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REACTOR & MATERIALS TECHNOLOGY MONTHLY REPORT
02 PROGRAM - NOVEMBER 1965

F. W. Albaugh

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REACTOR AND MATERIALS TECHNOLOGY DEPARTMENT

MONTHLY REPORT - 02 PROGRAM

NOVEMBER 1965

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REACTOR & MATERIALS TECHNOLOGY DEPARTMENT MONTHLY REPORT

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A. FISSIONABLE MATERIALS - 02 PROGRAM

METALLIC FUEL DEVELOPMENT

N-Reactor Fuel Element Failure

Examination and analysis of the 7th N-Reactor fuel element failure (Tube 2864, 9/3/65) have continued. Metallographic examinations of a suspect support weld, the end cap closure, and an unusual appearing fractured surface of the clad near the cap have not established the site of initial water entry. Several shallow indentations on the fuel side of the inner clad have been noted. These indentations coincide with major zones fuel cracking and outer clad tearing that occurred during rupture. Detailed metallographic examination through one of the indented areas reveals a layer of apparent foreign material that has reacted metallurgically with the clad. Attempts to identify the constituents of this material have not yet been successful.

A cursory examination of the 9th N-Reactor fuel element failure (Tube 2557, 9/26/65) has been made in the Radiometallurgy facility. The rupture occurred at the end of an outer component with considerable damage resulting from fuel fragmentation. There are three major pieces of the component: one piece contains the end cap and was separated from the component by a fracture plane on a 45° angle to the tube axis; the second piece is wedge-shaped, fitting the 45° fracture on the end cap piece and a transverse fracture on the balance of the element; and, the third piece is the balance of the component including the outer supports. The site of initial water entry was not apparent, and the examination is continuing.

N-Reactor fuel failure No. 11 occurred in Tube 1864 on October 25, 1965, in a fuel element that had been irradiated to an exposure of less than 100 Mwd/ton. Basin examination confirmed the failure and revealed a single blister under a support on the outer surface of the outer component. A wire-shaped object appeared to be protruding from under the support. The presence of the wire-shaped object was confirmed in Radiometallurgy, and examination showed that this object had penetrated the clad directly under the outer support by fretting corrosion, thus providing a site for water entry and causing the failure. The wire-shaped object; about 0.2" long, 0.03" wide, and 0.003" thick; was recovered and was identified by gamma spectral analysis as zirconium (probably Zr-2).

Target Element Development

Ninety aluminum canned target elements containing lithium aluminate ceramic cores were completed by the Metallurgy Development Section and delivered to N-Reactor Fuels Section on November 19. One-half of the targets contained sintered pellets. The other half contained vibrationally compacted particles obtained by bulk sintering of the ceramic powder, crushing, and then sizing to obtain the desired particle size distribution.

Thirty-four of the sintered pellet target elements, thirty-four of the bulk sintered and vibrationally compacted target elements, and thirty-four lithium aluminate elements fabricated by Ceramic Research and Development Section were assembled in pairs in Zr-2 cans, 24.4 inches long, by the N-Reactor Fuels Section, and these assemblies were autoclaved at 300°C in 1500 psi water for 72 hours. During autoclaving all of the assemblies sustained some degree of collapse of the Zr-2 cladding. The Zr-2 assemblies containing sintered pellets showed ovality from 3 to 5 mils. The assemblies containing vibrationally compacted particles developed sharp ridges in the Zr-2 with evidence of densification of the ceramic during autoclaving; these assemblies are considered unacceptable for irradiation. Plans are being made by the N-Reactor R&E Section to proceed with the irradiation of one column of 17 Zircaloy-2 clad assemblies (34 Al canned targets) of the sintered pellets.

Plans are being made for a replacement column of sintered pellet targets using a heavier Zr-2 cladding which will be free standing under the autoclave and reactor coolant pressures. A sufficient number of sintered pellets are now available for this replacement change.

In addition to the above target elements for irradiation, twenty aluminum canned elements of sintered pellets were fabricated and shipped to the Savannah River Project for test runs through the extraction process.

Target Element-Closure Weld Development

The use of the alternating current tungsten inert arc welding process for making the circumferential closure welds on the aluminum clad target material has proven to be very successful. Approximately 150 pieces were fabricated with no closure weld rejects.

In order to produce the required quality of circumferential welds, it was necessary to provide a pressure relief hole in the second end cap. A very successful method was developed for sealing the hole which produced approximately 150 pieces without one weld reject. The process involves placing an aluminum pin in the vent hole in a boss and upsetting the edge

of the hole against the pin. The top of the pin is slightly above the bottom of a "V" countersink in the boss on the cap. After the pin is sealed into the hole, the boss is fused into a hemispherical shape by filling the "V" by melting the top portion of the boss and continuing to melt the remainder of the boss with the alternating current tungsten inert gas shielded arc process. It takes approximately one and one-half seconds to fuse the boss. The process is very sensitive to operator response. A much less sensitive method of making the closure is under development and has reached the point that it appears successful. The boss on the cap has a small straight vent hole. A straight pin the same diameter as the boss and having the end that comes in contact with the boss rounded with a radius equal to one half the diameter of the pin is used. By using a magnetic force resistance welder a fusion zone is produced that is at least equal in length to the wall thickness of the boss. Helium leak checking of 25 consecutive pieces has shown no leaks. Metallographic examination of the sections of 35 seal welds has disclosed that the thickness of the fusion zone is approximately 0.010 inch, and no evidence of unbonded or void areas has been found. The pins on 25 caps were broken off, and all the fractures occurred at the base of the boss on the cap.

Alternate Target Element Closure Systems

Closure of the target element vent hole by resistance welding has been investigated and found to be very satisfactory. The vent hole on previous tests has been closed by resistance welding of a small disc over a hole in a flat cap. This time the caps were modified to have a projection and a bullet nosed stud was welded to it.

Initial development was done on a Sciaky 3-phase 100 KVA welder using a forge impulse at the end of the weld. Although good closures could be produced with caps supported on a base plate, the forge force proved to be too heavy for actual elements and dished the ends or otherwise deformed them. A spring mounted low inertia electrode was used with much lighter pressures and the process was successful. A small number of pieces were welded to demonstrate the process before it was adapted to single phase magnetic force resistance welding machines.

Alternate Uranium Compositions

A NaK capsule irradiation test of 120 specimens of 17 uranium base alloys containing small additions of Zr, Mo, Nb, Al, Si, Fe, P, and C is being conducted to evaluate the irradiation performance of uranium alloys in terms of composition, heat treatment, fabrication history, corrosion behavior, and irradiation history. Opening of the NaK capsules and examination of approximately 60% of the fuel samples irradiated to exposures ranging from 2200 Mwd/ton to 3400 Mwd/ton has been completed. Of the 60 Zr-2 clad samples

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examined thus far, 17 have narrow longitudinal splits in the cladding. The splits vary in length from about 1/4 inch to the full length of the 3-inch long samples. Diameter measurements have been completed on the unfailed samples and on the less severely split samples. Irradiation temperatures are now being calculated to determine the relative irradiation performance of the fuel alloys. Volume measurements on all undamaged samples and burnup determinations on a selected few are planned.

Recrystallization temperatures and hot hardness measurements are being obtained on unirradiated samples of certain of these alloys. The object of this study is to determine whether or not the irradiation performance can be correlated with the properties of the unirradiated alloys.

N-Fuel Support Development

The design of supports for N-Reactor target elements was undertaken with the goal of providing a tested support design for use in a 20-tube test. In view of the tight schedule it was judged advisable to adapt the buggy-spring concept for this test rather than attempt to develop a new design. Requirements for the supports are nominally the same as for the N-inner supports, except they are to be used in a 0.190 inch annulus. This reduced annulus places a severe forming requirement on the Zircaloy-2 strip to be used for the support.

The initial information needed on this work concerned the fabrication and ductility of 0.025-inch thick Zircaloy strip as compared to the 0.036 inch thick strip now in use. Experimental strip was formed by rolling a small lot of material using a slight modification of the production rolling schedule. Adequate bond ductility was maintained in the finished 0.025 inch sheet. An investigation of tooling and forming schedules was begun, and test fixtures for status and fatigue tests were completed.

Hot Die Size Fuel Cladding

The temperature and forming conditions used in the "hot die size" (HDS) cladding process have resulted in sharp changes in grain size and hardness in some regions of the fuel cladding. Examination has been made of claddings processed with variations in sizing reduction, end bonding temperature, and prior condition of cladding component to determine process conditions necessary to reduce or eliminate the regions of critical grain growth. The work hardening, recrystallization, and grain growth characteristics of an alternate cladding alloy, HDA-4, 10% Si, 1% Ni, 0.6% Fe were determined.

Improved Closures for ThO₂ Elements

The tools necessary for applying the "cold closure" technique to ThO₂ elements have been procured. A 400-ton press and its immediate area have

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been set up as a ThO_2 control zone in which the closure development will take place. The elements will be loaded with ThO_2 and necessary aluminum caps in production facilities. After closure in the 306 Building, the element will be shipped to 303 Area for sectioning and evaluation.

N-Reactor Meltdown Tests in the IRP

The large amount of radioactivity that was released to the IRP loop and supporting framework during a rupture test with a defected irradiated fuel element has been removed and the loop readied for further testing. The element was defected by induction heating to 1950°F (1065°C) to simulate an N-Reactor loss of coolant accident. Testing in the IRP with high temperature water at 570°F (300°C) caused the bulk of the core to be washed out after about 8 minutes. An entire month was then spent reducing the activity levels in various pieces of loop tubing and fittings, the loop filter, the cell floor and loop framework, cask and dolly, and drain lines coming from the loop. Most of the activity has been reduced to an acceptable level (up to 200 mR/hr) with the exception of the drain lines. These still have readings up to 1 R/hr through 1 ft of concrete and 2 inches of lead. Further attempts to clean the drain lines will be postponed until the entire series of meltdown tests are completed.

Additional meltdown product release tests are planned. Two elements defected by melting have been received during the past month and are scheduled for testing in the IRP.

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CORROSION AND WATER QUALITY STUDIESDownstream Pitting of Aluminum Process Tubes

Severe internal pitting is occurring in a region extending from 1/2 inch to 3 inches from the downstream end of many of the aluminum process tubes in D-Reactor. Sharp, deep pits are located mostly along the tube ribs; the majority of severely pitted tubes are original tubes, but many new tubes are also deeply pitted. The pitted region on these tubes coincides with the location of a Van Stone seal insert (to prevent flange leaks) which restricts flow in this downstream region. Other factors that may contribute to this pitting include: galvanic effects (1100 alloy tube in contact with 356 cast aluminum alloy insert), the water chemistry changes (silicate addition, pH adjustment) that have been tested at D-Reactor, and the frequent use of a diatomaceous earth paste mixture to plug small tube leaks.

It has been recommended that the seal inserts either be removed or re-designed to assure adequate flow in the rear of the tube. Laboratory polarization studies were also suggested, in order to define pitting corrosion parameters.

Preliminary measurements of galvanic potential indicate that the choice of aluminum alloy may influence this pitting. Measurements of relative potentials were made on unetched samples of aluminum, and indicated that 6061 and X-8001 alloy samples were about 10 to 30 mV anodic to 1100 alloy (stainless steel sample was 360 mV cathodic) in pH 6.6 filtered water, but that 6061 alloy samples were 200 mV cathodic to 1100 alloy in de-ionized water (X-8001 aluminum and stainless steel samples remained about 30 mV anodic and 360 mV cathodic, respectively, to 1100 aluminum in deionized water). Galvanic potential and polarization studies of this pitting corrosion are continuing. Samples of the 356 cast aluminum alloy are being obtained for use in these studies.

Hydride Formation in Zircaloy Tubes

Zircaloy-2 samples have been removed after 14 days of exposure in the out-of-reactor TF-20 facility in tests to help determine mechanisms and find control measures for the severe hydriding that has occurred in Zircaloy process tubes in the K-Reactors. Four different pretreatments were used for these samples:

- Etching only,
- Etching plus insertion of aluminum pin, plus scribed defecting,
- Etching plus insertion of magnesium pin, plus scribed defecting,
- Etching plus insertion of zinc pin, plus scribed defecting.

Thin oxide layers (showing interference colors) formed on all of the samples during this exposure. Slight corrosion of the aluminum and zinc pins occurred; magnesium was corroded completely away. These samples are now being processed for hydride measurements.

Groove and Ledge Corrosion of Aluminum

Aluminum samples are being exposed in the out-of-reactor TF-20 facility to filtered Columbia River water at pH 6.6 (with no inhibitor added) to determine the influence of turbulent high-velocity water on groove corrosion of aluminum, and to determine the relative susceptibility of different alloys to this type of corrosion. Flow constricting weir pieces are used to provide turbulent flow patterns in this facility. Initial samples removed after 14 days of exposure show film patterns corresponding to expected flow patterns. Aluminum Alloy 6063 samples show a slight erosion pattern; X-8001 and 1100 alloy samples show no apparent surface effect of flow after this initial exposure.

Pitting of Carbon Steel in Filtered Water

Pitting corrosion of the N-Reactor carbon steel nozzles has been observed. A two-week test was made in TF-8 (an out-of-reactor loop) to further study the severe erosion-corrosion attack recently found on carbon steel in flowing water containing high concentrations of iron oxides (crud). The extreme erosion-corrosion reported in October was reproduced with crud levels of 0.8 to 16 ppm in 540°F (280°C) water moving past test specimens with velocities of 1, 8, 29, 43 and 58 fps. The edges of the upstream coupon in each row of coupons at velocities of 29 fps and greater were eroded to knife edges. Increased velocity increased erosion-corrosion. Velocities of 1, 8, 29, 43 and 58 fps led to upstream coupon weight losses corresponding to 0.37, 0.39, 26, 29 and 40 mils, respectively. Penetrations rapidly decreased with distance away from the upstream sample. One row of specimens, for example, showed weight losses corresponding to penetrations of 28, 15, 17, 0.5, 0.08 and 0.4 mils.

Storage Basin Studies

A laboratory study of the corrosion and precipitation behavior of borax solutions in storage basin water has been completed. This study was conducted to determine the source of flocculent material during the tests to determine the compatibility of 17.6 g/l borax with N-Reactor storage basin materials. The corrosion penetrations of 6061 aluminum, Zircaloy-2, 304, 410, 440-C and 416 stainless steels, lead Oilite bronze, commercial brass, Admiralty metal, A245 carbon steel and weathered concrete were all less than 0.02 mil after 22 days of exposure in filtered Columbia River water treated with borax. The penetration of copper was 0.11 mil. The borax significantly reduced the corrosion of carbon steel as compared to corrosion in filtered water without borax.

Additions of 20 ml concentrated ammonium hydroxide/liter to the borax solutions accelerated the corrosion of copper alloys. Small amounts (0.028 ppm) of ammonium hydroxide in the borax solution did not affect the copper alloy corrosion.

Extensive flocculent material was formed by a reaction of the borax with concrete. Green concrete forms voluminous flocculent material, weathered concrete only small amounts. Commercial borax reacts more aggressively with the concrete than chemically pure borax.

Examination of K Reactor Process Tubes

Examination of hydride levels in the nonirradiated downstream portions of process tubes 3181 KE and 4372 KE has been completed. Process tube 3181 KE was in service for 2 years, process tube 4372 KE for only 5-1/2 months. Process tube 4372 KE was autoclaved prior to reactor service. The downstream five foot portion of both process tubes was sampled for hydrogen at approximately 1 ft intervals. Half inch wide rings were used for hydrogen analyses on tube 4372 KE. On tube 3181 KE four punchings around half the circumference of the tube were cut and analyzed for hydrogen at each 1 ft interval.

Samples from the examined sections of process tube 3181 KE were heavily hydrided. The total hydrogen content diminished as the distance from the rear flange increased. The total hydrogen concentration at 6 in. was 1160 ppm as compared to 260 ppm 60 in. from the rear flange. The hydride concentration appeared to be uniform around the circumference of the tube.

Hydrogen analyses of the tube sections after etching away the hydride case show that the base metal was also hydrided to some extent. A hydrogen concentration of 91 ppm was found at 6 in. This concentration diminished to 45 ppm at 60 in.

Metallographic sections along the length of the process tube confirm the hydrogen analyses. The hydride case on the ID surface of the tube is approximately 3.5 mils thick at 6 in. from the flange compared to 1 mil at 60 in.

Total hydrogen concentrations for tube 4372 KE decreased from 65 ppm 6 in. from the rear flange to 26 ppm at 59 in. Hydrogen analysis at the upstream end of the tube was 22 ppm. Metallographic sections of this process tube show the hydriding to be very spotty and associated mainly with scratches through the original autoclave oxide layer. Spots of hydride that cannot be associated with apparent scratches may be associated with defects in the oxide.

Zircaloy Tube Hydriding Studies

Continuation of the studies reported last month on charging Zircaloy-2 with hydrogen in reactor process water at 90°C by 3 volt (rms) AC potentials showed the metal did pick up hydrogen. Although analyses of the coupons at 283 hr. showed no significant amount of pickup, continuing the test to 667 hr. produced hydrogen absorption by 5 of the 7 coupons in test.

A study on the effect of various water qualities on the hydriding of Zr-2 with galvanic, i.e. coupled to 6063 aluminum, and applied potentials, 3 volts DC, has yielded preliminary results. The refreshed electrolyte was compounded from deionized steam condensate with addition of sodium bicarbonate, pH adjusted to 6.6 with either H₂SO₄ or CO₂. Another electrolyte was the same as the above except some of the bicarbonate was replaced with arsenate to produce 1 ppm As. The electrolytes had a specific resistivity at 90°C similar to process water, 2500 ohm-cm, at 90°C. The 30 mil Zircaloy coupon in each electrolyte made 3 volts cathodic to a platinum electrode for 211 hr. at 90°C, picked up 33 ppm hydrogen each. The current efficiency was about 0.3%. Coupons galvanically coupled to 6063 aluminum in the three different electrolytes showed substantial differences in galvanic current and hydrogen pickup. The solution with 1 ppm As was the highest current and the highest pickup (17 ppm). The solution neutralized with H₂SO₄ produced less current flow and less pickup (9 ppm) while the solution neutralized with CO₂ had the smallest current and no apparent pickup. The test is continuing.

Nickel Plated Aluminum

Three processes for nickel plating 6063 aluminum followed by precipitation heat treatment of the aluminum to obtain the T-6 condition were investigated. None of the methods has given completely satisfactory adherence of the nickel to the aluminum. Tensile testing of specimens prepared by the three processes showed that the tensile strength obtained was approximately that of 6063 aluminum in the T-6 condition when specimens were quenched in cold water after solution heat treatment. Somewhat lower values were obtained in specimens quenched in boiling water.

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Effect of Beryllium and Uranium on the Corrosion of Zircaloy-2

It is known that the brazed and welded end closures on Zircaloy-2 clad fuel elements may contain varying amounts of beryllium and uranium contamination. The effect of these contaminants on the corrosion characteristics and integrity of the end closures is under investigation. A test has been designed to investigate these effects under both in-reactor and out-of-reactor conditions.

Triplicate coupons were cut from Zircaloy-2 castings fabricated by Oregon Metallurgy Corp. and containing various amounts of beryllium and uranium. The Zircaloy-2 castings contained 1, 2, and 3% beryllium and 0.1, 0.5, 1, 2, and 4% uranium for a total of 15 ternary alloys. Each ternary alloy was cast in triplicate. Those coupons cut from each button were bright etched in a standard $\text{HNO}_3\text{-HF}$ bath and divided into three groups. One group was exposed to 300°C, pH-10 (NH_4OH) water; the second group to 360°C, pH-10 (NH_4OH) water. The third group is being held for in-reactor exposure.

Following two weeks in 360°C water and 4 weeks in 300°C water, an increase in the corrosion of Zircaloy-2 with increasing beryllium and uranium concentrations is apparent. Of the two contaminants, uranium has the most pronounced effect on the corrosion of Zircaloy-2. The corrosion weight gains for two weeks exposure to both 300 and 360°C, pH-10 water show a factor of at least 8 increase for the 4% uranium samples and a factor of at least 6 increase for the 2% uranium samples over the normal corrosion of Zircaloy-2. Normal corrosion weight gains for Zircaloy-2 were 10 mg/dm² and 20 mg/dm² for two week exposures in 300 and 360°C water, respectively. The actual increase in corrosion is probably greater than a factor of 6 to 8 increase. All of these samples are spalling, both in the form of powder and flakes. This is exemplified by the 3% Be-4% U samples which had gained 89.7 mg/dm² after two weeks exposure but had a net weight loss of 114 mg/dm² after four weeks exposure to 300°C water.

The exposure temperatures and water conditions used in this test are comparable to those in NPR. The results obtained in this test are probably representative of irradiated fuel end closures similarly contaminated with beryllium and uranium. These autoclave tests will be continued in order to determine whether the more dilute ternary alloys will slough on longer exposure.

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Examination of Zircaloy-2 Graphite Cooling Tube

Two Zircaloy-2 graphite cooling tubes were removed from N Reactor service for hydrogen analysis. Four 3 in. sections of tubing were cut from tube 0788 (2 upstream and 2 downstream), and two 3 in. sections from the downstream portion of tube 0899. Following crud removal, the sections were analyzed for hydrogen. Neither Zircaloy-2 tubes had increased significantly in hydrogen concentration over the preservice hydrogen concentration. The analyses for tube 0788 were from 10 and 12 ppm for the downstream portion and 12 and 15 ppm for the upstream portion. The two samples from tube 0899 analyzed 18 ppm. Analyses of similar Zircaloy-2 tube sections which have never been in reactor service have shown hydrogen concentrations ranging from 12 ppm to 25 ppm.

The crud levels measured on these two tubes were 19.8 mg/dm^2 for 0788 and 12.0 for 0899.

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THERMAL HYDRAULICS STUDIESBoiling Burnout Experiments for N-Reactor

Burnout experiments for N-Reactor co-producer fuel elements were resumed. The proposed co-producer element consists of a nonpower-producing cylinder of target material inside a fuel tube which in turn is centered inside of a pressure tube. Currently, the burnout experiments are being conducted with a model of the flow annulus between the target and fuel tube, consisting of an unheated tube inside an electrically heated outer tube. In this model, the dimensions of the coolant annulus are 1.775 in. OD and 1.220 in. ID which are not the same as the planned co-producer fuel dimensions but which are identical to those used in earlier experiments with both tubes heated. In later experiments, a model with the actual co-producer fuel dimensions will be used. The primary purpose of the present model experiments is to bridge the gap between the earlier two-heated surface model experiments and the experiments with the one-heated surface model having the co-producer dimensions.

These experiments will allow separation of the effects of the two variables - dimension changes and heated vs. unheated opposing surface. This information will be useful in estimating the effects of changes in co-producer dimensions which might be proposed in the future.

Experiments with the current model were interrupted earlier by physical burnout of the model after 14 sets of burnout conditions had been determined. These 14 sets did not provide enough information for establishment of good quantitative relationships. Consequently, the model was repaired and reinstalled in the heat transfer facilities of the Thermal Hydraulics Laboratory. Zero power runs at 16 combinations of coolant flow rates and temperatures have been completed with the repaired model. Data from these zero power runs will be used to determine instrument accuracies and heat losses for use in analyzing burnout experiment results. Experiments to investigate 14 more sets of burnout conditions have been scheduled and are ready to begin.

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Heat Transfer Experiments for K-Reactor

Laboratory experiments were completed for the first phase of a laboratory investigation of the capabilities of I&E fuel elements to operate at higher power densities in K Reactors. The objectives of this phase were to determine boiling burnout data and the pressure drop-flow characteristics of a full length fuel column positioned concentrically within a process tube at tube powers from 1000 to 3300 kw. The experiments were also meant to compare data with outlet fittings of the type in current use (standard outlet fittings) with data obtained with outlet fittings modified to give a minimum of hydraulic resistance (open outlet fittings).

The test section consisted of an electrically heated rod 0.470 in. ID, 1.512 in. OD, and 28.1 ft long. The length of the rod was made up of four different metal alloys to provide an approximation of a cosine power generation along the length with a peak-to-average specific power of 1.40. The rod was positioned inside of a 1.724 in. ID Zircaloy-2 process tube with short ceramic pieces which represented the "arch rail" type of supports in current use. The 0.090 in. height of these supports allows some clearance between the fuel and the process tube and thus the supports do not position the fuel pieces quite concentrically in the tube.

Two hundred thirty data points were obtained in the experimental program. A typical run consisted of maintaining a constant power generation while the flow was reduced in small steps from an initial condition where the exit water was subcooled to a final condition where film boiling existed at some point on the electrically heated test section. The onset of film boiling (referred to as boiling burnout in these tests) was detected by a rapid excursion of one of the surface thermocouples of at least 100°F (55°C). During each run, data were taken of flow, power generation, pressures, pressure drop, surface temperatures and water temperatures. The water temperature data included readings in the annulus at eight points around the circumference and three different locations in the downstream half of the test section.

A preliminary examination of the data obtained with the standard outlet fittings indicates that boiling burnout occurred when the flow rate was reduced to 47% of the flow required to produce a normal outlet temperature of 130°C for all tube powers between 1500 and 3000 kw. These flows are 6 to 8% higher than the burnout flow rates determined in previous tests with the rod supported by "bridge rail" supports. In addition, it was found that the amount of circumferential mixing as determined with the present test section using "arch rail" supports was not as great as the mixing found previously with "bridge rail" supports.

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When the test section was equipped with the open rear fittings, it was found that the flow reduction required to cause boiling burnout was less than that when the test section was equipped with the standard outlet fittings. The probable reason for this is that the more restrictive flow passage in the standard outlet fittings caused pressurization of the test section during two-phase flow. This would restrict the amount of steam formation over the heated surface and require a lower flow rate to cause burnout. On the other hand, the pressurization which took place in the standard outlet fittings makes that case much more susceptible to hydraulic instability than the test section equipped with open outlet fittings. In fact, it appears that instability could not occur in a tube using the open outlet fittings at tube powers below 3000 kw.

At the conclusion of this phase of the study, the test section was removed and the supports modified to allow the rod to be 54% eccentric within the process tube.

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GRAPHITE STUDIES

Burnout Monitoring - N Reactor

The second burnout monitoring charge in 1648-N was removed on 10-8-65 after 142 operating days in the reactor. The majority of the monitors showed burnout rates below 0.10% per thousand operating days (%/KOD). The maximum burnout observed (one sample) was 0.20%/KOD. There was an apparent peak at ~5 ft from the reflector front face, and the burnout rate decreased thereafter with distance down the channel.

There were a total of 126 burnout monitoring samples in the channel (18 boats x 7 samples per boat). The monitors were all of TSX graphite 0.426 in. in diameter and of 3, 4, or 6 in. lengths. The lengths of approximately half of the samples were measured before charging into the reactor. The average length change on these samples during the operating period was -0.01%, with ~90 per cent of the samples falling in the range from 0.00 to -0.02%. Exposure information will be available after flux-monitor activities are determined.

Graphite Contraction

Tests have indicated that large bars used in reactor construction contract at a significantly greater rate than the small samples used in irradiation tests. A pertinent question is whether the ratio of contraction rates continues to all exposures, or whether the reversal of contraction (turn-around) which has been observed for the small transverse test samples occurs at a lower exposure for the large bars with the total amount of contraction being a fixed quantity. It is now possible to provide a partial answer to this question.

Central depression of the K Reactors has exceeded 7 in. without indication of a reduction in depression rate. This represents a contraction of the TSGBF central filler bars of more than 2.0% at an exposure of approximately 4.5×10^{21} nvt ($E > 0.18$ MeV). In comparison, the maximum contraction at turn-around for small CSF transverse samples at 600°C is about 1.3% at an exposure approaching 10^{22} nvt. From the limited data available for TSGBF transverse samples, it appears that turn-around in TSGBF may occur at much lower exposures than for CSF, but the maximum contraction at turn-around may be the same or slightly less than for CSF. Thus, present data indicate that the ratio of contraction rates for large bars and small samples prior to turn-around will apply to the maximum amount of contraction at turn-around.

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