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IN-REACTOR MEASUREMENT OF FUEL ELEMENT CLADDING TEMPERATURES

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IN-REACTOR MEASUREMENT OF FUEL ELEMENT CLADDING TEMPERATURES

A Thesis

Presented in partial fulfillment of the requirements for the

Degree of Master of Science in

Mechanical Engineering

in the

University of Idaho Graduate School

by

Donald Ross Doman

1960

[redacted]

The thesis of Donald Ross Doman, "In-Reactor Measurement of Fuel Element Cladding Temperatures," is hereby approved:

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BIOGRAPHICAL SKETCH OF THE AUTHOR

Donald Ross Doman was born at Parma, Idaho, on May 26, 1932. He attended the Montpelier High School, Montpelier, Idaho, and was graduated in 1950. That September he enrolled in the University of Idaho and received the degree of Bachelor of Science in Mechanical Engineering in 1955. He was employed by the General Electric Company in the Hanford Operations at Richland, Washington, following graduation. In September, 1956, he enrolled in the Graduate School of the University of Idaho through the General Electric School of Nuclear Engineering at Richland and continued his studies there when the University of Washington assumed direction of the school in September, 1958. In June, 1960, he completed the requirements for the degree of Master of Science in Mechanical Engineering, of which this thesis is a part.

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Gratitude is expressed to the following individuals: Dr. J. A. Ayres of Coolant Systems Development, Hanford Laboratories Operation, under whose direction the experimental work was performed and who aided greatly in the analysis of data; Dr. J. M. Batch of Thermal Hydraulics, Hanford Laboratories Operation, for serving as the on-site advisor for this thesis; and Professor W. P. Barnes of the University of Idaho Department of Mechanical Engineering for serving as campus advisor for this thesis.

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ABSTRACT

A design was developed for leading thermocouples from a high-temperature, pressurized water reactor - coolant system of such integrity that no reactor shutdowns were caused by its use. Using this design, measurements of the fuel-element-cladding temperature and its variation with time were made in three tests on elements clad in type X-8001 aluminum alloy.

The following conclusions were reached from the test results:

(1) The cladding temperature of a fuel element operated at low heat flux in high bulk-outlet temperature water did not increase with time and was slightly lower than predicted by the Sieder-Tate equation.

(2) Cladding temperatures of fuel elements operated at high heat flux in either high bulk-inlet or outlet temperature water increased 40°C higher than predicted by the Sieder-Tate equation with initial temperatures equal to the predicted temperatures.

(3) The rate of temperature increase appeared dependent only on fuel-element heat flux and location with respect to the front and rear faces of the reactor.

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IN-REACTOR MEASUREMENT OF FUEL ELEMENT CLADDING TEMPERATURES

BY

DONALD ROSS DOMAN

INTRODUCTION

Prediction of corrosion rates of fuel element cladding materials is one of the major problems in designing fuel elements for use in nuclear reactors cooled by water. To permit accurate prediction of corrosion rates, the temperatures to which the fuel-element jackets are subjected must be known. Since advanced reactor cooling systems are to be operated up to 300°C and 1800 psi, cladding temperatures of fuel elements operating near this range must be determined. This introduces the problem of making temperature measurements in a reactor at such temperatures and pressures with a system of such high integrity that no reactor operating problems or unnecessary shutdowns be involved. With such a system, the cladding temperature and its variation with time on an actual operating fuel element could be measured. These measurements could then be used as a basis for predicting corrosion rates by comparing them with values calculated from the Sieder-Tate equation, the present method of determining cladding temperatures.

Aluminum alloy cladding is frequently considered for use in advanced reactor designs because of its cheapness. Such jackets have been used since the startup of the present Hanford reactors at bulk-

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water temperatures up to 100°C with fairly satisfactory corrosion performance.

To aid in plutonium production, it is often desirable to raise reactor power levels by increasing coolant temperatures. Some advanced reactor designs are based on temperatures in the 200°C to 300°C range. As water temperatures increase, corrosion of the aluminum jacket, in general, increases. The type 1245 aluminum alloy used for present reactor fuel jackets could be used to about 200°C temperatures, but accelerated corrosion is experienced at temperatures much above this. From out-of-reactor corrosion tests, some types of new aluminum alloys do not experience the accelerated corrosive attack shown by type 1245 alloy even above 300°C if proper water quality conditions are maintained (5)*. From these tests it would appear that such new aluminum alloys could be used for jacketing with no corrosion problems.

Corrosion rates experienced in the reactor are increased over out-of-reactor rates from two effects: fuel-element heat flux and irradiation. In high-purity water systems specified for the high temperature advanced reactor designs, Lobsinger (26) reports effects from radiation only are small enough to be ignored for practical purposes. The heat flux is important since in the transfer of heat from the fuel element to the water, high temperature drops are experienced through the corrosion-product film and the semistagnant water film immediately

* Figures in parenthesis refer to references in the Bibliography.

adjacent to the element. These drops then raise the actual fuel element surface temperature substantially higher than the bulk water temperatures. This high surface temperature in turn increases the corrosion rate of the jacket. Since corrosion in the range of 300°C is very temperature dependent, the temperature drops across the corrosion and water films and their variation with time must be known to permit close prediction of corrosion rates. Accurate calculations of the drops cannot be made since actual values of thermal conductivity for the corrosion-product film are not known. Therefore, actual experimental measurements of one or both temperature drops and time variations were required.

The temperature drop through the corrosion film was selected as the easier for measurement. By inserting thermocouples in the jacket of a fuel element, the temperature increase with time caused by the corrosion-product film could be measured. The 1706 KER Recirculation Test Facility in the 105 KE reactor is well suited for making such studies in pressurized high temperature water coolant. Use of this in-reactor facility, however, introduced the problem of developing a design for making these measurements in a reactor coolant system using water up to 300°C and 1500 psi with such high integrity that no failures resulting in reactor shutdowns would arise. The development of a satisfactory design and the results of tests measuring the temperature drop caused by the corrosion-product film are described herein.

SURVEY OF THE LITERATURE

Determination of corrosion rates of fuel-element jacketing as influenced by film formation has been a problem since the earliest operation of the Hanford reactors. Woods (43) made one of the earliest investigations on fuel-element films by calculating their effect on local heat transfer coefficients as influenced by film thermal conductivity. According to monthly reports, Amos (2) also made some investigation of the influence of the corrosion films on heat transfer. Both of these investigations were concerned only with heat transfer aspects and not corrosion of fuel elements. Early attempts at predicting corrosion rates were concerned with the correlation of rates directly with the bulk water temperatures, but little success was realized. In 1952, Shields (39) corrected for the temperature rise across the water film giving somewhat higher cladding surface temperatures and resulting higher corrosion rates. These results were considerably more successful than previous attempts, but they still failed to account for the observed decrease in corrosion rates at the rear of the process tubes where there are high water temperatures but low fuel-element powers. His results also indicated a flow dependence of corrosion rates at a given surface temperature. Early in 1954, Goldsmith (17) gave a correlation taking into account both the surface temperature and slug power, but this still did not permit prediction of corrosion rates to better than an observed factor of 2.2. Finally, in late 1954, deHalas (10) produced a correlation accounting for the temperature rise across the corrosion-product film on the cladding. The inclusion of the temperature drop across the corrosion product film resulted in more accurate calculation of the fuel-element surface temperature and prediction of corrosion rates to a factor of 1.75. To substantiate these correlations, however, actual measurement of the temperature drops was required.

Consequently, investigations were started on the possibility of using an actual fuel element in the reactor with thermocouples in the cladding to measure the temperature changes as a corrosion film was built up.

The use of thermocouple fuel elements had long been considered, and a few actually were used at Hanford in heat transfer studies. In 1951, Jones (22) suggested their use in measuring fuel-element core temperatures to determine irradiation effects on uranium thermal conductivity. Five elements were charged, but only two did not fail before the end of the test periods, since the lead out system used was quite complicated. (23,24) In 1957, Marshall (28) developed an acceptable system for leading in-reactor thermocouples from probes used to measure temperature distributions around internally and externally cooled fuel elements in a standard reactor tube. No method existed for leading thermocouples from a reactor tube at the high temperatures and pressures involved in the experimental tests discussed here, therefore requiring the development of such a system.

Since thermocouples were to be used to measure the temperatures in these experiments, the literature was reviewed to determine the effects of irradiation on thermocouples. Palladino (33) and Giberson (15) both reported no noticeable effects of irradiation on thermocouple calibration and performance even at temperatures and neutron fluxes higher than would be encountered in these tests. Thermistors, which were also considered for use because of the high degree of accuracy attainable, were reported by Moody (30) as affected by irradiation.

DESCRIPTION OF APPARATUS

A schematic drawing of the experimental apparatus arrangement is shown in Figure 1. It consisted simply of the test fuel element in one of the process tubes on the 1706 KER facility in the 105 KE reactor with thermocouples imbedded in the fuel element cladding leading to a temperature recorder on one of the reactor experimental test levels.

The 1706 KER Test Facility, shown schematically as the reactor coolant portion of Figure 1, consists of four special test process tubes in the 105 KE reactor for fuel-element, corrosion, and water-chemistry studies. Each tube is individually supplied with pressurized-water coolant pumped from the 1706 KER building and, unlike the ordinary single-pass cooled reactor tubes, the coolant is recirculated. Recirculation permits pressurization for operation of the coolant at temperatures considerably above the present reactor limit of 100°C, thus increas-

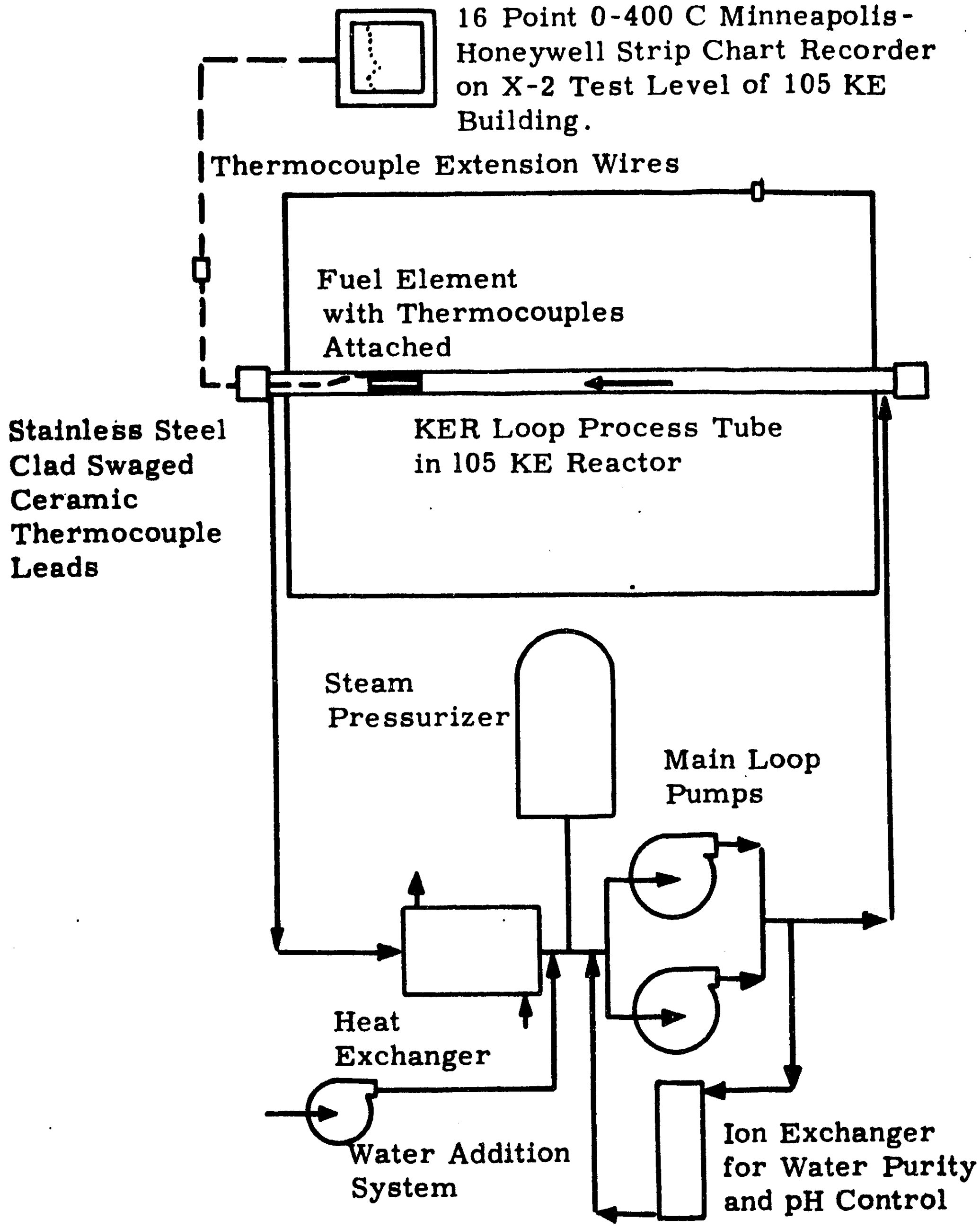

FIGURE 1

Schematic Diagram of Apparatus Arrangement ing reactor power with associated greater plutonium production. Recirculation also allows the use of higher purity water and better water quality control to help attain lower corrosion rates than would be possible on a single pass basis because of high costs. To give maximum flexibility for testing purposes, three of the KER loops use stainless steel piping, while the other uses carbon steel piping. All loops have in-reactor process tubes of Zircaloy-2, a zirconium alloy having good corrosion resistance with a low cross-section for neutron capture. All loops also have heat exchangers to permit operation at any desired loop inlet temperature and have equipment, such as ion exchangers, to control water conditions and quality at any desired level. All loops are pressurized by contained steam pressurizers.

The design of the in-reactor portion of the test equipment required a high degree of integrity from both the experimental and operational viewpoints. A failure of either the test fuel element or the thermocouple lead-out system from the high temperature, high pressure coolant would have caused a reactor shut-down resulting in plutonium production loss. Failure of the temperature monitoring system would have resulted in the loss of the experimental data.

A detail of the in reactor portion of the apparatus is shown in Figure 2. The thermocouple fuel elements had cores either of enriched uranium dispersed in aluminum, called "Doe metal", or solid natural uranium. The elements were fabricated by the "dip canning" technique in which the heated cores were pressed into a fuel element sheath closed

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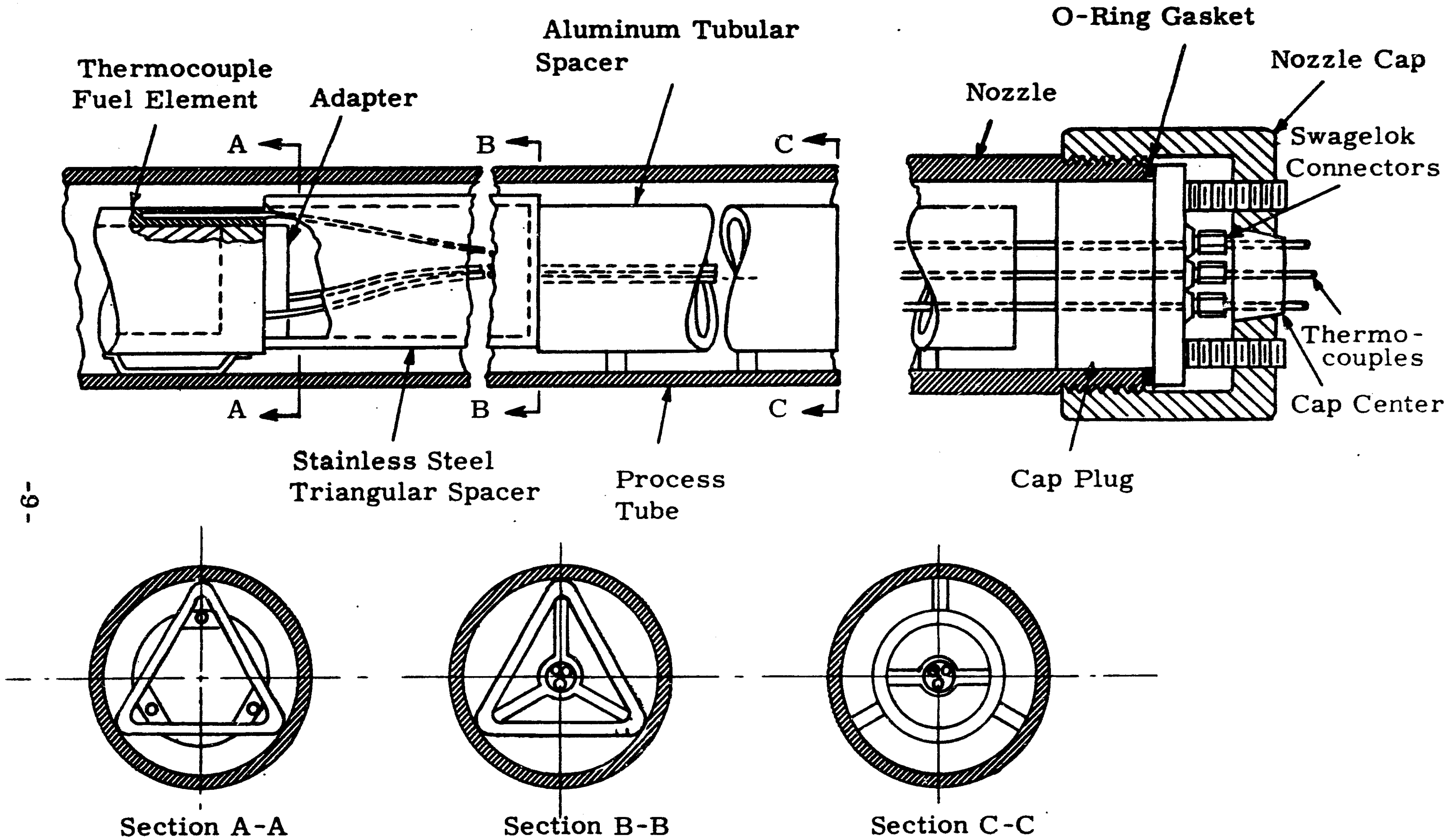

FIGURE 2

Detail of In-Reactor Portion of Apparatus

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at one end and filled with molten aluminum silicon alloy, usually abbreviated AlSi. Most of the AlSi is displaced by the core insertion but some flows into the space between the core and cladding to act as a heat transfer medium. After dip canning, an end cap approximately one-half inch thick was welded into the open end. In all thermocouple elements made for these tests, a type X-8001 aluminum-alloy sheath with a 1.44-inch outside diameter and a 0.12-inch wall thickness was used. After canning, three 0.065-inch holes 120° apart were drilled longitudinally one and one-fourth inches into the cladding so at least 0.030 inches of cladding remained between the bottom of the hole and the inside of the cladding, as determined by subsequent radiographic examination. After drilling and radiographic examination a special hexagonal adapter shown in section "A-A", Figure 2, was welded to the fuel-element end cap.

All thermocouples used were the stainless-steel sheathed, ceramic-insulated type fabricated by Aero Research or Thermo Electric Corporations. In the particular thermocouples used, two 30-gage chromel-alumel wires insulated from each other and the type 304 stainless-steel sheath by magnesium oxide or zirconium oxide powder were swaged to a one-sixteenth inch outside diameter. The thermocouple junction was made at the factory by removing some of the ceramic insulation from the end of the sheath, joining the two thermocouple wires, and inserting the junction into a weld applied to seal the sheathing end. This method insured minimum response time to temperature variations.

The in-reactor portion of the test apparatus was assembled into one unit or thermocouple train as shown in Figure 2. This train was made by passing three stainless-steel-clad thermocouple wires through the nozzle cap center plug and the Swagelok fittings on the cap plug, and down the middle of several tubular aluminum spacers to the fuel element. The nozzle-cap center plug permitted the cap to be screwed on the nozzle without twisting the wires. The wires were kept from whipping in the water flow by inserting them in the central hole of a cross-bar welded at one end of each aluminum spacer, as shown in Section C-C, Figure 2. The spacer adjacent to the fuel element was a triangular stainless-steel type with a triangular cross bar shown in Section "B-B", Figure 2, instead of an aluminum tubular type. This spacer was placed over the hexagonal plate welded to the thermocouple fuel element, as illustrated in Section A-A, Figure 2, so that it straddled the thermocouple wires. After the wires were run through all the spacers, they were inserted in the holes drilled in the fuel-element cladding. The cladding over the wires for one-fourth inch from the end of the fuel element was swaged with a hammer and punch to hold the wires in place. When the wires were secured in the fuel element, all slack except that left for thermal expansion was removed from the wires in the train between the fuel element and cap plug, and the Swagelok fittings were compressed into the thermocouple sheaths.

Installation of the train into the reactor was made by inserting it into the process tube by hand. After insertion, the nozzle cap was screwed onto the nozzle, and the process tube pressure seal made by tightening set screws in the cap to seal a solid stainless-steel O-ring on the cap plug. Connections between the stainless-steel sheathed, thermocouple-train wires and extension leads of ordinary 14-gauge chromel-alumel wire were made on the reactor face. The connections in the first test were made by soldering the leads together. Connections in all other tests were made with quick-disconnect plugs. All connections were waterproofed by wrapping with electrical tape and covering with neoprene hose. The extension wires were led from the reactor face to a 0-400°C, sixteen-point, Minneapolis-Honeywell strip-chart temperature recorder on the X-2 test level of the 105 KE reactor building.

METHOD OF PROCEDURE

The first requirement to make the desired temperature measurements was the development of a method for leading the thermocouples from the pressurized high-temperature water coolant with such integrity that the lead-out system would not fail causing an emergency reactor shutdown.

As mentioned in the literature survey, Marshall (28) had developed a method for leading thermocouples from a process tube operating at usual reactor outlet conditions of temperatures near 100°C with maximum pressures of 50 psi. The conditions intended for the high-temperature measurement tests in the 1706 KER Test Facility involved maximum temperatures and pressures of 300°C and 1500 psi, respectively. Because Marshall's lead-out method utilized a rubber O-ring sealant which would deteriorate in high temperature water, it could not be used for this application.

Three possible lead-out methods were considered: welding the stainless-steel-clad thermocouple wires directly to the process tube cap plug, and two methods of leading the wires out through tubing fittings attached to the cap plug. The system also had to incorporate minimum installation and discharge time, with the further stipulation that no problems of safety to personnel be incurred either at charging or discharging. "Swagelok" compression-type tubing fittings manufactured by Crawford Fitting Company had performed well on out-of-reactor corrosion test loops operating at similar conditions, so they were considered for use here. (14) The two methods of attaching the Swagelok fittings were: (a) threading the fittings directly to the cap plug using the one-eighth inch pipe threads supplied on the fittings, and (b) removing the threads and heliarc welding the fittings directly to the cap plug. In attempting to fabricate a cap plug for prototype testing using all three lead-out methods, it was found that the heat of the welding torch burned through the thermocouple wire before it could be welded, so this method was discarded. A cap plug using both the threaded and welded Swagelok fittings was successfully fabricated, but hydrostatic-pressure testing revealed leaks around the threads of the threaded fittings, so they were seal welded.

The fittings were tested on the out-of-reactor corrosion-test loop, ELMO-7, in the 1706 KE Building at 305°C and 1800 psi, conditions higher than the maximums anticipated in actual operation to give added safety. During a four-weeks test at these conditions, a small amount of leakage was detected by condensing steam on a shiny surface held near the nozzle cap. The leak was so small that attempts to determine an actual leakage rate were unsuccessful. Based on this test and previous operational experience with this type fitting, both the threaded and welded Swagelok fitting lead-out methods appeared satisfactory. The welded method was adopted, since seal welding was required anyhow to pressure seal the threaded type. Welding also permitted a more compact arrangement since the space allowed for screwing fittings into the cap plug could be eliminated. Removing the pipe threads reduced the over-all fitting height so that only an additional one-half inch increase in length of the nozzle cap was required. (14)

Three tests measuring the cladding temperature and the increase with time were made. In the first test, a fuel element with a natural uranium core was charged from the rear face into the reactor tube for test loop KER-2. The element was positioned just in the reactor neutron flux giving an element with a low heat flux in the highest bulk water temperature, which occurs at the downstream end of a fuel element charge.

Enriched uranium fuel-element cores were used in the other two tests to provide high fuel-element heat fluxes. In the second test, the thermocoupled fuel element was charged from the reactor rear face into the highest neutron-flux region of the reactor tube for test loop KER-4. This gave a high heat-flux fuel element but still at the highest bulk water temperature of the tube, since it was the last element in the charge. In the third test, the fuel element was charged from the reactor front face into the high neutron-flux region of the reactor tube for KER-3. This again gave a fuel element producing a high heat flux but this time at the coolest tube bulk-water temperature, since it was at the upstream end of the fuel charge. In charging from the front face an eleven-inch space instead of the usual one-inch space for thermal expansion was left between the last tubular spacer and the front cap plug to compensate for variations in fuel-element lengths and provide additional room for thermal expansion.

Water used in all three tests was demineralized, deoxygenated and maintained at 4.5 pH by addition of phosphoric acid as required. Water purity was maintained by passing a small portion of the total flow through ion-exchange columns in the 1706 KER Building.

RESULTS

Test No. 1:

The first thermocouple fuel-element train was charged into the rear of KER Loop No. 2, 105 KE reactor tube number 2864, on July 1, 1958, with the thermocoupled fuel element barely in the reactor neutron-flux field giving a low-power fuel element. Twenty-five type X-8001 aluminum-clad fuel elements with enriched uranium-aluminum alloy cores were charged upstream of the thermocouple fuel element.

Several minor difficulties were encountered in charging the tube. The first charging method used was by sliding the train into the tube from an angle iron support. This arrangement proved too unwieldy, so the train was removed from the angle iron and charged by hand. After the train was inserted, a hydrostatic pressure test of the system revealed a leak on the solid metal O-ring on the cap plug. After the train was removed from the tube, inspection of the O-ring showed it had dropped from the gasket seat and had become pinched between the cap plug and end of the nozzle. A new O-ring was installed, the train re-installed, and a successful hydrostatic pressure test of 2500 psi applied. Difficulties were also encountered in trying to solder the small 30-gauge thermocouple wires of the stainless-steel clad thermocouples to the relatively large 14-gauge extension wires. All three thermocouples were finally soldered successfully.

The actual temperatures measured, tube inlet and outlet temperatures, and calculated tube power are plotted in Figure 3, with the original data given in Appendix A. As noted, no response was obtained from thermocouple No. 3, indicating an apparently broken wire, and thermocouple No. 2 began drifting downward after about 200 hours of operation. This indicated a probable short in the thermocouple on the rear face causing it to indicate an average value between the cladding

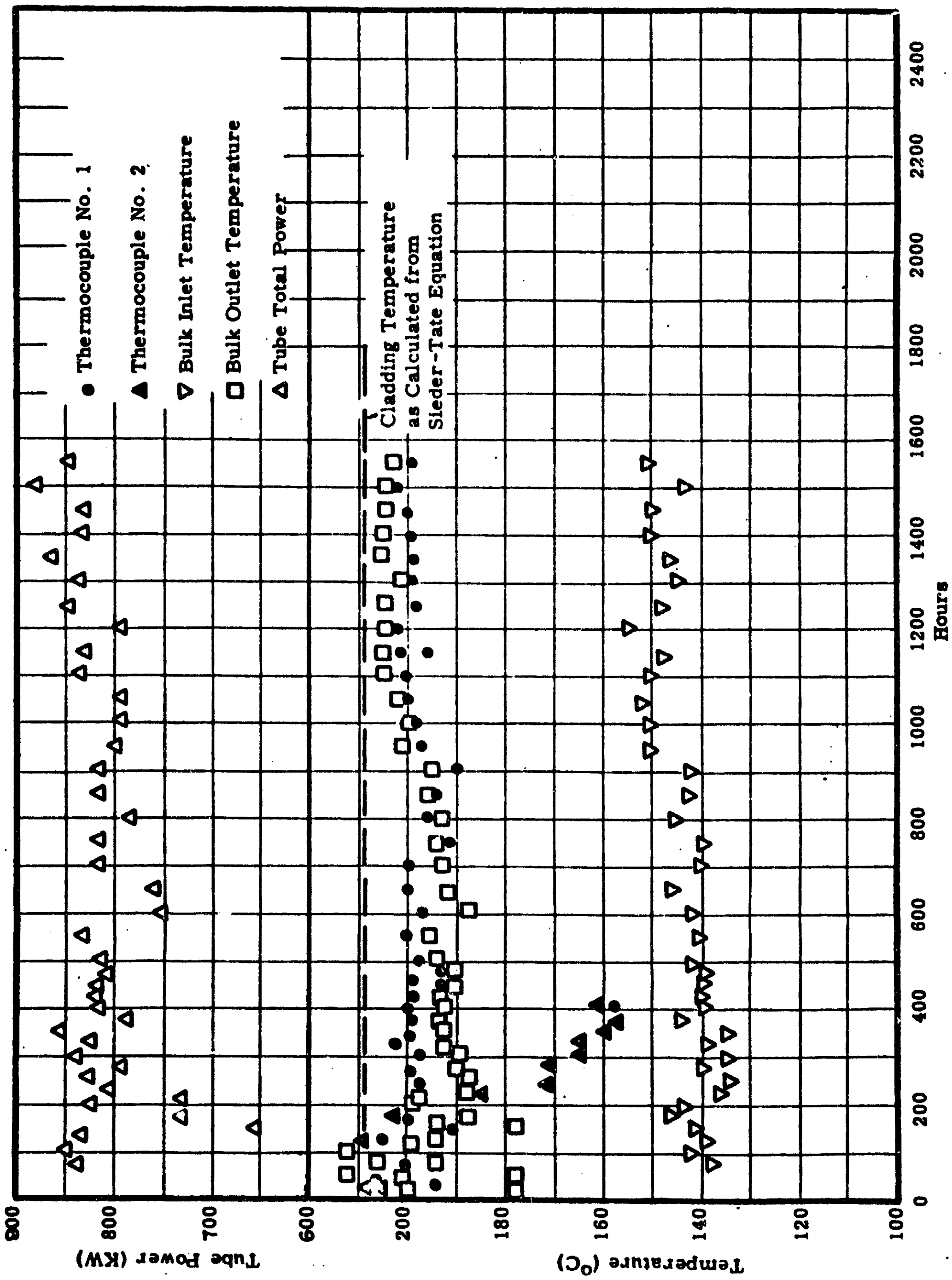

FIGURE 3

Test No. 1 - KER Loop No. 2 - Low Heat Flux with High Bulk Outlet Water Temperature

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and rear face temperatures. Consequently, after 400 hours of operation, records from this thermocouple were stopped. The remaining thermocouple apparently functioned well until the test was terminated with discharge of the process tube on October 18, 1958. No trouble was experienced in discharging the thermocouple train. The reactor had been down five times during the test period. No temperature readings were taken upon a reactor startup until test operating conditions of nominal tube-inlet and out let temperatures of 145° and 200°C, respectively, were reached.

In Figure 3, it should be noted that the cladding temperature recorded remained fairly constant at slightly above 200°C over the test period, averaging 201°C with an average bulk outlet temperature of 198°C. The tube power averaged about 770 KW.

After the tube was discharged, a heat flux of about 96,000 Btu/hr-ft^2 for the thermocouple fuel element was obtained by taking activity readings from the rest of the fuel elements in the tube. (27) Based on this heat flux, a temperature drop across the water film of 10.9°C was calculated from the Sieder-Tate equation, as shown for Test No.1 in Appendix B. Since the calculated drop across the can was small, it was neglected. With the average bulk outlet temperature of 198°C obtained in the test, the expected cladding temperature would be the sum of the average temperature and the calculated temperature drop or 198 + 10.9 = 209°C. From Figure 3 it can be seen that only at the beginning of the test did the cladding temperature ever exceed this value, with temperatures decreasing with time instead of increasing with time as would be expected. The most possible explanation for the observed discrepancy based on operation of the thermocouples in this test is that calibration could have changed. Since the rear face connections between the thermocouples and the extension leads were, of necessity, in such a location that water from discharging process tubes in the immediate vicinity could splash on them, moisture could possibly have entered through the intended waterproof insulating wraps of electrical tape. Moisture in the junction would account for the No. 2 thermocouple dropping downscale following the reactor startup after 150 hours of testing, causing it to read an average between the cladding and rear face temperatures. A small amount of moisture in the No. 1 thermocouple junction could possibly have caused a change in its calibration at this same reactor outage since it never again reached the temperatures as indicated before the outage.

Results from this test indicate that in operation of a fuel element at low heat flux and in high bulk-outlet temperature cooling water, the cladding temperature does not increase with time and the temperatures are lower than predicted from the Sieder-Tate equation. The results are somewhat inconclusive because of the problems encountered with the thermocouples in this test.

Test No. 2

The second thermocouple fuel element train was charged into the rear of KER Loop No. 4, 105 KE reactor tube No. 4268, on August 8, 1958, with the thermocoupled fuel element in the maximum reactor-neutron-flux field giving a high-power fuel element. Fifteen type X-8001 aluminum-clad fuel elements with enriched uranium-aluminum alloy cores were charged upstream of the thermocouple fuel elements. No difficulties were encountered in charging the thermocouple train. Quick-disconnect thermocouple jacks and plugs were used to connect the thermocouple wires to the extension leads to eliminate the problems encountered with the soldered connections used in Test No. 1. The connections were carefully wrapped with electrical tape and covered with hose to prevent water infiltration. The actual temperatures measured during the test are shown in Figure 4. As shown in the figure, low temperature operation of the loop for a 200-hour period after 300 hours total testing was required when failure occurred in one of the two canned motor pumps supplying the loop coolant. When the pump was repaired, the test again continued at the high temperature conditions until November 17, 1958, when the loop was discharged with no problems from the thermocouple train.

In Figure 4, it will be noted that one thermocouple followed the same pattern as the other two but read a lower temperature. This resulted from one thermocouple being inserted into the fuel element only about five-eighths inches compared with one and one-fourth inches for the other two. This put it nearer the end cap causing its temperature to be somewhat lower.

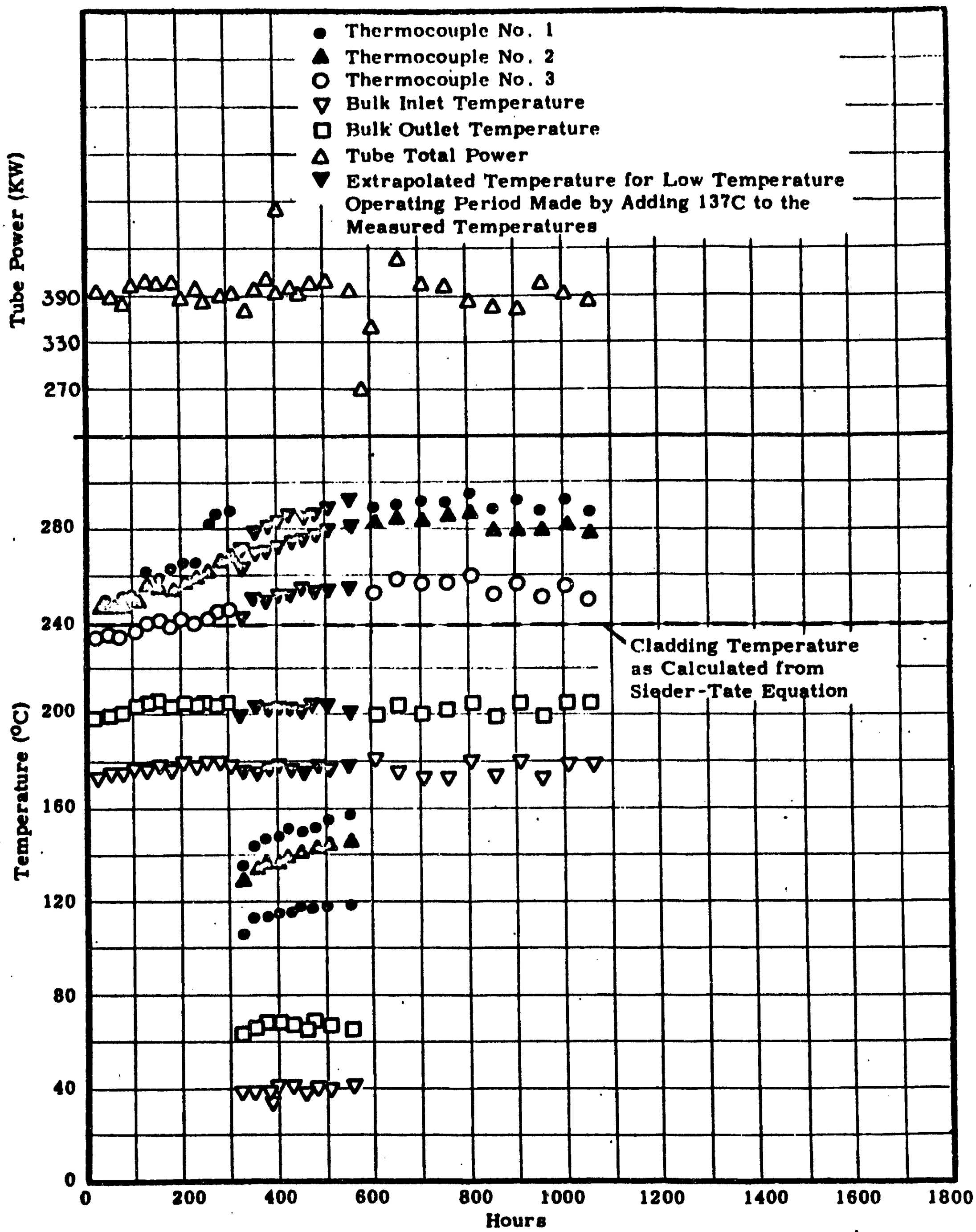

FIGURE 4

Test No. 2 - KER Loop No. 4 - High Heat Flux with High Bulk Outlet Water Temperature

The average bulk outlet-water temperature during the high-temperature operating portion of the test was 203°C. The tube power averaged 392 KW for the test period.

After the tube was discharged, a heat flux of 312,000 Btu/hr-ft^2 for the thermocouple element was obtained from activity readings from the rest of the fuel elements in the tube. (27) From this specific power and the average bulk-outlet temperature, a temperature drop across the water film of 35.5°C was calculated from the Sieder-Tate equation, as shown for Test No. 2 in Appendix B. With the average bulk-outlet temperature of 203°C, the expected cladding temperature would be 203 + 35.5 = 238.5°C, as indicated by the dotted line in Figure 4. From the figure, it can be seen that this was approximately the initial cladding temperature measured. During the test period, based on an average of the two thermocouples with the maximum readings, the measured cladding temperature increased some 40°C higher than this were it reached equilibrium. This increase was caused by the scale formation on the outside of the fuel element which acted as an insulator. The equilibrium temperature was apparently caused by the scale reaching an equilibrium formation-removal rate. By adding 137°C to all temperatures measured during the period of low temperature operation to obtain approximately the same bulk-inlet and outlet temperatures observed for high-temperature operation, it can be seen that the cladding temperature increased at about the same rate for both low and high temperature operation.

Test No. 3

The third thermocouple fuel element was charged into the front of KER Loop 3, 105 KE reactor tube No. 3565, on February 4, 1959. The element was positioned in the high neutron-flux region of the reactor giving a high power element at high inlet-temperature conditions. Twenty-five type X-8001 aluminum-alloy-clad fuel elements with depleted uranium cores were charged downstream of the thermocouple fuel element. As previously mentioned in the description of the test apparatus, an unsupported length of thermocouple wires about eleven inches long was left at the front of the thermocouple train to provide additional thermal expansion and compensate for uneven lengths of fuel elements and spacers in the rest of the charge. No problems were encountered in charging the train from the front face. As in Test No. 2, quick-disconnect thermocouple jacks and plugs were used to connect the thermocouple wires to extension leads. The connections were again waterproofed with electrical tape and covered with a hose to prevent water infiltration.

The actual temperatures measured in this test are shown in Figure 5. After 100 hours testing, one thermocouple began drifting downscale. After about 200 hours total testing, another thermocouple began drifting downscale. After 400 hours total testing, the remaining thermocouple began drifting downscale. Since the thermocouple junctions were on the front face of the reactor in this test, radiation levels were not too high to prevent examination of the junctions during reactor opera-

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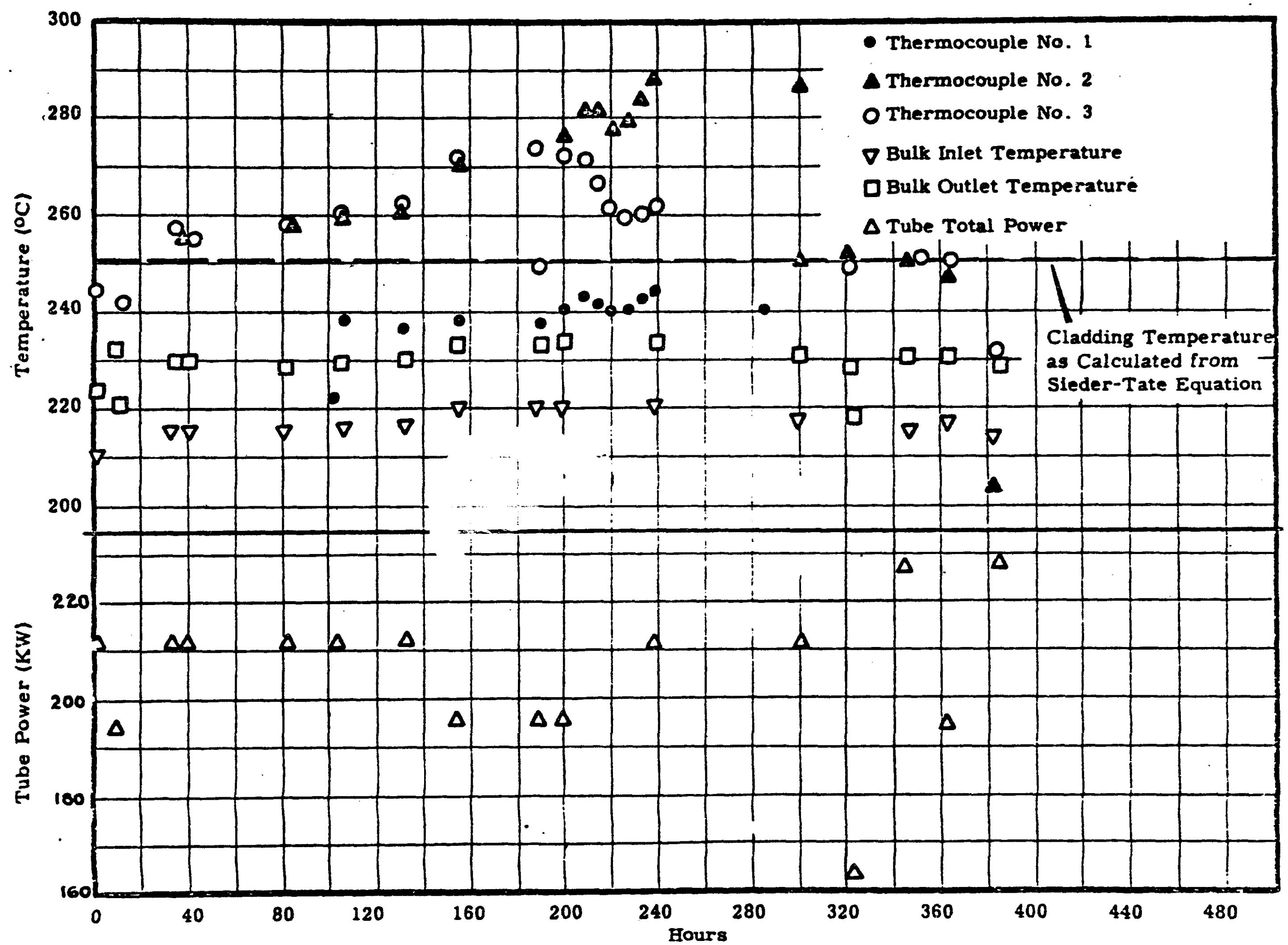

FIGURE 5

Test No. 3 - KER Loop No. 3 - High Heat Flux with High Bulk Inlet Water Temperature

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tion. When tested on February 26, 1959, all the thermocouples indicated an open-circuit so further readings on this test were stopped. When the tube was discharged two months later, the wires were found to have broken where the Swagelok fittings in the cap plug were compressed into the wires. A fatigue-type failure resulted at the point of high compressive stress in the Swagelok fittings apparently caused from vibration of the unsupported thermocouple wires by water entering the tube nozzle.

The average tube-inlet temperature during the test period was 216°C while the tube power averaged 206 KW. No activity readings to permit calculation of a specific power for the thermocouple fuel element were made since data was collected over such a short period. A specific power about the same as that obtained in Test No. 2 can be reasonably assumed, however, since the two fuel elements were in about the same nominal neutron-flux zone. Assuming a heat flux of about 310,000 Btu/hr-ft^2, a temperature drop across the water film of 34.7°C was calculated from the Sieder-Tate equation, as shown for Test No. 3 in Appendix B. With the average bulk-outlet temperature of 216°C, the expected cladding temperature would be 216 + 34.7 = 250.7°C, as indicated by the dotted line in Figure 5. From this figure, as was the case in Test No. 2, note that this was approximately the same temperature as measured toward the beginning of the test. As in Test No. 2, the temperature increased about 40°C higher than the calculated temperature but reached this temperature in about half the time. The tem-

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perature appeared to have reached equilibrium in this test, but since the temperature recorder had jammed in the forty-hour period just before the last thermocouple broke, this could not be said with certainty. This fast temperature increase probably resulted from the rapid formation of heat insulating scale caused by operation of the fuel element at the tube inlet. Heavier scales on specimens near the front of the reactor have been noted in other in-reactor experiments by Richman. (34)

SUMMARY AND CONCLUSIONS

A system was successfully developed for leading thermocouples from a reactor process tube operating up to 300°C and 1500 psi and with such integrity that no reactor shut-downs resulting in plutonium production loss occurred. The design consisted of using stainless-steel-clad, ceramic-insulated thermocouple wires with the high pressure, high temperature lead-out seals made with Swagelok compression-type tubing fittings welded to standard KER loop nozzle cap plugs.

This lead-out system was successfully used in three tests to measure the fuel-element cladding temperature and its variation with time on fuel elements clad in type X-8001 aluminum alloy. Temperature measurements were made by inserting one-sixteenth inch thermocouples into holes drilled longitudinally into the 120-mil thick wall of the fuel-element cladding. The tests involved operating thermocoupled fuel elements at both low and high specific powers in high bulk-outlet temperature cooling water and operating an element at high specific power but in high bulk inlet temperature cooling water. These experiments are the first known to have been made in which the cladding temperature and its variation with time were experimentally measured on fuel elements actually operating in a reactor in high temperature pressurized water coolant. The following conclusions were reached from the results obtained in these three tests:

(1) In operation of a fuel element at low heat flux (about 95,000 Btu/hr-ft^2) in high bulk outlet temperatures encountered at the rear of a reactor process tube, the measured cladding temperature did not increase with time and was slightly lower than predicted by the Sieder-Tate equation.

(2) Cladding temperatures of high heat flux fuel elements (about 310,000 Btu/hr-ft^2) in either high bulk-inlet or outlet temperature water rose to equilibrium temperatures about 40°C higher than predicted by the Sieder-Tate equation, but initial temperatures measured were about the same as predicted from this equation. Equilibrium temperature operation apparently resulted from the heat insulating scale on the fuel element reaching an equilibrium formation-removal rate.

(3) The rate of rise to the equilibrium temperature for fuel elements with high heat flux operated at the high inlet temperature was about twice as fast as that for the element operated at the high outlet temperature. This was due to preferential scale buildup toward the front of the reactor tube,

as has been noted in other in-reactor experiments.

(4) The rate of cladding temperature increase appeared to be influenced only by fuel-element heat flux and location with respect to the front and rear faces. It appeared to be independent of actual tube operating temperature since approximately the same rate of temperature increase was noted for a fuel element operated at both low and high temperatures but maintained at the same heat flux and location.

The results from these and subsequent similar experiments will be used as a basis for predicting corrosion rates and mechanisms of aluminum fuel-element cladding when used in high-temperature, pressurized-water reactor-coolant systems.

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APPENDIX A: DATA

TABLE I

TEST NO. 1: KER LOOP NO. 2

Hours	Thermocouple Temperatures (C) 1	2	3	Temp In (C)	Temp Out (C)	Flow (gpm)	Tube Power (KW)	Comments
25	206	208	No response from thermocouple	178	200	60	334	
50	206	212		178	201		350	
75	201	206		138	194		850	
100	206	213		143	200		865	
125	205	209		139	194		836	
150	192	195		142	178		500	Reactor down at 133 hours
175	200	203		146	188		637	
200	198	199		144	198		819	
225	198	186		136	188		790	
250	198	172		135	188		819	
275	200	172		140	190		750	
300	198	165		134	190		850	
325	203	165		139	193		819	
350	200	160		135	193		882	
375	200	158		144	193		745	
400	200	161		140	193		805	
425	200			140	193		805	Ceased taking No. 2 reading because of its obvious error
450	193			140	193		805	
475	193			139	191		790	
500	198			141	194		805	
550	200			140	195		836	
600	198			143	188		682	Reactor down at 580 hours

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TABLE I (CONTINUED)

Hours	Thermocouple Temperatures (C) 1	2	3	Temp In (C)	Temp Out (C)	Flow (gpm)	Tube Power (KW)	Comments
650	200		No response from thermocouple	146	192	60	699	
700	200			140	193		805	
750	192			140	193		805	
800	195			145	194		745	Reactor down at 790 hours
850	194			142	195		805	
900	190			142	195		805	
950	197			150	201		775	
1000	198			150	200		759	
1050	200			152	202		759	Reactor down at 1025 hours
1100	200			150	205		836	
1150	196			147	202		836	
1200	202			155	205		759	Reactor down at 1180 hours
1250	198			148	205		865	
1300	200			144	200		850	
1350	199			146	205		895	Reactor down at 1372 hours
1400	199			150	205		836	
1450	200			150	205		836	
1500	202			143	205		836	
1550	200			146	203		865	Reactor down at 1554 hours

Avg. Power = 770 KW
Avg. Outlet Temperature = 198°C
Avg. Fuel Cladding Temperature = 201°C

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TABLE II

TEST NO. 2: KER LOOP NO. 4

Hours	Thermocouple Temperatures(C) 1	2	3	Temp In (C)	Temp Out (C)	Flow (gpm	Tube Power (KW)	Comments
25	245	246	234	174	200	60	394	
50	249	246	234	175	200		379	
75	249	247	235	176	200		364	
100	252	247	236	177	204		313	
125	261	256	240	178	205		400	
150	259	256	240	179	206		400	
175	262	254	238	178	205		400	
200	266	256	241	180	205		379	
225	266	259	240	179	205		394	
250	283	262	241	180	205		379	
275	287	266	245	180	205		379	
300	287	267	245	179	204		379	Reactor down at 302 hours
325	136	128	106	40	63		349	Low temperature operation
350	144	134	113	40	66		394	
375	145	135	113	40	67		400	
400	148	137	115	42	68		394	
425	150	139	115	41	68		394	
450	150	140	117	40	66		394	
475	152	142	117	41	68		400	
500	154	143	118	41	68		400	
550	156	145	119	41	67		394	
600	289	283	253	180	201		318	Reactor down at 573 hours
650	290	286	258	175	205		455	
700	291	284	257	173	200		400	Reactor down at 671 hours

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TABLE II (CONTINUED)

Hours	Thermocouple Temperatures (C) 1	2	3	Temp In (C)	Temp Out (C)	Flow (gpm)	Tube Power (KW)	Comments
750	291	285	257	174	201	60	400	
800	294	288	260	180	205		379	
850	288	280	252	176	200		364	Reactor down at 843 hours
900	292	280	256	181	205		364	
950	288	278	250	173	200		400	
1000	291	282	256	179	205		298	
1050	288	278	250	180	205		379	Reactor down at 1069 hours

Avg. Power = 392 KW
Avg. Outlet Temperature = 203°C (excluding low temperature operation)

TABLE III

TEST NO. 3: KER LOOP NO. 3

Hours	Thermocouple Temperatures(C) 1	2	3	Temp In (C)	Temp Out (C)	Flow (gpm)	Tube Power (KW)	Comments
3	241	-	244	210	224	60	213	No. 2 thermocouple not connected
10	242	-	242	220	233		197	" " " "
34	256	255	257	216	230		213	
39	254	255	255	216	230		213	
82	257	258	258	215	229		213	
105	238	259	260	216	230		213	
131	236	261	262	216	230		213	
154	238	270	270	220	233		197	
189	237	274	274	220	233		197	
199	240	279	273	220	233		197	
208	243	282	271					No inlet and outlet temperatures recorded
214	242	282	266					
220	241	278	261					No. 3 thermocouple went downscale
226	241	280	260					
238	244	289	262	220	234		213	
298	247	287	250	218	232		213	
321	248	253	250	218	229		163	No. 2 thermocouple went downscale
336	250	250	251	215	230		228	
370	251	247	251	217	230		197	
382	229	204	230	214	229		228	No. 1 thermocouple went downscale

Avg. Power = 206 KW
Avg. Inlet Temperature = 216°C

APPENDIX B: CALCULATIONS

TUBE POWER

The tube powers in all tests were calculated from the following equation based on water properties at an average temperature of 200 C:

$$P = 0.253\ F\ (\Delta T)$$

where P is tube power in KW

F is flow in gpm

and ΔT is difference in bulk inlet and outlet temperatures in C

FILM TEMPERATURES DROP

The Sieder-Tate equation states:

$$\frac{h\ D_e}{K} = 0.027 \left[\frac{W\ D_e}{\mu}\right]^{0.8} \left[\frac{\mu C_p}{K}\right]^{0.333}$$

or rearranging:

$$h = 0.027 \left[\frac{W^{0.8}}{D_e^{0.2}}\right] \left[\frac{K^{0.667}\ C_p^{0.333}}{\mu^{0.467}}\right]$$

h = heat transfer coefficient ($Btu/hr\text{-}ft^2\text{-}F$)

D_e = equivalent diameter (ft)

K = thermal conductivity (Btu/hr-ft-F)

W = weight rate of flow ($lb/hr\text{-}ft^2$ of cross sectional flow area)

μ = viscosity (lb/hr-ft)

and C_p = specific heat (Btu/lb-F)

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With a specific fuel element power known (Btu/hr-ft^2), the temperature drop across the water film can be obtained by

$$\Delta T = \frac{q}{h}$$

where ΔT = temperature drop across the water film (F)

q = specific power (Btu/hr-ft^2)

and h = heat transfer coefficient (Btu/hr-ft^2-F)

<u>Test No. 1</u>

For operation of the loop at 198°C(388°F) average bulk outlet temperature, from tables on the properties of water the following values are obtained:

K = 0.383 Btu/hr-ft-F (from <u>Reactor Handbook 2</u>, pp. 22-23)

μ = 0.335 lb/hr-ft (from <u>McAdam's Heat Transmission</u>, p. 407)

C_p = 1.07 Btu/lb-F (from <u>Reactor Handbook 2</u>, p. 24-25)

For a 1.44 inch OD fuel element in a 2.1 inch ID process tube

$$D_e = 4r_h \frac{(D_t^2 - D_f^2)}{(D_t + D_f)} = D_t - D_f$$

where r_h = hydraulic radius

D_t = diameter of tube

D_f = diameter of fuel element

$$D_e = \frac{2.1}{12} - \frac{1.44}{12} = 0.175 - 0.12 = 0.055 \text{ ft}$$

At a flow rate of 60 gpm at 198°C with the cross sectional flow area between the 1.44 inch O.D. fuel element and 2.1 I.D. process tube,

$$W = \frac{F\rho}{A}$$

where F = flow rate (gph)

ρ = density (lb/gal) based on <u>Thermodynamic Properties of Steam</u> by Keenan and Keyes

and A = cross sectional flow area (ft^2)

$$W = \frac{F\,\rho}{(D_t^{\,2} - D_f^{\,2})}$$

$$= \frac{60(60)(7.29)}{\left(\frac{2.1}{12}\right)^2 - \left(\frac{1.44}{12}\right)^2}$$

$$= \frac{2.62 \times 10}{0.0127}$$

$$= 2.06 \times 10^6 \text{ lb/hr-ft}^2$$

$$\therefore \quad h = 0.027 \frac{(0.383)^{0.667}(1.07)^{0.333}(20.6\text{x}10^5)^{0.8}}{(0.335)^{0.467} \quad (0.005)^{0.2}}$$

$$= 0.027 \frac{(0.528)(1.0226)}{(0.601)} \frac{11.3\text{x}10^4}{0.56}$$

$$= 0.027\,(0.898)(2.02 \times 10^5)$$

$$= 4900 \text{ Btu/hr-ft}^2\text{-F}$$

The temperature drop across the water film for this fuel element with a measured specific power of 96,000 Btu/hr-ft^2 then is:

$$\Delta T = \frac{q}{h} = \frac{96000}{4900} = 19.6\text{F} = 10.9°\text{C}$$

Test No. 2

For operation of the loop at 203°C (398°F) average bulk outlet temperature, using the same reference tables as given for Test No. 1, the following values of properties of water at this temperature are:

$$K = 0.380 \text{ Btu/hr-ft-F}$$

$$\mu = 0.328 \text{ lb/hr-ft}$$

$$C_p = 1.080 \text{ Btu/lb-F}$$

$$\rho = 7.2 \text{ lb/gal.}$$

Since the same size fuel element, process tube and flow rate are used as in Test No. 1,

$$D = 0.055 \text{ ft}$$

$$\text{and} \quad W = \frac{(60)(60)(7.2)}{0.0127} = 2.04 \times 10^6 \text{ lb/hr-ft}^2$$

$$\therefore \quad h = 0.027 \frac{(0.38)^{0.667} (1.08)^{0.333}}{(0.328)^{0.467}} \frac{(20.4 \times 10^5)^{0.8}}{(0.055)^{0.2}}$$

$$= 0.027 \frac{(0.526)(1.026)}{(0.595)} \frac{(11.2 \times 10^4)}{(0.56)}$$

$$= 0.027 \, (0.908) \, (2.0 \times 10^5)$$

$$= 4890 \text{ Btu/hr-ft}^2\text{-F}$$

The temperature drop across the water film for this fuel element with a measured specific power of 312,000 Btu/hr-ft^2 then is:

$$\Delta T = \frac{q}{h} = \frac{312000}{4890} = 64.0°F = 35.5°C$$

Test No. 3

For operation of the loop at 216°C (421°F) average bulk inlet temperature, using the same reference tables as given for Test No. 1, the following values for properties of water at this temperature are:

$$K = 0.376 \text{ Btu/hr-ft-F}$$

$$\mu = 0.31 \text{ lb/hr-ft}$$

$$C_p = 1.095 \text{ Btu/lb-F}$$

$$\rho = 7.11 \text{ lb/gal}$$

Since the same size fuel element, process tube, and flow rate are used as in Test No. 1,

$$D = 0.055 \text{ ft}$$

$$\text{and} \quad W = \frac{60(60)7.11}{0.0127} = 2.02 \times 10^6 \text{ lb/hr-ft}^2$$

$$\therefore \quad h = 0.027 \; \frac{(0.376)^{0.667}(1.095)^{0.333}}{(0.31)^{0.467}} \; \frac{(20.2 \times 10^5)^{0.8}}{(0.055)^{0.2}}$$

$$= 0.027 \; \frac{(0.521)\,(1.0306)}{(0.58)} \; \frac{11.1 \times 10^4}{(0.56)}$$

$$= 0.027\,(0.929)\,(1.98 \times 10^5)$$

$$= 4960 \text{ Btu/hr-ft}^2\text{-F}$$

The specific power of the fuel element in this test would be about the same as that used in Test No. 2 and since it was not measured, a value of 310,000 Btu/hr-ft^2 is assumed. The temperature drop across the water film then is:

$$\Delta T = \frac{q}{h} = \frac{310000}{4960} = 62.5^\circ F = 34.7^\circ C$$

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