HW - 41877

Cover Sheet for a Hanford Historical Document Released for Public Availability

Released 1994

Prepared for the U.S. Department of Energy
under Contract DE-AC06-76RLO 1830

Pacific Northwest Laboratory
Operated for the U.S. Department of Energy
by Battelle Memorial Institute

Battelle

DECLASSIFIED

[redacted]

MAR 30 1956

300 AREA
CLASSIFIED FILES

FILES CIRCULATING COPY
RETURN TO:
TECHNICAL INFORMATION FILES

HW-41877

AEC RESEARCH AND DEVELOPMENT REPORT

COPY No. ~~41~~ A

COPY 1 OF 1, [redacted]

PILE TECHNOLOGY SECTION
ANNUAL REPORT FOR 1955

COMPILED BY

MEMBERS OF THE PILE TECHNOLOGY SECTION, ENGINEERING DEPARTMENT

MARCH 15, 1956

THIS DOCUMENT IS PUBLICLY
AVAILABLE

~~THI... MENT ... DE... IN THE ATO... ENERG... AL OR ... CE... OF IT... TS IN ... Y ... NER ... TH... ERSON IS ... BITED.~~

HANFORD ATOMIC PRODUCTS OPERATION

RICHLAND, WASHINGTON

GENERAL ELECTRIC

MASTER

DECLASSIFIED

[redacted]

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

DECLASSIFIED

[redacted]

HW-41877

Special Distribution

This document consists
of 58 pages. Copy No. ~~24~~
~~of 87~~ copies. Series ~~A~~

Classification Cancelled and Changed To
DECLASSIFIED
By Authority of OPR-24
DS Lewis 9-28-92
By PM Eck 4-26-94
Verified By JH Webb 4-26-94

COPY 1 OF 1, [redacted]

PILE TECHNOLOGY SECTION ANNUAL REPORT FOR 1955

Compiled by

Members of the Pile Technology Section, Engineering Department

March 15, 1956

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

Operated for the Atomic Energy Commission by
General Electric Company under Contract #W-31-109-Eng-52

~~[illegible] E [illegible] ED [illegible]~~

~~[illegible]his do[illegible] [illegible] Restri[illegible]ted [illegible] as [illegible]fined [illegible] Atomic En[illegible] Act of 19[illegible] Its transm[illegible] or the dis[illegible]re of its [illegible] any [illegible] to an u[illegible]orized per[illegible] is prohibited.~~

Route To:	P. R. No.	Location	Files Route Date	Signature and Date
N. M. Hardy	[illegible]	[illegible]	4-20-56	

[redacted]

DECLASSIFIED

[redacted]

DISTRIBUTION

Copy Number	
1	F. K. McCune
2	R. D. Bennett
3	W. E. Johnson
4 - 8	AEC - HOO - Attn: J. E. Travis
9	A. B. Greninger
10	H. M. Parker
11	W. K. MacCready
12	O. H. Greager
13	R. B. Richards
14	R. H. Beaton
15	J. S. McMahon
16	F. W. Albaugh
17	P. H. Reinker
18	V. R. Cooper
19	L. P. Bupp
20	J. J. Cadwell
21	F. W. Woodfield
22	W. J. Ozeroff
23	H. F. Zuhr
24	W. M. Harty
25	R. W. Benoliel
26	W. Sale
27	H. A. Paulsen
28 - 30	Technical Information Service, Oak Ridge
31 - 35	Extra
36	300 File
37	Yellow File

[redacted]

TABLE OF CONTENTS

[redacted]

[redacted]

DECLASSIFIED

TABLE OF CONTENTS (contd.)

[redacted]

DECLASSIFIED

TABLE OF CONTENTS (contd.)

[redacted]

PILE TECHNOLOGY SECTION

ANNUAL REPORT FOR 1955

PILE ENGINEERING SUB-SECTION

Technical restricting limits to power levels at B, C, D, DR, F and H Reactors were increased about 8 per cent. At year's end all piles except the K's were operating on a tube outlet water temperature limit of 105 C and all pile powers were established generally by this limit and the flow capacities of the water plants.

Considerable process assistance was rendered to the startup of the K Piles and to initiating the process tube replacement program.

A number of tests were carried out to check theoretical predictions of a positive component in the graphite temperature coefficient of reactivity due to plutonium buildup. Results in general confirm the trend predicted by theory; these predictions have been incorporated in nuclear safety studies.

A study was completed predicting the rates of shield masonite deterioration and of radiation leakage increase as a function of exposure conditions. A production test of a fringe poison loading was carried out in a study to devise a feasible shield protection scheme at minimum cost to production efficiency.

Alternate methods of producing tritium in large quantity at Hanford were evaluated with respect to nuclear safety, product yields and product costs. The E-N method utilizing slightly enriched uranium fuel was shown to be most attractive for modest production requirements immediately with a modified J-N approach (wrap-around) potentially most attractive over the long range, but requiring substantial development.

The nuclear lattice constants for the K Pile lattices were measured in detail giving the most complete comparison between experiment and theory yet available in graphite lattices.

The gamma spectrometer slug rupture detection systems were installed in the three older piles and demonstrated their capability to reliably give the early rupture indications required to increase operating efficiency.

A technical evaluation of the tamperproof safety fuse showed its use at Hanford to be technically possible but extremely expensive.

[redacted]

The prototype Physical Constants Test Reactor was completed and turned over to the operating group.

A new method of specifying trip-before-instability limits was devised. The method simplifies the limit calculations and is more general in application.

The electrically-heated heat transfer mockup was modified to permit study of flow, temperature, and pressure conditions during such transients as power surges, steam loss, cone screen plugging, and downstream plugging. Tests of increasingly severe abnormal conditions are being made. Modifications to the electrically heated mockup were completed, permitting boiling and burnout studies up to 2000 psi and 1100 kw. By the end of the year, preliminary tests had shown generally satisfactory operation.

Test irradiations of internally-externally cooled slugs at C Pile verified the applicability of theoretical analysis of flow and heat distributions. Paper and laboratory studies in support of I and E development continued.

Proposed disaster control systems were evaluated for effectiveness under conditions that would be expected as a result of an act of war. Because the systems would give only limited protection under special circumstances, and in the absence of a known need for such systems, it was decided to concentrate efforts on methods to prevent and limit disasters resulting from peace time catastrophies.

A system to quickly locate water leaks in reactor process tubes was developed.

An experimental program was initiated to determine the cause and result of slug column discontinuities such as slug cocking.

Construction of the KAPL-120 loop at H Pile was completed, and the loop was put into operation with fuel elements and construction materials for the Submarine Advanced Reactor in the in-pile portion.

PILE MATERIALS SUB-SECTION

Based on flow laboratory studies and a half-pile production test, the pH of pile process water was reduced during the year from 7.65 to 7.3 with a resulting reduction in aluminum corrosion rate by 25 to 50 per cent. Following completion of a current half-pile test employing pH 7.0 process water it is anticipated that the pH specification may be lowered again to realize a further reduction in aluminum corrosion rates.

[redacted]

A comprehensive study during the year of the corrosion of aluminum process tubes resulted in the conclusions that (a) the primary cause of tube leaks is internal tube corrosion which is dependent upon past temperature history, (b) the tube replacement program should be scheduled to replace tubes at 30 mil residual wall thickness, and (c) reduction of outlet water temperatures to decrease the number of required tube replacements would result in a new loss of production. It was also concluded that the recent increase in "belly type" fuel element ruptures is due to the reduction in process tube rib height by combined corrosion and wear at the rear of the active zone of the aluminum process tubes. An economic study of retubing an old reactor with zirconium tubes showed that the capital cost could be amortized over periods as short as 1.3 to 2.5 years depending upon aluminum corrosion and retubing rates achieved, and future pile power levels.

As a result of in-pile studies of the effect of high helium concentration in the pile gas on graphite stack distortion the process specifications were changed during the year to permit helium concentrations up to 55 per cent at B, D, F, and 65 per cent at DR and H Piles with the result that no pile power level is currently limited by graphite limits. Continued studies during the year of the properties of a number of alternative types of graphite by means of both in-pile and out-of-pile tests have resulted in a much better understanding of raw materials requirements and graphite production techniques which result in graphites with desirable properties for pile use at different projected pile operating conditions. The cooperative program with Battelle Memorial Institute has continued to provide new graphites with remarkable radiation stability even up to 3000 MD/CT. Graphites with low crystallite size (30 Å) are characteristic of graphite dimensional stability under low temperature irradiation while a crystallite size greater than 300 Å results in generally good stability under high temperature irradiation.

The new 1706-KE Water Studies Semiworks was started up during the year with the objectives of testing a wide range of water qualities for effects such as corrosion and film deposition. This facility provides water of controlled quality to eight process tubes in the KE reactor on a once-through basis. Construction of the 1706-KER Recirculation Test Facility was approximately 70 per cent complete at year's end.

The H Pile high pressure recirculation loop employing a zirconium tube was operated experimentally during the year employing standard fuel elements and demineralized water at outlet temperatures as high as 200 C. Operation of this loop has contributed significantly to the field of recirculation technology by providing reliable corrosion data under actual pile operating conditions and practical knowledge relating to activity problems in recirculation systems. Following a fuel element rupture in this tube which resulted in considerable contamination of the loop system, the system was effectively decontaminated and the loop was returned to experimental operation. This rupture experience demonstrated that with the present fuel element configuration, sufficient time is available for safe shutdown of a reactor at an outlet water temperature of 200 C.

[redacted]

PHYSICS RESEARCH SUB-SECTION

The Lattice Testing Reactor was put into operation and the experimental program for this reactor was begun. A second experimental reactor, the Thermal Test Reactor, was brought near completion.

A major portion of reactor core theory was reformulated in terms of the neutron blackness concept. The new methods allow more precise calculations; they have been used with success in computing temperature coefficients and resonance escape probabilities. The problem of the interaction of several near-critical assemblies was worked out.

The study of the nuclear constants of U-235 and Pu-239 continued with measurement of the variation with neutron energy of the values $\nu(25)$, $\nu(49)$, and $\eta(49)$. These quantities, respectively the number of neutrons per fission in U-235, the number of neutrons per fission in Pu-239, and the number of neutrons per capture in Pu-239, were found to be constant except for the last. This showed some variation near 0.3 electron volts, the energy of the resonance. In the exponential experiment program, lattices with cluster type slugs, lattices with enriched slugs, and water lattices were studied.

METALLURGY RESEARCH SUB-SECTION

Forty-eight pre-characterized specimens of uranium were examined after irradiation permitting the determination of dimensional instability as a function of exposure and of degree and type of preferred orientation. Post-irradiation x-ray examination has also been made for selected samples of this material.

Metallographic samples of uranium, extensively studied before irradiation, were exposed in NaK-bearing capsules in the MTR and are awaiting examination.

A replica was successfully prepared of a uranium specimen that had received 580 MWD/T exposure and was examined employing both the optical and electron microscopes.

Ex-pile diffusion studies of the U-Al system were completed and the initial U-Al couples exposed in the MTR are under post-irradiation examination.

Apparatus for studies of diffusion of krypton in silver and uranium was completed and initial experiments were performed.

An elevated temperature tensile test unit for use with radioactive materials was designed and installed. The tensile properties of uranium exposed to 1.8 x 10^{20} nvt (620 MWD/T) were determined at 285 C.

Additional room temperature tensile tests were made on uranium specimens exposed in the MTR to 1.4 x 10^{20} nvt at 125-150 C and 1.8 x 10^{20} nvt at 300-400 C.

A study of the effects of irradiation and post-irradiation annealing on the tensile properties of zirconium was completed. Impact tests of irradiated Zircaloy-2 containing hydrogen were made.

The reaction kinetics of sponge zirconium and Zircaloy-2 with dry air at elevated temperature were determined. Irradiations of zirconium and Zircaloy-2 specimens in normal pile atmosphere and in controlled atmosphere at elevated temperatures were completed. A study of the creep properties of cold-worked Zircaloy-2 was initiated and initial results obtained at 300 C.

The uranium-magnesium matrix fuel material was proven much more resistant to irradiation effects than other homogeneous fuel materials. Data were obtained for specimens of this material exposed to 1000, 5000, and 10,000 MWD/T in the MTR. Additional specimens were discharged after 20,000 MWD/T exposure, and elements of larger geometry had reached exposures of 3500 MWD/T at year's end.

Irradiation tests of thorium-2 w/o U-235 fuel material are under way to determine the irradiation behavior under high temperature, high power, and long exposure conditions.

Seventeen capsule specimens of compacted UO_2 were irradiated at the MTR at temperatures near the melting point of UO_2.

A cluster fuel element design has been studied, and fabrication of components developed so that in-pile tests of the assembly may be made.

The fuel element testing facility at the MTR was employed throughout the year for fuel element evaluation tests. The operation of the facility was enhanced by the design and installation of a basket-type rechargeable B-block.

Unbonded fuel elements, both solid and cored, of natural and enriched uranium were exposed to determine the necessity of an uranium-aluminum bond for heat transfer. Slugs exposed to approximately 1000 MWD/T showed no evidence of preferential corrosion nor of non-uniform heat transfer.

 [redacted]

Insulated fuel elements were studied to determine the in-pile stability of uranium slugs operating at elevated temperatures. Two elements were exposed to 850 MWD/T at 80 kw/ft in the MTR. The uranium of these pieces was essentially unchanged in over all dimensions, although the 3/8-inch axial core was completely closed. The calculated maximum temperature of these elements was approximately 1000 C.

A high temperature water loop was installed, and experimental studies of the kinetics of the water-uranium reaction at high water and metal temperatures were made.

The facilities in the Radiometallurgy laboratory were operated throughout the year for fuel element examination and in support of research and development programs.

Corrosion rates of several stainless steels in Purex 2WW waste acid concentrate were determined in static corrosion tests and in heat transfer testing equipment. A pressurized heat transfer unit was designed and constructed in which variables affecting the corrosion of heat transfer surfaces may be evaluated. A field corrosion test of AISi 1020 steel and low alloy-high strength steels, Mayari-R and Corten, in the Hot Semi-Works Purex waste self-concentrator was initiated.

A joint was developed for joining tubes to tube sheets in heat exchangers having the corrosive media on the outside of the tubes. An extensive comparison was made of three welded pipe butt joints, and the advantages of the "GE" joint were shown. Initial tests were made in the evaluation of Zircaloy-2 weld quality in terms of welding conditions. A "roto-arc" welding unit for welding caps or closures on cylindrical containers was built and successfully employed.

Techniques were developed for precision casting alloys of natural uranium, depleted uranium, and plutonium that will be employed as fuel material in the PCTR. A procedure for canning the fuel elements in aluminum cans by brazing the cap was developed. Building modifications and assembly of equipment for production of these fuel elements were almost completed at year's end. Plutonium-aluminum alloys were successfully cast and will be fabricated for use as PCTR monitor foils. Various sizes of electro-formed nickel tubing were prepared for use in PCTR monitor slugs.

FUEL TECHNOLOGY SUB-SECTION

Technical assistance efforts to the manufacturing process overcame various equipment and processing difficulties during 1955. A cooperative effort with FMPC made possible a reduction in core hydrogen content to the extent that by mid year the production outgassing at Hanford necessitated by braze layer porosity was discontinued, except on dingot uranium. A new method was developed for hydrogen analysis which should yield better correlations and more control of braze porosity.

More stringent acceptance levels were established for the ultrasonic transformation test to preclude the passing of incompletely transformed slugs. The relationship between Al-Si "spikes" in the weld bead and the lead content of the canning bath was established and remedial measures were initiated. Results from efforts to correct the processing difficulties encountered with the mechanical spray type slug pickling equipment indicates that rates and qualities comparable to those for dip pickling can be attained.

The hot press method was established as the standard manufacturing process for enrichment slugs. Three methods of sealing the ends of cored slugs were developed. Assistance was rendered in the fabrication of fifty "donut converter" slugs for the Belgian government.

Various programs were undertaken in the field of process development. An extensive evaluation was initiated at HAPO on dingot (direct cast ingot) uranium. Results, to date, indicate that, although this process produces an improvement in metal quality, the hydrogen content in the material produced by the present methods is such to make it unacceptable for the Hanford canning process. A large scale testing program is underway to establish the effect of uranium heat treating on the irradiation stability of slugs. Solid and tubular rods were successfully alpha extruded off-site in a cooperative program.

Hanford hot press canned fuel elements were irradiated and results were negative. Further process improvements have been made and a second irradiation test is now underway. Six tubes of unbonded fuel elements with point closures were irradiated with negative test results.

A modified lead dip canning process was developed for I and E (internally and externally) cooled fuel elements. An irradiation test on this process and element geometry is now underway. Equipment for Al-Si canning at reduced pressures was developed for I and E slugs.

[redacted]

Development work on new type fuel elements proceeded during the year. Wafer (segmented) fuel elements were designed, assembled, and evaluated by out-of-pile tests and by a MTR irradiation test. This type of slug fails in such a manner to minimize the probability of blocking the process tube water flow. Studies on grain size and orientation were made on uranium wafers, and methods of heat treating and of producing production quantities of wafers were developed.

Methods of supporting fuel elements in ribless process tubes were investigated. Various methods of attaching supporting projections to slugs are being developed. An irradiation test on self supported fuel elements prepared by one of the possible methods (percussion welded) is now underway.

In the field of materials development, development work on ribbed zirconium tubes has been moderately successful and has resulted in the production of two tubes of old pile dimensions and two lengths (6 to 13 feet) of KER dimensions. Charging studies have shown that anodized coatings on aluminum jackets prevents galling and slug misalignment during charging and should be effective in overcoming these same effects due to thermal expansion during operation. Since corrosion studies show that unsealed anodized films wash off rapidly in hot water, a new sealed film was developed and is now ready for in-pile testing.

A study on uranium water reactions yielded the rate relationship for temperatures up to 350 C. A technique was developed for grain size determination on macro etched uranium.

Development work proceeded on methods and equipment for pre-irradiation testing of fuel elements. A second prototype of the eddy current metal quality tester yielded a good correlation between test results and induction cycling results although one test to verify the correlation by in-pile testing was unsuccessful. The development of instrumentation for testing various qualities of fuel elements (among which are the sonic orientation test, the ultrasonic grain size test, the Al-Si penetration test, the ultrasonic bond test, and a test to detect lead or Al-Si in the central void of cored slugs) proceeded through various stages up to the design and installation of equipment for regular production use.

The irradiation behavior of fuel elements during 1955 was erratic. While failures of slugs due to core splitting decreased substantially, the over-all number of failures increased markedly due to jacket failures. The jacket failures have been attributed primarily to localized hot spots and accompanying intergranular corrosion, and to increased localized corrosion caused by worn ribs. A study of the warp of irradiated slugs showed that slug warp can be correlated and increases with tube power and exposure. Fracture testing of irradiated fuel elements has better defined irradiation induced splitting of solid and cored fuel elements.

DECLASSIFIED

Facilities for fuel development work were increased during the year. The 306 Building Fuel Element Pilot Plant was occupied although the installation of some equipment is still proceeding. The 105-C Metal Examination Facility was occupied, necessary equipment revisions on equipment in Basins I, II, and III were completed, and operator training was begun. Revisions to Basin IV equipment are proceeding.

DECLASSIFIED

[redacted]

DECLASSIFIED

PILE ENGINEERING SUB-SECTION

Power Lever and Capacity Increases

Continued gains were made during the year as a result of Research and Development efforts in the Pile Engineering Sub-Section and other Technical components. Technical restricting limits to power levels at the B, C, D, DR, F, and H Reactors were increased about 8 per cent. Ratios of average power levels in December 1955 and December 1953, to those in December 1954 are given below:

Pile	1953	1954	1955
B	0.7	1.0	1.0
C	0.7	1.0	1.2
D	0.8	1.0	1.1
DR	0.8	1.0	1.6
F	0.7	1.0	1.1
H	0.9	1.0	1.2
KE	---	---	0.9*
KW	---	---	0.9*

*Compared to revised design capacity. The K Piles are not presently on technical limits.

Power levels achieved exceeded the limit increases because of the following factors: (1) relaxation of an operational arbitrary 100 C limit, (2) installation of Venturis at H and DR, and (3) discharge of the J-N loading at DR. At year's end all piles except the K Piles were operating on a tube outlet water temperature limit of 105 C and the pile powers were established generally by this limit and the flow capacities of the water plants.

Research and Developments were curtailed by process assistance demands in starting the two K Piles and initiating a process tube replacement program.

Process Technology - Ruptured Slugs

Slug ruptures occurred in two hundred sixteen charges during the year. The number of failures incurred by various types of metal are shown in the following table.

[redacted]

DECLASSIFIED

Number of Failures by Metal Type

Type of Metal	Number of Failures
Normal uranium production metal	154
Aluminum-U^{235} Alloy "J" Metal	14
Aluminum-U^{235} Alloy "C" Metal	12
Production Test Material	36

Most regular slug failures were of two types, both attributed to corrosion. Uranium cleavage failures and cap failures, which have been noted in the past, occurred infrequently. Of the failures due to corrosion the "hot-spot" type was the most prevalent. Failures of this type are characterized by a hole in the jacket opposite, or nearly opposite, the rib marks; with elliptical or tear-drop shaped film markings around the hole. The appearance of these film patterns indicates that they may be the result of a coolant flow disturbance or uneven heat flow through the aluminum jacket. Radio-metallurgical studies have shown that intergranular corrosion, of the jacket, is present in the area surrounding the hole. It may be concluded that slug misalignment, slug column bowing or slug warping places the slug near or in contact with the tube wall. This results in local overheating of the jacket, which initiates intergranular corrosion, and results in an eventual penetration.

The other type of corrosion failure is characterized by a hole in the jacket between the rib marks, with the jacket appearing to have been corroded or thinned excessively in this region. These failures are felt to result from a reduction in rib height due to accumulated corrosion and rib wear in tubes which have been in pile for several years. This reduction allows the slugs to be positioned lower in the tube, decreasing the coolant flow between the ribs, increasing water temperatures in this region, and thus accelerating the corrosion of the jacket.

The incidence of slug ruptures increased sharply during the last quarter of the year. This increase is due to the increase in the number of badly corroded process tubes and the fact that the H and DR Piles were removed from the low concentration program.

Failure rates for "J" metal continued at about the same level as for 1954. The total number of failures is greatly reduced, however, due to the termination of the J-N program and the discharge of this material early in the year.

Of the "C" metal failures, nearly half are the result of vibration or "chattering" in the process tube. This action causes an eventual penetration by abrasion.

 [redacted]

Process Technology - Fuel Testing and Development

During the year, nearly all significant proposed fabrication changes were evaluated by irradiating a few charges to failure and applying statistical analysis techniques for the determination of improvement or unimprovement.

A considerable portion of the testing concerned itself with the evaluation of cored slugs. Experience to date is not adequate to definitely detect a difference in performance between slugs having a three-eights inch axial void and the standard solid uranium slug.

Eight tests were concluded during the year. Analysis of results for tests of hot press canned slugs, unbonded (C process) slugs and mechanically bonded slugs revealed that the test slugs were inferior to the regular production slugs which were used as controls. Tests of cored slugs and power metal compact slugs failed to reveal a significant difference between the test and control pieces. Preliminary irradiation of internally and externally cooled slugs were conducted but no measure of relative performance was determined.

Currently being irradiated under test conditions are:

1. Four-inch cored slugs at low powers.
2. Slugs from uranium cast by the dingot process.
3. Slugs having various modifications in the beta heat treating cycle.
4. Enriched, cored slugs with welded uranium end plug, pressed aluminum end plugs and 3/8 inch and 5/8 inch core diameters.
5. Eight-inch natural uranium cored slugs having pressed aluminum end plugs.

Process Technology - Process Analysis

Studies were completed which reveal that 0.9 per cent U235 enriched uranium can be substituted for "C" alloy metal in fringe enrichment tubes with considerable economic advantage. Additional advantages are:

1. A gain in production of about one per cent can be achieved by using the nonproductive power of the alloy columns for plutonium production as well as enrichment.
2. Enriched uranium slugs will not have the decreased reactivity or high burnout that characterize the present alloy columns.
3. Safer pile operation, reduced graphite temperatures and a reduction of tube damage from fuel vibration result from the substitution.

A production test is being prepared for the initial loadings of the enriched uranium.

[redacted]

Large savings in metal processing costs may be achieved by blending irradiated enriched material with normal low goal exposure depleted metal to obtain uranium of natural U^{235} concentration. Enriched metal, proposed as a substitute for "C" metal fringe enrichment columns, could be a source of blending material, if irradiated to low goal exposure.

The use of depleted uranium, of 0.2 and 0.3 per cent U^{235}, in poison columns was studied and was shown to increase plutonium production and to provide increased flexibility when compared to 10-66 irradiation. The use of a fringe blanket of depleted uranium to reduce shield damage was also evaluated. A production test is necessary to determine pile variables.

Studies were completed which revealed that metal goal exposure in excess of one thousand MWD/ton is not optimum from an economic point of view because of the high rate of plutonium burnout at higher exposures. Related studies were made to determine the economic losses incurred when metal is discharged at less than goal exposure. These losses are balanced against other operation losses for: (a) reduction of goal exposure due to high rupture rate, (b) early discharge of rupture prone lots of material and (c) discharge of metal at less than goal exposure to make use of available shutdown time.

An examination was made of the variables of reactor operation after water plant modifications describing the production of a modified reactor in terms of goal exposure and tube power. This examination reveals that:

1. Substantial increases in production can be attained by a decrease in goal exposure after water plant expansion as a result of the reduction of plutonium burnout at lower exposures and the achievement of increased powers at reasonable operating efficiencies.
2. Increased production can be attained after water plant modification by reduction of goal exposure even if rupture resistance is increased by a factor of ten.
3. Increases in production at present power levels can be obtained by reduction of goal exposure.

Pile Physics

Considerable progress was made in the field of pile safety. Analyses were performed and results reported on temperature coefficient measurements made in both the wet and dry K lattice at time of original loading; a series of graphite temperature coefficient measurements were made by deliberate atmosphere composition changes at constant power level as the KE pile loading increased in exposure; a careful evaluation was carried out relating the large reactivity changes noted during two otherwise similar pile outages to the effective plutonium concentration in the pile in the two cases;

[redacted]

and Hanford personnel participated in and initiated the analysis of a coefficient measurement in the moderately exposed loading in the dry graphite lattice at Brookhaven. Results of these tests and analyses to date tend to confirm the theoretically predicted positive reactivity component of the graphite temperature coefficient due to plutonium buildup.

An experimental measurement was made of the ultimate control capacity of the 51 vertical channel Ball 3X system of the enriched K Piles, and a complete K Pile horizontal rod calibration by period measurements was obtained. Water loss excursion calculations were extended to include the effects of the use of various numbers of faster acting safety rods. A safety summary of natural uranium loadings was issued presenting the best available data on the Hanford reactor nuclear safety problem in a series of explanatory graphs; conclusions were drawn from these studies and recommendations were made concerning construction budget items which would improve pile safety.

The physics constants for the 7-1/2 inch natural uranium-graphite lattice derived from critical size determinations during the K Pile startups were evaluated and published. The approximate reactivity effects of poison and enriched materials commonly used in single column loadings and of special fuel and target and new slug geometry loadings were also obtained and evaluated. Calculations were made in support of scoping studies for special isotope production concerning safety system control capacity and operational transient behavior. Both theoretical and empirical studies of U^{233} formation in Hanford irradiated thorium were completed.

Comprehensive reports on shield masonite deterioration studies and on the attenuation measurements on high density concretes were completed during the year. An experimental program to ascertain the degree of shield protection which could be attained by fringe poisoning and its cost in terms of production efficiency was carried out first by low level flux distribution measurements during the K Pile startup and second by a production test loading in the far-side of the H Pile. Test results combined with an economic analysis indicate that approach to be technically feasible and economically within reason. Attenuation measurements were taken on high density concretes which had previously been baked at 320C; these temperatures cover the range to be expected in a recirculating design for production piles.

Physics Development

The gamma spectrometer slug rupture detection systems replaced the beta sensitive systems on three reactors as part of the scheduled Project CG-578 and 579 activities. The improved sensitivity and reliability is as expected and should markedly decrease the "stuck" ruptures and rupture induced water leaks. Gamma spectrometer units to replace the K Pile beta systems were developed as were delayed neutron sensing systems for use with recirculating coolant systems. Systems to provide increased sensitivity in isolating the tube containing a rupture were also developed and demonstration experiments initiated for those designed for remote, automatic operation.

The Physical Constants Test Reactor was completed and turned over to Physics Research for operational responsibility. The electrical, instrument and mechanical features, many of which were original with this reactor, have all performed in practice as designed. This reactor is expected to largely replace the exponential pile program in giving rapid, economic support to studies determining the reactivity and conversion ratio in graphite lattices, e.g., massive cored uranium, internally and externally cooled, or cluster type uranium fueled lattice data are critically needed for both present and future reactor programs.

The development of the sub-critical pile neutron flux monitor, an instrument which (1) provides a direct measure of neutron density during all phases of shutdown, thus permitting an experimental extrapolation to critical, (2) low level period trips, and (3) continuous monitoring of the power from sub-critical to full power, has been completed and action to install these systems on all piles initiated. The development of high temperature and flux ionization chambers, safety circuit activation mechanisms, and period trip systems continued in support of reactor operational safety. Systems to place outlet water temperatures in the safety circuit continuously were scoped and specified and the degree of protection afforded evaluated to demonstrate justification.

Complete lattice experiments in the cold, clean KW Pile lattice were analyzed to yield the nuclear lattice constants of that uranium-graphite system in detail and the initial cold, clean conversion ratio. These results have clearly indicated that areas of theory which require development are currently underway. Studies of the lattice characteristics of internally and externally cooled slugs and cluster elements have been partially completed. Several alternate methods of producing tritium in large quantity were scoped and the lattice physics and nuclear safety associated with the E-N approach (uranium fuel enriched to 0.94 weight per cent U^{235} and 3.5 weight per cent lithium in aluminum N slugs) were developed to show the E-N approach to be technically feasible and economically attractive.

The program, costs, protection afforded, and limitations involved in utilizing tamper-proof safety fuses in the Hanford reactors were evaluated and continuing assistance rendered in support of the development program. The studies showed that fuse utilization is technically possible but is very expensive. Methods of experimentally determining rate of change in the volume and distribution of water in a process tube following pressure loss were developed in support of water boilout and power excursion studies.

[redacted]

Heat Transfer

During 1955, experimental and analytical studies to broaden the understanding and improve the application of trip-before-instability limits continued. Experimentally, boiling curves were redetermined for B, D, DR, F and H Piles and were run for the first time for K, using the electrically heated mockup with three generators and pre-heater to simulate tube powers up to 1400 kw. The data showed a previously unobserved effect. At flows somewhat greater than those corresponding to the minimum point of the boiling curve, there is a moderate increase in the tube demand over that expected for single-phase flow; pressure and temperature measurements indicate that bulk boiling is not occurring. The effect is believed to be due to surface boiling near the end of the active section and results in somewhat decreased Panellit sensitivity.

The analytical studies of trip-before-instability data resulted in a major modification in the methods of determining and specifying instability limits. Correction of an erroneous characterization of instability flow resulted in a reduction of limits by as much as 8 C in venturi zone and an increase in limits in the double orifice zone. A new method of specifying limits was devised. This method is based on generalizing the experimental boiling curves; it is technically sound for filmed and corroded tubes and eliminates the "G" factor calculations.

The electrically heated mockup was modified during the year to permit study of flow, temperature, and pressure conditions during such transients as power surges, steam loss, cone screen plugging, and downstream plugging. In each test, with the tube operating at the desired flow and power, the abnormal conditions were introduced and the power decayed following a curve corresponding to or more conservative than the Borst-Wheeler curve. The testing and analysis program is continuing with more and more severe conditions being imposed. The most extreme test run to date was a simulated fast cone screen plug on a tube at 1000 kw power and 145 C outlet temperature. The flow was dropped almost instantaneously to about 55 per cent of the original value and the power decay initiated only after the high Panellit trip which followed the onset of flow instability. Flow was less than 5 gpm for over 20 seconds, but examination of fusible metal plugs in the heater rod indicated the melting point of aluminum was not exceeded at the surface.

A few steady state boiling tests were performed which resulted in net steam formation at pressures up to 1250 psi, completing a program begun in 1954. Equipment difficulties prevented more extensive testing.

During the year, modifications to the electrically heated mockup were completed to permit boiling and burnout studies up to 2000 psi and 1100 kw. By the end of the year, preliminary testing had shown generally satisfactory operation.

[redacted]

Analytical, laboratory, and in-pile studies of internally-externally cooled slugs continued during the year. Two irradiations were made at C Pile. The first was a single tube with a full charge of hot-press-canned slugs. This charge was in the pile a month before individual annulus and hole outlet temperatures could be measured; it was then found that the hole stream was running much hotter than expected and the tube was discharged. To date, no satisfactory explanation has been found. The second irradiation used three tubes with hot-press-canned slugs of three different hole diameters. On this test, the heat and flow distributions checked the theoretical values satisfactorily. The slugs were discharged after the nozzle insert on one of the tubes broke loose and washed downstream, cutting off the hole flow and scramming the pile. Laboratory studies have been made to find flow characteristics in support of the in-pile tests. Some work has been done on optimization studies.

Laboratory and paper studies were made to determine flow characterisitics of fittings, optimum Panellit pressure, and the correct venturi and orifice throat diameters for CG-558.

Slug heating effects associated with the 4669-KW incident of January 5, 1955, were studied to ascertain whether the predictions regarding the consequences of lack of tube cooling were substantiated. It was concluded that the effects can be explained reasonably on the basis of heat transfer calculations. Diffusion of uranium into the aluminum was more rapid than expected, but this has a minor effect upon the predictions.

Revised water shut-off charts were prepared for rupture discharge conditions at C Pile. A preliminary analysis was made of the K Pile operating procedures in order to establish safe procedures for operation of the cross-tie, the rear riser cross-under line, and related equipment. A detailed study of the K emergency water supply was initiated. Studies were made of the consequences of water loss or steam power loss to the reactors.

Support of fuel element development studies continued with extensive calculations being made for a number of slug and test specimen designs.

Mechanical Equipment Development

Development work was completed for the new horizintal control rods and seals that were installed in B, D, F, DR and H Piles during the year.

Development and laboratory testing of the poison spline supplementary control system has progressed to the point where the spline is ready for in pile testing.

 [redacted]

A system to accelerate vertical safety rods with compressed air was tested. Drop times for the K Pile rods were decreased 40 per cent. Refinement in design should provide increased accelerations. Additional testing must await construction of a suitable test facility.

Disaster studies were continued during the year. Since the need for a war-time disaster control system is vague and the effectiveness of such systems limited, efforts were diverted toward defining the extent of damage that might be expected as a result of a major power excursion. In support of this program studies are being made to determine the extent of fission product escape that may be expected for various conditions. An experimental program to determine basic data is being developed.

A considerable number of slug ruptures have occurred because of intergranular corrosion of the aluminum can. A possible cause of intergranular corrosion is an unsymmetrical water annulus caused by slug misalignment. To determine the cause and effect of slug misalignment a series of laboratory experiments are being conducted. Thus far it has been discovered that cocked slugs cause misalignment of adjacent slugs in the column.

Process tubes of zirconium alloys are received in the fully work hardened condition and crack when on-site forming is attempted. Development of annealing and forming techniques have improved such that vacuum annealing and flanging of the tube in a lathe is no longer necessary. The present technique employs a short induction anneal in air and flanging the tube with a hand operated rolling tool.

Locating a leaking process tube in the past has required individual pressure testing of many tubes. A system was developed that would maintain a partial vacuum on a large number of tubes simultaneously. A leaking tube will allow helium to enter the water system and by monitoring for the presence of helium a leak is quickly detected. Necessary piping has been installed on D, F, and H Piles.

Experiments have been conducted to determine the feasibility of splitting process tubes with an electric arc prior to their removal from the reactor. These experiments are continuing.

A program to determine the effect of large pushing on reactor graphite was initiated. A detail analytical study was completed and plans were made to supplement the analytical results with experimental data.

Assistance was rendered in assemblying and pre-startup testing of the Physical Constants Test Reactor.

Testing of rubber compounds under pile conditions continued during the year.

 [redacted]

DECLASSIFIED

Many smaller developments were pursued. Included among these were the graphite miner, graphite core borer, process tube splitting saw, mechanical components of the sub-critical monitor, testing of process tube connectors, and charging machine improvements.

A number of design tests were performed in support of the plant modification and expansion programs.

Special Irradiations

Bluenose studies pertaining to the accurate determination of the integrated power generated in a column of slugs in a process tube of a Hanford pile was completed. This work included a very accurate weaseling of individual slugs to permit an assignment of integrated power generation to each slug in the irradiation.

Dismantling of the old ANL-140 Loop at H Pile was completed. The major phase of this operation was the discharge of the stainless steel in-pile tube from the A test hole. This discharge was accomplished remotely and the tube was drawn into a specially prepared water basin where it was cut up for metallurgical studies. The gamma activity of the tube was 10,000 r/hr at one foot in the air.

Construction of the KAPL-120 loop at H Pile was completed, and the loop was put into operation with fuel elements and construction materials for the Submarine Advanced Reactor in the in-pile portion. Operation of the loop has been successful.

With the start-up of the K Piles, the irradiation facilities in those units were tested out and preparations made for their use for experimental work. Equipment has been constructed for the use of the Snout facilities in KE Pile. Design of a quickie facility for future installation in one of the K Piles has been initiated. A gamma facility giving irradiation intensities of 10^7 r/hr over a 2 cubic foot volume has been designed, and steps are being taken for its construction.

Feasibility studies and preliminary designs were conducted in conjunction with the Design and Project Sections on an air-cooled facility in C Pile for the irradiation of fuel elements and construction materials for aircraft nuclear reactors, and on three high pressure, high temperature recirculating loops in DR Pile in support of the submarine reactors and other Atomic Energy programs. A third study was made on the amount of cobalt that could be produced as a by-product at HAPO piles without interference to the production of plutonium. It was estimated that in excess of 2,000,000 curies of Cobalt 60 could be produced from all piles exclusive of the K Piles. This cobalt would have an average activity of 2 curies/gram.

DECLASSIFIED [redacted]

Methods for the measurement of high intensity gamma sources were studied. Dosimetry methods which were investigated included the use of ionization chambers, calorimeters, and chemical techniques. Satisfactory agreement between these three methods was attained.

A program in support of the development of reactor safety fuses was carried out jointly with North American Aviation. A series of experiments were planned. Experimental assemblies in support of these experiments are being prepared by NAA. Instrumentation and other work in support of irradiating these assemblies is in progress at HAPO.

The production of isotopes in support of the isotope distribution program administered at Oak Ridge continued. Approximately 25,000 curies were shipped during the year. Of this quantity, cobalt 60 accounted for approximately 90 per cent, and sulfur 35 about 8 per cent. Twenty-five different kinds of isotopes were shipped.

A total of 34 irradiation requests were serviced in support of HAPO Research and Development and Process Technology studies. Thirty-five Hanford Irradiation Requests were serviced in support of offsite irradiation programs. The latter programs were aimed primarily at submarine reactor development, isotope production, and general research studies.

PILE MATERIALS SUB-SECTION

Pile Coolant Studies

Flow laboratory studies and a one-half pile production test (in 100-F Pile) demonstrated the desirability of reducing the pile process water pH specification from 7.65 to 7.3. This water quality improvement resulted in reduction of aluminum corrosion rates by 25 to 50 per cent. A one-half pile test of pH 7.0 process water was started in 100-F Pile and has operated satisfactorily for six months. It is anticipated that pH 7.0 process water will result in a further reduction of aluminum corrosion rates.

Examination of pile process tubes was accelerated to collect current information on the condition of process tubes in the old reactors. These data revealed the necessity of starting a large scale retubing program, requiring the replacement of approximately 3500 tubes in the twelve months starting November 1, 1955. A comprehensive study resulted in the conclusions that (a) the primary cause of tube leaks is internal tube corrosion which is dependent upon past temperature history, (b) the tube replacement program should be scheduled to replace tubes at 30 mil residual wall thickness, and (c) reduction of outlet water temperatures to decrease the number of required tube replacements will result in a net loss of production. It was also concluded that reduction of the rib height by combined corrosion and wear at the rear of the active zone causes accelerated corrosion of the tube wall and the slug jacket between the ribs. The recent increase in corrosion rupture rate (e.g., "belly", or "between-the-ribs" ruptures) is attributed to this mechanism.

The one-half pile test of the reduction of sodium dichromate concentration in pile process water from 2.0 ppm to 0.5 ppm was terminated. Although the slug corrosion data from this test indicated no increase in slug-jacket corrosion rate at the lower dichromate concentration, the majority of the tube leaks in the reactor occurred on the side supplied with 0.5 ppm dichromate water. IBM compilation of a more complete temperature history for all of the tubes in the reactor is in progress to reach a final conclusion regarding the influence of the lower dichromate concentration on rate of tube corrosion.

A ribbed Zircaloy-2 process tube was installed in a reactor to study the effect of a zirconium tube on the corrosion rate of aluminum-jacketed slugs. Orders were placed for 18 zirconium and Zircaloy-2 process tubes. Allegheny-Ludlum will fabricate by extrusion and tube-reducing techniques six ribless zirconium tubes with 1.61 I.D. Trent Tube Company will fabricate six zirconium and six Zircaloy-2 ribbed process tubes, 1.729 O.D., by roll-forming-welding techniques. An economic study of retubing an old reactor with zirconium tubes showed that the capital cost could be amortized over periods as short as 1.3 to 2.5 years, depending upon aluminum corrosion and retubing rates achieved, and future pile power levels.

Laboratory corrosion tests demonstrated that there is no great difference in the uniform corrosion rates of various aluminum alloys under simulated pile conditions. Based on strength characteristics, 63S and APMP-M257 are the most promising aluminum alloys for process tubes operating at future higher power levels. In line with this conclusion, 63S process tubes are being subjected to in-pile testing.

Determination of aluminum corrosion rates in low pH, 7.0 to 5.5, process water was initiated in flow laboratory mock-ups and in-pile tubes. These studies are expected to lead to reduced corrosion rates which will allow increased outlet water temperatures and higher power levels.

Pile Graphite Studies

Data, obtained from a number of in-pile as well as laboratory experiments, provided the basis for operating C Pile under test conditions to evaluate operation at 600 C maximum graphite temperature.

In conjunction with the above test, a full pile test at D Pile to determine the effect of high helium concentration in the pile gas on the stack distortion was completed. As a result of this test, process specifications were changed to permit helium concentrations up to 55 per cent at B, D, F, and 65 per cent at DR and H Piles. As a result of this change, no pile power level is currently limited by graphite limits.

Bowing traverses, used to monitor over all moderator distortion, have been obtained at all piles and indicate a continued contraction of the central zone. Tube curvatures, calculated from these data, continue to become more restrictible. Traverses and probing tests indicate the need for four-inch long fuel elements in the upper regions of F Pile.

The monitoring system devised to evaluate oxidation of the graphite stack has continued to confirm the low oxidation rates observed for operation under present Process Specifications 31:00 and 41:00.

Evaluation of stack damage at H Pile and KW Piles has resulted in a recommendation as to the maximum force allowable in discharging stuck ruptures and/or process tubes.

A compilation and evaluation of graphite temperature data from all piles and in particular from C and K Piles emphasizes the need for adequate graphite temperature instrumentation. A program for testing new designs, materials, and techniques has been initiated.

 [redacted]

Graphite and Materials Development

The Effect of Irradiation on Graphite Physical Properties - Irradiations of many types of graphite were performed throughout the year under a variety of experimental and operating conditions. Study of the behavior of these graphites resulted in correlations of irradiation effects with raw materials, manufacturing variables, and irradiation conditions.

For example, a series of graphites graphitized at temperatures from 1800 to 2700 C have been irradiated both in cooled test holes and at ambient pile temperatures. Physical distortion rates, crystallite growth rates, thermal and electrical conductivities, and coefficients of thermal expansion have all been determined as functions of graphitization temperature. A similar study was completed for graphite samples ranging in density from 1.31 to 1.83 g/cc exposed to 2880 MD/CT in cooled test holes.

Control of physical properties of graphite has been found possible by alterations in raw materials and manufacturing processes. The cooperative program with Battelle Memorial Institute has continued to provide new graphites with remarkable radiation stability even up to 3000 MD/CT. Graphites made from Korite asphalt coke and phenol formaldehyde resin, both characterized by low density, are stable under low temperature irradiation. Burnout in a moist atmosphere is higher than normal, but changes in binder material give higher density graphites which are still stable but less subject to oxidation. The preliminary technology for extruding these materials has been developed. A relationship has been found between stability and crystallite "c" dimension that indicates 30 Å to be the crystallite size characteristic of graphite dimensionally stable under low temperature irradiation. Graphites with large crystallites, greater than 300 Å, are predicted to be stable under high temperature irradiation.

The effect of irradiation temperature is also under study through the use of a variety of irradiation facilities. Physical properties changes of pile grade graphites have been observed to exposures as high as 5,240 MD/CT in cooled test holes. Simultaneously, exposures in empty process tube channels and hot test holes have given data at temperatures between 300 and 500 C with exposures to 4000 MD/CT. An uncooled test hole facility was installed at C Pile and charged with both experimental and pile grade graphite samples. These are being irradiated at ambient C Pile temperatures and atmosphere.

In order to obtain radiation damage data for proposed graphite temperatures a facility was designed, constructed and test-operated in the MTR L-42 position. Irradiations can now be performed at temperatures up to 1000 C and fluxes ten times those in Hanford piles.

All unclassified data on irradiation of graphite were accumulated into a paper that was presented by a Hanford representative to the Geneva Conference on the Peaceful Uses of Atomic Energy.

Physical Properties Measurement - Several hundred graphite samples were measured for changes in physical properties as a result of irradiation. The backlog of samples awaiting measurement was reduced almost to zero. A new faster thermal conductivity apparatus was tested and put into use. The stored energy calorimeter was made capable of measurements to annealing temperatures of 1250 C, and a refined model which will allow routine operation has been designed.

Annealing of Graphite - Methods of annealing the distorted fringe graphite in the Hanford piles were investigated through the use of J-N and J-Q loadings under production tests. While annealing was observed, the use of these methods is being withheld until the testing of heat flux distortion methods is completed.

Thermal annealing of cooled test hole samples was performed in a laboratory furnace at successively higher temperatures. The annealing of samples exposed up to 2500 MD/CT caused them to contract below their pre-irradiation length. Maximum contraction occurred at 1400 C. Between 1400 and 2300 C a minute expansion was observed.

The kinetics of the annealing process have been studied by observing the shift of the C_o X-ray peak with time as annealing occurs. Activation energy spectra of the C_o damage have been calculated by applying the Vand method of analysis. Because long exposure samples present extreme experimental difficulties, pulse annealing techniques are being studied for the extension of this work.

Alternate Pile Atmospheres - Production tests have been performed to determine the feasibility of alternate pile atmospheres. Mixtures of CO with CO_2 have been studied for the purpose of reducing the graphite oxidation rates. Preliminary results indicate the use of CO to be an unsatisfactory means of reducing oxidation. Nitrogen has been investigated for use as a temporary poison during startups. Preliminary data show no adverse effects of using nitrogen from a pile materials standpoint since the corrosion of aluminum and steel coupons exposed to water-saturated nitrogen in the pile during the test was not excessive.

Plastic Materials for Radiation Service - Literature surveys were conducted to determine those plastics and elastomers that could be used in radiation zones without losing their desirable properties. Experimental testing of those which appear feasible has begun but results are not yet available. These materials are needed for flexible electrical insulations and various types of hydraulic seals.

Recirculation Technology

The 105-D Flow Laboratory was shut down in March 1955 and development activities were transferred to the newly completed 1706-KE Water Studies Semi-Works.

The new Semi-Works provides extensive facilities for mock-up and experimental in-pile testing of a wide range of water qualities for corrosion and film deposition effects. Acceptance testing, training of operating personnel, and start-up of the facility were accomplished during the year. However, continued restriction of the KE Pile power level during the year (for operating-safety reasons) has limited the temperature and flux density at which the 1706-KE experimental in-pile tubes are operated.

Information obtained from the in-pile H Loop facility contributed significantly to the field of recirculation technology. The loop operated with demineralized water at 200 C outlet temperature, using standard fuel elements in a zirconium process tube. Seven runs were made at exposures up to fifty-two days. Reliable corrosion data under actual operating conditions were thus obtained. Information relating to activity problems in recirculation systems was obtained by correlation of operating data.

Although a fuel-element rupture occurred in the recirculation tube resulting in some contamination of the 100-H Area environs and gross contamination of the H Loop equipment, chemical decontamination of the loop was accomplished effectively and the loop was returned to experimental operation. The rupture demonstrated that with the present fuel element configuration sufficient time is available for safe shutdown of the reactor at an outlet water temperature of 200 C.

Further exploration of rupture effects in high temperature systems was conducted by means of a series of out-of-pile simulated pinhole rupture tests. Thirty rupture tests were performed, including solid, cored, internal-externally cooled, cluster, and wafer-configuration fuel elements. In addition to massive uranium, several other core materials were examined with this technique, including uranium-magnesium matrix, uranium oxide, and thorium cores. These tests demonstrated the rapidity and potential severity of rupture effects of the various elements in high temperature demineralized water.

Several out-of-pile corrosion tests were conducted as back-up data for in-pile experimentation and to develop fundamental information on aluminum corrosion. In high purity water, a pronounced effect of 2S aluminum corrosion reduction was measured in going from pH 6.5 to 5.5.

Construction of the 1706-KER Recirculation Test Facility was approximately 70 per cent complete at year's end. A test program for the facility was developed, and a preliminary operating manual was prepared. A study was completed to determine modifications required to a KER Loop to permit extended in-pile boiling tests.

 [redacted]

PHYSICS RESEARCH SUB-SECTION

Theoretical Physics

During the past year, a major portion of reactor core physics has been reformulated in terms of the neutron blackness concept. Since neutron blacknesses can be accurately calculated through mathematical methods developed here at Hanford, this reformulation has resulted in an improved quantitative ability for reactor analysis and design.

A major success of these new methods was the calculation of the dry temperature coefficients of the Hanford reactors. These results portrayed a coefficient which increases in magnitude with exposure, i.e., with Pu-239 content. It was further shown that even under present operating conditions this effect is considerable. Experimental tests conducted through 1955 at KE pile have tended to confirm the theoretical predictions.

An additional achievement of the blackness approach to reactor core calculations was the development of a method of calculating resonance escape probability from basic neutron cross-section data alone. Resonance escape probability measures the probability of a neutron's capture in U-238 while slowing down, and is thus intimately connected with the plutonium productivity, and reactivity, of a nuclear reactor.

A perennial Hanford problem concerns itself with the interaction, through neutron leakage, between depositions of fissionable material physically separated in space. This problem arises in chemical plant operation where the occurrence of a neutron chain reaction is intolerable. During the past year, a mathematical means has been perfected to calculate the criticality conditions for interacting systems. Existence of this means will allow safe 200-Area operation without the necessity of restricting efficiency through over conservatism.

Work directed towards a basic knowledge of irradiation damage to materials was conducted. In particular, a more accurate value for the screening constant of the screened coulomb potential was derived. This is a parameter, extensively studied in the literature, which is related to the rate with which an ionizing particle transmits energy to the medium it is traversing.

Experimental Physics

Lattice Testing Reactor - The Lattice Testing Reactor became a chain reacting unit on October 25, 1955. At that time the fuel inventory in the reactor was 9.1 kilograms of U-235. Following startup, a series of experiments were conducted with the LTR to determine the operating characteristics of the reactor and the worth of the various safety features incorporated therein. These experiments included measurements of control rod strengths, safety disk strengths, rates of change of reactivity associated with the removal of one face of the reactor, sensitivity of the reactor to additions of fuel, pressure coefficient of reactivity, and the lifetime of prompt neutrons in the reactor.

Laboratory studies were conducted to facilitate the design of an oven to heat the core of the LTR. This design was completed and an oven was fabricated for this use. The oven will be an integral part of an LTR experiment designed to measure the temperature coefficient of Hanford piles operated to exposure levels of 1000, 2000, and 4000 MWD/T.

Thermal Test Reactor - An oven to heat the TTR thermal column was designed. This oven is to be used to study the change in the distribution of energy of the neutrons which enter the uranium upon loss of water. Such information is needed to predict the magnitude of the reactivity gain which would be encountered upon loss of water from one of the production piles.

Nuclear Physics - A measurement of the neutron absorption properties of Neptunium 239 was made as a joint experiment with Chemistry Research. This study was needed to determine whether the amount of Pu-240 formed in the production piles through capture of neutrons by Np-239 would increase if the flux level in the piles were to be raised. The result of the experiment showed that this particular nuclear process will not increase the Pu-240 concentration in the plutonium product if power levels are raised. In the course of doing this experiment, two new gamma rays were found in the decay scheme of Np-239. This result was published in the "Physical Review".

Measurements of relative values of ν^{25}, ν^{49}, and η^{49} were made as a function of incident neutron energy using the crystal diffraction neutron spectrometer. These quantities are respectively the number of neutrons per fission in U-235, the number of neutrons per fission in Pu-239, and the number of neutrons per capture in Pu-239. The quantities are important in studies of temperature coefficients of graphite-moderated piles. The results showed ν^{25} and ν^{49} to be constant over the energy band covered. A variation in η^{49} was observed in the vicinity of the 0.3 ev resonance in Pu-239.

A paper on "The Total and Fission Cross Sections of Plutonium" was prepared by B. R. Leonard, Jr., for presentation at the Geneva Conference. Contributions were made to four other Geneva Conference papers presented by other authors.

[redacted]

Exponential Physics

Experiments during the year 1955 involved the use of natural uranium fuel elements, enriched uranium (1.007% by weight U-235) fuel elements, and completely enriched (aluminum-U-235 alloy pieces) C slugs.

Solid natural uranium slugs of diameters 0.925", 1.175", and 1.36" were used; also a cored 1.36" slug with a 1/2" hole. (The cored slug is of interest for reducing the slug rupture problem since it should be subject to less stress than a solid slug.) The application of lattice theory was applied to the 0.925" and 1.175" slugs where the parameters were "carried over" from the earlier correlation of experiment and theory using the 1.36" slug. The agreement between predicted and measured values was quite good although minor discrepancies exist between theory and experiment.

Several measurements were taken to obtain more data on the effect on the critical buckling of a lattice from increasing the water annulus size.

A series of experiments were undertaken in the small 4' x 4' exponential piles using C slugs (Al-U-235 alloy fuel elements) in order to obtain further data on the reactivity behavior of lattices which use enrichment and also to obtain additional information for testing the lattice theory as applied to these cases. It is of interest to note that it is possible to define a safe lattice, i.e., one that loses reactivity upon the loss of coolant even though the fuel elements are completely enriched. For the C slugs, the "cross-over" point occurs at a lattice spacing of about 7.5" and the buckling is 830×10^{-6} cm^{-2}.

An experimental and theoretical study was made on neutron streaming in air channels. The results of this study indicated the validity and necessity of applying streaming corrections to the migration area in order to predict the measured critical bucklings of the operating piles.

In view of the increased emphasis on dual purpose power reactor designs, measurements were undertaken using cluster-rod fuel assemblies. The buckling was determined for a seven-, six-, and five-rod cluster of 0.925" rods in a 3" tube where the lattice spacing was 14". Cluster assemblies are difficult to treat from a theoretical standpoint. Preliminary results indicate that the main effect in assembling the bundles over a single rod is to increase the resonance capture. These measurements are being extended to other lattices.

A series of exponential experiments were made with enriched uranium rods (1.007% by weight U-235) in light water. These experiments were designed to yield the necessary data required to fix critical limits for plant dissolvers. Two different rod sizes were used: 0.925" and 1.66".

Four different water-to-uranium-volume ratios were used with the 0.925" rods and three with the 1.66" rods. From the measurements the maximum critical buckling for the 0.925" rod was estimated to be about 3450×10^{-6} cm^{-2}; that for the 1.66" rod at about 2750×10^{-6} cm^{-2}.

It was also observed that water and uranyl nitrate solution (uranium enrichment 0.68%) are essentially equivalent reflectors.

The fission resonance integral of U-235 was measured in the exponential piles by means of a small fission chamber. From this measurement it was estimated that about 5 to 6 per cent of the total fission in a typical Hanford lattice is produced by epi-cadmium neutrons, i.e., by neutrons of energy greater than 0.5 ev. It was concluded that the assumption that U-235 behaves like a 1/V absorber will be adequate for use in most Hanford type lattice calculations.

A summary paper "Exponential Experiments in Graphite Systems" was submitted to the Geneva Conference.

Experimental Reactors

The 305-B Building, to house the Lattice Testing Reactor and the Thermal Test Reactor, was completed in the spring and assembly of the reactors was then begun.

The LTR electrical and mechanical components were installed and tested and fuel loading was completed to reach critical on October 25. Operators were assigned and training initiated before startup. For the training program and for establishing operating conditions, the LTR Manual was prepared and the sections issued by the end of the year covered: Operating specifications, Startup Procedure, Check Sheets for pre-startup, startup, and shutdown, Safety Rules including fuel storage specifications, Radiation Monitoring Procedures, and Temporary Revision to the Specifications authorizing specific changes required for particular experiments.

Additions or changes to the building equipment included the installation of a remotely controlled personnel safety barrier, revisions to the ventilating system and mechanical interlocks on the trip circuit controller which limit the operating range between low and high trip within a preset maximum. Plans were prepared and wiring started to enable heating the LTR core and the TTR thermal column.

The TTR mechanical components were erected and wiring between Reactor and Control Room was started. Fuel allocation was received, specifications prepared and off-site manufacture of the elments was begun.

[redacted]

DECLASSIFIED LASSIFIED

METALLURGY RESEARCH SUB-SECTION

Fundamental Studies

Forty-eight specimens of uranium irradiated to 130 to 512 MWD/AT (8.8 to 35.4×10^{19} nvt) were examined to determine the dimensional instability as a function of exposure and of degree and type of preferred orientation. Specimens with predominant (010) orientation increased in length and decreased in diameter. Specimens with a duplex (010)(110) orientation with the (110) slightly predominant were essentially stable. Specimens with either (110) or (100) type orientation shortened in length and increased in diameter. Post-irradiation x-ray examination revealed that rho values for (020) planes tended to increase and values for (110) planes decreased during the irradiation.

Three capsules containing pre-characterized metallographic samples of uranium with NaK as the heat transfer medium were exposed in the MTR. Specimens of Zircaloy-2 and uranium with metallographically prepared surfaces were exposed with NaK at 600, 700, and 800 C for three weeks and extensively studied to aid in the interpretation of structures observed after irradiation. Studies were continued on the use of Faxfilm replicas for the metallographic examination of radiation damage in uranium. A replica was successfully prepared of a uranium specimen that had received 580 MWD/T and was examined employing both the optical and electron microscopes. Specially constructed manipulators have been constructed to aid in these studies.

Ex-pile diffusion studies of the U-Al system in the temperature range 200-390 C were completed. Initial U-Al diffusion couples were irradiated in the MTR to a total atom burnup of 0.02 per cent at approximately 275 C and are under examination to determine the extent of diffusion. Ex-pile diffusion studies of the systems U-AlSi and U-Zr have been continued in support of irradiation tests.

A series of experiments were undertaken to determine the diffusion rate, activation energy, and mode of diffusion of krypton through silver and uranium. These experiments will serve as a basis for the interpretation of fission product mobility. Assembly and installation of equipment for these studies was completed. Bombardment with krypton ions employing a cylindrical silver cathode proved successful in providing a uniform distribution of the gas in silver. A sensitive method of measuring very small quantities of krypton was developed employing a gamma spectrometer. Measurement is made of the 0.150 Mev gamma radiation generated in krypton by neutron bombardment.

DECLASSIFIED [redacted]

The deformation and fracture modes in uranium are being studied employing motion pictures taken during controlled tensile deformation and metallographic examination at different stages of deformation. Room temperature deformation of beta heat treated dingot uranium was found to be predominant by twinning rather than by slip and the fracture is transgranular. Alternate tensile and compressive loading was found to generate and eliminate twins in a cyclic fashion.

Mechanical and Physical Properties of Fissionable Materials

An elevated temperature tensile test unit for use with highly radioactive materials was installed in the Radiometallurgy laboratory. The unit consists of a specially designed vacuum furnace installed on a Baldwin universal testing machine and an optical strain measuring system. The unit enables tensile tests to be made at temperatures to 800 C with control to ± 1 C over the specimen gage length. Elongations to 50 per cent may be measured with the optical strain measuring system. Employing this unit, the tensile properties of uranium, irradiated to 1.8×10^{20} nvt (620 MWD/T), were determined at 285 C. The strength values obtained were nearly the same as for material with the same exposure tested at room temperature. The elongation to fracture was only slightly more than for irradiated material tested at room temperature. The elongation observed was 0.7 per cent as contrasted to an average of 33 per cent for non-irradiated, beta heat treated uranium at 285 C. A tensile specimen of uranium irradiated to 1.8×10^{20} nvt (620 MWD/T) and vacuum annealed at 800 C was tested at room temperature. No property data were obtained from the test. Pronounced faceting, typical of a cleavage fracture, was observed. Tensile specimens irradiated to 1.4×10^{20} nvt (475 MWD/T) at 125-150 C and to approximately 1.8×10^{20} nvt (620 MWD/T) at ca. 300-400 C in the MTR in NaK-filled capsules were tested at room temperature. These data indicate essentially the same drastic decrease in ductility, decrease in ultimate strength, and increased yield strength observed for specimens exposed at HAPO to 1.8×10^{20} nvt at 125-150 C. Tensile specimens in NaK filled capsules have been irradiated in the MTR to approximately 700 MWD/T at an estimated temperature of 800 C. Irradiation of mechanical and physical property test specimens of uranium at HAPO to 0.4, 0.85, and 1.8×10^{20} nvt was completed.

Tensile specimens of U-0.35 a/o Cr, U-0.5 a/o Ti, epsilonized U-23 a/o Si, and natural uranium were irradiated in the MTR in NaK-filled capsules to about 2.5×10^{20} nvt.

Uranium electrical resistivity and tensile specimens were irradiated to approximately 1×10^{16} and 1×10^{17} nvt at 50 C. The electrical resistance of one specimen irradiated to 1×10^{17} nvt was measured over a period of five days following exposure. No decrease in resistance indicative of recovery of damage was observed.

[redacted]

Measurements made within the thermal conductivity apparatus have been reproducible with ± 5 per cent of a calibrated standard over the range 20-200 C.

Reactor Structural Materials

Continued research has led to a firmer understanding of the potentialities and limitations of zirconium and zirconium alloys for reactor components. A study of the effect of neutron irradiations to 5.7×10^{19}, 1.5×10^{20}, and 2.4×10^{20} nvt on the tensile properties of Bureau of Mines zirconium in a range of cold work states was completed. Increases in strength and hardness and moderate decreases in ductility were observed. These effects were essentially saturated in the exposure range studied. Recovery of this damage was determined by post-irradiation annealing and from this study the influence of exposure temperature could be approximated.

The notch bend test was extensively studied and proven a satisfactory method for investigating embrittlement of zirconium and zirconium alloys.

Studies of the reaction kinetics of sponge zirconium and Zircaloy-2 with dry air at 500, 600, and 700 C were completed. Extrapolation of these data to lower temperatures shows that the service life of structures fabricated from these metals amounts to several years at temperatures below 400 C.

Zirconium and Zircaloy-2 process tube sections exposed to pile atmosphere at 410 C to 10.4×10^{19} nvt (345 MWD/T) displayed weight gains lower than those observed in out-of-pile tests. Specimens of zirconium and Zircaloy-2 were irradiated in air approximately four months in a controlled facility operating at 350-450 C to determine weight gain, dimensional changes, and notch bend characteristics. Similar specimens of zirconium and Zircaloy-2 were exposed fourteen months in normal pile atmosphere at 410 C.

Impact tests were completed on Zircaloy-2 specimens containing about 130 ppm of hydrogen that were exposed in and out of the flux field of a 150 C recirculating water loop. Exposure to an integrated neutron flux of 1.9×10^{10} nvt shifted the transition temperature 35 C. This exposure did not change the impact properties of material containing only 20 ppm of hydrogen.

Investigation of the creep properties of cold worked Zircaloy-2 was initiated. Second stage creep rates were determined at two stress levels at 300 C. A phenomenon similar to strain aging was observed.

Fuel Materials

The uranium-magnesium matrix fuel material was subjected to pile exposure and the excellent radiation resistance proven. Data were obtained for specimens of this material exposed to 1000, 5000, and 10,000 MWD/T in the MTR. Additional specimens were discharged after 20,000 MWD/T exposure and elements of larger geometry had reached exposures of 3500 MWD/T at year's end. This fuel material is nearly ideal for high burnup reactor application with the exception of its aqueous corrosion behavior. Uranium-magnesium samples corrosion tested in boiling (255 C) "Dowtherm A" (an eutectic of biphenyl and diphenyl oxide) remained clean and bright after two weeks, indicating the possible application of organic coolants.

Studies of uranium oxide as a fuel material were continued and a broader program aimed at evaluating ceramics and cermets was initiated. Sixteen capsule specimens of compacted UO_2 were irradiated in the MTR in an experiment designed to study the dependence of core behavior on flux, exposure time, packing density, and 235/238 ratio at temperatures near the melting point of UO_2. Experiments with swaging as a method of fabrication of uranium oxide fuel elements were conducted. Uranium oxide, tamp-packed in stainless steel tubes with welded end plugs, was cold-swaged, resulting in oxide densities as high as 9.9 gm/cc.

Included in fuel-bearing materials investigated for higher temperature, higher power, and long exposure operating conditions are thorium-uranium alloys or ternary alloys of thorium, uranium, and zirconium. Zirconium additions of 6 w/o and 10 w/o did not effect a decrease in the solid solubility of uranium in thorium. Specimens of thorium - 2 w/o U-235 alloy were charged in the MTR for exposures as high as three per cent burnup to determine irradiation effects on the mechanical properties and microstructure of this fuel material.

Processes for the reduction of uranium halides and oxides to metal for the recycle of depleted HAPO uranium were studied. Promising yields of metallic uranium were obtained by fused salt electrolysis of uranium salts prepared from UNH solutions and initial success was obtained in producing metallic uranium particles by electrolysis of U_3O_8 in molten calcium fluoride.

Studies were conducted on methods of producing uranium shot or pellets for use in the matrix fuel element. Initial success was obtained in the preparation of uranium shot by decomposing and sintering slip-cast pellets of uranium hydride and by calcium bomb reduction of UO_3. At the close of the year experiments were under way to determine the feasibility of producing uranium metal shot by pouring molten metal on a rotating graphite disc.

[redacted]

Fuel Element Evaluation

A cluster fuel element design was proposed that is made up of a bundle of small rods, individually cooled, and held in a cluster by spacers. Test elements were fabricated for flux measurements in the 305 Pile, testing in the MTR, and testing in the KE reactor recirculating loop.

A coaxial tube fuel element design has also been studied. This design consists of concentric uranium tubes of appropriate dimensions to have complete elastic behavior in the outer tube and complete plastic behavior in the inner tube. A preliminary stress analysis for this design was completed. The importance and effects on fuel element temperatures of a gap between the fuel piece and the cladding was more clearly pointed out by this study, and experimental work is planned for investigation of the effects of this design variable.

The fuel element testing facility at the MTR was employed throughout the year in support of fuel element evaluation programs. A basket-type rechargeable B-block was designed and fabricated which permits re-use of the B-block and greatly simplifies removal of slugs distorted by dimensional instability or rupture.

Unbonded fuel elements, canned by the room temperature point closure technique, of natural and enriched uranium in both the solid and cored geometries were irradiated to determine their rupture resistance and to check the necessity of a bond for heat transfer requirements. Examination of solid unbonded slugs irradiated to 1000 MWD/T reveals no evidence of preferential corrosion at the closure nor of non-uniform heat transfer. A slight amount of U-Al interdiffusion was observed at the jacket-uranium interface. The uranium of a cored, unbonded slug, irradiated to 200 MWD/T was found to be cracked longitudinally without noticeable localized deformation of the can wall.

Insulated fuel elements were subjected to irradiation to determine the in-pile stability of uranium slugs operating at elevated temperatures. Two elements were exposed in the MTR fuel element testing facility to 850 MWD/T at 80 kw/ft. This higher than anticipated power resulted in calculated surface temperatures of 480 C and a maximum temperature of about 1000 C. Measurements made on these slugs after each operating cycle indicated that after initial start-up the slugs were dimensionally stable. Post-irradiation examination showed that the uranium was unchanged in overall dimension, although the 3/8-inch axial core was completely closed. The insulating layer of anodized aluminum appeared to have suffered no serious deterioration. A second irradiation of insulated elements was stopped after eight hours of full power operation when the high power level in the facility was verified. Examination of these elements substantiated calculations that some melting of the core would occur.

[redacted]

Cored fuel elements, one with a longitudinal notch on the surface and one with a longitudinal notch along the core were irradiated to rupture in the MTR fuel element testing facility to test the effect of stress-raisers on fuel element performance. Preliminary examinations indicate that the prefabricated stress-raisers promoted the early (120 MWD/T) rupture. A group of longitudinally notched, cored, unbonded slugs were fabricated for irradiation to study the nature and consequences of split-type failures of unbonded fuel elements and to test the relative importance of internal and external surface stress-raisers.

A pressurized water loop was installed to permit experimental study of the kinetics of the water-uranium reaction at high water and metal temperatures. Initial experiments indicate that the reaction between 300 C water and 660 C bare uranium metal is considerably less severe than had been anticipated. In subsequent tests, corrosion rates were determined in 200 and 300 C water with both unheated metal and with metal preheated to 700 C. The rates were about 0.1×10^{-3} and 1.5×10^{-3} inches per minute penetration, respectively, for both unheated and preheated metal specimens. A series of tests were conducted using canned, unbonded specimens that had been defected. The attack was much more rapid and non-uniform in these specimens due to attack by hydrogen.

Radiometallurgy Examination

The facilities of the Radiometallurgy laboratory were employed to capacity throughout the year for fuel element examination and for detailed metallurgical testing of irradiated materials.

Additional equipment installed during the year included the elevated temperature tensile test unit; apparatus for determination of fission product distribution; apparatus to measure rate of evolution, radioactivity, and amount of gas produced when an irradiated sample is heated in a vacuum furnace; and a parting lathe and associated equipment to aid in opening production test capsules and to permit the handling of NaK-filled capsules.

An entire slug column and process tube from the HAPO KW reactor start-up incident was examined to determine the slug and tube damage, the processes resulting from the heating, and the temperatures that resulted throughout the slug column.

Separations Plant Structural Materials

The major portion of investigations during the year were concerned with corrosion problems with materials of construction in the Redox and Purex separations plants.

Factorially designed static corrosion tests and heat transfer corrosion tests were conducted upon types 304L, 308L, 347, and 329 stainless steel in boiling 2WW Purex waste acid concentrate. Quantitative heat transfer corrosion data indicate that type 308L is as good as, or superior to, type 347 at average metal temperatures between 110 and 160 C. A pressurized heat transfer unit was designed and constructed in an attempt to correlate corrosion rates obtained from the factorially designed static corrosion tests with rates obtained from heat transfer units. Specimen temperature, solution temperature, and heat flux may be separated and evaluated as factors in the corrosion of heat transfer surfaces. Initial experiments with this unit indicate that corrosion is a function of the surface temperature of the metal and is independent of the heat flux through the metal.

Data were obtained that show the behavior of type 347 stainless steel during exposure to boiling synthetic Purex 2WW waste acid concentrate under heat transfer conditions. Over the temperature range 115-165 C, type 347 weld metal exhibits greater general corrosion resistance to this media than does wrought type 347 stainless that has been given a sensitizing heat treatment.

A Hot Semi-Works field corrosion test was initiated to be conducted in the self-concentrator containing neutralized Purex waste solution. 1020 steel and two low-alloy, high-strength steels, Mayari-R and Corten, in both the stressed and unstressed condition were inserted for an estimated two years' exposure. AISi types 318, 309, and 304 sintered stainless steel filters were tested for corrosion resistance in synthetic 231-Z Building process streams. No significant differences in resistance were observed.

An investigation of several possible corrosion inhibitors for use in the Purex acid fractionators was performed, using Purex 2WW waste acid concentrate as the corroding media. AISi type 304L stainless steel that had been given a sensitizing heat treatment was employed with the following inhibitors: 0.2M, 0.1M, 0.01M, 0.001M urea; oxygen sparge; helium sparge; and air sparge. A slight inhibitory effect was noted in the cases of 0.2M and 0.1M urea, and a slight increase in corrosion rate was observed using the oxygen and air sparges. Subsequent tests with 0.2M, 0.1M, and 0.01M urea in boiling 65 w/0 nitric acid did not show any significant reduction in corrosion rate.

 [redacted]

A study was initiated of the corrosion of types 304L, 347, and 312 stainless steel and type A-55 titanium exposed to liquid and vapor nitric acid compositions in a mock-up nitric acid fractionator.

The effect of carbon content upon the corrosion rate of type 312 stainless steel in nitric acid was evaluated. Samples from five heats of type 312 stainless steel, with carbon contents from 0.027 to 0.039 w/o, were exposed to solutions of 40 and 65 w/o nitric acid at temperatures of 80 to 165 C. Carbon content appeared to have little or no effect on the corrosion rate of the material.

Several pile graphites were found suitable for process lubricated bearings from a standpoint of resistance to chemical attack. Tests made of the effect of gamma irradiation on the chemical resistance indicated that these materials cannot be exposed to nitric acid after irradiation with any degree of success.

Samples of Resistoflex connectors were tested in boiling 42 per cent $MgCl_2$ to determine the susceptibility toward stress corrosion cracking. The tests indicated that this type of connector was extremely susceptible to stress corrosion cracking. Laboratory examination was initiated to study the cause of failure of flange studs used for connections in Purex separations plant equipment.

Welding Development

A joint was developed for joining tubes to tube sheets in heat exchangers which have the corrosive media on the outside of the tubes. The joint effectively eliminates crevice corrosion.

Corrosion tests on weld deposits made using small percentages of oxygen in the argon shielding gas during MIG welding of some 300 series stainless steels indicated no increase in corrosion rate with added oxygen.

A comparison of the cracking tendency of types E-347-15, E-308L-15, and E-308L-16 covered welding electrodes was made. The tests indicated that if the chemical composition of the welds made by an electrode is within the limits of ASTM A-298-55T and the chromium/nickel ratio of the weld deposit is above 1.8, the welds are crack resistant regardless of the type coating used on the electrode.

An extensive comparison was made of three welded pipe butt joints; the "GE" joint, the EB joint, and the conventional bevel joint. The results indicate the distinct advantages of the "GE" joint.

[redacted]

Specimens of Zircaloy-2 were welded by the TIG process under varied welding conditions designed to produce various degrees of shielding gas purity. Results of bend and corrosion tests of these specimens indicate no significant difference in ductility or corrosion resistance between welds made in an argon-filled gloved box and those produced with normal TIG equipment.

A TIG cutting torch was designed and tested that would facilitate process tube removal in graphite reactors. A slot 1/16 to 3/8 inch wide is cut in aluminum tubing at the rate of 60-75 inches per minute. The torch cuts from the inside of the tube and is self-guiding along the ribs.

A "roto-arc" welding unit for welding caps on small cylindrical containers was fabricated. The unit employs a cylindrical electrode and a moving arc driven by a magnetic field. Initial experiments show that rapid, sound closures with uniform penetration are produced.

Plutonium Metallurgy

It would be of great value to have single crystals for study of the kinetics of the alpha-beta and beta-alpha transformations in metallic plutonium. Due to the number of transformations between the melting point and the alpha and beta phases, electrodeposition was selected as the most feasible method for production of single crystals. A number of solvents have been investigated in the search for a suitable non-aqueous ionizing solvent for plutonium halides.

Techniques were developed for the precision casting of alloys of natural uranium, depleted uranium, and plutonium that will be employed as fuel loadings in the PCTR. A procedure for canning the fuel elements in aluminum cans by brazing the cap was developed. Building modifications and assembly of equipment for the production line assembly of these fuel elements were nearing completion at the close of the year.

Low plutonium-aluminum alloys were successfully cast for fabrication to PCTR monitor foils.

Long, small diameter, nickel-coated holes are required for flux monitoring purposes in some U-Pu alloy PCTR slugs. The holes have to be precisely positioned parallel to the slug axis and at varying radial distances. Drilling could not be employed due to the small size required, and extensive investigation was made of methods of casting the alloy around wires of the correct size and subsequent removal of the wire. Success was obtained in casting the alloy around bare 0.014-inch diameter graphite and clay rods and later drilling the rods out.

[redacted]

DECLASSIFIED

Various sizes of electroformed nickel tube were produced for the Experimental Physics Unit. Three feet of tubing 0.022 inch I.D. by 0.025 Inch O.D., two feet of 0.037 inch I.D. by 0.040 inch O.D., and one foot of 0.035 inch I.D. by 0.040 inch O.D. were made by plating nickel on dag-coated copper wires.

[redacted]

DECLASSIFIED

FUEL TECHNOLOGY SUB-SECTION

Process Technology

The number of canned slugs rejected for porosity, much of which is attributed to the higher hydrogen content of the uranium cores received at HAPO in late 1954, has steadily decreased during 1955 to the extent that production outgassing was discontinued, except for dingot uranium, in mid-year 1955. The decrease in surface hydrogen was apparently related to the lower hydroxyl ion concentrations in the FMPC heat treating bath. The hydroxyl concentration was shown to be directly proportional to the absolute humidity of the bath environment. FMPC has reduced the hydroxyl content from 1.6 per cent to less than 1.0 per cent since September 1954 by daily purging the salt bath with carbon dioxide. This operation is sufficient to maintain the hydrogen content of ingot slugs at about 2.0-3.0 ppm (the rod entering the salt bath contains about 1.0 ppm hydrogen). Associated studies revealed: (1) long time outgassing treatments (vacuum annealing) removes the hydrogen but is costly for production use, and (2) induction heat treating reduces the hydrogen content of uranium.

A new method of analysis for hydrogen in slug cores is being developed which will permit measuring the amount of hydrogen evolved from the surface layers of the uranium cores during a heating cycle similar to that of the production canning operation. This measurement is expected to permit better correlation between braze porosity and hydrogen content.

Because a small number of incompletely transformed slugs are being accepted by the ultrasonic transformation test, a more stringent test level was put into effect.

A lead dip canning line was operated for several months with lower maximum bath impurity limits to test the hypothesis that impurities, particularly lead in the canning bath Al-Si, were responsible for Al-Si spikes (Al-Si stringers projecting up through the weld bead in the finished slugs). When the lead content of the canning bath was maintained at less than 0.2 per cent, spike defects were low. Bailing schedules were altered to keep the impurity content low (maximum lead values of about 0.23 per cent).

Processing difficulties were encountered with the use of the mechanized spray type slug pickling equipment provided for the expanded manufacturing facilities. The cores re-oxidized faster than the dip pickled cores, some slug cores were not adequately pickled with one cycle through the machine, and the fumes from the nitric acid exceeded the biological tolerance in the operating area. Corrective measures were taken and results indicate that rates and qualities comparable to those for dip pickling can be attained.

[redacted]

Three methods for sealing the ends of cored slugs prior to canning in the Al-Si bath were developed. The three methods were (1) welding uranium end plugs into counterbored holes in the slug, (2) crimping uranium end plugs into place, and (3) pressing aluminum end plugs into special counter-bored holes. Sizeable quantities of fuel elements closed by all three methods are in the pile. None of the three methods appear to offer significant advantages over the other two from a pile performance standpoint. With one exception, irradiation results do not show any superiority of cored elements over solid elements. The one exception concerns two tubes in the F pile of four-inch drilled cored slugs which have reached an exposure of 2000 MWD/T; the solid elements in the control tubes failed at 1500 MWD/T.

Fifty "donut converter" slugs were fabricated for the Belgian government. These were lead dip canned, four-inch I and E type fuel elements with an outside diameter of 1.440" and an inside diameter of 0.875". Some difficulty was experienced with limited cracking of the compound layer on the inner surface. As there was no non-destructive test equipment available for checking the condition of the individual slugs, a recommendation that the exposure conditions should not be more severe than recent HAPO exposures were forwarded to the AEC at time of shipment.

Bonded uranium$_{235}$-aluminum fuel elements were produced by the hot press method starting in January 1955. The hot press method replaced the "C" Process (for unbonded slugs) as the standard manufacturing process for enrichment slugs. To decrease chattering of these lighter slugs in the process tube, tru-line end contours were formed on the slugs starting in April. About 700 bonded tru-line slugs were charged to the piles by the end of the year.

Process Development

As a further step toward improvement in uranium metal quality, dingot (direct cast ingot) uranium has been prepared by Mallinckrodt Chemical Works. Extensive evaluation of dingot material is underway at HAPO. Out-of-pile test results comparing dingot and ingot uranium revealed: (1) dingot uranium contains 99.94+ per cent uranium; (2) dingot uranium is higher in density; (3) hydrogen contents are two or more times greater in dingot uranium, hence, it is more difficult to can by present methods; (4) the average grain size on a macro basis was as large or larger than regular production uranium; on a micro basis the grain structure was a combination of clusters of small grains intermixed with large grains; (5) the mechanical properties of dingot uranium were comparable to those for ingot uranium; and (6) the orientation (by sonic orientation testing) approximates that which is found in rolled ingot uranium after salt bath beta heat treatment in rod form; however, a critical examination of the data revealed that a small group of slugs (15 per cent of those tested) may not have the irradiation stability of ingot uranium. It was also found that the differences in fabrication histories of the dingot metal had no significant effects on mechanical properties, grain structure, or orientation.

Uranium heat treating variables were explored using two methods of approach; i.e., laboratory scale studies and full scale in-pile testing. The laboratory tests indicated that faster cooling rates (from the transformation temperature) produce a finer grain structure; also, it appears that the particle size and distribution of uranium hydride is related to cooling rate. Subsequent annealing studies revealed that a critical cooling rate from the beta phase is required for recrystallization to take place. Large scale pile testing is being performed on slugs from vertically quenched rods, and material which has been heat treated in both slug and rod form with varying time delays between the heat treating bath and the quench.

Solid and tubular rods for cored and I and E fuel elements have been successfully alpha extruded. Extruded solid fuel elements warped significantly less than alpha rolled production slugs after an exposure of 550 to 750 MWD/T. Cored slugs, produced from extruded tubular elements, have had insufficient irradiation experience for a comparison to be made.

The initial Hanford irradiation of hot press canned fuel elements was completed. Both the solid and cored rupture tubes had failures at about 600 MWD/T (versus the lead dip canned control material which was discharged at about 1000 MWD/T without rupture). Subsequent to the pile charging of the first group of hot press material, development work had produced a hot pressed fuel element which, on the basis of out-of-pile testing, was considerably improved. Prior to the end of the year a second production test for hot pressed fuel elements was approved and the fuel elements were canned. The canned assemblies will be anodized to prevent jacket damage, and tru-lined to prevent misalignment in the pile.

Experiments with direct casting of uranium into zirconium cans were conducted; on some two dozen of these elements, ends were sealed by welding and the elements were charged into a reactor prematurely. One of the elements failed after 4 hours; the second after 15 days. Difficulty has arisen in determining the method of sealing the end closure.

Six tubes of unbonded fuel element assemblies with point closures were irradiated. Half were discharged after 300 MWD/T, due to cleavage type of fracture in uranium; the other half were discharged after 500 MWD/T, due to a jacket failure. Apparently, the jacket failure was due to a combination of diffusion of uranium into the aluminum jacket and a flaw (fold) in the jacket.

[redacted]

Development of New Fuel Elements

In support of experimental programs, engineering studies were made of the physics and heat transfer aspects of two fuel element concepts - the compartmentalized (wafer) type and the internally-externally cooled (I and E) type. Preliminary dimensions, subject to change as the experimental work proceeds, have been established for the I and E slugs for use in Hanford reactors. In the case of solid wafer slugs, the studies have indicated that small amounts of aluminum or zirconium (up to 6 or 7 per cent of the present slug volume) might be used as spacer material without excessive neutron cost. Nickel was found to be too poisonous as a spacer material even below 1 volume per cent.

During the year, equipment for Al-Si canning of I and E fuel elements at reduced pressures was developed. Equipment capable of canning one fuel element at a time at pressures in the range of 10^{-4} microns was assembled and is being tested.

A modified "F" Process (lead dip) is also being developed for canning I and E fuel elements. Using a three piece jacket (cap, can, and tube) about 600 pieces were canned for an irradiation test at C pile. At the end of the year pieces were being processed through the various non-destructive tests.

One tube of hot press I and E fuel elements was charged in C pile to determine outlet water temperatures. The hole water temperature of the fuel elements with the 3/8" internal flow passage was about 130 C compared to 90 C for the annulus water temperature. For this reason, the tube was discharged at an exposure of 150 MWD/T. A second test was performed in which fuel elements with three sizes of internal flow passages were monitored for temperature balance; anomalously, the element with the 3/8" hole gave the most satisfactory temperature distribution.

During 1955 wafer (segmented) fuel elements were designed, assembled, evaluated out-of-pile, and irradiated in the Materials Testing Reactor. It was established, by various out-of-pile tests, that wafers 1/8" to 3/16" in thickness are optimum for this type of fuel element. MTR in-pile tests revealed that these elements after failure, deteriorate by transverse splitting. This indicates that with this design the hazard of blocking water flow in a process tube is minimized. The type of failure observed is independent of type of assembly, i. e., with or without aluminum spacers between uranium wafers, thickness of wafers (1/32", 1/8", or 3/16") or type of canning (lead dip or single dip). Two dip canned and one hot pressed fuel element were irradiated in the MTR at a power level of 80 to 90 KW/ft and to an exposure of 800 to 900 MWD/T. The hot pressed element failed. All three of these elements were returned to HAPO for study.

o

 [redacted]

Uranium wafers for an eight tube production test were ordered from National Lead Company. Prior to the placement of this order, all work was performed with wafers made at HAPO. These wafers were either machined from uranium slugs or stamped from sheet. Heat treating techniques for wafers have been developed. The effect of cold working, due to rolling and stamping, on the grain size and orientation was studied; it was found that this effect can be controlled by heat treatment. Proper shearing tools and dies have been designed and developed.

Methods of supporting fuel elements in ribless process tubes were investigated. These methods include canning in ribbed cans and welded studs, rails, or runners by conventional (arc) welding, ultrasonic welding, percussion welding, or brazing methods. Approximately 300 canned fuel elements were shipped to Argonne National Laboratory where six studs 1/4" in diameter were percussion welded onto the circumference near each end of these elements; subsequently, these elements were inspected at HAPO and autoclave tested. Twenty-four elements were submitted to flow laboratories for flow and heat transfer studies; these tests are currently underway. Also, studies of elements which were made to fail in autoclaves are being made to determine the extent of possible damage to process tubes during in-pile service.

Materials Development

Two tubes of H pile cross-section dimensions and two lengths (6 and 13 feet) of KER cross-section have been produced from Zircaloy-2 by the extrusion and tube reduction process. The extrusion step in the fabrication of this material has been fairly well defined; however, many difficulties are still being encountered during the tube reducing operation. Laboratory experiments indicate that both zirconium and Zircaloy-2 strain harden rapidly. Translation of these data to tube reducing operations leads to the conclusion that the material is being worked to its ultimate capacity.

The Flow Cup Laboratory was shut down in the early part of 1955 after having determined the low temperature (up to 90 C) corrosion characteristics of a group of aluminum alloys. Since present pile outlet water temperature is over 100 C and plans for future operation include even higher outlet temperatures, and since extrapolations to higher temperatures are extremely inaccurate, further Flow Cup Laboratory studies would be of limited value.

Many slugs failed during the year as a result of non-uniform corrosion which was found to contain areas of intergranular attack. It was concluded that the failures were the result of localized high temperatures which, in turn, were caused by misalignment of the slugs in the tubes or by worn tube ribs. A survey of all commercial alloys and of many new alloys made

[redacted]

DECLASSIFIED

specifically to resist intergranular corrosion was completed. Pure aluminum corrodes intergranularly at temperatures below 100 C, 2S aluminum disintegrates rapidly at temperatures above 230 C, but the new alloys M-388, M-400 and X-2219 corrode at a uniform rate even at temperatures above 350 C. Out-of-pile studies gave corrosion rates of 10 mils per year at 350 C for M-388 (1% nickel).

Laboratory, out-of-pile, and in-pile charging studies have shown that anodizing the outer slug surface prevents galling, sticking, cocking, or bowing during normal charging. Slugs slide freely on the ribs and this should also prevent cocking and bowing as a result of thermal expansion and contraction during operation. In addition, anodizing protects the slug surface during autoclaving, handling, storing, and charging which prevents gouges, scratches, and other marks which in many cases lead to accelerated and non-uniform corrosion. Since preliminary production tests of unsealed films had shown that these films washed off rapidly at temperatures above 60 C, many new sealing techniques were tried. A new sealed anodic film was developed which appears to afford increased corrosion resistance and satisfactory abrasion resistance. Equipment for anodizing 100 fuel elements per eight hour day is now in use; equipment for operating at a rate of 650 pieces per eight hour day will be put in service in the near future.

A survey of possible electroplates which might be of value as a corrosion barrier or diffusion barrier for fuel elements was completed. The conclusion was reached that nickel plating from a modified Thompson bath was the best for use with hot pressing. Iron plating appeared to show some promise for zirconium cladding. The work on anhydrous nickel plating baths, copper plates, brass plates, tin plates, aluminum plating, and other procedures was terminated and reported. The plate distribution for fuel elements of various shapes and sizes was studied and tentative methods for securing uniform plate distribution were designed.

The reaction of uranium and water was studied in static autoclaves and a flow loop. The studies showed: (1) no change in mechanism occurs between 25 C and 350 C; (2) no protective oxide of uranium is formed at any temperature up to 700 C; (3) the rate of reaction between 25 C and 350 C is not affected by rate of flow, time, nor initial temperature of the uranium and can be expressed by the equation $\log R = 3480/T + 6.4$; and (4) when canned unbonded uranium is heated to a high initial temperature (300-700 C) and uranium comes in contact with water at temperatures from 250-350 C, rapid swelling occurs.

A technique was developed for grain size determination at 20 X using bright field illumination on a macro etched uranium specimen. The observed grain size is compared with either (1) a grain size chart, or (2) a gridded micrometer eyepiece. This technique results in a greater accuracy, better reproducibility and a shorter time per determination. The method has been adopted by several off-site laboratories.

DECLASSIFIED

[redacted]

Testing Methods and Equipment

The second prototype of the eddy current metal quality tester (MIZ-2) was available for study at the beginning of the year. An excellent corrēlation has been obtained between the instrument readings and those from induction cycling (the woodsplitter). However, an effort (Production Test 7-MR) to carry this correlation over to performance in the pile was inconclusive. To determine whether a correlation does exist, a large scale sorting test is now being carried on.

Development and design of the instrumentation and mechanical components necessary to perform the sonic orientation test on a production basis has begun. The ultrasonic grain size test, which is used to check completeness of transformation, has been re-designed and three more production units have been built. During 1955 the Al-Si penetration test was brought into 100 per cent production with good results. Five production line units were designed and built and are now being used on the two finishing lines. At the end of the year the ultrasonic bond test was being used on a part time basis with reasonably good results. The rejection rate of the equipment was reduced (by installing a count rate circuit) to the point where it will selectively reject only those slugs with unbonded areas having a circumferential dimension of one centimeter or more. A production model of the ultrasonic bond test was designed and five units are being built. Design of an automatic conveyor for the Al-Si penetration test and ultrasonic bond test equipment was completed. Test equipment was developed to detect the presence of canning bath material (lead or Al-Si) in the central void of cored slugs. This equipment uses a cobalt 60 gamma source, a scintillation detector, and commercial instrumentation. Testing methods for Al-Si penetration and bonding on the tube wall of internally and externally cooled slugs have been developed. Adaptations of instruments which were developed for the same test on the external surface are being used. Design of mechanical components is in progress.

Development studies during 1955 included an investigation of eddy current techniques and internal friction in uranium. The study of internal friction was undertaken to obtain information about the effect of irradiation on the structure of uranium.

Irradiation Behavior

Failure of slugs due to core splitting decreased substantially below that of 1954; however, overall slug failure rates were as high as twenty-fold above those experienced in 1954, and failures continue to occur predominately in high power tubes. The major portion of the failures have been attributed to localized intergranular type corrosion of the aluminum jacket and to localized corrosion caused by worn process tube ribs.

[redacted]

Analysis of irradiated slug warp data reveals that warp can be correlated with both average tube exposure and tube power. The per cent of slugs expected to warp 80 mils (necessary condition for stuck charges in the old piles) increases by a factor of ten for each 200 MWD/T exposure. At an exposure of 900 MWD/T a rise in tube power from 500 to 1000 KW doubles the per cent of slugs warping 80 mils.

Routine fracture testing of irradiated fuel elements, including photograph of the fractured surfaces, has better defined the irradiation induced splitting in solid and cored fuel elements. Slugs containing both longitudinal and transverse internal cracks were found in tubes irradiated to high goal exposure (900 MWD/T).

Facilities and Equipment Development

The 306 Building Fuel Element Pilot Plant was occupied during 1955. Essentially all of the development work on uranium metal forming and fabrication and assembly process are being carried out in this facility. Construction is proceeding on the installation of slug pickling, can and cap cleaning, and anodizing equipment. The mezzanine floor of the building is being converted to office space for approximately 40 technical personnel.

Both projects C-431-C and CG-589 were completed for the new 105-C Metal Examination Facility. The necessary revisions to equipment installed in Basins I, II, and III were completed. Operator training and shake-down runs have begun. Revisions to equipment in Basin IV are progressing satisfactorily. It is anticipated that this basin will be in routine operation sometime during 1956.

[redacted]

CONTACT ENGINEERING UNIT

Project Activities

CA-431 - Replacement horizontal rods were received during June but have not been installed.

CA-512-R - Construction of the two K Piles was completed during the year. Numerous modifications were found to be necessary after start-up and such recommendations as required were made by the Project Representatives and forwarded to the Design Council. Most notable of these revisions were (1) Modifications and additions to the Beckman flux monitoring system, (2) Teflon connectors on the front face, (3) Replacement of the temperature sensing elements in the temperature monitoring system, (4) Replacement of the beta effluent monitor with a gamma sensitive system, (5) Addition of Boron steel ball to the Ball 3-X systems, and (6) Revision of auxiliary equipment in the 165 Buildings.

CG-558 - Scope and detailed design were essentially completed during the year. Phase I (B, D, DR) construction was 40 per cent complete and Phase II (F and H) was just starting at year end. There were several minor modifications to the scope of this project during the year as follows: (1) Teflon pigtails were substituted for aluminum, (2) New crossheader check valves were added at DR and H, (3) Filter sampling systems were included, and (4) Zone temperature monitoring was added to the project for budgetory reasons.

The installation of new horizontal rods was completed at all areas and new panellit gages were installed at D, DR and H Areas.

CG-578 and 579 - Gamma monitoring equipment was installed at D, DR and H with installations continuing at the other piles.

CG-600 - There were two major changes in the scope of this project during the year. First, it was decided to install new pumps which would deliver more water with the present motors, because of their higher efficiency, and second, new front face nozzles and pigtails were deleted from the project since it was felt that they would not pay out before they would be replaced with a new design to facilitate tube removal.

CG-642 - Scoping of the continuous charge-discharge project for C Pile was just getting started at year end. No scope drawings had been approved.

[redacted]

RDS Group Representation

Technical representation was furnished on the Mechanical Development and Reactor Development Study groups. Items considered by these groups included zone temperature monitoring, flow monitoring, process tube removal equipment, canning mechanization, charge-discharge, pressurization, and connector development.

Process Studies

Several process studies were conducted as an aid to other organizational components in planning their R and D programs. A study of pressurization established the minimum out-of-pocket costs of plutonium production, as a function of maximum tube outlet temperatures, goal exposures, and corrosion rates, in the older piles. Another study evaluated the economic effects of gross changes in allowable maximum heat transfer rates and lattice spacing on power production costs in a dual purpose reactor. Also investigated was the economic value of incremental power increases.

Special Assignments

The following unrelated assignments were accomplished:

1. Documented the 4669-KW incident.
2. Assisted the Design Planning Unit in preparing the FY-57 and 58 budgets.
3. Furnished technical consultation to visiting power study groups.
4. Established a liaison function with the Phillips Petroleum Company for eventual installation of fuel element testing facilities in the E. T. R. at ARCO.
5. Assisted in preparing a chapter for the New Edition of the Reactor Handbook.

[redacted]

APPENDIX

PILE TECHNOLOGY SECTION ORGANIZATION

Organization and Personnel

Total personnel for the Pile Technology Section increased at years end from 379 to 420. The main reasons were as follows:

(1) The personnel and functions of the Technical Administration Unit were transferred to Pile Technology from Engineering Administration on July 1, 1955.

(2) There was an increase in the number of Rotational Technical Graduates assigned within the Section at December 31, 1955.

(3) A general increase in program emphasis in the Metallurgy Research and Fuel Technology Sub-Sections resulted in additional personnel being added to the rolls as noted under the Section organization totals.

The functional organization of the Section remained constant during the year with the exception of minor changes made in some of the names of the Units as noted in the Sub-Section organizations and the addition of the newly formed Experimental Reactors Unit in Physics Research on June 1, 1955.

Pile Technology Section	12-31-54	12-31-55
Pile Engineering	97	97
Pile Materials	59	64
Physics Research	35	38
Metallurgy Research	72	81
Fuel Technology	110	127
Contact Engineering (100)	4	1
Technical Administration	--	10
Section General	2	2
Total	379*	420**

*Includes 7 Rotational Technical Graduates
**Includes 14 Rotational Technical Graduates

[redacted]

	12-31-54 Totals	12-31-55 Totals	12-31-55 Tech.	12-31-55 Non-Tech.
Pile Engineering Sub-Section				
Physics Development Unit	12	15	11	4
Equipment Development Unit	14	-		
Mechanical Equip. Dev. Unit	-	13	11	2
Heat Transfer Unit	14	16	9	7
Pile Physics Unit	12	11	8	3
Process Technology Unit	18	19	9	10
Special Irradiations Unit	25	21	10	11
Administration	2	2	1	1
Pile Materials Sub-Section				
Graphite Development Unit	11	-		
Graphite and Materials Dev. Unit	-	13	8	5
Pile Graphite Unit	10	-		
Pile Graphite Studies Unit	-	9	6	3
Water Plant Development Unit	20	-		
Recirculation Technology Unit	-	23	11	12
Pile Coolant Studies Unit	16	17	10	7
Administration	2	2	1	1
Physics Research Sub-Section				
Theoretical Physics Unit	4	6	6	0
Experimental Physics Unit	12	16	14	2
Exponential Physics Unit	13	11	6	5
Experimental Reactors Unit	-	3	2	1
Administration	6	2	1	1
Metallurgy Research Sub-Section				
Pile Metallurgy Unit	16	12	11	1
Physical Metallurgy Unit	14	15	11	4
Radiometallurgy Unit	17	21	9	12
Product Metallurgy Unit	7	15	9	6
Corrosion and Welding Unit	7	8	5	3
Metallographic Lab. Unit	8	7	3	4
Administration	3	3	1	2

[redacted]

	12-31-54	12-31-55		
	Totals	Totals	Tech.	Non-Tech.
Fuel Technology Sub-Section				
Fuel Assembly Unit	29	42	20	22
Fuel Element Development Unit	26	26	13	13
Fuel Evaluation Unit	23	26	12	14
Coatings and Corrosion Unit	18	19	10	9
Testing Methods Unit	11	11	7	4
Administration	3	3	1	2
Contact Engineering Unit	4	1	1	0
Technical Administration Unit	-	10	8	2
Section General	2	2	1	1
Total	379*	420**	246**	174

*Includes 7 Rotational Technical Graduates
**Includes 14 Rotational Technical Graduates.

[redacted]