakage. This reduces ampere turn requirements and coil current requirements.

Each coil set is constructed as a hermetically sealed unit so that a seal failure will allow the organic reactor coolant to enter only one set of coils. The electrical penetration is accomplished with a stainless steel sheathed mineral insulated cable. The sheath is welded to the coil case and Advac single-conductor seals are welded to each end of the MI cable. The current is brought in through the MI cable conductor. The MI cable sheath is used as a grounded return.

The power coil case is formed by welding the unitized coil assemblies together. Wear-sleeves for the movable armature are installed in power coil cases in the assembly sequence.

(b) Indicator Coils

The basic design of the original indicator coil is maintained. The wire-insulation system is changed to the double fiberglas-served copper wire encapsulated with the AI-1 high temperature encapsulant. The space factor of the new wire-insulation system requires a reduction in number of turns. The output from the indicator coil will still be adequate for the readout system requirements. The electrical leads are provided by two MI cable assemblies seal-welded to the coil case.

(c) Long Position and Thermistor Coils

The original long position and thermistor coils will be rebuilt to the original wet design except that the winding termination configurations will be improved. Two MI cable penetrations will replace the electrical feed-throughs in the original thermistor design.

(d) Upper Drive Plug

The upper drive plug modification retains the use of lava insulators. The lava insulator is made of two-piece construction to allow bench assembly of the conductor to pin termination. The modified upper drive plug is attached to the upper end of the long position coil, and is free to slide within the drive upper housing. This feature provides for differential thermal expansion during thermal transients.

(2) In-Vessel Jumper Wiring

The two-piece jumper modification described in the previous report* is retained. Additional quality control measures are imposed on the fabrication to gain additional assurance of termination quality.

(3) Power Supply Modifications

The power supplies are modified to provide better circuit isolation. Fuses are provided in the leads to each coil set, so that in the event of short the redundant set will remain in operation. The fuses will also limit the maximum current that can be carried by each lead. Thyrector diode over-voltage protection is provided on the power coil side of the fuses to limit transient voltage spikes to 150 volts.

(4) Position Indication Readout System Modifications

The position indication system will be modified to obtain greater accuracy and stability. The bridge circuits used with the long position coil will be rebuilt with more stable components to assure drift free operation. A digital voltmeter will be added to the system to eliminate operator readout error. Lights will replace the meters for the up-and-down indicator indication to increase the accuracy and visibility of indication.

c. Test Program

The test plan proposed for the redesigned Piqua control rod drives is determined by a very tight schedule and the availability of testing facilities. The objective of the test program is to reveal at the earliest possible date any deficiencies that remain in the design which would prevent the rod drive units from

*NAA-SR-Memo-12148

meeting the Piqua Recovery Program requirements. The test program is broken down into essentially four phases: (1) component testing, (2) proof testing of a complete unit, (3) dry testing of each unit prior to shipment to Piqua, and (4) wet testing in the unfueled Piqua reactor. A summary of the test program is shown in Table 1.

(1) Component Testing

The first satisfactory production quality item of critical components, procured or fabricated, will be used in the component testing program. The component testing program will consist primarily of accelerated-type tests to reveal design margins and potential failure modes that cannot be accurately determined by analysis or through the use of conservative design techniques. Typical of the types of component testings required are: thermal cycling of coils at accelerated rate, thermal cycling of swaged cables and terminations, and circuitry mockups to test techniques for improving accuracy of position indication. In addition to the environmental tests, circuitry mockups of modifications to the power supplies and position indication system will be made to verify performance.

(2) Performance Tests

The design lift capabilities of the modified Piqua control rod drive will be verified and drive coil current levels established for both dual and individual coil set operation. The lift capabilities of the original design will be obtained from drive S/N 11 for comparison.

(3) Production Tests

Each production control rod drive will be tested to demonstrate proper stepping action and lift capability.

(4) PNPF Tests

The final wet test of the rod drives will be made in the Piqua reactor using dummy fuel elements in the control rod positions. The rods will be operated through the full range of operating conditions using the in-vessel wiring and the external drive and position readout circuitry. Temperature will be cycled between 350 and 550°F, using pump heat from the main reactor coolant system.

Pressure will be cycled between atmospheric and 120 psig using the pressurizing loop. The position indication system will be calibrated for the range of operating conditions.

In addition to the operating conditions, the capability of the drives to withstand the thermal transients occurring during their insertion into the reactor and removal from the reactor will be demonstrated.

D. SPECIAL STUDIES

Hydraulic tests were conducted to provide assurance that the design modifications will result in satisfactory hydraulic performance of the restart core. Full-scale models were used to obtain accurate pressures and flow patterns. The tests performed are separated into two basic categories; fuel element tests to evaluate effects on hydraulic performance due to fuel element design changes, and moderator tests to evaluate flow patterns and velocities in the outer moderator tests to evaluate flow patterns and velocities in the outer moderator region. The fuel element tests are separated into two categories: (1) single-element tests to obtain pressure drops for sizing the orifices; and (2) tests containing an array to evaluate flow interactions between elements and flow distributions within the lower fuel element housing. The moderator region tests were performed on a full-scale quarter section of the reactor core. Flow patterns and coolant velocities were investigated for the 41 and 61 modified element core with downcomers.

In addition, thermal-hydraulic parametric studies were conducted to evaluate the performance of the 41-element restart core and fuel element examinations were conducted at the AI Hot Laboratory as part of the Piqua Recovery Program. A study also was performed to measure the fouling rate on PNPF fuel elements. A safety evaluation of the PNPF modifications was completed and published during this report period.*

1. Hydraulic Testing, Quarter Section of Core

A full-size quarter section hydraulic test assembly was designed and fabricated to evaluate the hydraulic performance of the restart core. The quarter section test section is shown on Figure 7. The following hydraulic tests will be described.

*"Safety Evaluation of PNPF Modifications," NAA-SR-MEMO-12103 (December 1966)

	Poison Section Travel Requirements	Operating Temperature and Cycles	Operating Pressure and Cycles	Transients Above Operating Conditions	Transients Below Operating Conditions	Coolant Analysis	Power Coil Currents
Reactor Design Conditions	2140 full travel cycles, up and down, in 20 yr based on 100 full travel cycles, 4 scheduled test scrams and 10 unscheduled scrams per year.	350°F refueling to 650°F maximum operating 100 cycles in 20 yr 2 scheduled plus 3 unscheduled per yr. Rate 50°F/hr.	0 to 135 psig 100 cycles in 20 yr 2 scheduled plus 3 unscheduled per yr Rate 100 psig/min	Operating to 750°F and 300 psig. 5 cycles in 20 yr. Arbitrary design value. Rates 50°F/hr, 300 psig/min	400 to 100°F at atmospheric pressure, 10 cycles in 20 yr. Design insertion rate 5 ft/sec. Arbitrary design value.	Ash 10 ppm max MST Valve 500 max Carbonyl 600 ppm max Water Content 2000 ppm max High Boiler 0 to 30%	To be established
Mockup Coil Assembly Test	Not Applicable	Obtain coil temperatures at 350, 550, and 650°F coolant temperatures	135 psig	Not Applicable	Not Applicable	Distilled Piqua coolant (nominal)	Lift 1 6.36 amp dc Lift 2 6.36 Grip 1 3.62 Grip 2 3.62 Pull 1 5.87 Pull 2 5.87
Representative Coil Assembly Test	Not Applicable	100 cycles 350 to 650°F at 100°F/hr – 1000 hr at 650°F.	100 cycles 0 to 200 psig at 350°F 1000 hr at 200 psig Rate of change 300 psig/min	10 cycles – 600 to 750°F at 300 psig Rates 100°F/hr 300 psig/min	20 cycles between 400 to 100°F at atmospheric pressure – Rate equivalent to design insertion rate of 5 ft/sec	Distilled Piqua coolant	Lift 1 6.36 amp dc Lift 2 6.36 Grip 1 3.62 Grip 2 3.62 Pull 1 5.87 (nominal design Pull 2 5.87 effect current)
Defected Coil Test Unit 1	Not Applicable	1000 hr in 650°F organic. Coil temperature determined by power density	Saturation pressure corresponding to organic temperature	None	None	Distilled Piqua coolant 400 ppm H_2O	5.87 amp dc
2							8.32
3							10.00
4	Not Applicable	1000 hr in 650°F organic. Coil temperature determined by power density	Saturation pressure corresponding to organic temperature	None	None	1800 ppm H_2O	5.87 amp dc
5							8.32
6							10.00
MI Cable Test	Not Applicable	100 cycles 350 to 650°F at 100°F/hr – 1000 hr at 650°F	100 cycles 0 to 200 psig at 350°F 1000 hr at 200 psig Rate of change 300 psig/min	10 cycles – 600 to 750°F at 300 psig Rates 100°F/hr, 300 psig/min	20 cycles – 400 to 100°F at atmospheric pressure – Rate equivalent to design insertion rate of 5 ft/sec	Distilled Piqua coolant	15 amp dc at 150 volts potential between conductor and sheath
Defected MI Cable Test	Not Applicable	1000 hr in organic at 650°F	135 nominal design pressure	None	None	Distilled Piqua coolant 400 ppm H_2O for 1000 hr, 1800 ppm for 1000 hr	50 volts between conductor and sheath
Prototype Test in Reactor	1000 full travel cycles, up and down, plus 60 scrams	350 to 550°F 4 to 10 cycles	0 to 135 psig 20 cycles	None	10 cycles – Reactor at 350 to 400°F to control rod thimble in fuel storage pool at 100°F.	Piqua values during test period.	To be established
Production Drive Tests in Reactor	12 full travel cycles, plus 10 scrams at 350°F, 12 full travel cycles, plus 10 scrams at 550°F.	350°F to 550°F 2 to 9 cycles	0 to 135 psig 8 to 18 cycles	None	None	Piqua values during test period.	To be established

Mockup Coil Assembly Test – An assembly shall be constructed consisting of the upper pole piece, lift coils 1 and 2, grip coils 1 and 2, and pull coils 1 and 2. Developmental coils may be used if they meet the requirement that all coil case assembly interfaces are representative of the production drive. This test will verify the capability of the assembly fixture to maintain the required alignments and develop the assembly welding procedures. The assembly will be subjected to mechanical loadings simulating the loads imposed during drive assembly, shipping, and handling operations. The test assembly shall then be inserted in organic coolant at 650°F and 135 psig. The coils will be energized at equivalent steady-state d-c currents to verify calculated coil heating and hot spot temperatures.

Representative Coil Assembly Test – An assembly will be made consisting of one No. 2 lift coil, one No. 1 grip coil, one No. 1 pull coil and simulated top and bottom pole pieces and armature. This assembly shall be representative of the production drives with respect to the coils and thermal characteristics. This assembly will be subjected to an accelerated environmental test to demonstrate the drives ability to retain its structural integrity and hermetic sealing of the coil.

Defected Coil Test – Six coil assemblies of pull coil No. 2 configuration will be defected to allow the entrance of organic. Three of these coils will be operated in distilled reactor coolant with 400 ppm H_2O, and three will be operated in distilled reactor coolant with 1800 ppm H_2O. The coils in each set of three will be operated at constant equivalent d-c currents corresponding to design maximum power, 200% design maximum power, and at the 10 amp maximum power supply capability. The objective of this test is to establish the compatibility of the wire-insulation system employed in the coils with the reactor coolant.

MI Cable Test – The MI cable test will be run in conjunction with the representative coil assembly test. Two MI cable assemblies will penetrate the test vessel top flange in a manner simulating the attachment of the cable to the coils. The coiled lengths of cable and end seals will be immersed in the distilled reactor coolant with the representative coil test assembly and will be subjected to the same environmental test conditions.

Defected MI Cable Test – Two MI cable assemblies, defected in a manner similar to that which could be expected in service, will be operated in organic coolant energized with 50 volt potential between conductor and sheath. One specimen will be operated in distilled reactor coolant with 400 ppm H_2O for 1000 hr and one other with 1800 ppm H_2O for an additional 1000 hr. The objective of this test is to determine consequences of a failure which allows organic to saturate the mineral insulation.

Prototype Test in Reactor – The prototype drive will be tested in the Piqua reactor. The initial testing will verify drive and position indication system performance. The prototype drive will then undergo an endurance run to verify the wear characteristics of the modified coil case. During this run, temperature and pressure will be cycled to simulate operating conditions within the limitations imposed by pump heating. The prototype will also be subjected to the most severe thermal transients which occur during insertion into or removal from the reactor.

Production Drive Tests in Reactor – The production control rod drives will be tested in the reactor in groups of three to five depending on delivery rate. Drive performance and position indication will be verified at 350°F and 550°F with full reactor coolant flow. Calibration of the position indication system and checkout of the internal drive circuit modifications will be accomplished concurrently.

Nonenvironmental Performance Tests – In addition to the environmental testing tabulated above, the following dry ambient temperature performance tests will be carried out. Control rod drive SN 11 of the original configuration will be tested to verify its load capability with the full complement of coils and with each functional set of coils. The prototype drive will be tested to verify its load capability. Coil sequencing will be optimized to obtain maximum load capability versus current level and tentative operating current levels will be established. The position indication transducer outputs and linearity will be verified and calibration requirements established. Load capability versus current level and position transducer output characteristic requirements for the production drives will be established. Each production drive will be shimmed to achieve step length and tested to demonstrate its compliance with the performance requirements established by the prototype test.

Table 1. Piqua Control Rod Drive Performance Requirements and Associated Component Development Tests

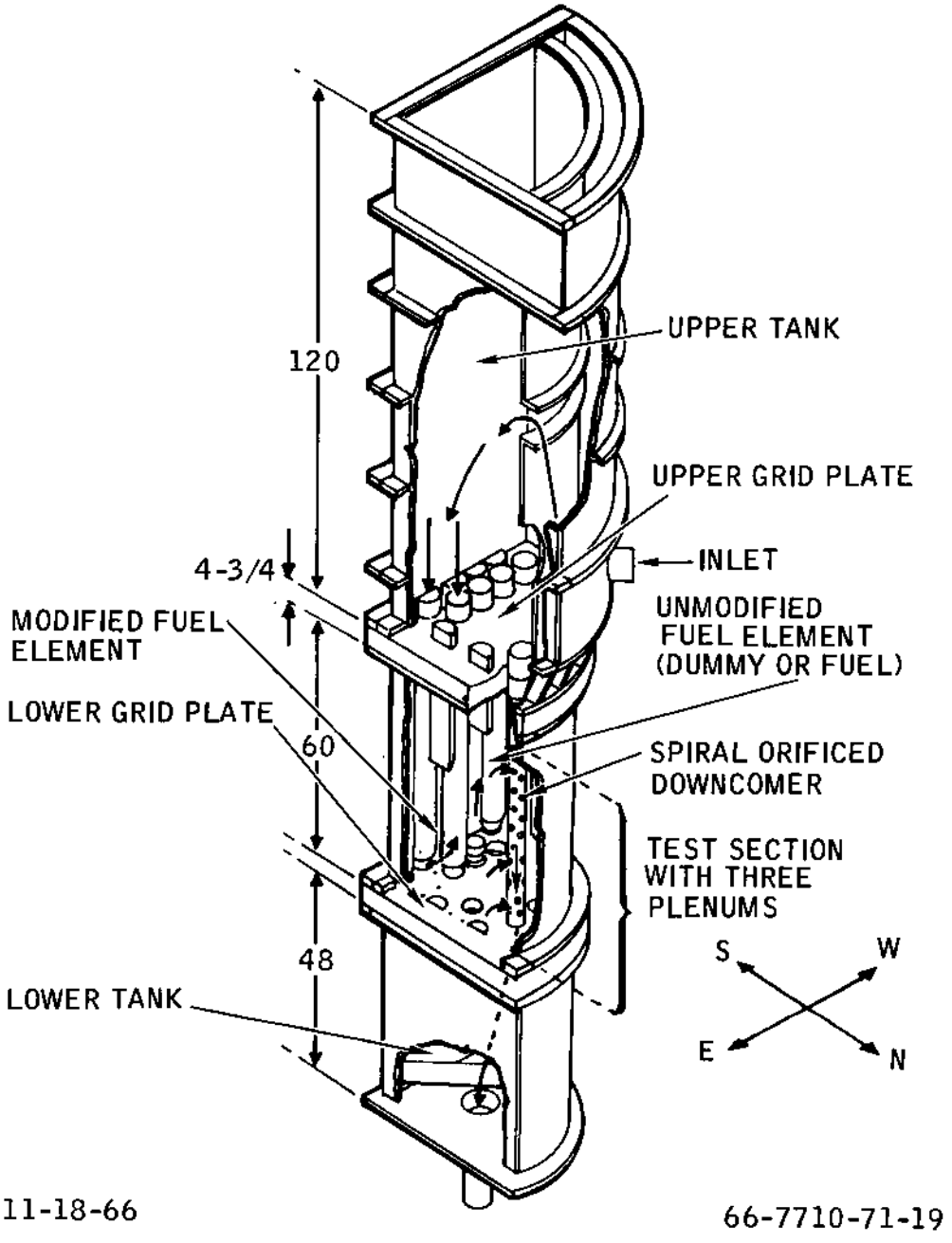


Figure 7. Quarter Section Hydraulic Tests

1) Vertical downcomer openings of 15 and 30 in.

2) Vertical downcomer openings of 50 in.

3) No downcomer test

4) Orificed downcomer test

5) Baffle test

6) Final testing of the orifice downcomer

7) Orificed downcomer test for 61-element core

The following single-element tests will be described below:

1) Four-hole orifice tests

2) Fuel element elbow tests

Additional tests described in this section are:

1) Flotation and buoyancy tests

2) Seven-element test

a. 41 Modified Element Core with Downcomers

(1) Vertical Downcomer Openings of 15 and 30 In.

The first of three tests with the equivalent of 41 modified elements and with the six downcomers (3-1/2 by 15-in. opening) facing outward toward the annulus region showed a greater ball velocity over that for the 31 modified element test with the eight 3/4-in.-diameter orifice flow holes with flow at 1100 gpm. The eight orifice flow holes for this test were enlarged to 1-1/8-in.-diameter, thus permitting an increased flow of 2100 gpm to the quarter section moderator plenum. The fluid (and ball) velocity in the center of the outer moderator plenum was of the order of 5 ft/sec, whereas the annulus area indicated a velocity gradient in the axial direction. In the annulus, the approximate velocities in the lower, middle, and upper region were estimated to be 1/3, 1, and 2 to 3 ft/sec, respectively.

The second test had the downcomers modified with an additional opening of 2 in. wide by 15 in. in height, placed directly below and joining the 3-1/2 by 15 in. existing opening. The results indicated that the annulus velocity had increased over the 15-in. downcomer opening figure.

(2) Vertical Downcomer Opening of 50 In.

The third and final test in the downcomer series involved a downcomer opening of 50 vertical in. that extended from 5/8 in. below the upper grid plate to 9-3/8 in. above the lower grid plate. Quarter section flow was established at 2220 gpm corresponding to 46 gpm through the inner process tube, plus flow through the eight 1-1/8 in.-diameter orifice flow holes. The general observed conclusions of the test are as follows: (1) coolant velocity in the center of the outer moderator plenum was about 4 ft/sec or greater; (2) annulus velocities in all regions were in excess of 1 ft/sec; and (3) there were no stagnant balls in any region of the plenum. The downcomer openings appear to be near optimum for balancing of flow distribution in the moderator plenum.

(3) No Downcomer Tests

A test was conducted on the quarter section with no downcomers in the core peripheral positions. The orificed downcomers were removed from the core and were replaced by plugs in the upper grid plate. The lower grid plate holes remained open. Flow was set at the limit of 2600 gpm for the quarter section. Preliminary conclusions of the test were:

1) Flow in the central moderator region of the plenum was axially upward except for some minor cross-flow across the dummy fuel elements.

2) Flow in the annulus was downward and uniform with velocities in the order of 1 ft/sec or greater.

3) No stagnant balls were observed.

Next, the no-downcomers configuration was also run at a flow of 200 gpm. The flow velocities and flow distribution were quite similar to those experienced in the orificed downcomer design at the same flow.

(4) Orificed Downcomer Tests

A flow test was conducted utilizing the orificed downcomers at a flow of 2600 gpm to investigate conditions that would develop in the event that fuel elements were not properly seated in the grid plate. For this test, Elements D-7, D-9, E-8, E-10, and F-9 in the quarter section were elevated from the upper grid plate by 2 in. to simulate this condition. The flow in the lower and middle

plenums remained essentially unchanged from the normal condition. In the upper plenum, however, opposing flows were observed, the flow from the lower fuel element housing being met directly by the bypass flow from the upper grid plate. Great agitation and turbulence occurred in the upper plenum. The polypropylene balls were observed to move almost radially outward toward the annulus region of the moderator plenum because of the opposed flows. No regions of stagnation were noted; in fact, the created leakage caused increased turbulence in the upper plenum.

In a subsequent test, the orificed downcomers were reinserted in the core and flow was set at 200 gpm to correspond to decay heat-flow conditions in the Piqua core. Since the flow was so low, some balls that entered the downcomers in the upper plenum drifted upward toward the "blind-alley" end of the downcomer. This is not necessarily an indication of stagnant flow. If the balls are slightly lighter than the water, the upward drift of the balls could be attributed to the buoyancy effect on the balls and not stagnation of the fluid. At these low flows, it is difficult to obtain good resolution of ball velocity.

(5) Baffle Test

The quarter-section test with the equivalent of 41 modified elements in the core was conducted with the baffle in place at the reference flow of 2220 gpm. The general conclusions of the baffle test were that: (1) coolant velocities in the central region of the moderator plenum were estimated to be 5 ft/sec; (2) coolant velocities in the annulus region outside the baffle were of the order of 1.5 ft/sec; and (3) the pressure drop across the baffle was 0.43 psi. The total pressure drop from the inside of the baffle to the lower tank plenum was measured to be 0.93 psi. The equivalent organic pressure drop is 0.83 psi, which compares favorably with the 0.8 psi estimated by calculation prior to the test run. The pressure drop due to the baffle design is approximately 50% greater than that associated with the latest slotted downcomer design.

(6) Selection and Final Testing of the Orificed Downcomer

The selection of downcomers over a flow baffle as a part of the modified reactor was made during the report period. Since both designs provide adequate flow throughout the reactor, this selection was based primarily on the maintainability and flexibility of operation of the downcomer design; the downcomers can

be easily removed from the reactor for inspection or replacement, and the baffle would be difficult to fabricate and install in the reactor.

After this selection had been made, the downcomer design was modified to include a continuous pattern of orifice flow holes rather than slots, to avoid the necessity for special downcomer orientation and possible flow distribution problems associated with downcome rotation.

A quarter-section hydraulic test was performed with the equivalent of 41 modified elements with orificed downcomers in the core. Each downcomer was perforated with 64 holes arranged in a spiral array which essentially made one revolution in 4 ft. The downcomers were placed in the quarter section in random fashion, with no orientation with respect to the grid plate holes. Hydraulic calculations called for a net moderator inflow of 11,012 gpm for the reactor or 2,753 gpm for the quarter section test. In conducting the test, 2,753 gpm was unobtainable due to limitations on upper tank height. To increase the hydraulic head of the open system, additional piping was installed at the exit nozzle and a flow of 2,550 gpm was obtained. The general observed conclusions from the test are as follows.

1) Central region velocities were high, as expected, and probably exceeded 5 ft/sec.

2) No stagnant balls were observed in the entire moderator plenum.

3) Flow in the annulus region was downward and relatively good for about the upper 2/3 to 3/4 of the plenum. The lower 1-ft section of the annulus, where there are no downcomer holes, had upflow. Thus, opposing flows were experienced in the annulus region.

4) The moderator plenum back-pressure was 0.57 psi (corrected for organic) and indicates that the orificed downcomer design has a greater back-pressure than the slotted downcomer design, which would have 0.51 psi at this flow. The holes in the orificed downcomer had a total flow area of 117 $in.^2$ as compared to the slotted downcomer with 102 $in.^2$ The data as extracted from the movie film can be readily upgraded to 2753 gpm. Data typical of that generated by the quarter section test are shown in Table 2.

TABLE 2

QUARTER SECTION TEST COOLANT VELOCITIES FOR 41- AND 51-ELEMENT CORES WITH SPIRAL ORIFICED DOWNCOMERS

Elements in Core	Type of Operation	Quarter Section Flow (gpm)	Test Section Plenum	Channel Velocity			
				Center (1,6,7,11,16)	Middle (3,8,21,25,30)	Outer (4,10,38,42)	Annulus* (5A,45A)
41	Normal	2590	Upper	3.6	5.6	2.9	-2.0
			Middle	6.4	5.8	3.2	-1.9
			Lower	8.4	6.0	4.9	1.3
41	Decay Heat	200	Upper	0.38	0.44	-	-0.22
			Middle	0.48	0.40	0.44	-
			Lower	0.68	0.55	-	0.11
41	Anomaly	2600	Upper	2.6	2.1	3.1	-
			Middle	4.9	3.9	4.0	-
			Lower	5.6	5.4	5.3	-
51	Normal	2330	Upper	3.1	4.9	2.5	-1.7
			Middle	5.6	5.1	2.8	-1.7
			Lower	7.3	5.2	4.3	1.1

*Negative velocity indicates downward direction.

b. 61 Modified Element Core with Downcomers

(1) Orificed Downcomer Test

Hydraulic tests were conducted to correspond to a 61-element core with a flow of 2180 gpm utilizing the orificed downcomers. Coolant velocities were observed to be proportional to the 41-element core configuration flow and the flow distribution was observed to be similar.

c. Single-Element Tests

A single-element test was run to determine the flow split and pressure drops associated with the 4-hole plate orifice (holes + 0.988 in. diam) and the eight 1-1/8-in.diameter orifice flow holes. These tests were conducted with total flows of 204, 251, and 397 gpm.

Flow splits determined at element flows of 204 and 251 gpm are subject to considerable experimental error. This is due to the low water head in the upper

tank, which resulted in erratic pressure instrument ratings. At an element flow of 397 gpm, the upper tank water level was sufficient to provide accurate pressure measurements. Thus, the 65/35% flow split measured for this test is considered valid. The flow split by analysis was within 2.7% from that measured by experiment. This small error establishes that flow splits for this configuration as flows above 394 gpm can be accurately calculated for other flow conditions.

The flow split in the single-element test was determined with a newly manufactured orifice plate containing four 5/8-in.-diameter flow holes. The flow was at 250 gpm for this test. The test flow split thus determined was 82.5/17.5% between the orifice flow holes and the orifice plate.

d. Fuel Element Elbow Tests

The fuel element elbow with the venturi diffuser and a throat diameter of 0.600 in. was tested to determine flow-pressure drop relationships. In the initial test, an O-ring was inserted between the drain valve housing and elbow to prevent leakage. The test data indicated a water flow of 47 gpm at a net pressure drop of about 57 psi. Since a flow of 46 gpm was expected at a ΔP of 36.6 psi for water (being equivalent to an organic flow of 46 gpm at 33.1 psi), the test results indicated the throat diameter was too small. A discharge coefficient of 0.579 was calculated from the data at hand, and a new throat diameter of 0.665 in. was calculated to yield the desired flow-pressure drop relationship.

In a subsequent test, with a new throat diameter, a water flow of 49.8 gpm was obtained at a pressure drop of 36.2 psi. In this test, leakage was permitted between the drain valve housing and elbow to duplicate reactor conditions. The test data indicated the throat diameter to be too large. Again using the data, a discharge coefficient of 0.627 was obtained and used to determine a throat diameter of 0.637-in., taking into account leakage present. Because the desired organic pressure drop of 33.0 psi was bracketed by the test results at the rated flow, it will not be necessary to retest the elbow with the throat diameter of 0.637 in. This throat diameter will yield an organic flow of 46 gpm at 33.1 psi.

A hydraulic test of an opposed elbow flowing directly into a 1-1/8-in.-diameter orifice flow hole of the single element was also conducted to determine if elbow flow would enter an adjacent fuel element and flow downward through the

lower orifice plate. Since the elbow flow in the reactor is at coolant inlet temperature, the potential problem exists that elbow flow could affect the temperature reading of the coolant outlet thermocouple in an adjacent element. A total of 350 1/8-in.-diameter polypropylene balls were injected through the elbow; of these, 82 or 23.4% entered the adjacent fuel element and came through the 4-hole orifice plate. If the balls were assumed to be an absolute indicator of elbow flow going through the 4-hole orifice plate, this would correspond to 10.8 gpm of inner process tube flow at a reactor inlet temperature of 543°F. The hydraulic testing indicated the flow split is unaffected by the presence of elbow flow. Therefore, since the flow split remains at 83/17%, the potential degradation in the outlet thermocouple reading is 10.8/44.4 or 24.3%. The conclusion of the hydraulic test is that the elbow or adjacent fuel element must be modified to divert the flow so that elbow flow does not affect the coolant outlet thermocouple reading. One solution being investigated is the redesign of the lower fitting to eliminate the high velocity jet coming from the elbow. At the end of this report period a number of design configurations were being investigated.

e. Flotation and Buoyancy Tests

A flotation test was conducted on the 5/16-in.-diameter polypropylene balls to determine the density variation and the velocity of the balls due to positive and negative buoyancy force effects. The temperature of the water was 95±1°F during the test. The specific gravity of the water was determined by a proof hydrometer which read 5 proof, and is equivalent to a specific gravity of 0.996. This roughly corresponds to the pure water density at 95°F having a specific gravity of 0.994. In the flotation test performed, more than 95% of the balls floated, indicating their specific gravity to be less than water at that temperature. There were only about a dozen balls of neutral density out of about 2100 balls. Less than 5% were heavier than water. Of the balls that floated, a selective process was conducted to determine the very lightest of balls and thus give an upper bound to the density variation. The balls were injected at the bottom of the plexiglass dummy element tube and their velocities noted over a 4-ft measured interval. Only the fastest rising balls were noted. An average of 7 light ball velocity readings indicated the upward velocity to be 0.21 ft/sec. This would correspond to a 5% error in ball velocity for a fluid velocity of 4 ft/sec, and is considered to be within the realm of experimental error. The

heavy balls were injected at the top and their downward velocity was noted over a 2-ft span. An average of 10 readings indicated an average downward velocity of 0.079 ft/sec. This corresponds to a 2% error for a 4-ft/sec fluid velocity. Of the floating balls, 18 were selected at random and their velocities indicated an average upward velocity of 0.189 ft/sec.

It has been determined from the above tests that the error in fluid velocity as measured by the balls was within plus 0.2 ft/sec and minum 0.08 ft/sec. These values lead to a more general conclusion that the fluid velocities as measured by the 5/16-in.- diameter balls in the quarter section tests are valid and are within a normal experimental error band.

The buoyancy and velocity of the 1/8-in.-diameter polypropylene balls was also determined (in stagnant water at 95°F). The specific gravity of the water was determined by a proof hydrometer. During the test, the water temperature was in the range of 94 to 95°F. The specific gravity of the balls was labelled as 1.000 ± 0.002 by the manufacturer. In the flotation test, most of the balls sank. It was necessary to add a small amount of detergent to the water to wet the surface of the balls so that the attached air bubbles could be disengaged. It was concluded that all balls were heavier than water at 95°F, and that a characteristic velocity of the balls sinking in water should be determined. Balls were therfore injected in the water, and the time in seconds for them to sink over a 3-ft path was noted. An average of 9 such heavy ball descents indicated a maximum measured velocity of 0.04 ft/sec, or for a 4.0 ft/sec fluid velocity, an error of 1%.

f. 7-Element Test

The design of the proposed 7-element test was completed and a design review was held. The objectives of the test were reviewed together with the testing sequence, and a schematic of the test set up. A test plan was written and received. As a result of the design review, the test plan was revised to include a description for determining the coefficient of discharge from the 7 bell-mouth orifice holes and the 4-hole plate orifice. To determine the two CD's, it will be necessary to instrument the test fixture with piezometer taps, one before and after the 7 orifice holes and one before the 4-hole orifice plate. These pressure taps will provide the necessary data to determine the discharge coefficients. A schematic diagram of the 7-element test assembly is shown in Figure 8.

g. Conclusions

The evaluation of the hydraulic design of the PNPF 41-element restart core has demonstrated that the data generated as a result of the hydraulic tests are representative of the expected restart core behavior. A single-element test was used to define orifice sizes, and was found acceptable in determining flow splits. Tests performed on the quarter-section verified the design of the downcomers, and confirmed that the velocities of the coolant in the outer moderator region were significantly greater than those required. These tests were performed for normal core flow conditions. In the above tests, the uncertainties were evaluated and it was found that they did not obviate the validity of the test results. Additional tests will be conducted with the 7-element test assembly to evaluate the new fuel element hydraulic design. The results of the new tests will be puslished in the next semiannual report.

2. Thermal-Hydraulic Parametric Studies of a 41-Element Core

Parametric studies applicable to a 41-element core based on thermal and hydraulic considerations have been completed and are summarized herein. It should be noted that many of these analyses are preliminary in nature and may be further modified and upgraded subsequent to this report.

Figure 9 gives element power as a function of relative radial power (peaking factor) φ_R, for various reactor thermal powers in a 41-element core.

Figure 10 shows predicted film growths on fuel element surfaces during modified Core I life as a function of coolant velocity for an average element and for a higher-powered center element. Assumed burnup is about 3000 Mwd/ MTU (41 elements generating 20 Mwt for 3,600 hr, followed by the installation of 20 additional elements and operation of the 61-element core at 20 Mwt for a subsequent 12,400 hr). If the film growth is linear with burnup, about 1/3 of the total film would be deposited during reactor operation in the 41-element configuration. Data are for a foulant particle concentration, C_o, of 1×10^{-4} lb/ft^3.

Figure 11 illustrates dependence of the effective heat-transfer coefficient, h_e, on the foulant film thickness. Data are plotted for two representative coolant velocities. The increasing film thickness may also be interpreted to illustrate the degradation of h_e with time.

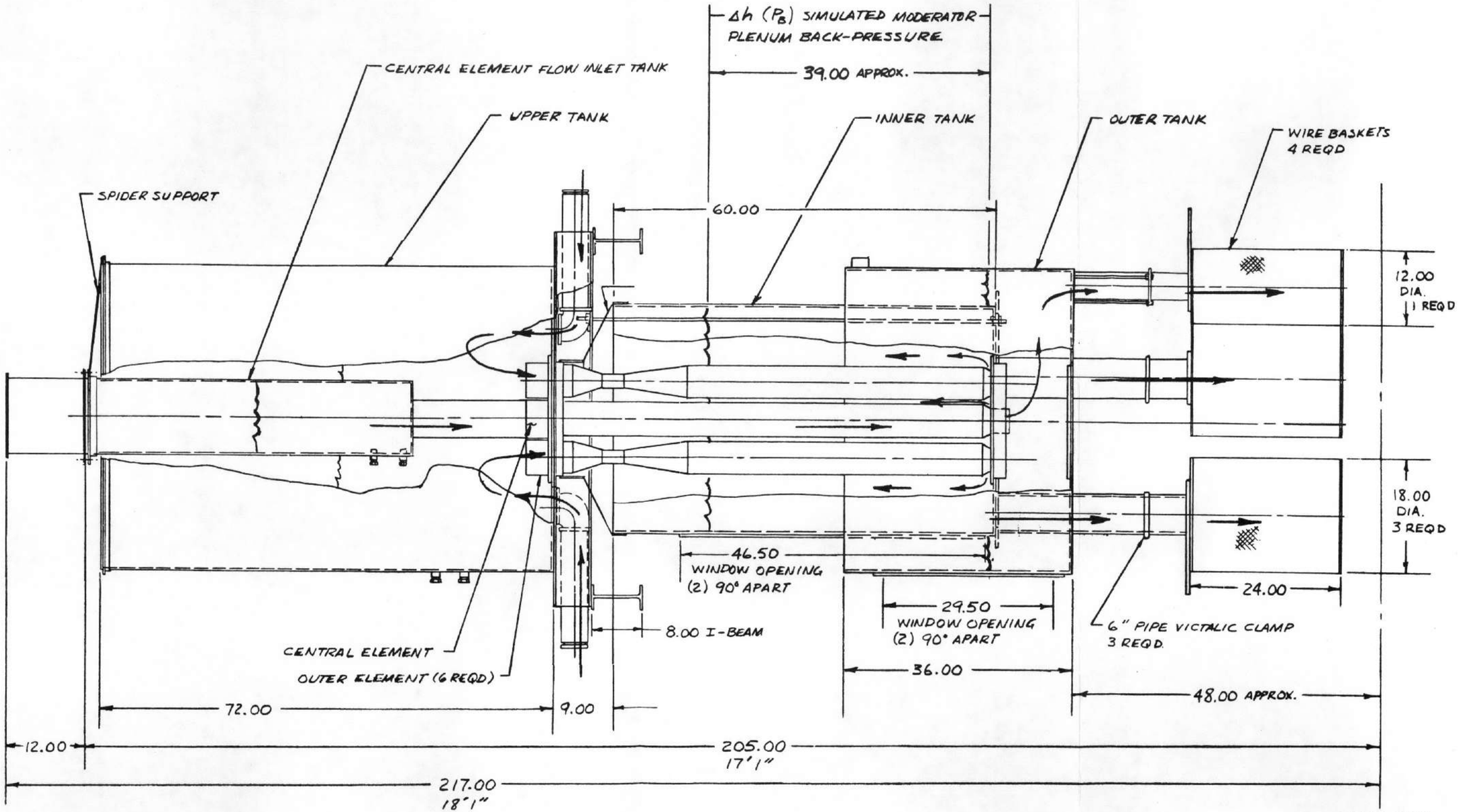


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Figure 8. Schematic Diagram of Seven Element Test Assembly

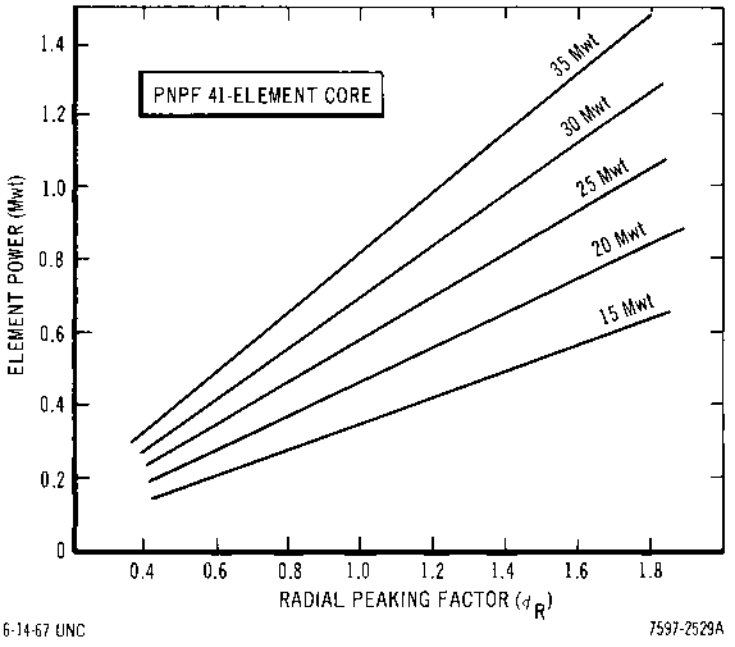


Figure 9. Element Power as Function of Radial Peaking Factor for Various Reactor Thermal Powers

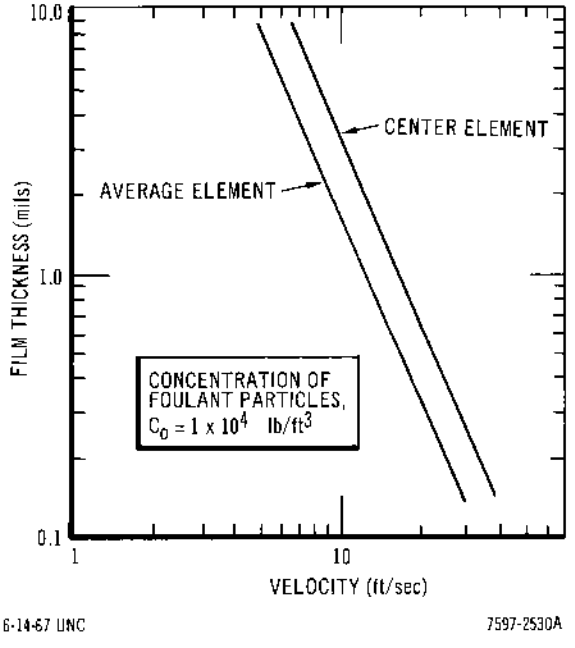


Figure 10. Predicted Film Growth on Fuel Element Surfaces During Modified Core I Life

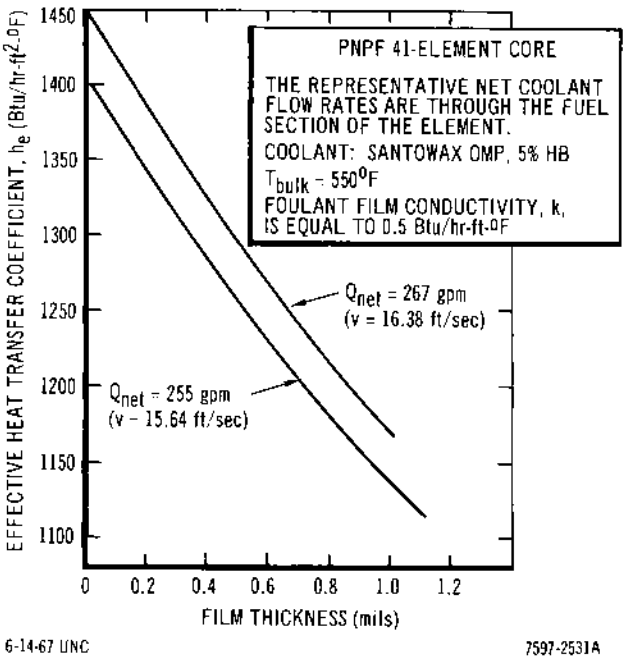


Figure 11. Film Heat Transfer Coefficient as a Function of Foulant Film Thickness

Figure 12 presents maximum reactor power capability for a 41-element unorificed core as a function of foulant film thickness for two representative maximum powered elements. Limitations and criteria are listed in Table 2. Two maximum conditions are considered: (1) center element F-11 with control rod withdrawn (ϕ_R = 1.8), and (2) the surrounding ring of elements, F-9, E-10, E-12, F-13, G-10, and G-12 (ϕ_R = 1.66). These peaking factors are slightly conservative.

Figure 13 presents hot-channel maximum fuel cladding surface temperatures. T_s (max), as a function of foulant film thickness at a reactor thermal power of 30 Mwt for the two representative maximum-powered elements.

3. Fuel Element Examinations at AI Hot Laboratory

As part of the Piqua Recovery Program, postirradiation destructive examinations of selected fuel elements have been conducted at the AI Hot Laboratory (AIHL). Presented here are summaries of observations about two elements, P-1050 (F-3) and P-1114 (H-9), examined during the report period. More complete and detailed reports on the elements will be published in the future.

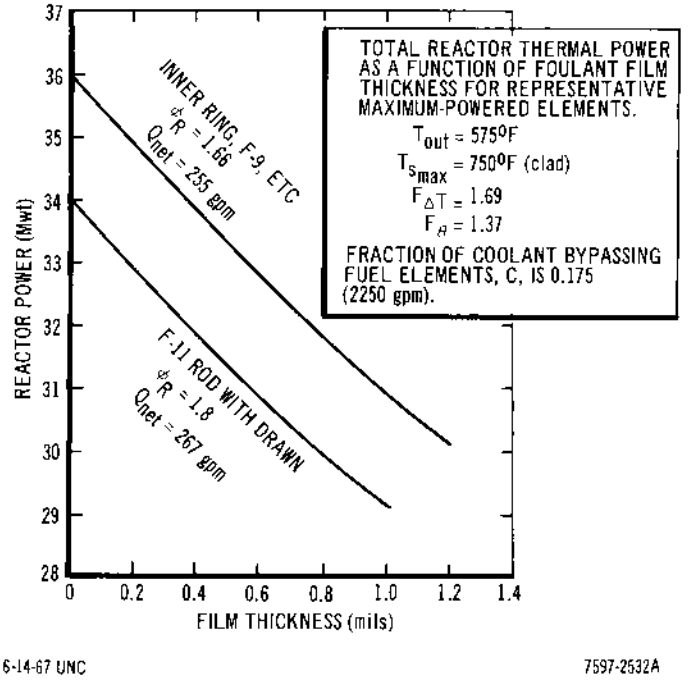


Figure 12. PNPF 41-Element Unorificed Core Maximum Power Capability

TABLE 3

THERMAL-HYDRAULIC CRITERIA

Parameters	Values
Reactor peak/average axial power	1.5
Coolant	Santowax OMP + 5% HB
Coolant temperature at which properties are evaluated (°F)	550
Reactor mixed mean outlet temperature (°F)	575
Limiting hot channel maximum cladding temperature, including fouling allowance (°F)	750
Total coolant flow rate (gpm)	12,860
Flow through fuel section (net flow) of control element (gpm)	267
Flow through fuel section (net flow) of standard fuel element (gpm)	255
Coolant flow area per element fuel section ($in.^2$)	5.23
Effective unit heat transfer area of fuel cladding (ft^2/ft)	11.65
Conductivity of foulant film (Btu/hr-ft-°F)	0.5
Fraction of heat generated in moderator	0.03
Fraction of heat generated in coolant	0.01
Hot channel factors	
a) Coolant temperature rise ($F_{\Delta T}$)	1.69
b) Film temperature drop (F_{θ})	1.37

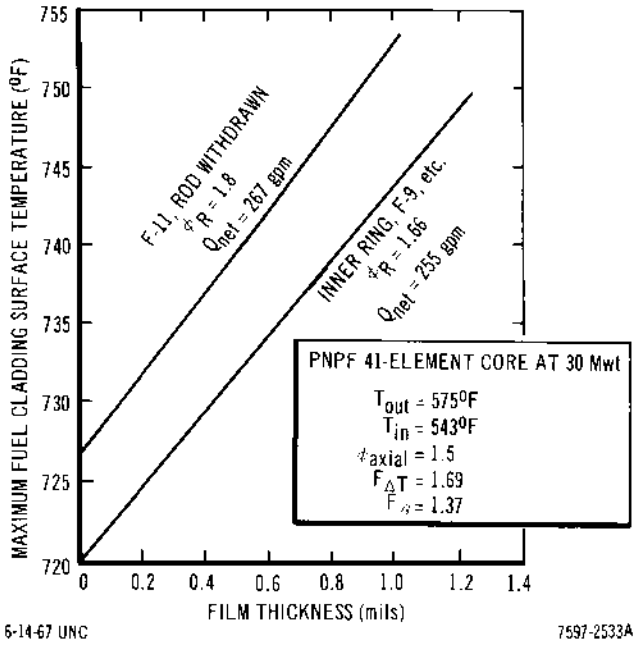


Figure 13. Hot Channel Maximum Fuel Cladding Surface Temperature

a. P-1050 (F-3)

Piqua fuel element P-1050 was the 13th fuel element removed from the core as part of the Piqua Recovery Program. This element was removed from core position F-3, where it originally had been inserted, after it had attained an average burnup of 1045 Mwd/MTU. Relative flow calculations based upon the coolant outlet temperature, showed that this element had ~91% of its original flow. The main reason for the examination was to investigate the possible reusability of elements having similar burnups and flow behavior histories. The determination of coke buildup inside the inner process tube was an integral part of the evaluation. The element was also chosen since its outer process tube had a dull surface finish prior to its insertion into the core.

The examination of the inner process tube showed that it had not been deformed at the bottom like the H-5 fuel element. However, there was some coke deposit inside which was fairly solid for the bottom one inch or less, and was porous above this area. This porous coke deposit was not extensive. At the interface of the bottom, two fuel cylinders light brown solid coolant was nearly all that was present. Some coolant was present in the valve body below the inner

process tube at the time of removal. Comparing deposits found on this element with the H-5 element (which had the same residence time but a higher relative power), shows that with a lower neutron flux and temperature there was still insufficient coolant flow in the inner process tube to avoid excessive decomposition of the coolant. However, the degree of the coolant coking was more severe at the higher temperature and radiation levels.

There was a uniform deposit on both sides of the outer process tube, but it appeared to be thicker on the outside. The film on the fuel cylinders did not appear to be very thick or adherent, and there was considerable flaking on the higher burnup cylinders. There were no coolant channels that were blocked. No film thickness measurements are available, since the metallographic examination has not been completed.

Dimensional measurements of the fuel cylinders showed that changes were small with a maximum outer diameter increase of 0.64% on the outer fuel cylinders. For complete information on postirradiation of this element, refer to NAA-SR-TDR-12131.

b. P-1114 (H-9)

Piqua fuel element P-1114 was the 48th fuel element removed from the reactor in the Piqua Recovery Program. This element was removed from core position H-9, where it had originally been inserted, after it had attained an average burnup of 3058 Mwd/MTU. The calculated flow through this element, based upon coolant outlet temperature, had not changed significantly during its irradiation (100%). This fuel element had received extensive preirradiation measurements so that it could be used as part of the Core I surveillance program. This element was chosen for examination because it was an evaluation fuel element with high burnup. For complete information on postirradiation of this element, refer to NAA-SR-TDR-12214.

Examination of the outer process tube, which had a shiny surface prior to insertion, revealed that there were five buckles present; one located near the screen retainer ring extending circumferentially from 225 to 30° orientation, a slight buckle 25 in. below the grid plate support bevel extending from 280 to 355°, a minor buckle 28 in. below the grid plate support bevel extending from 70 to 190°, a minor buckle 60 in. below the grid plate support bevel extending

from 80 to 220° and a slight buckle or impression of the fuel element support ring at the bottom of the outer process tube extending from 0 to 60°. There were also several small dents and some slight fin impressions. The majority of the surface was bare shiny metal with some small patches of thin black film, and a large area of flaky film near the top of the outer process tube. In addition, there were three localized regions on the outer process tube (when examined at Piqua, they appeared as fine cracks), which now appear to be merely stains on the metal. A metallographic examination of the process tube in these regions will be conducted to ascertain whether these marks are only stains.

Examination of the outer fuel cylinders showed that there was a relatively thin, adherent film on the aluminum fins. However, there were localized areas where the film had fallen from the outer process tube and was lying on top of the fin tips. These areas of film on the fuel cylinders corresponded to areas on the outer process tube where the film was no longer present. In addition, there were a few single coolant channels that were blocked with coke deposits for lengths up to 2 in. Probing these deposits showed that they were a solid mass. This observation is in contrast to the way in which the transfer films behaved; they flaked off very easily. The outer surfaces of the inner fuel cylinders were also covered with a thin, black, adherent continuous film.

In an effort to remove the inner fuel cylinders from the inner process tube, Cylinder 6 came off only after a substantial force was applied, and Cylinders 7 and 8 could not be freed. Examination of the inside surface of Cylinder 6 revealed what appeared to be coke deposits located near the bottom. It is probable that this deposit caused the difficulty in removing Cylinder 6, and is probably causing the problem in removing Cylinders 7 and 8. Outer diameter measurements of these two cylinders did not show any increase beyond that expected for an element with this burnup. These two fuel cylinders are presently being slit longitudinally with the elox machine so that they can be removed from the inner process tube.

4. Measured Fouling Rate on PNPF Fuel Elements

The objectives of this study were (1) to determine the fouling rates on PNPF fuel element heat transfer surfaces as a function of fuel exposure, and (2) to summarize film thickness data collected during hot cell fuel element examinations, plus analyses of carbonaceous deposits from the core outer moderator region (OMR) and out-of-core positions.

Cladding samples for the fouling rate determinations were taken from identical locations on four fuel elements, one orificed (F-3) and three unorificed (H-9, H-5, C-12). The sampling locations were on the outer fuel cylinder No. 3 and on inner cylinder 7, approximately 4 in. below the upper edge. These positions correspond approximately to the axial peak power location on the elements. Photomicrographs were taken of the base and sides of the fins, and film measurements made at five locations on each sample.

All four elements were installed prior to initial startup, and remained in the core until after the shutdown on January 13, 1966. The flow velocities in the orificed and unorificed elements were estimated to be 10 and 14 ft/sec, respectively, based on preoperational flow tests. Although the coolant impurity content varied widely during the overall interval, the coolant quality requirements outlined in the test specifications were generally satisfied during reactor operation; the one exception was the higher than desired carbonly-bound oxygen content in the reactor at startup.

The measurements made at each sample location on each of the three unorificed elements were subjected to a best linear fit as a function of fuel burnup (Mwd/MTU). This approach was made for measurements taken at the base as well as on the sides of the fins. The one set of data for the orificed type of element was incomplete in that no film could be detected on the sides of fins of three of the samples, but it appeared likely that the film in these locations had flaked off in the core or was scraped off during handling.

The fouling rates determined are as follows:

Unorificed Element 2.7×10^{-4} mil/Mwd/MTU

Orificed Element 6.0×10^{-4} mil/Mwd/MTU

In other words, the rate of fouling on heat transfer surfaces increased by a factor of approximately two when the flow was reduced from 14 to 10 ft/sec.

All the results indicate the film formation rate was highest at the inside of the inner fuel cylinder, and decreased toward the outer surface of the outer fuel cylinder. This observation held for both orificed and unorificed elements. Although no quantitative explanation is available at present, the differences may result from differences in hydraulic diameter and flow conditions in the inner and outer channels.

Summaries of fuel examination film results (Table 4) and carbonaceous deposit analyses (Table 5), both in-core (OMR) and out-of-core, are presented as well as a resume of coolant chemistry conditions determined throughout Phase I of Core I (Table 6).

5. Safety Analysis

A summary of the safety assessment for the PNPF modification follows.

A review of the PNPF safeguards features, in the light of an operating history of 2.5 yr shows that safety performance of the plant has been good. Coolant, air, and waste monitoring have verified the plant safeguards effectiveness.

The planned plant modifications do not directly affect the safeguards features. The new core loading results in similar or improved reactor stability and postulated nuclear incident response. Potential radiological hazards due to coolant activity, fission product inventory, etc, are similar or reduced.

Reactivity accountability procedures will be continued with improved accuracy to verify the conclusions of the reactivity anomaly studies.

Assessments of the consequences of postulated coke buildup in the new core indicate that the reactivity effects are minor. However, mechanical damage and augmentation of the consequences of nuclear incidents are theoretically possible. Consequently, in the absence of verification of the expected elimination of gross core coking, an intensified operational surveillance of core internal conditions will be instituted.

Conclusions derived from the Safety Evaluation of PNPF Modifications* follows.

1) Planned modifications, on the basis of investigative studies of the unloaded core and operating record, are fully expected to prevent the recurrence of the core "coking" effect and the attendant problems of sticking fuel elements, control rod binding, and reactivity accounting anomalies.

2) Planned modifications to the control rod drive circuitry will improve their operational reliability.

*"Safety Evaluation of PNPF Modifications," NAA-SR-Memo-12103 (Dec. 27, 1966)

TABLE 4

DESCRIPTION OF FILM AND DEPOSITS ASSOCIATED WITH PNPF FUEL ELEMENTS

Fuel Element	Sample Description	Total Core Exposure (Mwd)	Element Exposure (Mwd)	Thickness (mil)		Ash (wt %)	C/H Ratio
P-1091 (F-13)	Film flakes from xylene wash of all fuel cylinders	5,800	1,375	0.5 (avg) (0.2 - 1.0)		8.9	-
	Film brushed from Cyl. 1			0.4 (0.1-0.5)		12.5†	-
	Film brushed from Cyl. 3 at 0°			0.40		5.0	2.35
	Film brushed from Cyl. 3 at 180°			0.40		5.1, 6.0†	1.96
	Film brushed from Cyl. 7			1.0 (0.3-1.7)		-	1.79
	Film brushed from Cyl. 8			0.4 (0.2-0.7)		-	1.69
	Film brushed from inside of OPT			0.08-0.14 (some of film may have abraded)		8.0, 11.3†	-
	IPT deposit					0.14	2.1-3.1
	Valve at bottom of IPT (hard + porous coke)					0.16	2-2.3/1
P-1071 (D-9)	Film from outer surface of outer Cyl. 3	3,240	796	0.07		110*	
	Film from outer surface of inner Cyl. 7			0.05		80*	
	IPT deposit					3-4	
	Film on outside of OPT			0.4		27.2	
P-1050 (F-3)	OPT metallographic samples			Inside	Outside		
	(1) 9 in. from top of OPT weld	10,659	896	<0.1	0.4		
	(2) 22-1/2 in. from top of OPT weld			0.2	8.0		
	(3) 36 in. from top of OPT weld			0.2	4.5		
	(4) 49-1/2 in. from top of OPT weld			0.1	1.0		
				Base of Fin	Side of Fin		
	Film from inside surface of Cyl. 7			1.1	<0.1		
	Film from outside surface of Cyl. 7			1.0	<0.1		
	Film from inside surface of Cyl. 3			1.0	<0.1		
	Film from outside surface of Cyl. 3			0.7	0.7		
				Base of Fin	Side of Fin		
P-1073	Film from inside surface of Cyl. 7	10,695	2,107	1.2	1.5		
	Film from outside surface of Cyl. 7			1.6	0.9		
	Film from inside surface of Cyl. 3			0.6	0.7		
	Film from outside surface of Cyl. 3			0.7	0.6		
	IPT deposit					0.03-0.8	2.0-3.1
P-1114 (H-9)	Flakes from inside surface of Cyl. 5 at the bottom	10,695	2,454	0.2-1.0		7	-
	Flakes from inside surface of Cyl. 6 at the bottom			0.2-1.0		0.8	2.5
	Flakes from inside surface of Cyl. 7 at the bottom			0.2-1.0		2.6	2.5
	Flakes from inside surface of Cyl. 7 at the top			0.2-1.0		2.1	2.3
	IPT deposit					<0.1-1.3	2.4-4.2
				Base of Fin	Side of Fin		
	Film from inside surface of Cyl. 7			1.4	1.7		
	Film from outside surface of Cyl. 7			1.0	1.3		
	Film from inside surface of Cyl. 3			1.2	1.6		
	Film from outside surface of Cyl. 3			1.0	1.1		
				Base of Fin	Side of Fin		
P-1063 (H-5)	Film from inside surface of Cyl. 7	10,695	1,418	0.9	1.0		
	Film from outside surface of Cyl. 7			1.1	0.7		
	Film from inside surface Cyl. 3			0.6	0.6		
	Film from outside surface of Cyl. 3			0.5	0.9		
	OPT film measurements:			Inside	Outside		
	Top of No. 1			0.0	<0.1		
	Center of No. 2			<0.1	5.6		
	Center of No. 3			0.3	7.9		
	Center of No. 4			<0.1	3.1		

*Elemental Al (scraped off clad) only crystalline material – oxidation of Al during ashing procedure resulted in high ash values.
†Sample ashed in platinum.

TABLE 5

CARBONACEOUS DEPOSITS FROM THE OUTER MODERATOR REGION AND OUT-OF-CORE POSITIONS

Sample Description*	Ash (wt%)	C/H Ratio	Total Oxygen (wt%)	C/O Ratio	Total C + H (wt%)	Density (gm/cm^3)	Total Fe (wt%)
(1) Out-of-Core Reactor Deposits							
Deposit from E-6 in-vessel filter housing – removed August 1965	1-14	2.1-2.2	-	-			
Porous coke-like deposit removed from purification still bottoms heater after its failure in April 1966	0.5	4.2	3.3	38			
Solid-appearing layer from still bottom heater deposit, removed April 1966	0.7	3.6	3.7	34			
Deposit removed from P-2 main pump section strainer in October 1965. (This deposit plugged the strainer during period when contents of tank T-2 was being transferred to the system.)	0.7	1.6	13	8.4			
Sludge deposit removed from tank T-1 in May 1965	1.3	1.6	12-14	9.0			
(2) Outer Moderator Region Deposit							
Samples of the outer moderator region deposit; the ranges given result from 64 samples taken from 21 core locations (top number); the lower number is a typical figure	<0.1-1.1 <0.1	2.1-5.9 3.1	0.45-1.06 0.7	122-287 -	91-100 98	1.3-1.6	300-1100 700

*Before analysis, all samples were extracted with benzenes to remove coolant.

TABLE 6

PNPF COOLANT CHEMISTRY

	HB (wt%)	Ash (ppm)	Carbonyl-Bound Oxygen (ppm)	MST (A/mg x 10^5)	Fe (ppm)	Mn^{56} (dpm/gm/Mwt)	Na^{24} (dpm/gm/Mwt)
<May 16, 1964	6 (before Mar, was 10-12)	2 (0-5)		20-50 (0-90)	0.2 ⁄ 0.4 (0-1.0)	steady decrease from 400 in 1963 to 65 in May 1964	decrease from 4 in 1963 (60, Mar 1964) to average of 8 in May 1964
May 17-Aug 10 (SD)	<0.1	4 (0-8)	-	30 ⁄ 40 \ 20 (10-50)	1.0 (June 0.2-1.8 \ 0.4 (July)	-	-
Aug 11, 1964-Jan 27, 1965	6 ⁄ 9 (Sept-Oct) \ 6 (Oct)	2-3 \ 1 (Oct) (0-4)	-	20 ⁄ 30-35 (sept was in range 15-50)	0.2 (0-0.8)	50 ⁄ 240 (Sept) \ 60 (Dec SD); 150 after Dec SD; \ 60 (Jan)	7 ⁄ 20 (Sept) \ 5 (Dec SD); 10 after SD; 7 (Jan)
Jan 28-Feb 12 (SD)	3 \ 0.5	5	-	5 ⁄ 35	0.4	-	-
Feb 13-Apr 2	3 ⁄ 6	2.5 \ 1-2	30 (18-46)	35 \ 7 ⁄ 35 at Mar SD; 11 after SD	0.2 ⁄ 0.5 (0.1-1.5)	225 \ 70 ⁄ 100 at March SD; 45 after SD	20 4
Apr 3-26 (SD)	<0.1	4 (2-7)	40 ⁄ 175 \ 75	10 ⁄ 25 \ 10	-	-	-
Apr 27-July 18	4 ⁄ 7	1-2 (0-4)	55 \ 25	10 (20 during May SD)	-	4200 \ 70 ⁄ 170 (June)	1000 \ 25 \ 8 (June)
July 19-Sept 6 (SD)	<0.1	6 (1-9)	160-325	120 (60-185)	-	-	-
Sept 6, 1965-Jan 13, 1966	4 (Sept-Oct) ⁄ 5-6 (Nov-Jan)	1	150 \ 50 (Oct) steady \ 20 (Jan)	15 \ 10 (Oct) constant until ⁄ 17 (Jan)	-	2100 \ 40 (Sept) ⁄ 110 (Nov) \ 80 (Dec) \ 60 (Jan)	225 \ 8 (Sept) steady \ 4-5 (Nov)
>Jan 14 (SD); Jan 14-Mar 22	<0.1	1	10 ⁄ 50	(10-86) \ 4 (Feb)	-	-	-
Purif. System Down		(1-2) ⁄ (5-15)	130 ⁄ 750	70 ⁄ 250	-	-	-
Purif. System Back (Apr)		7 \ 2 (May) \ 1 (June)	500 \ 270 (May) ⁄ 400 (June) \ 250	200 \ 20 (May) ⁄ 250 (June) \ 25 (June)	-	-	-

(1) SD - Shutdown

(2) 50 240 (Sept) 60 (Dec. SD) - the concentration was 50; it increased to 240 in September and then decreased to 60 at the December Shutdown
⁄ = increase to
\ = decrease to

(3) 2 (0-5) - the average concentration was 2; the range of values was from 0 to 5.

3) The modified facility will have similar or improved reactor dynamic and static safety characteristics compared to those described in the PNPF Final Safeguards Report.* The facility safeguards features remain unchanged.

4) Studies of carbonaceous deposit accumulation effects reveal that the significant safety implications of coking are mechanical damage to core internal structures and possible augmentation of postualted nuclear accidents through increased fuel temperatures. The coking effect is not judged to be a direct causative agent for reactor incidents.

5) An operations surveillance program is used to continuously verify the predicted facility performance. During operation of the modified reactor system, added surveillance program activities will include improved reactivity accounting procedures and carbonaceous deposit accumulation monitoring.

6) A review of postulated nuclear incidents and their potential radiological sequences has not revealed any significant departure from those presented in the original safeguards report.†

*"Final Safeguards Summary Report for the Piqua Nuclear Power Facility," NAA-SR-5608 (August 1, 1961)
†Ibid.

II. OPERATIONS ANALYSIS

A. RELATIVE FLOW COMPUTER CODE

To provide improved surveillance of fuel element behavior, a new computer program has been developed to calculate the relative flow for a particular channel.

In the initial core, the entire fuel inventory consisted of new fuel elements, so that the flow calibration curves could be used with the constraint,

$$\Sigma w_i \text{ (orificed)} + \Sigma w_i \text{ (unorificed)} = \text{total element flow} \quad , \qquad \ldots(1)$$

to establish the original orificed and unorificed element flows, w_o. With these values and measured ΔT's, the element Q_1's were calculated. The flow calibrations showed a manimum of ± 10% band about the calibration curves, due to inherent variances in the physical geometry of the fuel elements. There is also some variance about the true value of ΔT_i. The calculated values of Q_i were therefore not equal in spatially symmetrical locations. Since it was anticipated that real variations in Q_i for symmetrical positions should be small, the values of Q_i were smoothed by averaging symmetrical locations and adjusting the averaged values to the expected J_o-shaped radial distribution. With these smoothed Q_i's, the initial relative power distribution was determined. Once this distribution was established, all temperature responses were viewed as changes in element flow conditions, so that w_i could always be calculated from the relationship,

$$RP_i = \frac{Q_i}{\Sigma Q_i/N} = \frac{w_i \Delta T_i}{P \times f/N} \quad , \qquad \ldots(2)$$

where

RP_i = relative power of element i,
P = reactor power,
f = fraction of reactor power generated in the fuel,
N = number of elements in the core,
w_i = mass flow rate in element 1, and
ΔT_i = temperature rise across the element.

Summation of the element coolant flows must satisfy,

$$\sum_{i=1}^{N} w_i = \text{total coolant flow through the fuel} \quad . \qquad \ldots(3)$$

If this constraint is not met, ΔT_i must be adjusted by a correction factor T_{corr}, since the validity of the power calculation is insured by the steam-organic power comparison. The ΔT_i's are referenced to the mixed mean temperature on an independent ΔT recorder, so that recording errors may be introduced. T_{corr} is a constant offset applied to all recorded ΔT_i's, since the correction is due to reference temperature level error. With the new values of w_i plus w_o, relative flows can be calculated from the relationship

$$\frac{w_i}{w_o} = \text{relative flow of element i, } RF_i \quad . \qquad \ldots(4)$$

The values of ΔT_i and w_i fix the thermal performance of the fuel elements.

This new surveillance program consists of two computer programs. The first provides a daily surveillance through calculations of relative flows to determine any major short-term flow degradation in a channel. The second program uses the daily output to correlate graphically the relative flows with individual element as well as total core exposures, in order to detect long-term trends. The behavior of an element is plotted on an individual graph with three graphs per sheet. In this way, overall trends in element performance are easily recognized. In addition, this long-term surveillance program contains a fuel management program to maintain a record of fuel element exposure.

During the month of June, it became apparent from investigating the early core life data that a discrepancy existed in the reference thermocouple used in the fuel element outlet temperature readout. The computer calculation was modified to make provision for the discrepancy encountered. The modified calculation serves a dual purpose: (1) it corrects for the reference thermocouple error, and (2) it provides a means of detecting temperature recorder drift.

During initial investigations of the transition from 61 to 67 elements, it became evident that a further modification to the program was required. The total element flow of Equation 1 above was obtained originally from an averaged value of total coolant flow rate over the first few days of operation; the flow was treated as invariant thereafter. Consideration of the operating data showed that the flow reduction resulting from increasing in-vessel filter and core pressure drop were of sufficient magnitude to warrant special treatment in the program. The necessary changes were made, and code results reflect the inclusion of this refinement.

With this new code, the relative flow at the start and end of each core designation (except the short duration Core 1-E) has been established (Figures 14 through 28), as well as the relative power distribution (Figures 29 through 32).

B. STARTUP TEST PROCEDURES

1. Test Plan Summary

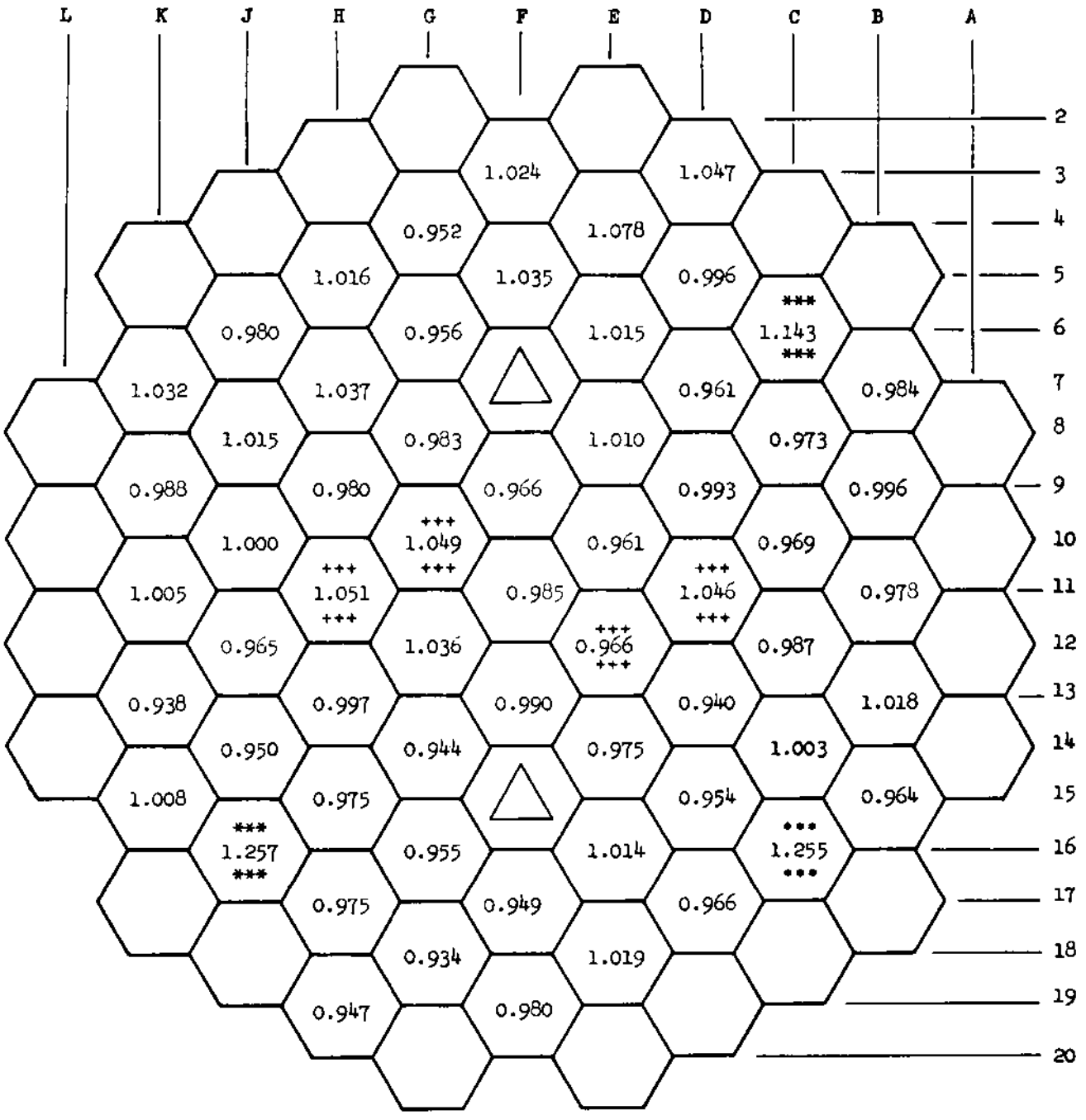
A Test Plan Summary was written which establishes and briefly describes the tests which will be conducted as part of the PNPF – Phase III Recovery Program. The basic objectives of the test program are to load and test the reactor, and checkout the associated systems prior to returning the PNPF to normal service. The program will do the following.

1) Demonstrate the functional operability of the nonnuclear systems, provide performance data of these systems, and establish base values from which subsequent operation can be evaluated. (Systems maintained in an operating status during the recovery program will be excluded from the formal test program.)

2) Perform the initial critical testing (including fuel loading), and measure nuclear and physical parameters for the system.

3) Perform the initial power operation of the facility and measure power dependent nuclear and physical parameters.

Three series of tests define the sequence of testing, as follows.



*** Data Recorded on the Tdr 384 Recorder

+++ Data Recorded on the Tdr 385 Recorder

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Figure 14. Relative Flow Distribution at Start of Core IA

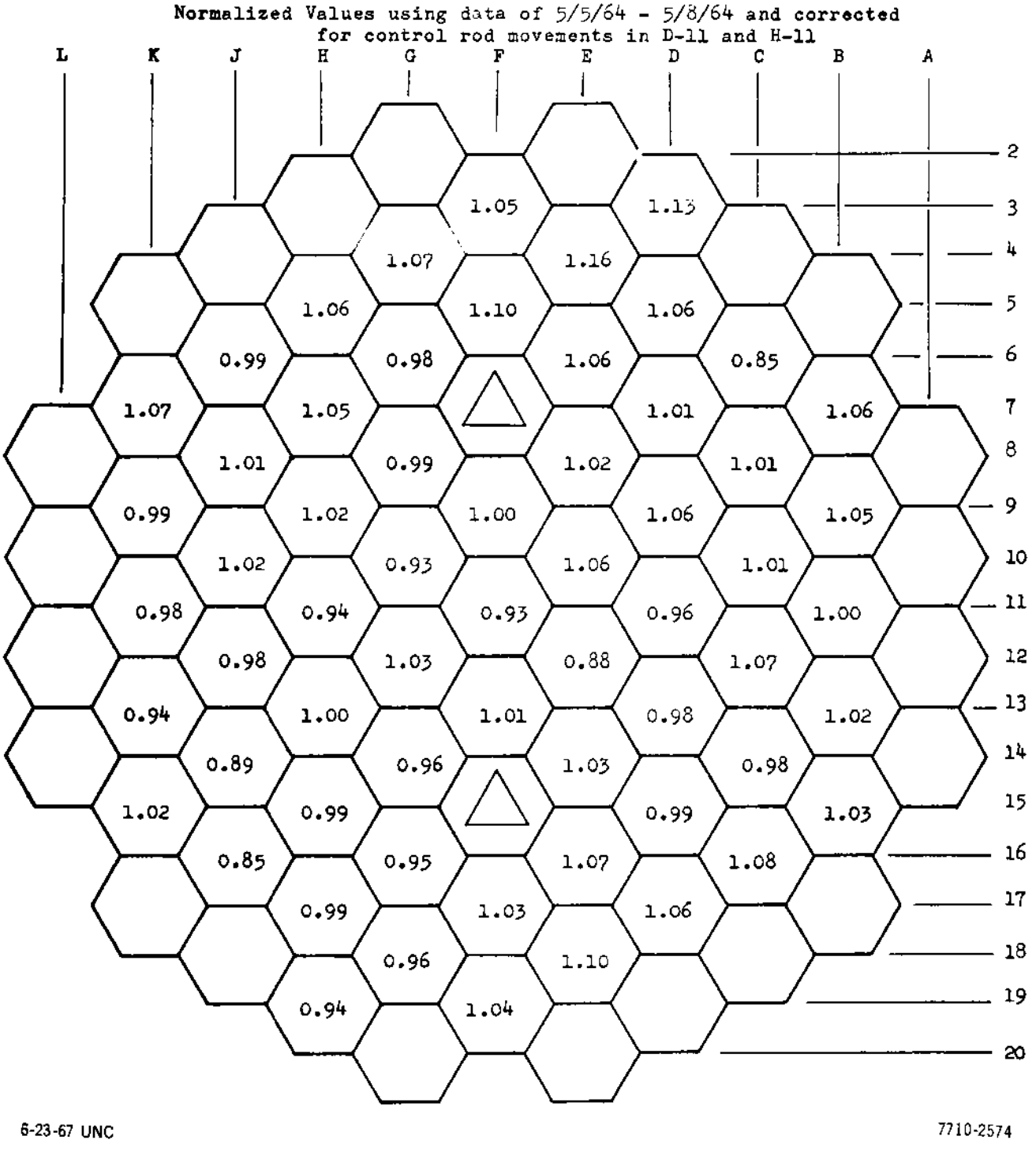

Figure 15. Relative Flow Distribution at End of Core IA

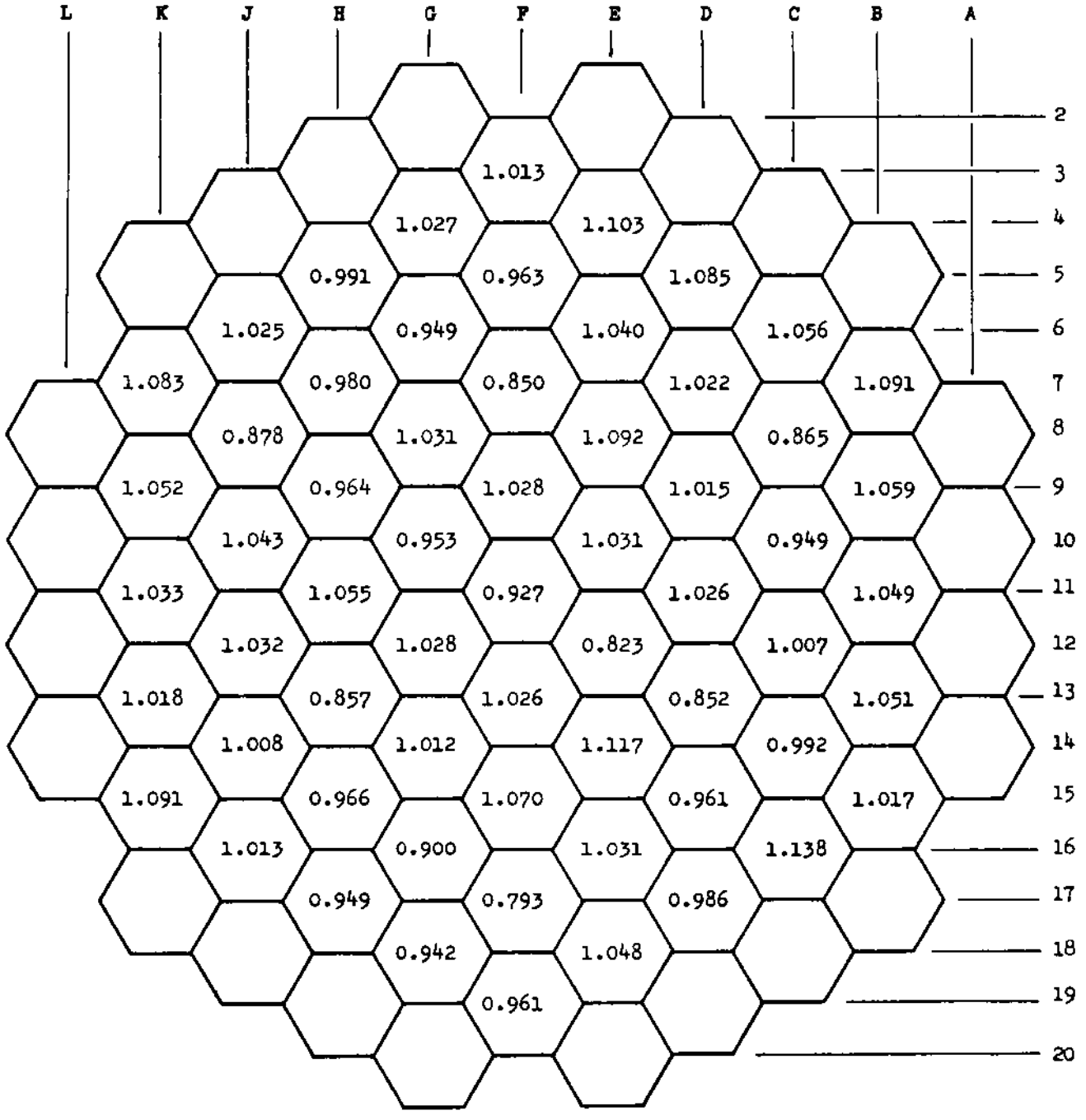

Figure 16. Relative Flow Distribution at Start of Core IB

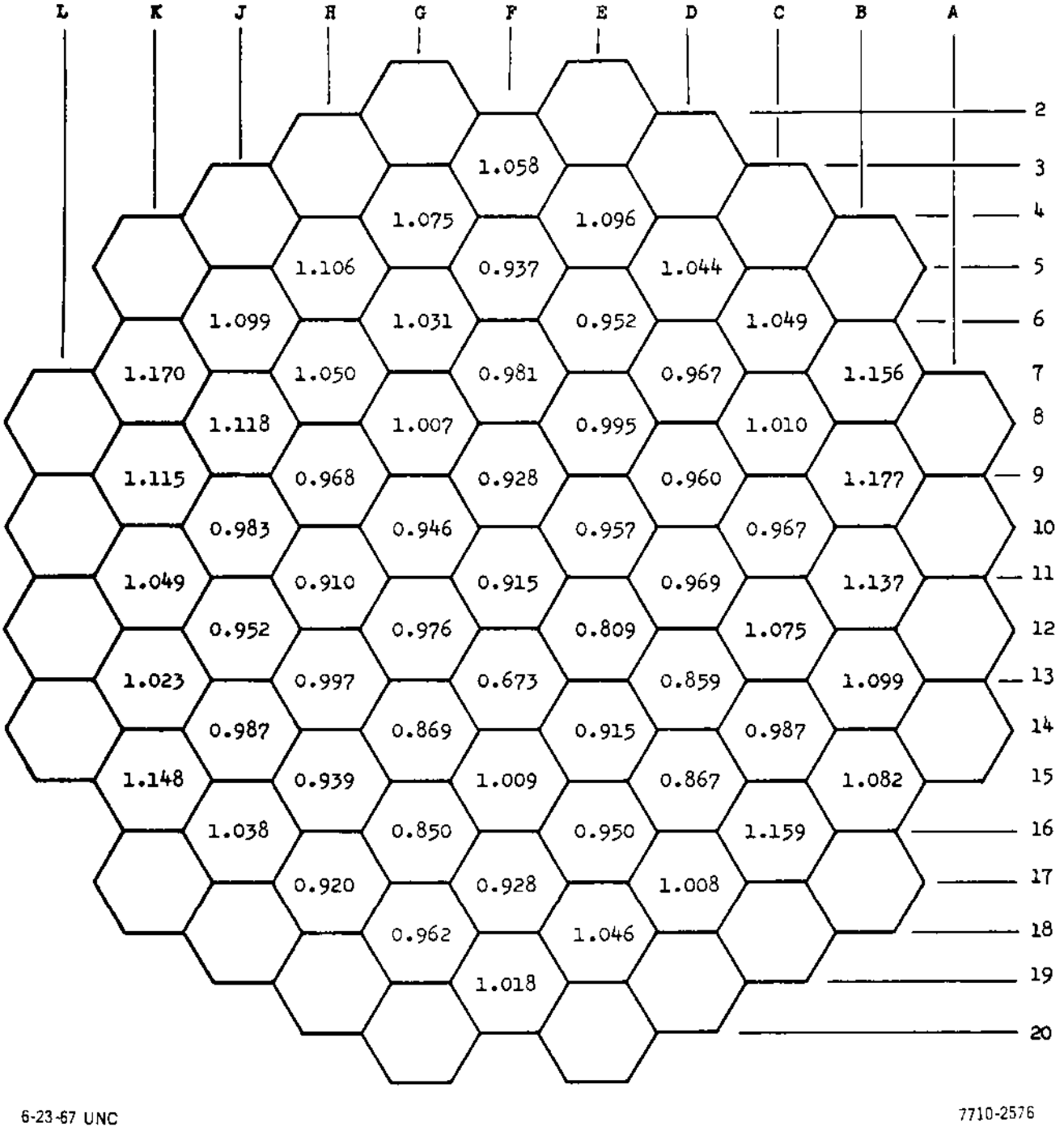

Figure 17. Relative Flow Distribution at End of Core IB

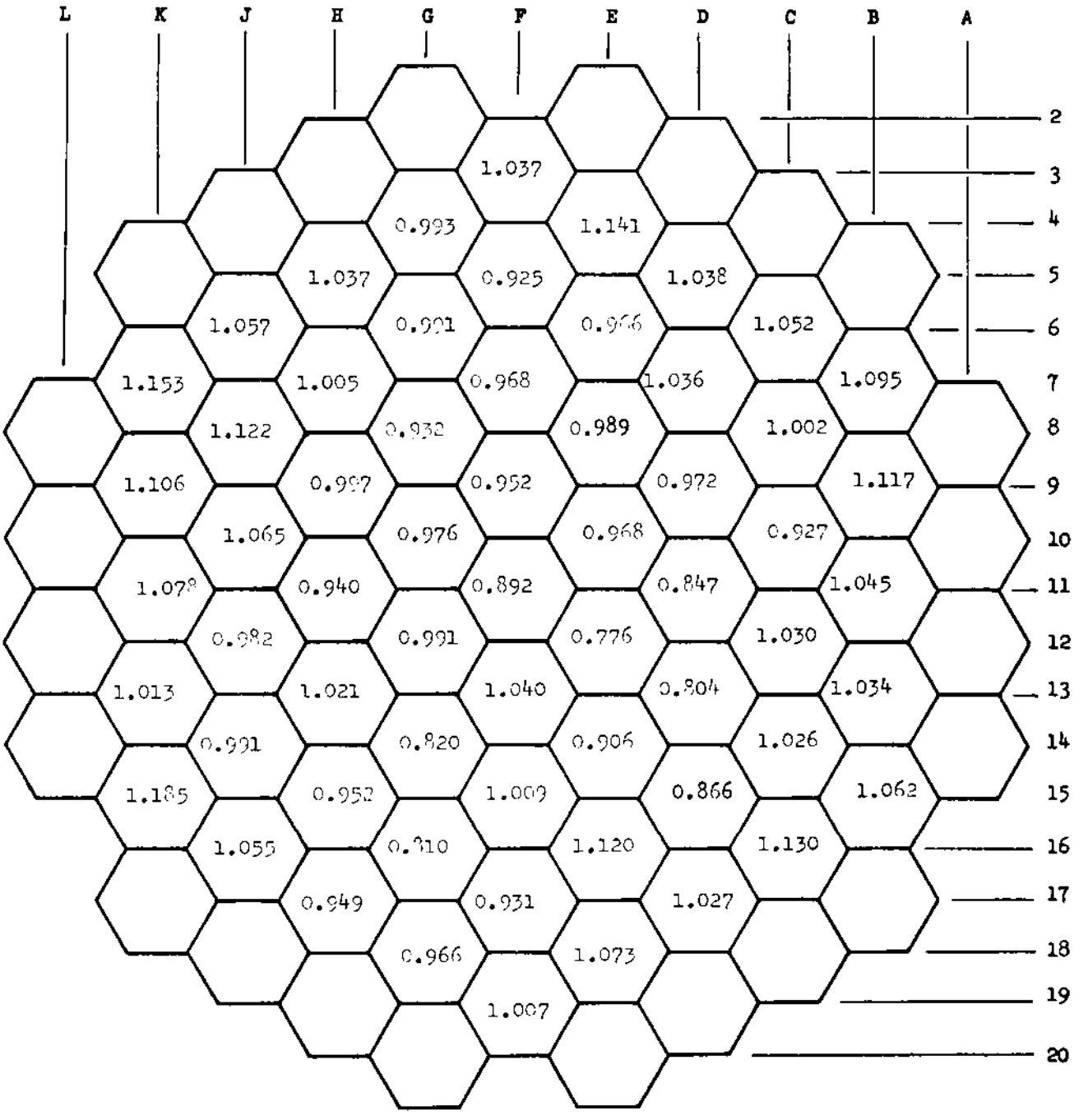

Figure 18. Relative Flow Distribution at Start of Core IC

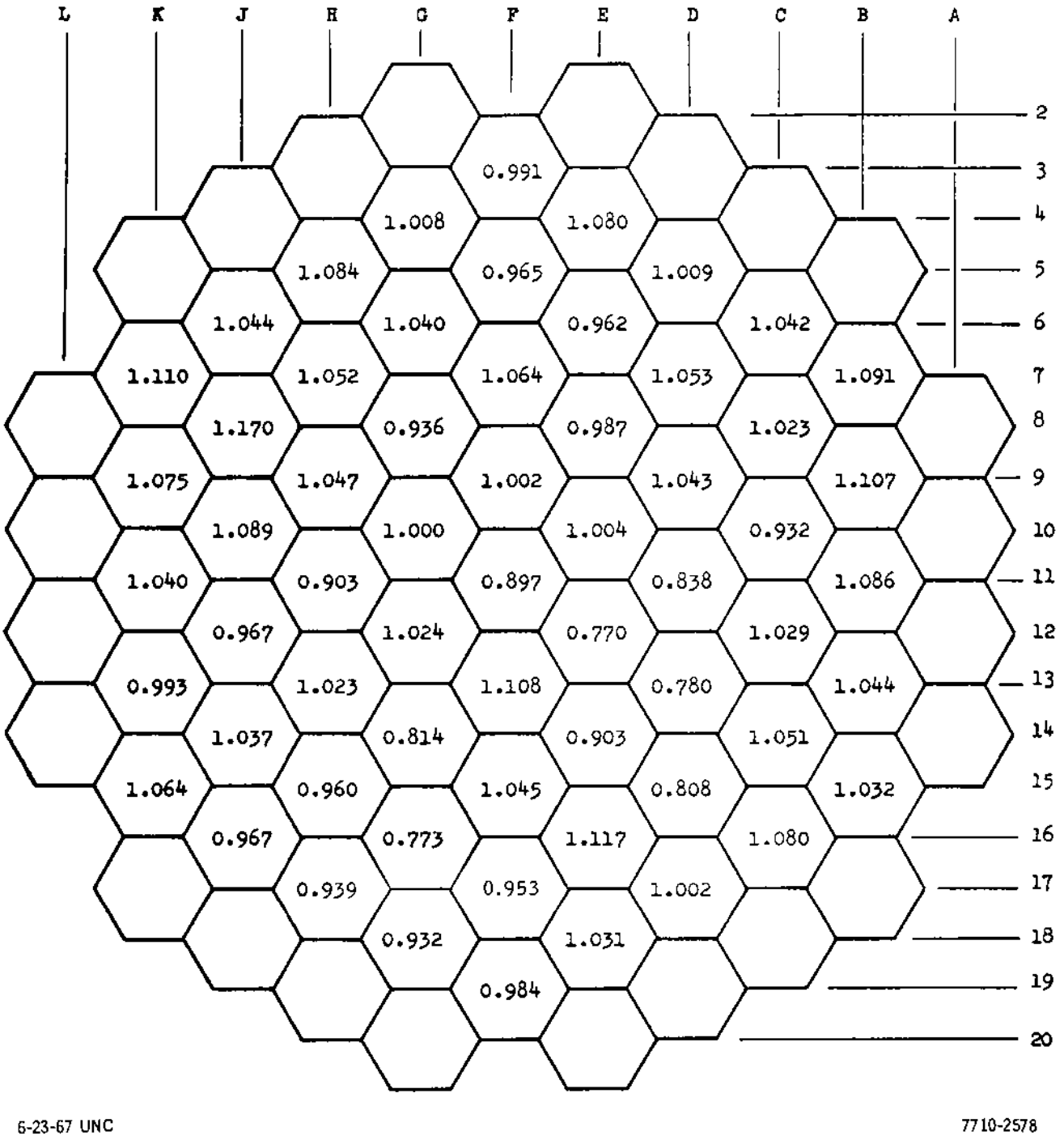

Figure 19. Relative Flow Distribution at End of Core IC

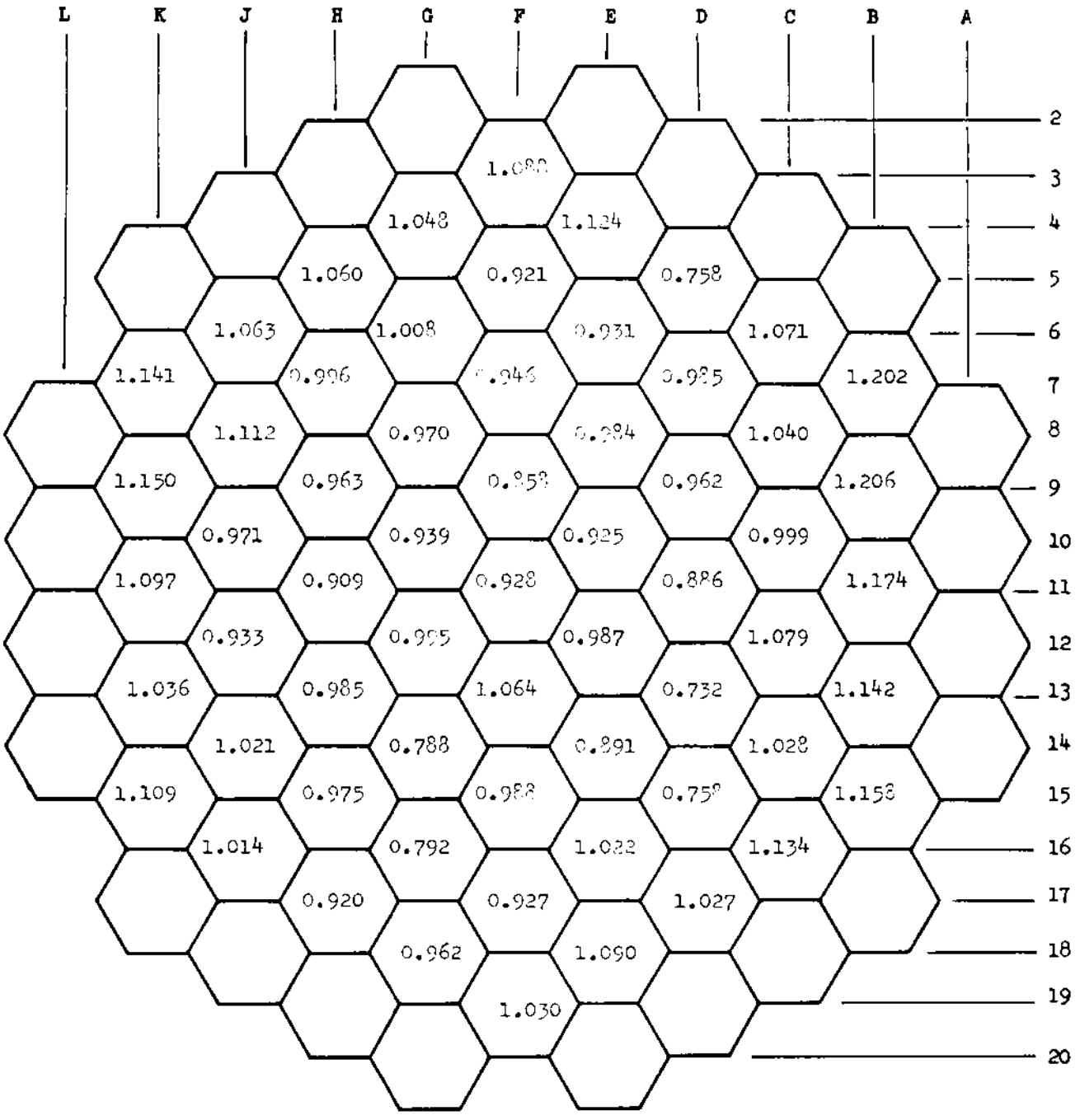

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Figure 20. Relative Flow Distribution at Start of Core ID

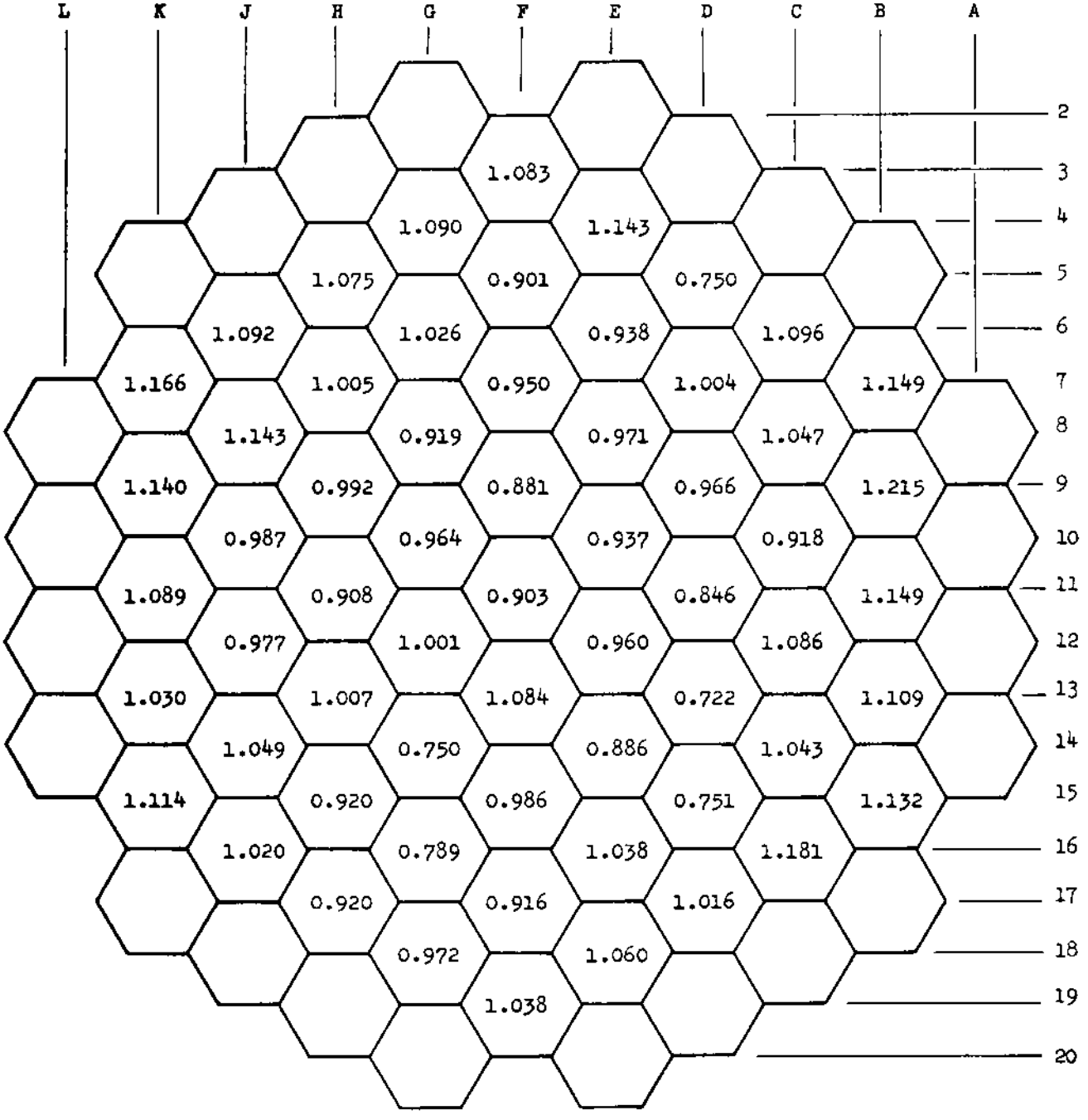


Figure 21. Relative Flow Distribution at End of Core ID

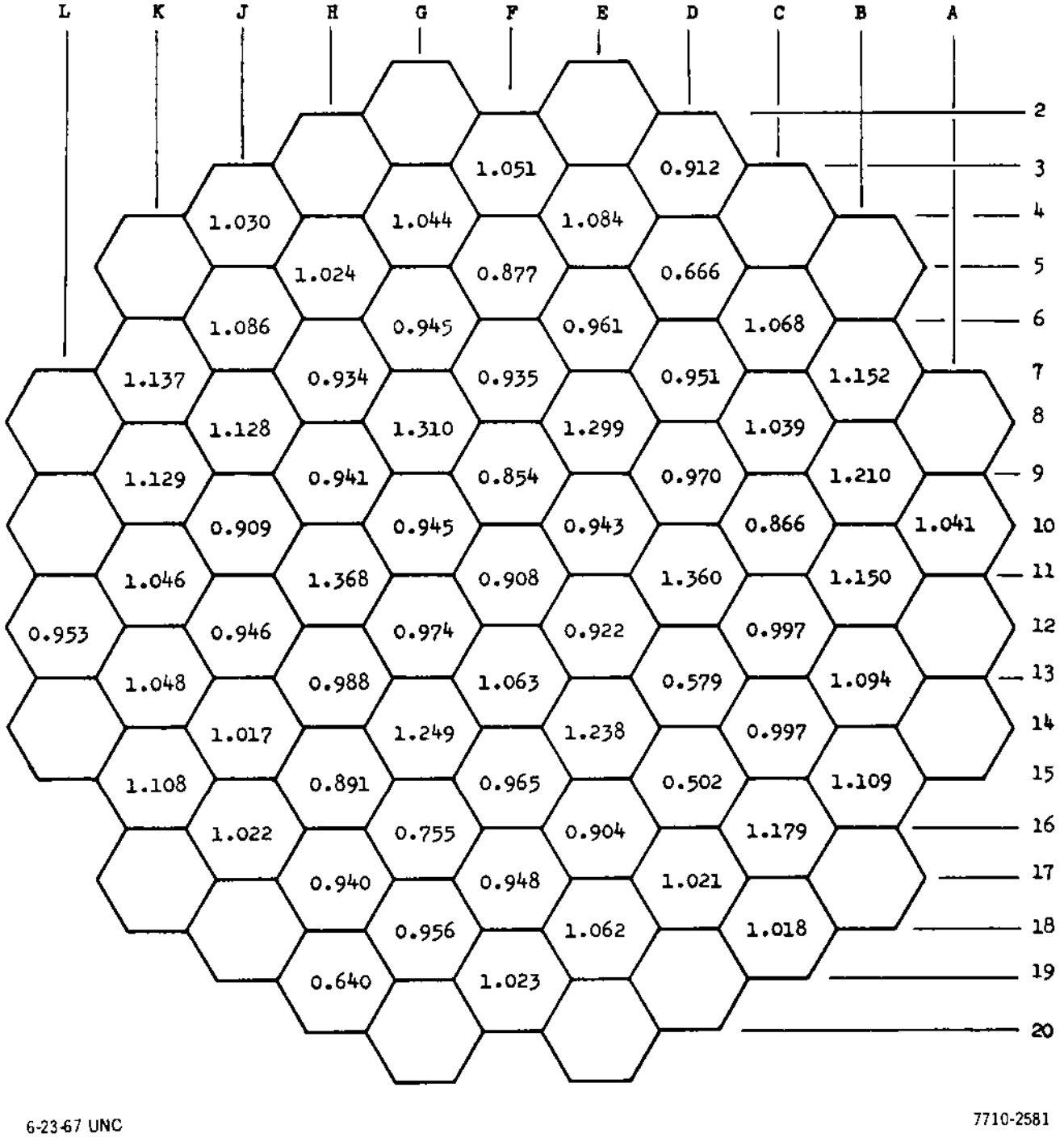


Figure 22. Relative Flow Distribution at End of Core IE

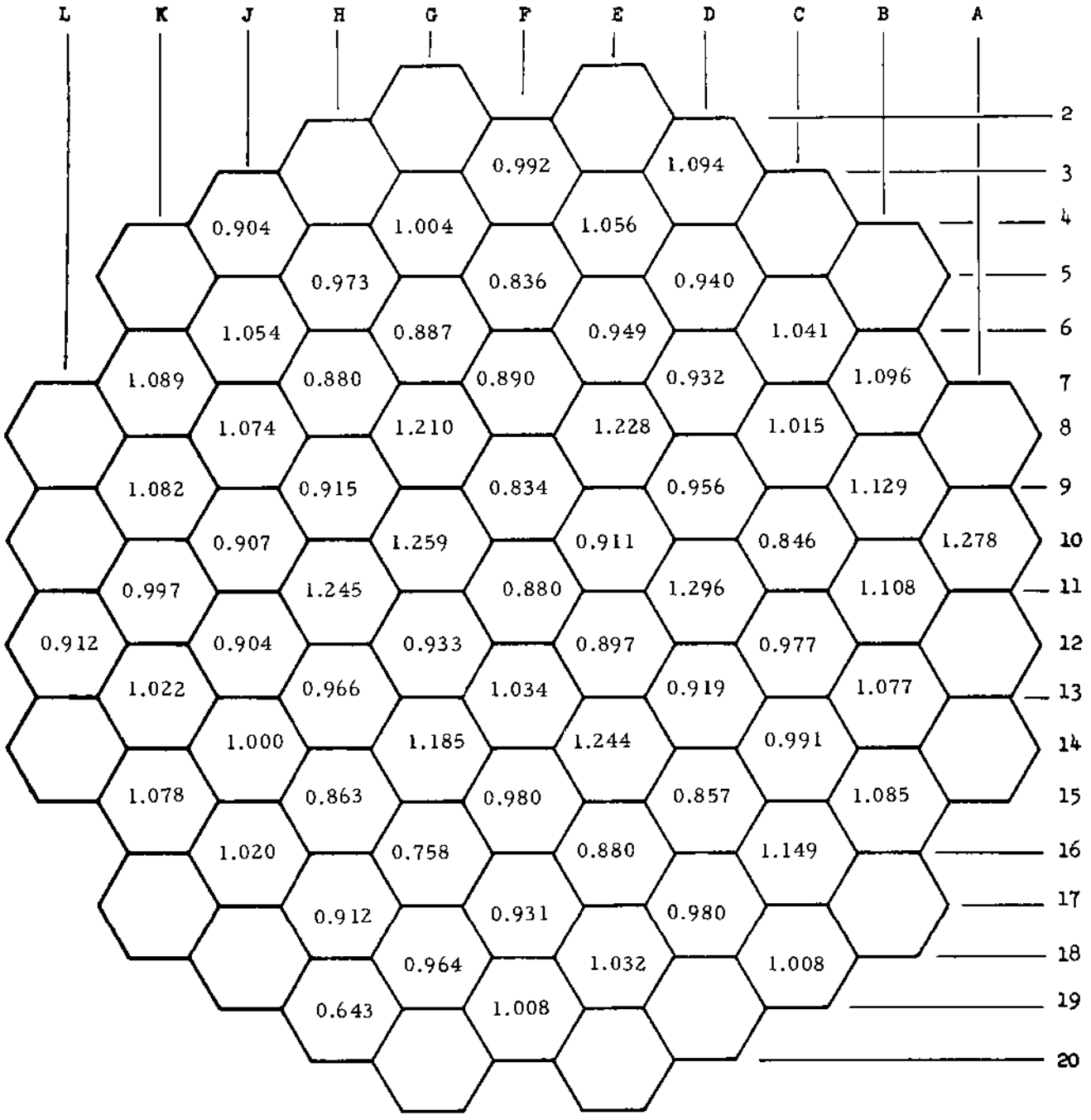


Figure 23. Relative Flow Distribution at Start of Core IF

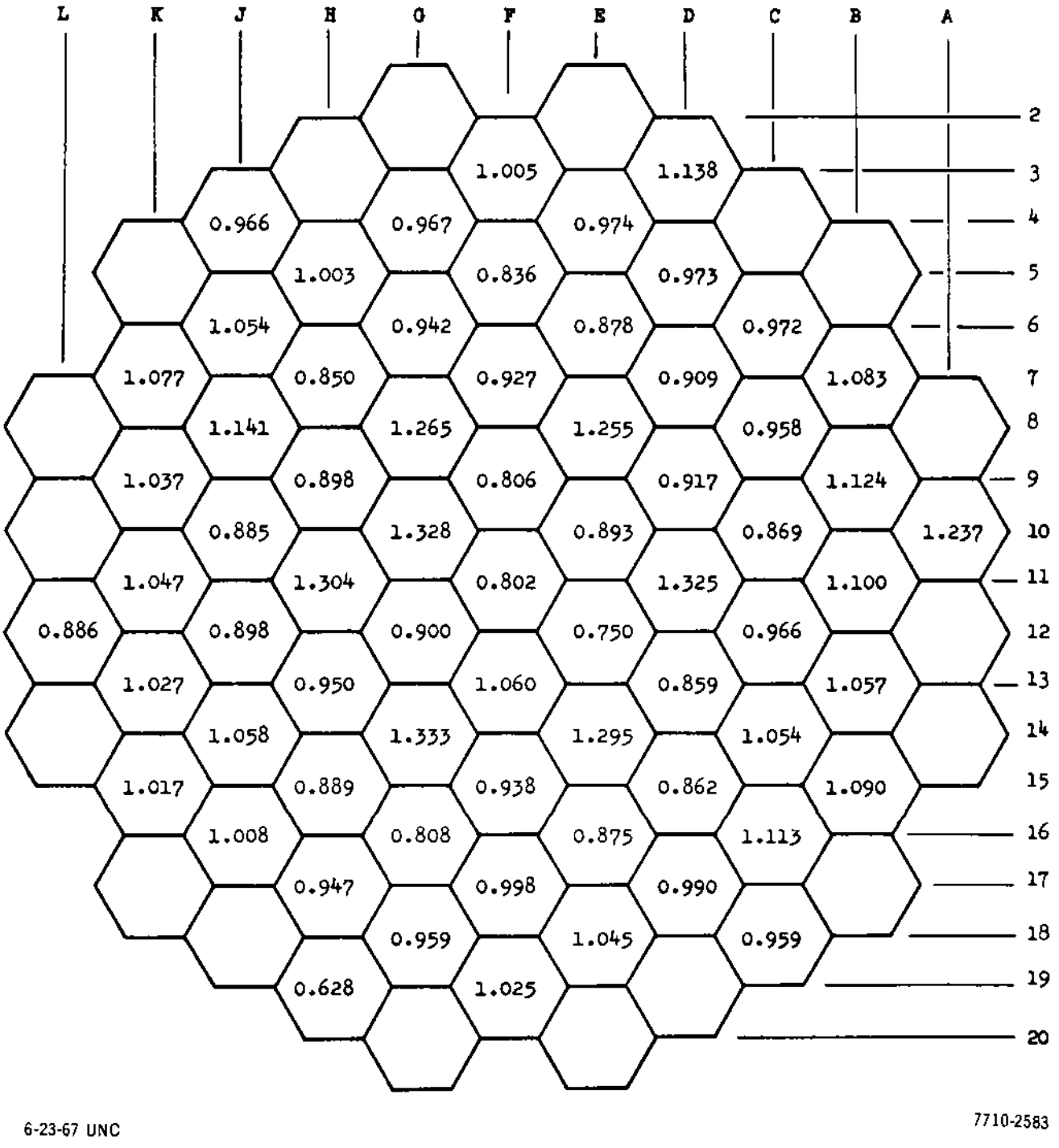


Figure 24. Relative Flow Distribution at End of Core IF

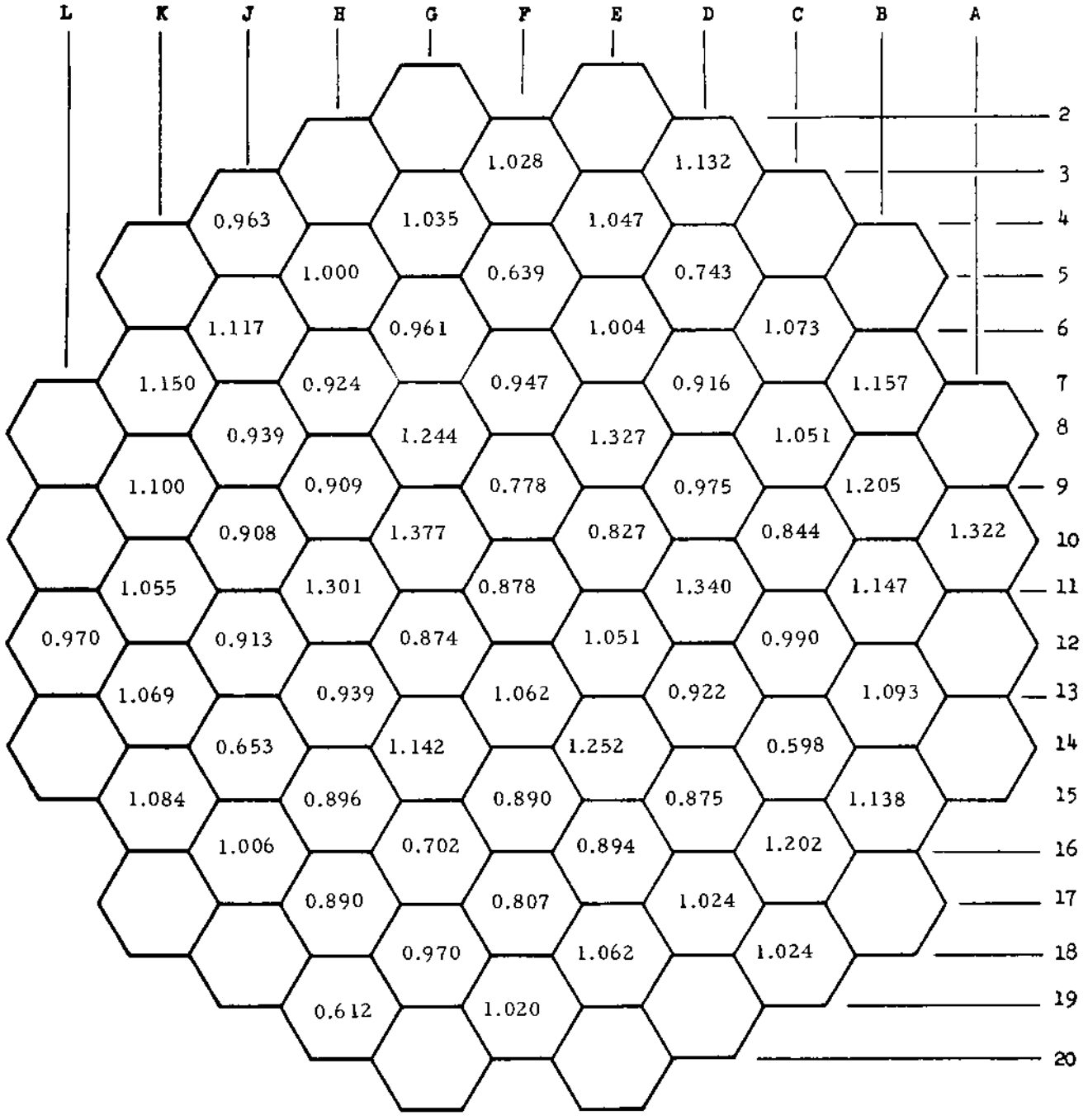


Figure 25. Relative Flow Distribution at Start of Core IG

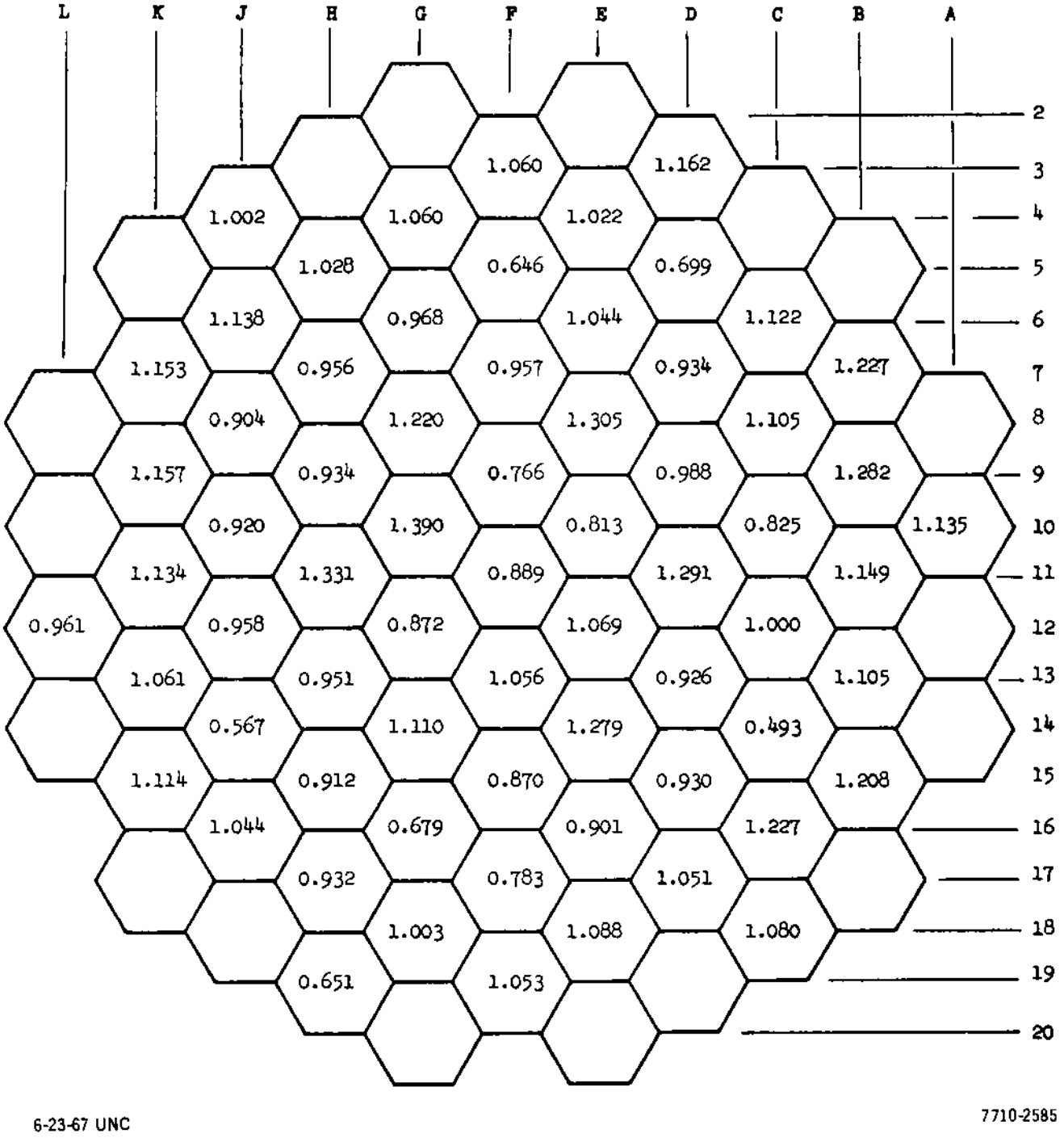


Figure 26. Relative Flow Distribution at End of Core IG

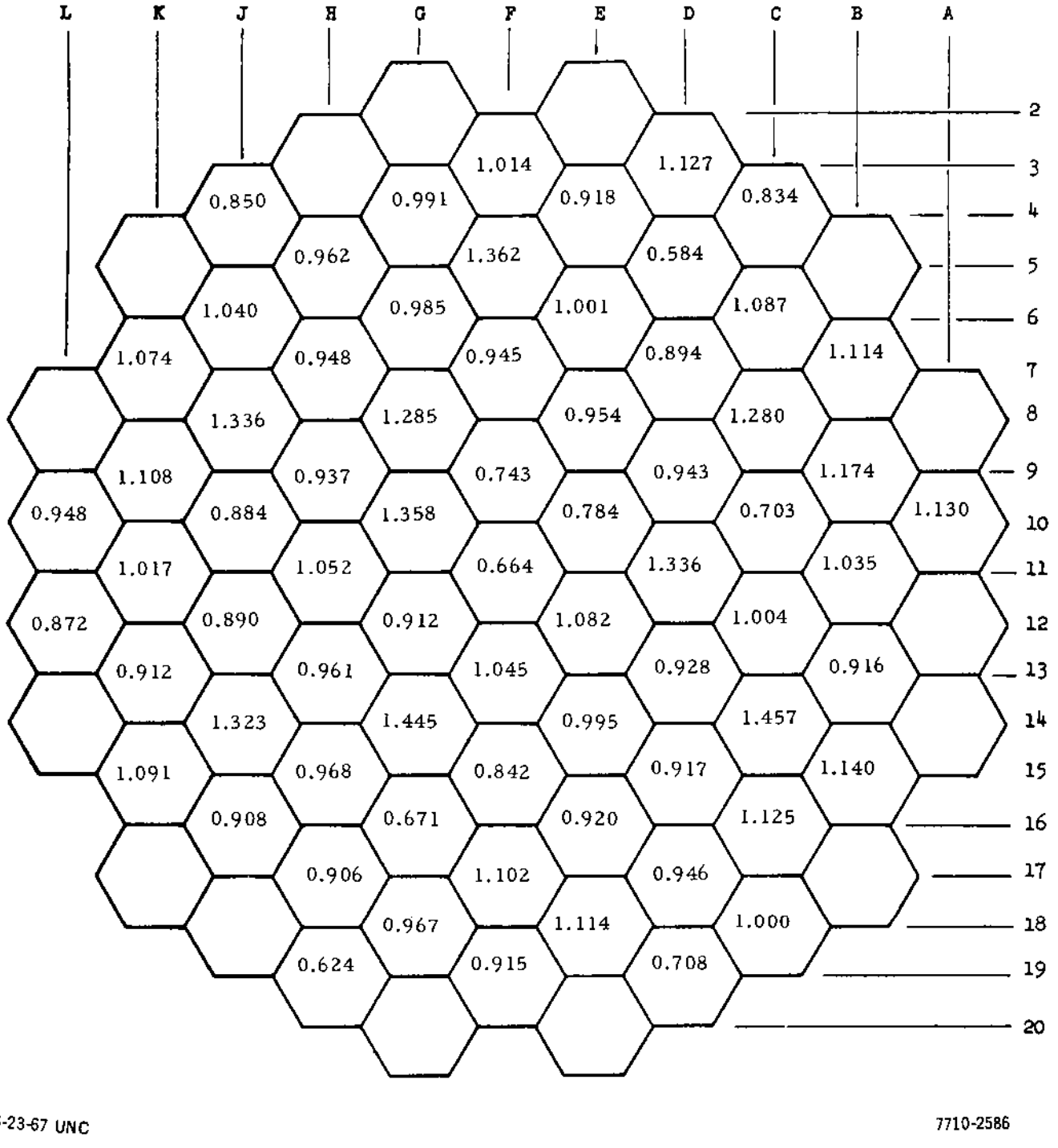


Figure 27. Relative Flow Distribution at Start of Core IH Tilted

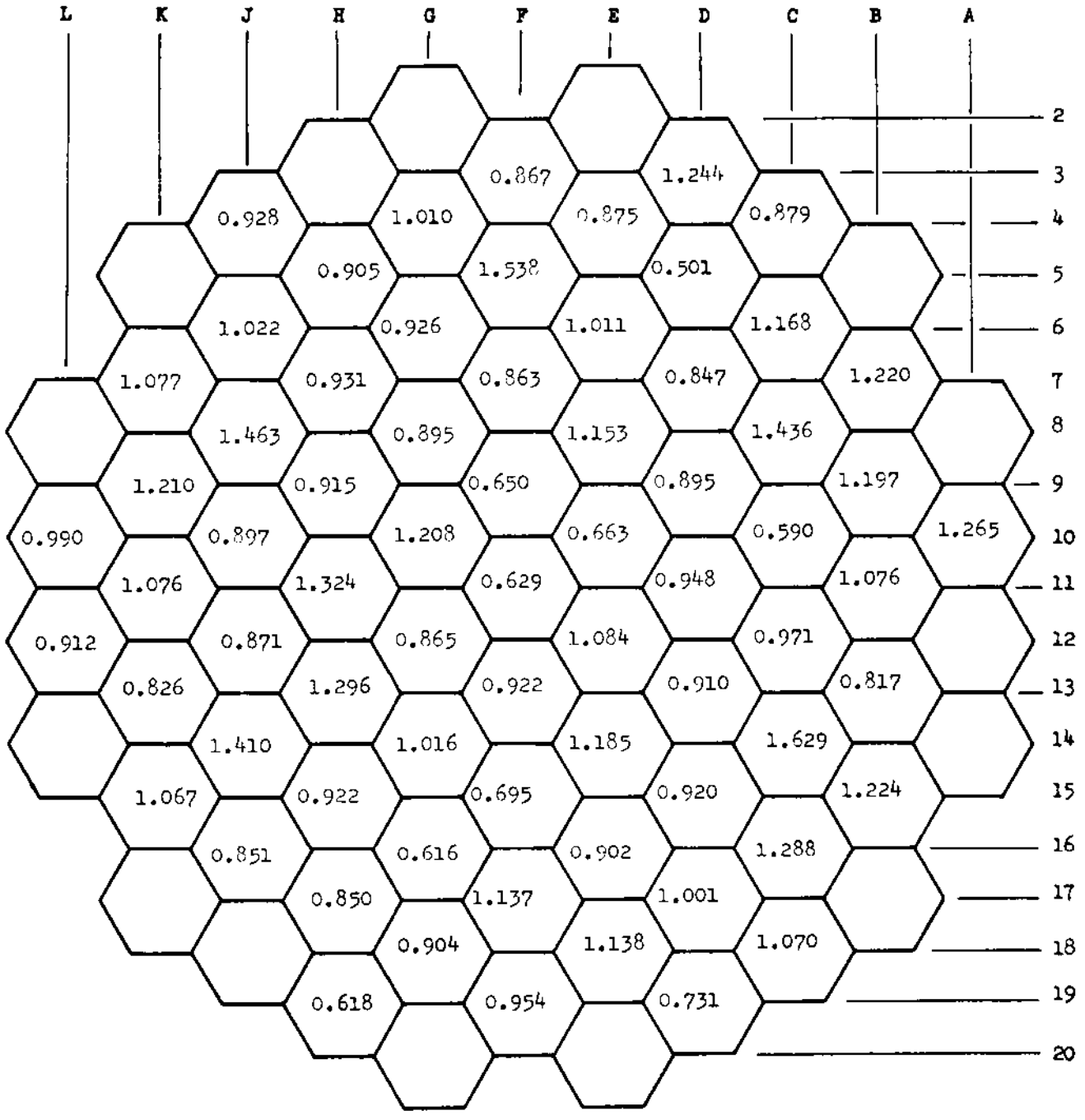


Figure 28. Relative Flow Distribution at End of Core IH Tilted

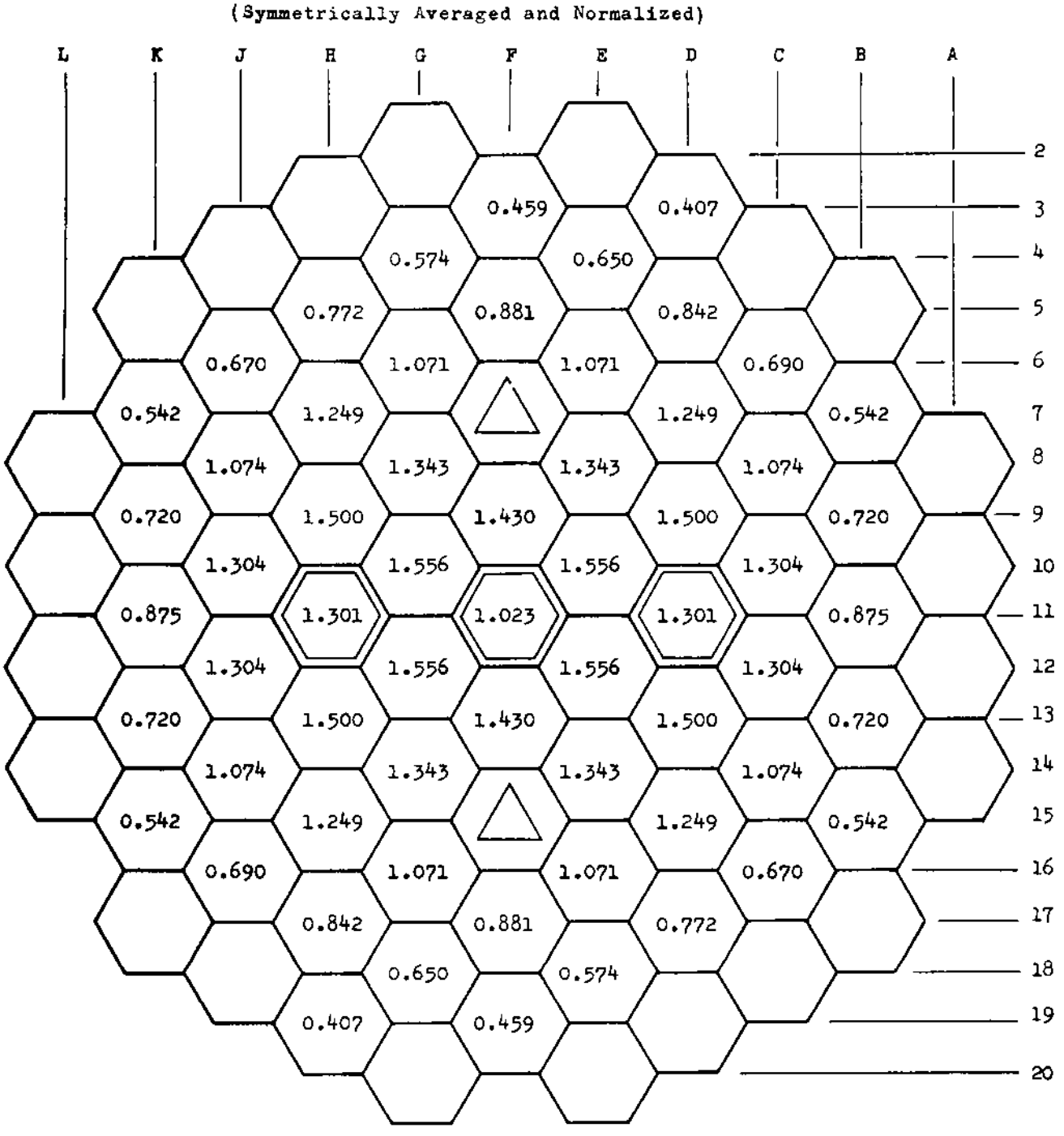


Figure 29. Relative Power Distribution of Core IA

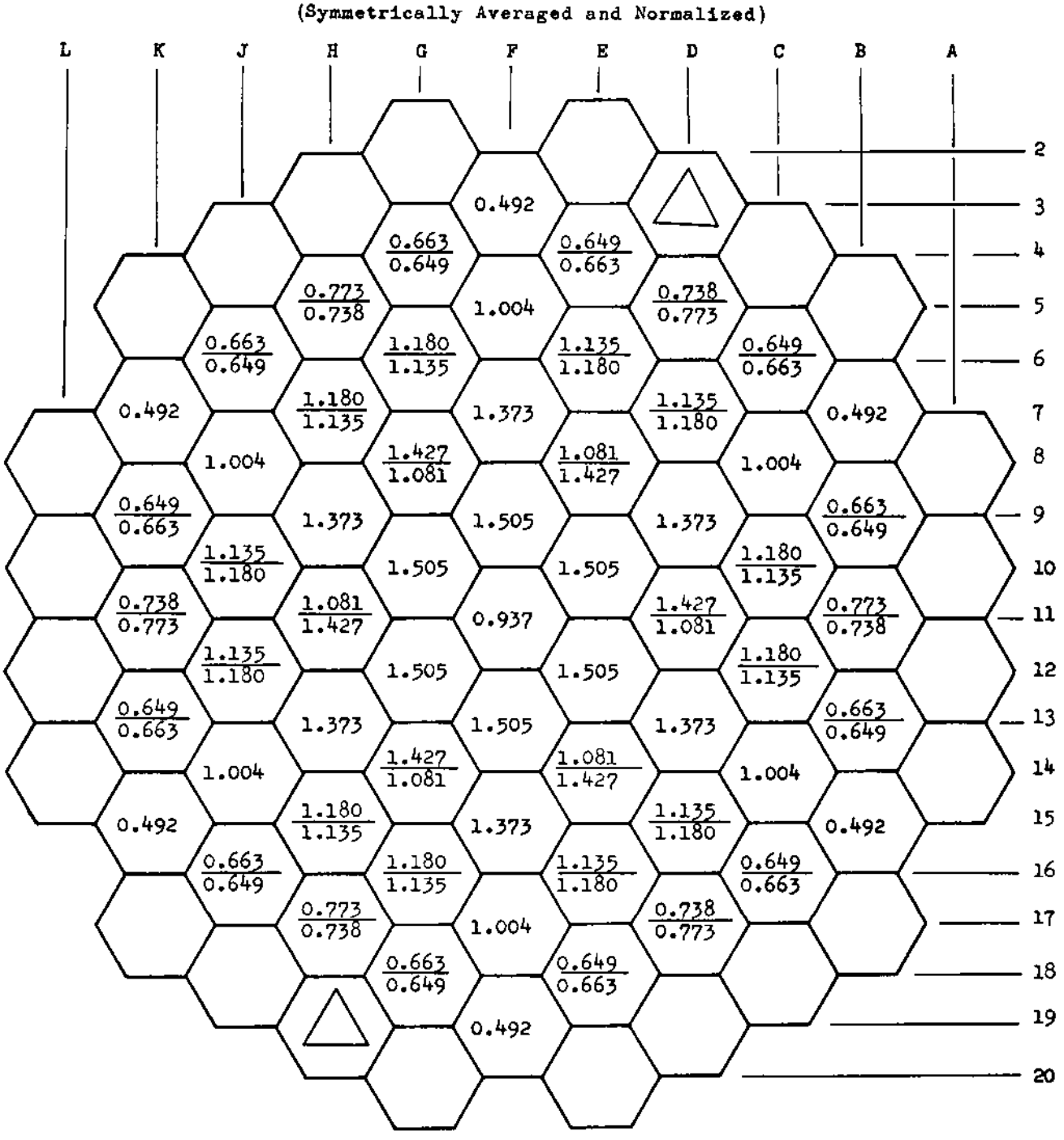


Figure 30. Relative Power Distribution of Cores IB, IC, and ID

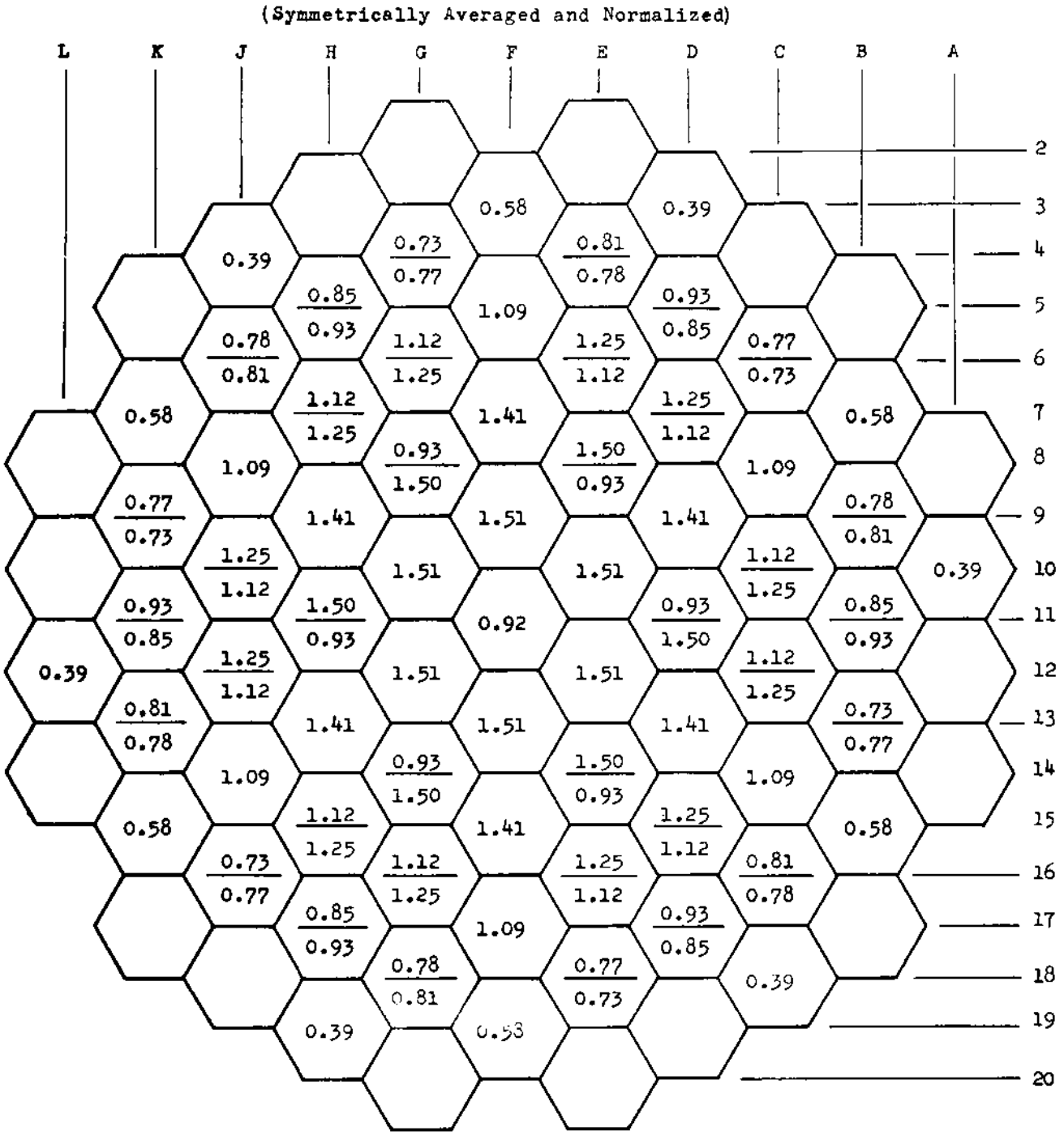
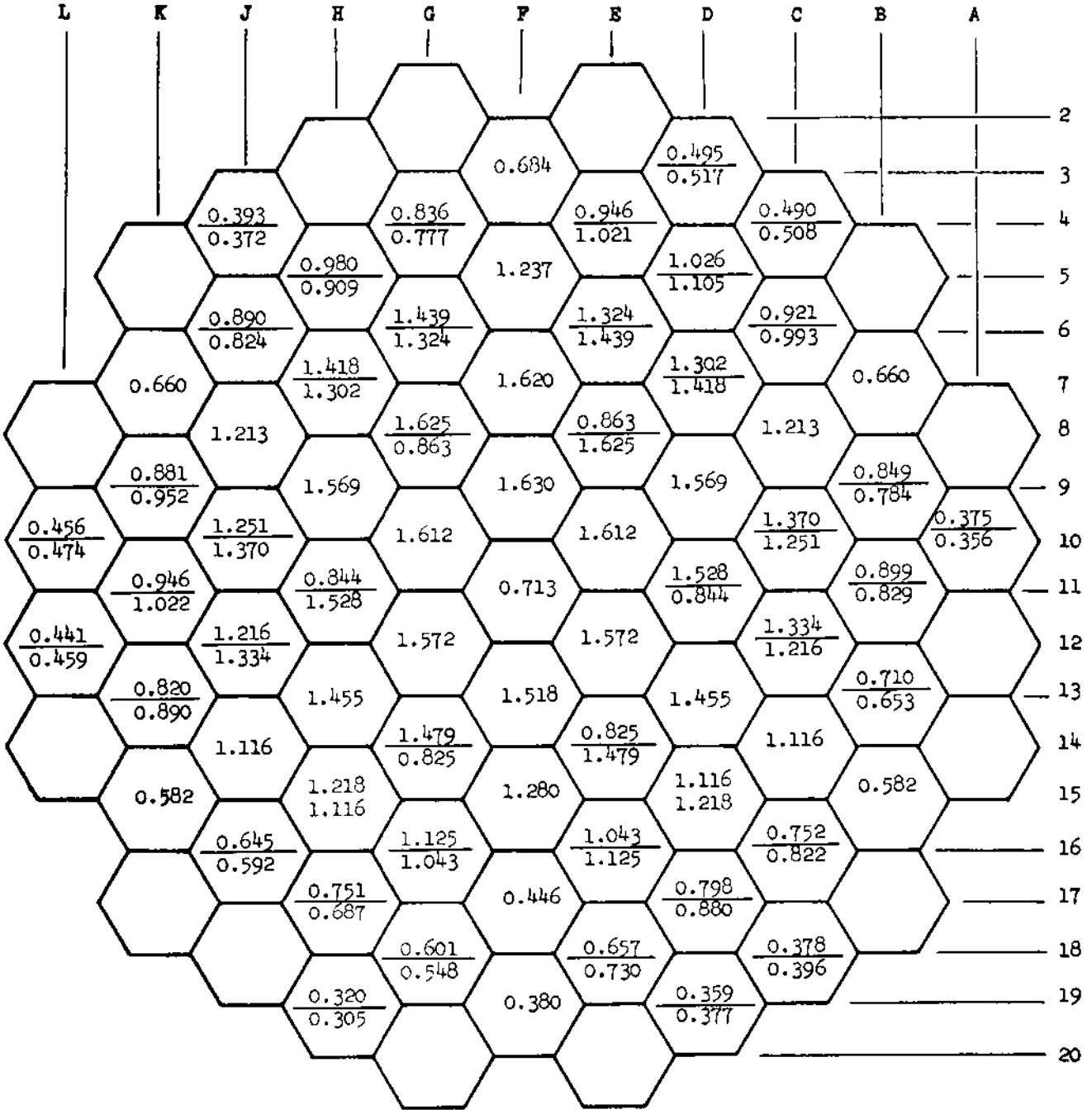


Figure 31. Relative Power Distribution of Cores IE, IF, and IG



Relative Powers
EVEN Gang Controlling
Relative Powers
ODD Gang Controlling

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Figure 32. Relative Power Distribution of Core I-H, Tilted

a. Preoperational Tests

1) Fuel Handling System

2) Nuclear Instrumentation System

3) Plant Protective System

4) Coolant Chemistry

5) Superheater Tube Bundle

6) Control Rod Drive Assemblies.

b. Initial Nuclear Tests

1) Critical Loading

2) Excess Reactivity Loading

3) Isothermal Temperature Coefficient Measurement

4) Flow Coefficient of Reactivity and Core Pressure Drop Measurement

5) Control Rod Calibrations

6) Main Heat Transfer System Temperature Instrumentation Calibration.

c. Postcritical Operational Tests

7) Rise-to-Power Test

8) Steady State Power Coefficient

9) Transient Power Coefficients

10) Xenon Test

11) Pressure Coefficient

12) Core Thermal Hydraulic Tests

Test procedures to cover these tests have been worked on and discussed during the report period. Operations Analysis is responsible for Tests 7 (Rise-to-Power), 8 (Steady State Power Coefficient), 9 (Transient Power Coefficients), and 12 (Core Thermal-Hydraulic). The first drafts of these procedures have been written and are described in brief in the following sections.

2. Test Procedure No. 7 – Rise-to-Power Test

The primary purposes of the test are as follows.

1) Coordinate the following tests to be conducted during the rise to power: power coefficient (steady state and transient), xenon transient, pressure coefficient, and thermal-hydraulic tests.

2) Determine the initial system steady state characteristics (flows, pressures, and temperatures) at different power levels up to "limited" full power.

3) Determine the power to which the reactor will be limited by the requirement not to exceed the maximum fuel cladding temperature of 750°F.

4) Operationally calibrate the nuclear instrumentation.

5) Calibrate the element outlet thermocouples and obtain at-power data to be used as a basis for future comparisons of thermocouple operability.

The reactor outlet temperature will be maintained at approximately 570°F throughout the test. At each power level, a complete system thermal analysis will be performed to establish system operating characteristics, and to insure safe operation of the plant at the next power level. The temperature information will be used to calculate hot-channel values of cladding temperatures which will, in turn, determine the "limit" on reactor power (approximately 30 Mwt with 2-pump flow). In addition to the thermal analysis, a number of tests will be performed at each power level, e.g., pressure coefficient, steady state and transient power coefficients, and a radiation survey. Other measurements, notably flow and xenon transient test information, will be taken only at higher power levels. All the tests will be performed with two-pump flow. When the series has been completed, the power coefficient tests will be performed with one-pump flow up to the "limiting" power level (approximately 20 Mwt).

3. Test Procedure No. 8 – Steady State Power Coefficient

The purposes of this test are to determine the magnitude of the steady state power coefficient and the effects of one-pump operational flow on the magnitude of the coefficient.

The test will start at a reactor power level to 5 Mwt and 2-pump flow. The power will be increased by increasing the organic coolant flow through the boiler. This produces a drop in inlet temperature, an addition of reactivity to the system (through the isothermal temperature coefficient), and the resultant power increase. If equilibrium power levels before and after a transient of a few minutes duration are considered, the reactivity change between the two cases, after correcting for inlet temperature variation, control rod movement, and xenon effects, must be attributed to the steady state power coefficient.

For steady state conditions (Δ_{ex} equal to zero) the power coefficient can be defined in terms of other variables and associated coefficients, a change in any of which would result in a change in excess reactivity,

$$-\alpha_p = \frac{\alpha_R \Delta R + \alpha_T \Delta T_{ave} + \Delta \rho_{xe}}{P}$$

where

ΔR = change in control rod position,

ΔT_{ave} = change in average core temperature, and the coefficients are determined by

1) $\Delta \rho_{xe}$ from the xenon transient test,

2) α_R from the differential rod worth curve, and

3) α_T from the isothermal temperature coefficient measurement.

At least three sets of power changes from 5 to 10 Mwt, 10 to 20 Mwt, and 20 to 30 Mwt will be used to measure the coefficient. The tests will be performed in both the one- and two-pump operational modes.

4. Test Procedure No. 9 – Transient Power Coefficient Test

It is possible to separate the temperature reactivity effects of the power coefficient components by investigating reactor transient performance. The investigation consists of two steps: (1) the transient response of reactor power to a specified reactivity input is determined, and (2) the coefficients of an analog computer simulation of core parameters are adjusted to obtain the same performance. The effects of individual components can be inferred from the fitted coefficients of the analog model.

The two major parts of the computer program are concerned with the nuclear kinetics and heat transfer phenomena occurring in the core. To investigate adequately the magnitude of the temperature coefficient and temperature changer per unit of power change, three different control rod inputs are used. The flux is recorded on a visicorder oscillograph with the intention of later simulating the flux output with the analog computer program.

As the nuclear effects are assumed to be wellknown, the flux is approximated by manipulation of the heat transfer portion of the program. The transient responses of the fuel, coolant, and moderator regions and the coupling between them can be found to a good approximation by a theoretical calculation of the quantities affecting the heat flow in the reactor. The determination of the solutions is enhanced by first finding the theoretical time constants of the four regions (fuel, coolant, inner and outer moderator) and the coupling betwen them. With these, the temperature responses can be solved by using determinants. Then remains the problem of simulating the flux responses to the three reactivity (control rod) inputs by means of the analog computer. The control rod ramp changes will be made at 10 and 20 Mwt.

5. Test Procedure No. 12 – Core Thermal-Hydraulic Tests

The objectives of the test are (1) to observe the changes in overall core flow characteristics as a result of fouling, (2) to determine the effect of flow rate on the fuel element heat transfer coefficients, and (3) to measure a base curve to be used in observing changes in the Instrumented Fuel Element (IFE) friction factor. At 20 Mwt and two-pump flow, the core flow rate will be reduced in three 10% increments by throttling main loop flow. The core pressure drop, flow and ΔT; element outlet temperatures; and all IFE temperatures will be recorded at each level of reduced flow.

The first objective is attained through an accurate determination of the core ΔP as a function of core flow rate. These data are related by the equation, $\Delta P \sim fv^2$, where $f \sim N_{Re}^{-0.2}$ for a clean core in the region of interest for the Reynolds number, N_{Re}. As roughness develops, the dependence of f on N_{Re} reduces, and the ΔP function approaches a v^2 relationship. A log-log plot of ΔP vs flow for different core exposures should demonstrate the effects of fouling on core performance through increased roughness and a change in slope.

The periodic performance of reduced flow testing will permit the determination of changes in individual fuel element hydraulic characteristics (i.e., surface roughness and channel blockage). The theoretical analysis has been given,* with only the final equation included here,

$$\log\left(\frac{\Delta T_c}{\Delta T_i}\right) = (n-1)\log W_c + \log K \quad ,$$

where

ΔT_c = core temperature differential,

ΔT_i = individual element, i, temperature differential,

W_c = core flow rate,

K = constant, and

n = a number which characterizes the deviation from smooth tube conditions.

The experimental data will be plotted and the slope determined; in this way n will be evaluated. Though the change in heat transfer coefficient is not determined explicitly, the daily analysis will be amplified by the test results indicating which elements have experienced gross changes.

The analysis involved with objective (3) above is also presented in detail.† Through utilization of the Wilson plot technique,§ the following information on the condition of the IFE is gained.

1) An indication of the degree of surface roughness caused by film formation.

2) A measure of the fouling thickness in a fuel channel.

3) A value of the heat transfer coefficient h.

*M. Huntsinger and R. S. Hart, "Safety Evaluation of PNPF Modifications," NAA-SR-MEMO-12103 (December 27, 1966)
†Op. cit., NAA-SR-MEMO-12103
§Wilson, Trans. American Society of Mechanical Engineers, Vol. 37, p 47 (1915)

III. COOLANT CHEMISTRY AND ANALYSIS

During this reporting period sampling of the core obstruction (outer moderator deposit) was completed, the deposit was removed from the core, and core cleanup operations were begun. Support was provided by the Coolant Chemistry and Analysis Project in characterizing the moderator deposit samples by physical and chemical means, and in determining the particle size distribution of deposit fragments in the coolant during the system flushing operations in November as an aid in assessing the effectiveness of system cleanup.

A. COOLANT ANALYSIS SUMMARY

A summary of the coolant quality data obtained during the reporting period and the general status of the reactor system is shown in Figure 33.

1. July

The reactor vessel remained open during most of July to permit sampling of the core obstruction and other work to be carried out. Coolant ash and MST values remained less than 2 and 36, respectively; during the month, however, a slight increasing trend was apparent among the MST values from July 1st to 21st which may be related to slight coolant-air contact. Because of an instrument malfunction, no carbonly-bound oxygen values were obtained.

Core obstruction sampling was completed on the 19th, following which the coolant in the reactor vessel, main heat transfer system, and pressurizing loop was replaced with "HB-40" fluid (Monsanto Chemical Company tradename for hydrogenated terphenyl mixture), in preparation for cooling the system to ambient. The coolant which was replaced by HB-40 was processed through the purification still following its startup after being out of service since June 24 for replacement of the H-4 bottoms heater.

2. August

The reactor vessel remained filled with HB-40 during the first part of the month. No analyses were performed. Because of the presence of some Santowax OMP coolant in the HB-40 as a slurry, circulation was established through the pressurizer-degasifier circuit and the system was heated to a temperature (263°F) above the liquidus point of the mixture before draining the reactor vessel

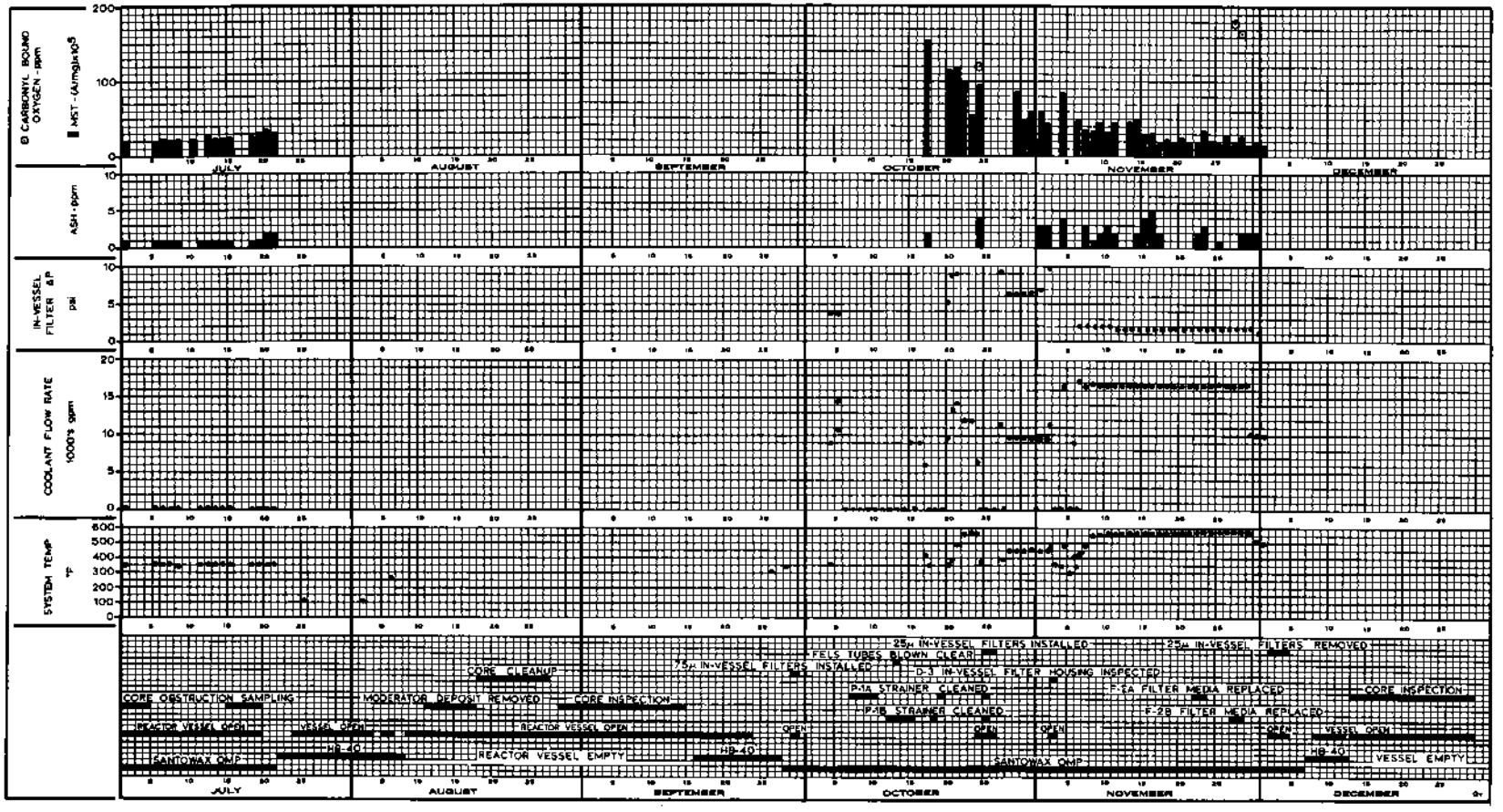

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Figure 33. PNPF Coolant Analysis Summary, July Through December, 1966

on the 8th. Following inspection, core cleanup operations were initiated on the 11th. The Coolant Chemistry and Analysis Project provided basic distillation data which aided in establishing conditions in the PNPF purification still which resulted in separation of the HB-40 and Santowax.

3. September

The reactor vessel remained empty of coolant during the first half of the month, while inspection of the area below the lower core grid plate was conducted. The reactor vessel was filled with HB-40 during the period from the 16th to 18th, and circulation through the pressurizing loop was initiated on the 24th. The HB-40 was heated to 332°F, drained from the reactor vessel on the 28th, and was replaced by 340°F Santowax OMP coolant. No coolant analyses were carried out during the month.

4. October

Organic coolant was circulated through the main heat transfer system and through 75-micron nominal pore size full flow filters as part of the system cleanup operation. The decreasing trend apparent in the MST values reflects the cleanup achieved in the system.

5. November

System cleanup via coolant circulation through full-flow filters, side-stream (F-2) filters, and the purification still was continued during most of the month. Support was provided in assessing the progress of system cleanup through the determination of the size distribution of moderator deposit fragments (as black particles) in coolant samples taken from the main loop, ahead and downstream of the degasifier filters.

System cleanup was achieved from November 11 through the 19th as indicated by the following trends: (1) decreasing MST values, (2) decreasing numbers of black particles suspended in the coolant, and (3) decreasing radioactivity levels on the particle filters. That particles were still being released to the coolant from parts of the system on the 20th was shown by increasing values of MST, particle counts, and particle filter activity.

Particle size distribution data obtained on coolant samples taken upstream and downstream from the degasifier filters aided in assessing their effectiveness

in achieving cleanup. Data obtained on the 14th, 16th, and 21st suggest that these filters were not performing effectively, and both sets of filter media (F-2A and F-2B) were changed out in the latter part of the month. Several days after the F-2A unit was placed in service (1 to 3 micron nominal pore size glass spool-type filter media), particle size distribution data suggested that the unit was effective in removing particles larger than about 20 microns from the coolant.

6. December

The system flushing operation was terminated early in the month, and the coolant was drained from the main heat transfer system in preparation for inspection of the reactor vessel and components.

B. SEPARATION OF HB-40 AND SANTOWAX OMP

The inadvertent admission of Santowax OMP into HB-40 in the reactor system towards the end of July resulted in a need for information concerning certain properties of such mixtures. The specific questions were as follows.

1) Is the mixture single phase and transparent at 110°F?

2) At what temperature does the mixture become clear and how much does it supercool without precipitation or separation into layers?

3) On distillation, does any solid separate which might plug the condenser, liquid lines, etc.

Some data obtained in the AI laboratories on a synthetically prepared mixture of HB-40 and Santowax OMP are summarized as follows.

1. Sample Preparation

An HB-40 Santowax OMP mixture was prepared by adding 25 gm of Santowax OMP at 360°F to 155 gm of HB-40 held at a temperature of 110°F, thus simulating the conditions of mixing in the reactor. The mixture which resulted contained 14% Santowax OMP by weight, and consisted of an opaque slurry of Santowax crystals in the HB-40.

2. Melting Characteristics

Upon heating the slurry, the last traces of crystals were observed to disappear at 230°F. Upon cooling the liquid, the first traces of crystals appeared at 196°F.

3. Distillation Behavior

In a batch distillation carried out at 26.5 mm pressure, boiling was observed when the liquid temperature reached 352°F. During the essentially single-plate distillation, the overhead temperature varied from 383°F initially to 460°F. At the latter temperature the concentration of Santowax in the distillate coming over was sufficient to crystallize in the previously clear liquid in the receiver. Following termination of the distillation at an overhead temperature of 460°F, it was found that 83% of the initial charge was recovered as distillate. The distillation data are summarized in Table 7.

TABLE 7

BATCH DISTILLATION* OF HB-40 SANTOWAX OMP MIXTURE AT 26.5 mm PRESSURE

Pot Tem. (°F)	Overhead Temp. (°F)	Wt % of Initial Pot Charge in Distillate†
352	(Initial boiling observed)	
432	383	0
442	428	20
450	433	40
464	444	60
489	460	83

*Single plate distillation.
†Initial pot charge consisted of HB-40 containing 14 wt % Santowax OMP

C. MODERATOR DEPOSIT SAMPLING AND ANALYSIS

1. Sampling

The extent of the obstruction in the PNPF core outer moderator regions, as determined from profiling data obtained as the fuel elements were removed, was established to be confined to the vicinity of fuel elements which operated with relative powers greater than 1.0 between November, 1965 and January 13, 1966. The observation suggests that a "flux profile effect" should be found in which more extensive deposition is associated with regions of greater flux.

It is apparent that samples of the outer moderator obstruction fall into two general categories (see Figure 34) which correspond to:

1) The region of closest approach between adjacent fuel elements,

2) The center of the tri-cusp region between three adjacent fuel elements.

Samples in these general categories are thought to represent the extremes of temperature conditions which may have existed in the outer moderator obstruction, since maximum cooling would occur at the process tube wall (Category a); and maximum radiation heating in an immobilized solid would be expected in the interstitial positions (Category b). The AI recommendation for sampling the outer moderator obstruction included samples of both categories from locations covering the variation in flux distribution at midplane as well as vertically at additional elevations.

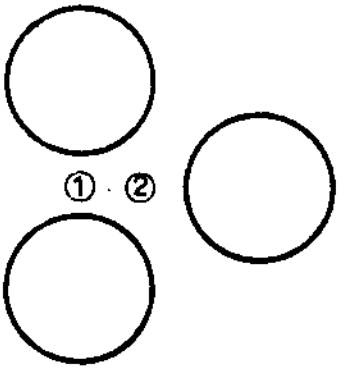

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Figure 34. PNPF Core Obstruction Sample Categories

1- In area of closest approach between two adjacent fuel elements

2- In center of tri-cusp region between three adjacent fuel elements

2. Analysis

The total information yield from each sample depends in part on the size of the sample. Several of the larger samples were analyzed on a layer-by-layer basis in an attempt to establish factors relating to the pattern of deposition. However, in only one case was it possible to establish the orientation of the sample with respect to the outer process tube. Analyses basic to the

characterization of the moderator deposit include the determination of density, crystal parameters, specific activities of beta- and gamma-emitting radionuclides, elemental analysis for carbon, hydrogen, and oxygen, inorganic constituents as ash remaining upon combustion, and an identification of the inorganic constituents by means of atomic emission spectroscopy.

The concentrations of inorganic impurities and oxygen in the moderator obstruction are related to the nature of the surface active species which initially underwent deposition. The extent of decomposition of the deposited material, as assessed by the carbon-to-hydrogen atom ratio, may be a function of both the total radiation dosage received by it and any contribution of pyrolysis due to local heating. A "higher-than-normal" C/H ratio would indicate a probable pyrolytic contribution to the overall decomposition of the sample.

The production of graphitic structure in carbonaceous deposits is achieved at high temperatures. The detection by x-ray diffraction of the characteristic graphite crystal pattern in the moderator deposits would certainly be indicative that very high temperatures were attained during reactor operation.

The density of the moderator deposit, its hydrogen content, and rate of change after deposition are factors which are related to reactivity. Any change in density may also be related to the extent of decomposition of the deposit and consequently to its molecular structure. For example, decomposition of the species deposited in-core involves hydrogen atom loss which can lead to cross-linking within the deposit and to a correspondingly increased density.

The identities and specific activities of the radioactive isotopes contained in the core obstruction are important in assessing health and safety aspects of handling, shipping, and disposal of this material. The specific activities of the isotopes will enable a relative comparison to be made among the samples which may be helpful in revealing "concentration of impurities" effects.

3. Results to Date

The results which have been obtained to date in the analyses of the outer moderator deposit samples are shown in Tables 8 and 9. Upon visual inspection of the samples following the benzene wash (to remove coolant), it was found that several samples possessed both black and whitish, or shiny parts. Where possible, these were analyzed separately. Several samples were also of sufficient

TABLE 8

SOME PROPERTIES OF PNPF OUTER MODERATOR DEPOSITS

Core Location	Orientation, Degrees From North	(Mean) Elevation, Inches From Top of Upper Core Grid	% Wt Loss After Benzene Wash	Density (gm/cc)	Total C + H (wt %)	Total Inorganic as Ash (wt %)	C/H Atom Ratio	Total Oxygen (wt %)	C/O Atom Ratio	Gamma-Emitting Nuclides – dpm/gm						Date of Gamma Activities	Tritium Content (pc/gm)	XRD Pattern [a]
										Ag^{110m}	Co^{60}	Fe^{59}	Mn^{54}	Sb^{124}	Zn^{65}			
C-8	210	44	40.8	1.37-1.44	98.7	< 0.1	2.8	0.82	156	2.5×10^5	1.3×10^6	4.3×10^6	2.3×10^6	-	2.1×10^6	7-12-66	-	A
C-10	-	-	-	-	98.2	< 0.1	2.6	-	-	-	-	-	-	-	-	-	3.9	-
C-10	270	50	24.1	1.44-1.51	94.9	0.17	3.2	-	-	0.71	1.4	-	2.3	-	0.83	7-12-66	-	A
C-12	270	56	-	1.33-1.37	100.0	0.14-0.18	2.3	1.0	128	7.8	1.9	3.5	3.4	3.7×10^5	2.5	2-17-66	-	A
C-14	300	50-I [b]	32.4	1.44-1.51	98.4	< 0.1	2.4	-	-	2.2	1.5	-	3.8	-	1.8	7-12-66	-	A
C-14	300	50-II		1.44-1.51	98.3	< 0.1	2.8	-	-	2.2	1.2	-	2.3	-	1.2	"	-	A
C-14	300	50-III		1.44-1.51	98.2	< 0.1	2.9	-	-	1.7	1.2	-	2.3	-	0.8	"	-	A
D-9	90	36	47.7	I [c]	97.2	< 0.1	5.3	I	-	0.47	1.1	2.1	1.8	-	0.9	"	I	A
D-9	210	36	26.6	-	97.3	0.15	3.9	0.65	195	2.5	2.9	8.9	7.2	-	1.5	"	0.98	A
D-9	210	49.5 Black	32.7	1.44-1.51	97.5	< 0.1	3.0	-	-	0.33	0.96	-	1.5	-	1.0	"	-	-
D-9	210	49.5 Shiny		1.44-1.51	97.9	< 0.1	2.7	-	-	-	-	-	-	-	-	-	-	-
D-9	240	36	41.6	-	98.8	< 0.1	4.6	1.06	122	0.95	1.0	-	1.7	-	2.2	7-12-66	0.73	A
D-13	300	30	45.8	1.44-1.51	97.5	0.21	3.3	-	-	1.2	1.6	-	2.2	-	0.80	"	-	A
D-13	300	42	25.1	1.44-1.51	97.5	0.08	4.0	-	-	3.4	2.5	-	4.2	-	1.4	"	-	A
E-8	210	38	37.2	1.44-1.51	96.7	1.12	3.4	-	-	1.2	1.3	-	3.3	-	0.94	"	-	A
E-12	300	42	32.6	1.44-1.51	97.9	< 0.1	2.6	-	-	1.4	1.2	-	2.6	-	1.3	"	-	A
F-7	180	44	35.9	1.44-1.51	97.4	< 0.1	2.8	-	-	1.8	1.5	-	3.8	-	2.0	"	-	A
F-9	90	48	33.7	1.51-1.55	96.7	0.2	3.8	-	-	1.6	2.0	-	3.3	-	1.8	"	-	A
F-9	210	36	44.9	I	96.9	< 0.1	4.7	-	-	1.3	1.6	-	2.8	9.2×10^4	1.7	"	I	-
F-9	210	48 Black	28.6	1.44-1.51	98.7	< 0.1	2.4	-	-	0.85	1.7	-	2.4	-	0.92	"	1.4	A
F-9	210	48 Shiny		1.44-1.51	96.9	< 0.1	3.6	-	-	-	-	-	-	-	-	-	-	A
F-9	270	48 Black	22.1	1.51-1.55	98.4	0.27	4.6	-	-	3.7	3.1	-	6.0	-	0.56	7-12-66	-	-
F-9	270	48 Shiny		1.44-1.51	96.4	< 0.1	3.1	-	-	2.7	2.1	-	4.2	-	0.92	"	-	-
F-11	60	36	26.0	-	99.8	< 0.1	3.4	0.75	173	1.2	1.6	2.1	3.0	-	1.1	"	-	A
F-11	60	48 Black	24.2	1.44-1.51	95.9	< 0.1	5.9	-	-	0.68	0.58	-	1.3	-	1.7	"	-	-
F-11	60	48 Shiny		1.44-1.51	97.3	< 0.1	2.8	-	-	-	-	-	-	-	-	-	-	-
F-11	90	36	19.0	-	91.2	< 0.1	4.7	0.56	213	0.53	0.57	1.7	1.5	-	0.78	7-12-66	0.17	A
F-11	150	28	43.1	1.37-1.44	97.2	0.12	2.7	-	-	2.3	1.4	-	4.1	-	1.3	"	-	-
F-11	180	36	36.4 [d]	-	100.5	< 0.1	3.1	0.72	180	1.4	1.1	-	1.9	-	1.5	"	0.83	A
F-11	210	42 Adj. OPT		1.44-1.51	97.3	0.14	2.8	-	-	1.7	1.6	-	2.3	-	1.2	"	-	A
F-11	210	42 3" From OPT	35.1	1.44-1.51	98.2	< 0.1	3.2	-	-	1.9	2.8	-	4.2	-	2.5	"	-	A
F-11	210	42 6" From OPT		1.44-1.51	97.7	< 0.1	3.7	-	-	2.6	4.9	-	4.6	-	3.4	"	-	A
F-11	300	48	46.7	1.44-1.51	96.3	0.15	3.4	-	-	1.5	1.8	-	2.4	-	1.5	"	-	-
F-13	150	36 Black	30.9	1.44-1.51	98.1	0.28	2.8	-	-	1.3	2.4	-	4.1	-	2.4	"	2.8	-
F-13	150	36 Shiny		1.44-1.51	98.4	< 0.1	2.6	-	-	-	-	-	-	-	-	-	-	-
F-13	150	48	39.7	1.44-1.51	96.4	< 0.1	3.4	-	-	0.74	1.7	-	1.9	-	1.3	7-12-66	0.95	-
F-13	180	36	22.0	-	98.7	< 0.1	4.6	0.45	287	1.3	2.0	-	2.2	-	1.8	"	1.2	A
F-15	180	48	29.9	1.44-1.51	96.5	< 0.1	3.4	-	-	0.91	1.6	-	2.4	-	2.7	"	1.8	-
F-13	180	48		1.44-1.51	95.9	< 0.1	5.2	-	-	-	-	-	-	-	-	-	-	-
F-15	300	45	41.5	1.44-1.51	95.7	< 0.1	4.1	-	-	1.2	2.9	-	2.7	-	1.4	7-12-66	-	A
F-15	330	41	42.7	1.44-1.51	98.0	< 0.1	2.7	-	-	3.5	0.98	-	2.4	-	1.9	"	-	A
G-8	150	38	18.3	1.44-1.51	98.1	< 0.1	4.3	-	-	2.4	1.6	-	4.7	-	0.51	"	-	A
G-10	30	46	19.6	1.44-1.51	97.9	< 0.1	2.5	-	-	1.2	1.3	-	2.2	-	1.1	"	-	A
G-10	210	41	36.9	1.44-1.51	96.2	0.99	3.1	-	-	1.8	1.2	-	3.1	-	3.0	"	-	A
G-10	300	28	37.1	1.30-1.37	99.2	< 0.1	2.3	-	-	1.1	0.81	2.3	1.8	-	1 2	"	0.37	A
G-10	300	36	29.7	1.44-1.51	99.3	< 0.1	2.3	-	-	1.7	2.0	3.0	3.3	-	1.2	"	0.6	A
G-10	300	44	20.3	1.44-1.51	99.5	< 0.1	3.1	-	-	1.4	1.1	2.8	2.3	-	0.86	"	1.5	A
G-10	300	48	18.1	1.44-1.51	99.8	< 0.1	3.2	-	-	0.92	1.5	-	2.0	-	0.78	"	0.57	A
G-10	330	36	31.3	1.44-1.51	97.0	< 0.1	2.9	-	-	0.86	0.66	-	0.79	-	1.4	"	0.65	-
G-10	330	44 Black	31.0	1.44-1.51	97.4	0.12	3.2	-	-	0.88	0.95	-	1.5	-	1.7	"	-	-
G-10	330	44 Shiny		1.44-1.51	97.1	< 0.1	3.0	-	-	-	-	-	-	-	-	-	-	-
G-12	90	38	48.8	1.44-1.51	96.6	0.46	3.1	-	-	1.5	2.7	-	2.7	-	1.5	7-12-66	-	A
G-12	210	30	24.0	1.44-1.51	97.7	< 0.1	2.9	-	-	2.0	1.9	-	3.4	-	2.4	"	-	A
H-13	90	50	8.7	1.51-1.55	98.3	< 0.1	4.6	-	-	3.6	4.4	-	1.3	-	-	"	-	A
H-13	210	38-I	19.4	1.44-1.51	98.3	< 0.1	2.5	-	-	1.3	1.4	-	2.5	-	1.7	"	-	A
H-13	210	38-II		1.44-1.51	98.3	< 0.1	2.8	-	-	0.93	1.3	-	2.0	-	1.4	"	-	A
H-15	330	44	35.4	1.51-1.55	93.5	0.25	2.8	-	-	1.5	1.8	-	2.6	-	1.3	"	-	A
J-12	90	35	39.8	1.44-1.51	97.3	< 0.1	3.0	-	-	2.2	1.7	-	3.9	-	2.1	"	-	A
J-12	90	41	43.1	1.44-1.51	97.1	0.34	3.0	-	-	2.0	2.3	-	2.8	-	1.4	"	-	A
J-14	30	44	27.2	1.44-1.51	97.0	< 0.1	3.4	-	-	2.1	2.3	-	4.7	-	1.8	"	-	A
J-14	90	39	32.4	1.44-1.51	97.4	0.16	3.0	-	-	2.4	2.2	-	5.2	-	2.3	"	-	A
J-14	90	44	32.0	1.44-1.51	97.5	< 0.1	3.0	-	-	3.3	2.1	-	5.0	-	2.2	"	-	A
K-11	30	44	49.2	1.44-1.51	98.6	< 0.1	2.1	-	-	1.8	1.3	-	2.9	-	1.4	"	-	A
K-11	60	44	38.1	1.44-1.51	98.3	0.44	2.4	-	-	3.7	1.9	-	5.8	-	2.2	"	-	A

Notes: a) A; amorphous as determined by x-ray diffraction.
b) Roman numerals refer to successive aliquots removed from a single piece sample.
Orientation with respect to process tube is not known.
c) I; insufficient sample to permit analysis.
d) OPT; fuel element outer process tube.

TABLE 9

CHLORINE, IRON, AND SULFUR CONTENTS* OF SOME PNPF OUTER MODERATOR SAMPLES

Core Location	Orientation Degrees from North	(Minus) Elevation Inches from Top of Upper Core Grid	Chlorine Content (ppm)	Iron Content (ppm)	Sulfur Content (ppm)	Cl/Fe Atom Ratio	C/H Atom Ratio
C-10	270	50	25	650	20-30	0.061	3.2
C-14	300	50I†	20	650		0.049	2.4
C-14	300	50II	10	700		0.023	2.8
C-14	300	50III	10	700		0.023	2.9
D-13	300	30	30	600		0.079	3.3
D-13	300	42	30	700		0.068	4.0
E-12	300	42	35	350		0.16	2.6
F-7	180	44	20	950		0.033	2.8
F-9	90	48	15	1100		0.021	3.8
F-9	270	48 Black	<10	800		<0.020	4.6
F-11	210	42 Adj OPT§	50	750		0.11	2.8
F-11	210	42 1/4 to 1/2 in. from OPT	25	550		0.072	3.2
F-11	210	42 1/2 to 3/4 in. from OPT	10	800		0.020	3.7
G-8	150	38	10	950		0.017	4.3
G-10	30	46	30	300		0.016	2.5
G-12	210	50	30	400		0.012	2.9
H-13	90	50	10	700		0.023	4.6
H-13	210	38I	25	350		0.11	2.5
H-13	210	38II	20	300		0.11	2.8
H-15	330	44	15	500		0.047	2.8
J-12	90	35	30	1050		0.045	3.0
J-12	90	41	25	400		0.098	3.0
J-14	30	44	20	800		0.039	3.4
J-14	90	39	25	800		0.049	3.0
J-14	90	44	30	650		0.073	3.0
K-11	30	44	15	450		0.052	2.1
K-11	60	44	20	600	20-30	0.052	2.4

*Chlorine, iron, and sulfur were determined in each sample by means of x-ray fluorescence.

†Roman numerals refer to adjacent parts of a sample which could not be identified as to orientation with respect to the process tube.

§OPT; outer process tube.

size that they could be split into two or three portions for analysis. The adjacent portions are indicated in the tables by Roman numerals. Only one of the larger samples, F-11, 210°, 42 in., was of such a nature that its orientation with respect to the process tube could be established.

4. Concentration of Impurities Effects

Chemical analyses of PNPF coolant and moderator deposit samples have shown that several coolant constituents are markedly concentrated in the deposit. These are summarized in Table 4.

TABLE 10

CONCENTRATION OF IMPURITIES IN PNPF MODERATOR DEPOSITS

Constituent of Property	Typical Concentration in		Concentration Factor
	Coolant	Moderator Deposit	
C	93.9 wt %	94%	1
H	6.1 wt %	2-3 wt %	~0.5
O	100-800 ppm	~7000 ppm	~10
ash	3 ppm	1000 ppm	~300
iron	1-2 ppm	~700 ppm	~300
Cl	0.1 ppm	30	~300

On the basis of elemental abundance, PNPF coolant is comprised of approximately 93.9% carbon and 6.1% hydrogen, by weight resulting in a C/H atom ratio of 1.28. In a "coking" process involving decomposition and loss of hydrogen as the most volatile component, the concentrations of trace impurities in the resulting carbonaceous matrix would not be expected to change greatly. For example, if 100 gm of coolant containing 3 ppm of inorganic impurities, measured as ash, were to be decomposed to the extent that 50% of its hydrogen were lost (C/H atom ratio = 2.6), the impurities would be concentrated in the remaining 96.9 gm of residue and their concentration, measured as ash, would be 3.1 ppm. The concentration effects displayed by the several coolant constituents are indicative of the fact that the in-core deposits were not the result of in situ degradation of whole coolant as such. Rather, they indicate that selective deposition

of certain impurities-bearing coolant constituents was a major factor contributing to in-core deposition.

5. The C/H Atom Ratio

The dissimilarities among the C/H atom ratios are related to the extent of decomposition of the sample. Decomposition of the deposit in core is normally a function of flux and time; however, if appreciable radiation heating is possible within a deposit, a pyrolytic contribution to decomposition is also possible. The higher C/H ratios may correspond to samples from locations within the deposit in which high temperatures were attained during reactor operation.

While the hydrogen loss rate from terphenyl can be calculated from the reactor off-gas rate and composition data, the uncertainties associated with a lack of knowledge of similar data for the hydrogen loss rate from an in-core deposit and the possible influence of pyrolytic effects on the C/H ratio of the deposit preclude the use of C/H ratio as an index of deposit age in-core.

6. Sulfur, Chlorine, and Iron

Analyses of a number of outer moderator deposit samples for their iron, chlorine, and sulfur contents were carried out by means of x-ray fluorescence techniques. These results are shown in Table 3. Iron was found to range from 300 to 1100 ppm, chlorine from about 10 to 50 ppm, and sulfur from 20 to 30 ppm.

The chemistry of sulfur compounds in organic coolants has not been extensively studied; however, the fairly uniform 20 to 30 ppm sulfur content of the outer moderator deposit suggests that the normal sulfur-containing species which have existed in the coolant have not been outstanding contributors to deposition.

A marked effect of chlorine in the transport of iron to the fouling deposit has been noted in many experiments of the Canadians. Abstraction of iron from the system walls followed by its transport as an iron-chlorine-organic complex to hot surfaces has been proposed. Thermal decomposition of the complex at the hot surface is thought to be the factor which deposits iron and releases chlorine to the coolant, where it can enter into another iron abstraction-transport-decomposition cycle.

The type of chlorine compound has been found to be important in the fouling process. Aliphatic chlorine compounds have been found to be more effective

than aromatic chlorine compounds in abstracting and transporting iron in organic systems. At present, no method exists for determining the various types of chlorine compounds which may be present in the coolant, or in reactor deposits; thus any assessment of a chlorine-initiated contribution to the Piqua moderator deposits must rely upon the characterization of the coolant and deposits in terms of their total chlorine and iron contents. Since the x-ray fluorescence method of analysis has provided self-consistent values for the iron and chlorine in each sample, a reasonable basis also exists for comparing the chlorine-to-iron atom ratios among the samples for which data exist.

a. AECL Data on the Effect of Chlorine

Canadian operating experience has shown that "an acceptably low fouling rate is obtained if the total chlorine is the equivalent of 3 ppm in the coolant."* This reference appears to be to the X-7 in-reactor loop data discussed by Bancroft, Charlesworth, and Duerksen† which indicate the relatively low fouling rates found to be associated with coolants having 1 to 3 ppm chlorine and 1 to 2 ppm iron content.

In out-of-reactor circulating loop experiments employing small in-line heated fouling probes,§ (results believed to apply in principal to in-core deposition) the observation of deposits rich in iron content is reported, ranging normally between 50 and 99% iron, and containing typically between 300 and 500 ppm chlorine in those cases where chlorine was determined. The high iron content is ascribed to the chlorine abstraction-transport-deposition-release mechanism. The primary iron compounds present in the deposits were determined by x-ray diffraction to be Fe_3O_4 and alpha-iron. The chlorine-to-iron atom ratio in all cases reported was of the order of 0.001. The low value of this ratio was taken to imply the release of chlorine from the complex by way of a thermal decomposition mechanism.

*W. M. Campbell, et al., "Development of Organic-Liquid Coolants," AECL-2015 (1965)

†A. R. Bancroft, D. H. Charlesworth, and J. H. Duerksen, "Impurity Effects in the Fouling of Heat Transfer Surfaces by Organic Coolants," AECL-1913 (May 1965)

§For a description of the SPFT fouling probe see A. R. Bancroft, "Equipment for Studying Fouling in Organic Coolant Systems," AECL-1706 (March 1963)

An apparent affinity of mild steel surfaces for chlorine was observed, and the rate of deposition was found to be more closely associated with the total chlorine content of the system (i.e., chlorine in the circulating coolant plus that associated with the steel surfaces), rather than with that in the circulating coolant only. Aliphatic chlorine compounds were found to have a greater effect on the deposition rate than aromatic chlorine compounds.

b. PNPF Data on Chlorine

By way of comparison, the Piqua outer moderator deposits have been found to contain typically about 0.07% iron by weight and 30 ppm chlorine resulting in a chlorine-to-iron atom ratio of about 0.06. The low concentration of iron species in the outer moderator deposits precludes the detection of either alpha-iron, or magnetic iron oxide by means of x-ray diffraction; however, very small fragments of these samples do consistently demonstrate weak magnetic properties thus suggesting the presence of these magnetic iron species.

The magnitude and variation of the values of the chlorine-to-iron atom ratio in the Piqua outer moderator samples are of interest. The magnitude of the Cl/Fe ratio is substantially greater in each of the Piqua samples than that found in the AECL deposits. The association of greater numbers of chlorine atoms per atom of iron in the Piqua deposit may be related to a lesser degree of thermal decomposition of the iron-chlorine-organic complex within the depositing layer resulting from lower temperatures in this region than the approximately 900°F temperatures which existed in the AECL tests. A detailed analysis of two samples (C-14 300°-50 in., and F-11 210°-42 in. locations) indicates the identical trend of decreasing chlorine-to-iron ratio with increased extent of decomposition (increasing carbon-to-hydrogen ratio). Proportionately, very much less variation exists among the values of iron content in these samples; thus these results lend support to the Canadian concept of chlorine-release from the chlorine-iron-organic complex.

Consideration of the coolant in which the Piqua moderator fouling species were formed shows that, during normal reactor operation, coolant ash values have been generally less than 5 ppm (implying an iron content less than 2 to 3 ppm), and that typical Cl^{38} activity levels have generally corresponded to a chlorine content of the order of 0.01 ppm. Several coolant samples removed

from the reactor during January-March, 1964 showed chlorine concentrations ranging from 1.8 to 2.8 ppm as determined by neutron activation off-site. Cl^{38} activity levels in the coolant during that period corresponded to about 0.01 ppm, and have not changed appreciably during reactor operation since that period. The reason for this apparent discrepancy is not clear; however, both the off-site analyses and Cl^{38} activities do suggest that reasonably low chlorine concentrations existed in PNPF coolant during periods of normal reactor operation; hence only a slight contribution to the in-core deposits might be expected to have resulted from the chlorine-promoted abstraction and transport of iron from the system walls. The generally low iron contents of the outer moderator deposit samples, when compared with iron contents of deposits formed in Canadian experiments, further suggest that coolant chlorine levels at Piqua have been low enough for chlorine-promoted corrosion and mass transport of iron in the system not to be a major factor contributing to deposition in-core.

c. Summary and Conclusions

1) The sulfur content of the PNPF outer moderator deposit has been found to range consistently between 20 and 30 ppm, suggesting that the normal sulfur-containing species which have existed in the coolant have not been outstanding contributors to deposition.

2) The chlorine content of PNPF outer moderator deposit samples has been found to range from 10 to 50 ppm, 30 ppm being fairly typical.

3) The iron content of PNPF outer moderator deposit samples has been found to range from 300 to 1100 ppm, 700 ppm being fairly typical.

4) The PNPF coolant chlorine content has remained at fairly low levels (perhaps 2 to 3 ppm at most) during periods of normal reactor operation as indicated by Cl^{38} activity levels in the coolant, and by several independent analyses.

5) The Canadians have reported greatly enhanced deposition rates in out-of-reactor tests as the result of chlorine compounds in the system. The deposits have been found to contain appreciable amounts of iron (~50 to 99%), thought to have deposited from a thermally unstable chlorine-iron-organic complex. Coolant chlorine contents less than 3 ppm have been reported to give "acceptably low" deposition rates.

6) The role of chlorine compounds in the organic system, as postulated by the Canadians, involves:

a) The abstraction of iron from the system piping by chlorine compounds,

b) The transport of iron in the coolant as a chlorine-iron-organic complex,

c) Thermal decomposition of the chlorine-iron-organic complex at heated surfaces resulting in deposition of iron on the surface, and

d) Release of chlorine to the coolant where it can participate again in the iron abstraction-transport steps.

7) In two PNPF moderator deposit samples, a trend was found in which the Cl/Fe atom ratio decreased with increasing extent of sample decomposition (i.e., with increasing C/H atom ratio). The iron content in the samples did not vary correspondingly, thus lending support to the Canadian concept of chlorine-release from a chlorine-iron-organic complex.

8) Greater values of the Cl/Fe atom ratio were found in the PNPF moderator deposit than appeared in the Canadian deposits. This fact may be related to a temperature-dependent rate of decomposition of the chlorine-iron-organic complex in the depositing layer, as proposed by the Canadians. (i.e., lesser chlorine loss, or greater Cl/Fe ratios would be expected in deposits formed under conditions where the temperature in the depositing layer is less than the approximately 900°F temperatures achieved in the Canadian experiments.)

9) The low concentration levels of chlorine in the coolant during normal reactor operation, and the low concentrations of iron in the outer moderator deposit, when compared with similar Canadian data, appear to indicate that chlorine-promoted corrosion and mass transport of iron in the PNPF system have not been a major factor contributing to deposition in-core.

D. SYSTEM PERFORMANCE TESTS

Deposition of film-forming species on surfaces in the inner and outer moderator regions of the Piqua core at a rate substantially greater than that on the heat-transfer surfaces is a phenomenon which has probably occurred since initial

power production in 1963. Buildup of the film and its radiolytic decomposition have led to a deposit having the properties necessary to result in flaking from the moderator surfaces under certain conditions. These conditions may be related to system upsets during reactor "scrams" and shutdowns. Lack of appropriate conditions and the absence of flaking may have led to the deposition of a substantial amount of material in the moderator regions of the core. Two tests designed to detect flaking from the core region and buildup of an extensive deposit in core have been recommended for application to the PNPF system following its startup.

1. Proposed Method of Detecting Flaking of an In-Core Deposit

This test makes use of the fact that all in-core deposits, heat transfer surface deposits and moderator deposits, have associated with them significant amounts of tritium and Fe^{55}. Direct monitoring of sample activity using a 2π proportional flow counter has shown that very small samples can give extremely high count rates. The direct monitoring method is only qualitative, due to large shielding effects in the sample itself. Tritium assays have revealed that a large portion of the sample activity observed in direct monitoring is due to Fe^{55} x-ray emission. The tritium-beta and Fe^{55} x-ray emissions behave similarly in the 2π proportional flow counter.

Flaking which may occur from any surface deposit in the reactor system may release particles of varying size to the coolant. If sufficiently large, the particles may be detected visually. Both small and large particles will contribute to increased Membrane Stain Test values through deposition on the MST filters. The high tritium and Fe^{55} activities associated with the in-core deposits affords a convenient means of assessing whether a discharge of particles into the coolant took place in the core region, or out-of-core. The test which will make this distinction is necessarily empirical in nature, and consists simply of monitoring each of the daily MST filters under conditions of constant geometry in a 2π proportional flow counter. To demonstrate that such a technique is feasible and produces interpretable differences, several MST filters sent to the AI labs from Piqua were counted in the 2π counter. A difference count, ascribed to the tritium and Fe^{55} emissions, was obtained by counting each filter with and without a 1/4-mil "Mylar" absorber. The difference counts which were obtained are summarized in Table 11.

TABLE 11

COUNTS OBTAINED ON MST FILTERS IN
2π PROPORTIONAL FLOW COUNTER

Sample	Date	MST (A/mg) x 10^5	Δcpm*
OC-1509	1-21-66	189	820
OC-1518	1-28-66	16	16
OC-1567	3-11-66	60	38
OC-1581	3-23-66	78	50
OC-1655	5-27-66	128	230

*Δcpm is the difference count obtained when the filter is counted with and without a 1/4-mil "Mylar" absorber.

The high MST value and difference count obtained on the filter from January 21 appear to be associated with control rod movements made on the previous day. The high values obtained on the May 27 sample are probably associated with the core unloading operations being carried out at that time.

It has been recommended that a daily difference count as herein described be determined for each MST filter during future operation of the PNPF, and that the results obtained be included with the coolant chemistry data for transmittal to AI and subsequent evaluation.

2. Proposed Method of Detecting Gross Deposition of Material In-Core

The conversion of hydrogen to deuterium and deuterium to tritium by means of neutron capture is aided by a 100% residence time in the neutron flux. Thus, the conversion of hydrogen, chemically bound to carbon in a polymeric deposit in core, to deuterium and tritium should result in the production of higher concentrations of the heavier isotopes in the deposit than those which exist in the circulating coolant, although the net rate of formation of these species does not change significantly. The storage of the heavier isotopes in the in-core deposit is important, and results in the finding of greater-than-normal concentrations of deuterated and tritiated compounds in the reactor off-gas, as the deposit becomes more extensive and suffers accelerated decomposition from radiation heating effects. In a "clean core" situation, the gaseous deuterated and tritiated

compounds which form in the coolant result in concentrations in the reactor off-gas which are less than the normal limit of detection. Only limited experimental data are available on this effect.

The composition of reactor off-gas samples has from time to time been determined by means of mass spectrometry. The most recent determination of off-gas composition was made on a sample taken January 7, 1966. For the first time, di-deuteroethylene was indicated to be present. Its presence was also suggested in the gas recovered from a "bomb" sample of coolant taken from the reactor on January 6, 1966. Under higher resolution, traces of other deuterated ethylenes and methanes were found along with traces of HD, HT, and T_2. Quantitative results were not obtained, due to the lack of suitable calibration standards. These isotopically labeled gases were not observed in previous reactor off-gas samples, the next most recent of which was taken on July 7, 1965.

There is some possibility that erroneous mass spectrometer cracking patterns (obtained by subtraction) could confuse ethane and di-deuteroethylene in the complex mixtures. Further clarification is impossible on these samples but will be obtained on more recent samples. Specifically, gas chromatographic analysis will be done on the samples to clearly identify the peak as ethane or di-deuteroethylene.

As the basis of a test relating to the extent of deposition of material in core, the sensitivity of detection of di-deuteroethylene would have to be maximized. The maximum sensitivity of the method would then be compared with the concentrations of deutero-compounds expected, or actually measured in the reactor off-gas under "clean core" conditions. Samples of reactor off-gas would be monitored frequently (probably every 5000 to 6000 Mwht of reactor operation) for the deutero-compounds. Any significant increase in the concentrations of these components may be related to the buildup of a deposit in-core.

It has been requested that a bomb sample of reactor off-gas and a bomb sample of reactor coolant be taken on the third day after resumption of power operation, and approximately every 5000 Mwht of reactor operation thereafter. These samples will be analyzed in the AI laboratories for the presence of deuterated and tritiated compounds. Results will be evaluated on a continuing basis by the PNPF Coolant Chemistry project.

E. COOLANT QUALITY, PRESENT AND FUTURE

The present system cleanup operation, the forthcoming fuel element loading, and reactor startup are factors which call for the special consideration of several aspects of coolant chemistry. These are discussed in the following sections, as are requirements for achieving and maintaining good quality coolant. The discussion is based on the assumption that future PNPF reactor system operation will exhibit behavioral trends in coolant chemistry which will be closely similar to those observed in past operation.

1. The Present

a. Gross System Cleanup

The purpose of the coolant circulation through full-flow and side-stream filters is to flush the system and cause deposition on the filters of the larger fragments of the core obstruction material. The flushing operation is efficiently removing fragments of the moderator deposit, as evidenced by the material found on the pump strainers and in the filters. It may be expected that removal of fragments down to some small size is accomplished by filtration alone. Smaller fragments which remain in the circulating coolant may be removed by operation of the purification still.

The coolant used in the flushing operation was that which was fairly extensively oxidized during April, May, and June, 1966. Though this coolant has been distilled, it is most likely that abnormal concentrations of oxygen compounds are present, since past experience has shown that their removal in the purification still is incomplete.

Some types of oxygen compounds, notably phenols and carboxylic acids (both of which have been shown to be formed in irradiated, oxidized polyphenyl coolants), are acidic in nature and may cause accelerated corrosion in the system. Corrosion of structural materials exposed to the coolant results in the entry of extremely small (perhaps atomic or ionic size) inorganic species into the coolant, where they may be transported to the various parts of the system and deposited on the available surfaces according to their natural affinities for the types of surfaces which are present. Once deposited they may be released during subsequent operation, contribute to coolant activity, and become available for deposition in-core. Continued operation of the purification still offers the primary means of removal of these species.

b. Coolant Quality During Core Loading

During core loading, and at any other time when the reactor vessel lid must be removed, the use of an effective nitrogen purge above the coolant is essential so that coolant-air contact may be minimized and good coolant quality maintained. Assistance in this area could be obtained through use of a suitable monitor to continuously measure the oxygen content of the atmosphere directly above the coolant, during periods when the reactor vessel must remain open. This instrument could be equipped to provide an alarm when the oxygen concentration reached preset level. The greatest tolerable oxygen concentration will probably have to be determined experimentally. As a guide, conditions during the month of February, 1966 were such that oxidation of the coolant, as indicated by carbonyl bound oxygen values, proceeded at a very slow rate.

c. Coolant Quality at Reactor Startup

Prior to startup, it is important to characterize reactor coolant: (1) ash, (2) MST, (3) HB, (4) major coolant components, (5) HB-40 content, (6) water content, (7) carbonyl bound oxygen content, (8) total oxygen content, (9) total chlorine content, (10) total sulfur content, (11) iron content, and (12) particle size distribution. The initial characterization of the coolant will provide a base of reference for subsequent reactor operation with the modified core, and will permit comparison with initial reactor startup data. If virgin coolant is used at atartup, no particular problems should arise in meeting any of the established coolant quality limits.

2. The Future – Coolant Quality After Reactor Startup

Following startup, the basic coolant quality parameters (ash, MST, carbonyl bound oxygen, HB, radionuclides, and water content) should be monitored as during previous periods of reactor operation. Further tasks are as follows.

1) The activities of the daily MST filters should be determined in the 2π flow proportional counter with and without a 1/4-mil Mylar absorber. The count rates with and without the absorber should be reported regularly with other coolant data. The basis of this test has been described in Section D.

2) A reactor off-gas sample should be taken on the third day after resumption of power operation, and every 5000 to 6000 Mwht thereafter. These

samples should be analyzed for the presence of deuterated compounds as mentioned in Section D.

3) Daily coolant ash samples should be saved and composited during each month for quantitative analysis of inorganic constituents. In the event of a reactor shutdown, the composite sample during the shutdown period would be desired.

In addition to the routing monitoring activities which were previously applied, consideration is being given to the incorporation of additional instrumentation for on-line determinations of various parameters of interest. One such device is an on-line coolant water monitor. The feasibility of such a monitor has been demonstrated in tests at the U. S. -operated organic test loop in Chalk River. In addition to giving a rapid indication of water inleakage in the event of a superheater tube failure, the monitor would provide valuable data to aid in our understanding of the relationship between coolant water content and film deposition rate.

Consideration is also being given to use of a conductivity cell of the current Canadian design for the purpose of continually monitoring the electrical conductivity of the coolant. Some evidence has been gained in the U. S. tests at Chalk River that suggests a possible relationship between the coolant electrical conductivity and MST values. Such information may help to increase our knowledge of the behavior of the fouling species in irradiated organic coolant, and would help to provide rapid indications of possible system upsets.

Additional work is required to establish the usefulness of a coolant quality test based upon measurement of a light-scattering parameter. Such a test would provide information related to the nature of submicron particles in the coolant.

Additional work is also required to establish the feasibility and basic parameters of an on-line monitor of total oxygen content of the coolant. Such a monitor could be based on determination of the 7-sec activity of N^{16} resulting from the fast neutron activation of oxygen species. The existing multichannel analyzer could be used along with a suitably placed detector adjacent to one of the main coolant lines. Possible utilization of a small-probe fouling test of current Canadian design in the flowing PNPF coolant stream is also being investigated.

Deposition rate information could be obtained in this test on a timely basis, and could be compared directly with similar results obtained in other organic systems (X-7 and U-3 loops, and the WR-1 reactor). Finally, an in-line viscometer may be of value. Such an instrument would provide data of importance to in-core heat transfer calculations. Correlations with other measured coolant quality parameters might also be expected.