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HEAVY WATER ORGANIC COOLED REACTOR

EXP-NRU-305 EXPERIMENT DESIGN AND IRRADIATION

By

G. E. CULLEY
J. G. LeBLANC
A. F. LILLIE

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ABSTRACT

The initial fuel irradiation test of the HWOCR program, designated the EXP-NRU-305, was designed to establish the irradiation performance capabilities of the reference and backup fuels for the HWOCR, UC and UO_2 , respectively, under conditions representative of HWOCR design. The irradiation test was conducted in the organic cooled U-3 loop of the NRU reactor at Chalk River, Canada, between July 1965 and April 1966.

The fuel string consisted of three 18-rod clusters of SAP (Sintered Aluminum Powder) clad hyperstoichiometric uranium carbide and two 18-rod clusters of Zircaloy-4 clad UO_2 , mounted in tandem. The uranium carbide portion was designed to operate at a peak specific power of 40 kw/kgU at an initial central fuel temperature of 2000°F, with a cladding temperature of 850°F. The UO_2 portion of the string was designed to operate with a maximum temperature of the cored- UO_2 pellets of 3000°F, and a cladding surface temperature of 820°F. The experiment coolant was Santowax OM with 20% high boiler compounds at a nominal 600°F temperature and a 30 ft/sec velocity in the test section.

The irradiation test was terminated on April 16, 1966, following 263 days in-pile and 206 days at the design string power of 1.7 Mw, as a result of the release of rare fission gases from the rupture of several Zircaloy 4 clad UO_2 rods. At the time of discharge, the peak UC burnup was approximately 10,000 Mwd/MTU, and the peak UO_2 burnup approximately 8500 Mwd/MTU.

During the irradiation test, fuel string cladding temperatures, coolant temperatures, test assembly hydraulic and coolant chemical characteristics were closely monitored. Analysis of operating characteristics indicated in general that the string operated in a stable manner during its irradiation with an increase in string hydraulic pressure drop of about 10%, and with formation of a heat transfer surface film with a maximum thickness of approximately 0.002 in. Calculated film thicknesses were corroborated by post-irradiation examination.

I. INTRODUCTION

The Heavy Water Organic Cooled Reactor (HWOCR) advanced converter concept has the inherent advantages of the high degree of fuel resources conservation, flexibility in the types of fuel cycles which it can use and in its potential as a dual-purpose plant. These advantages accrue from the characteristics of excellent neutron economy, compatibility of the coolant with fuel materials of interest, simplicity of the fuel element design, and low fuel cycle cost. Based on capital charge rate prevalent in the United States, this reactor appears economically competitive, by using natural or slightly-enriched fuel. To profit from these advantages and economics, an HWOCR fuel irradiation testing program has been initiated to obtain results on which to base the reactor fuel element design. The EXP-NRU-305 experiment (hereafter designated as the U-305 test) was the first experiment of this program.

The U-305 test was designed and fabricated as a part of the joint venture between Atomics International and Combustion Engineering (AI-CE) on the HWOCR program. The U-305 test was the first of three fuel strings constructed for the purpose of materials evaluation as applied to the HWOCR concept. The objectives of the U-305 test were to:

- 1) Evaluate the performance capabilities of both the SAP-UC and Zr- UO_2 fuel systems under conditions appropriate to HWOCR,
- 2) Provide data for the UC fuel growth model,
- 3) Furnish data for the UC fuel fission gas release model,
- 4) Provide data for the SAP cladding behavior model, and
- 5) Develop information on coolant chemistry behavior and fouling film formation.

Fulfillment of these objectives through the U-305 test was intended to verify and extend the state-of-the-art in HWOCR fuels systems technology, to provide a basis for design of subsequent tests involving fuel assemblies, and to provide engineering data for the design of the HWOCR.

The ability of the U-305 experiment to meet its objectives is to a large extent dependent on the similarity of its design, operating conditions, and performance with the reference HWOCR demonstration plant design. This requirement was fulfilled as can be seen from examination of Table 1 which compares

TABLE 1
COMPARISON OF HWO CR AND U-305 DESIGN
AND OPERATING PARAMETERS*

Parameter	HWO CR	U-305
Physical Configuration	37-pin hexagonal array	18-pin circular array
Fuel and Cladding	SAP-UC	SAP-UC and Zircaloy-UO ₂
Coolant	Santowax OM + 10% HB	Santowax OM + 10-20% HB
Fuel U ²³⁵ Enrichment	1.3%	1.5%
Pressure Tube In-Core L/D	54.6	33.3
Fuel Area/Pressure Tube Area	0.435	0.438
Fuel Surface Area/Fuel Volume	0.706 ft ² /ft ³	0.669 ft ² /ft ³
Peak Unperturbed Neutron Flux	2.8×10^{14} n/cm ² -sec	2.4×10^{14} n/cm ² -sec
Peak Initial Specific Power	48.1 Mw/MTU	46.8 Mw/MTU
Peak Cladding Heat Flux	298,000 Btu/ft ² -hr	326,000 Btu/ft ² -hr
Peak Fuel Temperature	2100°F	2000°F
Peak Cladding Temperature	870°F	850°F
Bulk Average Power Density	4.11 Mw/ft ³ core	3.96 Mw/ft ³ core
Bulk Average Core Coolant Temperature	662°F	661°F
Coolant Velocity	30 ft/sec	30 ft/sec
Average Core Coolant Pressure	217 psi	200 psi

* For U-305, all parameters are for UC elements unless otherwise stated. Information was obtained from References 1 and 2.

the basic design, and operating parameters of the HWO CR and U-305 systems. Details of these systems can be found in References 1, 2, and 12.

The organic cooled U-3 loop in the NRU Reactor at Chalk River, Ontario, Canada, was the irradiation facility used for the U-305 test. The loop was made available to the HWO CR fuel irradiation testing program under a memorandum of agreement between the USAEC and the Atomic Energy of Canada, Ltd. (AECL) who own and operate the NRU Reactor. The U-305 fuel string was installed in the loop in late July 1965 and the irradiation of this fuel string was terminated in mid-April 1966.

II. EXPERIMENT DESCRIPTION

A. DESCRIPTION OF THE U-305 FUEL STRING

1. The U-305 Assembly

The U-305 fuel string assembly consists of three SAP (Sintered Aluminum Powder) clad UC fuel bundles mounted in tandem with two Zircaloy-4 clad UO_2 fuel bundles, as illustrated in Figure 1. The fuel bundles are mounted on a 3/8-in. stainless steel hanger rod passing through the center of each bundle. The end-plates of adjacent bundles are keyed together and the top end-plate of the uppermost bundle is keyed to the upper retainer assembly to prevent bundle rotation. The hanger rod is shrouded by a dummy cladding tube, where it passes through the fuel bundles, to preserve hydraulic characteristics and to house instrumentation leads running from the test section. An Inconel X spring (Figure 2) is positioned at the bottom of the fuel string assembly to maintain a compressive load to assure nesting of the bundles, and to accommodate differential thermal expansion between the fuel bundles and the hanger rod.

The test section is vertically supported in the U-3 loop pressure tube by a hanger tube welded to the upper retainer assembly. The breech block assembly, which is welded to the hanger tube, seats on the pressure tube extension. The instrumentation leads from the fuel string pass through the breechblock and are sealed with a graphite compression gland. The instrumentation leads are terminated in a connector assembly above the breechblock, and a cable with mating connector connects the instrumentation with the appropriate loop and data retrieval systems. This assembly is shown in Figure 3.

2. Fuel Bundles

a. SAP-UC Fuel Bundles

The nuclear fuel was arc-cast hyperstoichiometric UC enriched to 1.5 wt % U^{235} . The fuel slugs were of random (1-in. to 6-in.) lengths, with a finished OD of 0.4995 in. $\pm$ 0.0005 in. Each fuel slug was dished on one end to a depth of about 0.055 in., with a spherical radius. These fuel slugs were then loaded into cladding tubes to give a fuel stack length of about 13.6 in. Loading was done according to carbon content, so as to permit postirradiation evaluation of the

Figure 1. U-305 Test Assembly

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Figure 2. U-305 Compression Spring and Lower Hardware

Figure 3. U-305 Assembly Drawing.

influence of carbon chemistry on fission gas release; the loading schedule is given in Table 2. Complete preirradiation data for the UC fuel may be found in Reference 3.

TABLE 2
U-305 UC FUEL LOADING SCHEDULE

Bundle Identification and Position in U-305	Fuel Rod Cladding Material	Fuel Column Length (in.)	Number of Fuel Rods per Fuel Compositional Range	Fuel Rod Identification	Fuel Compositional Range (wt % C)
LRA (top)	XAP-001	13.6	3	B, C, D	4.82 to 4.92
			3	E, F, H	4.92 to 5.00
			3	J, K, L	5.00 to 5.10
			3	M, N, P	5.10 to 5.20
			6	R, S, T, W, X, Y	4.92 to 5.20
LSA	XAP-001	13.6	3	B, C, D	4.82 to 4.92
			3	E, F, H	4.92 to 5.00
			3	J, K, L	5.00 to 5.10
			3	M, N, P	5.10 to 5.20
			6	R, S, T, W, X, Y	4.92 to 5.20
LTA (bottom)	XAP-001	13.6	3	B, C, D	4.82 to 4.92
			3	E, F, H	4.92 to 5.00
			3	J, K, L	5.00 to 5.10
			3	M, N, P	5.10 to 5.20
			6	R, S, T, W, X, Y	4.92 to 5.20

The cladding material was SAP containing 5 to 8 wt % Al_2O_3 . The material was extruded into a finned tube geometry, as indicated in Figure 4a. The tubes were twisted to give a $90^\circ/\text{ft}$ spiral to the fins to provide rod-to-rod spacing and promote intersubchannel coolant mixing. The cladding ID was 0.5045 in. $\pm$ 0.0005 in., which yielded a nominal 0.005-in. diametral gap between the fuel and cladding. Complete preirradiation data for the cladding may be found in Reference 4.

a) FINNED CLADDING TUBE CROSS-SECTION

b) ASSEMBLED SAP-CLAD UC FUEL ROD

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Figure 4. U-305 Test Series SAP Cladding and Fuel Rod Geometry
(All dimensions shown are in inches)

The fuel tube closures were formed from 1100 S aluminum end plugs, approximately 3/4 in. long, which were silver-eutectic-bonded to the fuel tube. After the fuel was loaded in the tubes, the final closures were made in a helium atmosphere to improve the heat transfer between the fuel and cladding. The assembled fuel rod is shown in Figure 4b.

The fuel rods were then assembled into bundles which consisted of 18 rods, arranged in a circular array around a central dummy cladding tube housing the hanger rod. The end plates were flat 1/16-in. stainless steel, and the fuel rods were attached with a screw and pinned to prevent rotation. The completed fuel bundle was 16-5/8 in. long with a diameter over the fin tips of 3.175 in. $\pm$ 0.005 in., and 3.220 in. $\pm$ 0.002 in. at the spacer rings. A typical SAP-UC fuel bundle is shown in Figure 5.

b. Zircaloy-UO₂ Fuel Bundles

Two Zircaloy-UO₂ fuel bundles were used in the U-305 fuel string. One bundle consisted of fuel rods with six spacer wires attached to the tube wall by electron beam (EB) welding techniques. The other bundle contained fuel rods with spacer wires attached by spot welding techniques. The spot-welded-type bundle occupies the bottom-most position on the string with the EB welded type bundle mounted directly above it. The completed spot-welded bundle is shown in Figure 6a, and the EB welded bundle in Figure 6b.

The UO₂ fuel was in the form of cored pellets, each approximately 1/2 in. long, with a core ID of 0.184 in. and a pellet OD of 0.522 in. $\pm$ 0.0005 in. The fuel pellets were dished on one end similar to the UC slugs, and were procured to CEND Specification NL-100. The fuel was enriched to 1.12 wt % in the top UO₂ bundle and to 1.52 wt % in the bottom UO₂ bundle. Special solid end-pellets, approximately 3/8 in. long, each of depleted UO₂, were located on each end of the fuel stack in each rod, yielding an overall stack length of 24.375 in.

The Zircaloy-4 cladding material was procured to the requirements of CEND Specification NL-100. Measured tube ID's ranged from 0.531 to 0.534 in., yielding a helium gas bond diametral thickness of 0.009 to 0.012 in. The clearance between the top end-plug and the top of the fuel stack was a nominal 0.160 in. As with the finned SAP cladding, the Zircaloy-4 wire wrap tubing was twisted to give a 90°/ft spiral to the spacer wires to provide rod-to-rod spacing and promote intersubchannel coolant mixing.

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Figure 5. Typical SAP-UC Fuel Bundle

a. With Spot-Welded Wire Wrap

b. With EB-Welded Spacer Wires

Figure 6. CE-Fabricated Zircaloy- UO_2 Fuel Bundles
(Photographs provided by AECL)

Figure 7. Typical Thermocouple Installation
in SAP Cladding Fin

3. Description of Other Test Assembly Features

a. Instrumentation

The SAP-UC fuel bundles were instrumented with seven 0.020-in.-OD (swaged from 0.040 in.) stainless steel sheathed, grounded junction, Chromel-Alumel thermocouples. The thermocouples were embedded in the fins of selected fuel rods as shown in Figure 7, and were used to monitor the SAP cladding temperatures during irradiation. The thermocouple installation schedule for the three SAP-UC bundles is shown in Figure 8. The Zircaloy 4 - UO_2 bundles were not instrumented.

In addition to the cladding thermocouples, five thermocouples were located in the coolant stream at different bundle interfaces. The locations of these thermocouples are also shown in Figure 8. Figure 9 shows a completed bundle assembly depicting fuel rods, end-plate with spacer ring, cladding T/C, and coolant T/C.

b. Zirconium Specimens

Zirconium alloy specimens were attached to the hanger tube of the U-305 test assembly. The specimens were Zircaloy 4, Zircaloy 2, and Zr - 2.5 Nb, which were subjected to various surface treatments to determine relative rates of hydriding and other effects in organic coolant. Figure 10 is a photograph of the specimen installation, and the table in Figure 3 describes the treatment process for each specimen.

B. DESCRIPTION OF THE IRRADIATION FACILITY

The irradiation facility used for the U-305 experiment was the organic cooled U-3 loop of the NRU reactor. This facility is located at the Chalk River National Laboratory (CRNL), Chalk River, Ontario, Canada, which is owned and operated by the Atomic Energy of Canada Limited (AECL).

1. The NRU Reactor

The NRU was originally a 220-Mwt natural uranium fueled, heavy water moderated and cooled reactor. In 1964, the reactor was converted to use a fully enriched uranium fuel and is presently operating near 100 Mwt. In addition to its primary product of isotopes, the NRU contains three separate fueled test

Figure 8. U-305 Fuel Rod, Bundle, and Thermocouple Schematic

Figure 9. A Completed SAP-UC Instrumented Fuel Bundle

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Figure 10. Zirconium Specimens Installed on U-305 Experiment

loops which enjoy a relatively high thermal neutron flux and a high plant factor of about 90%. The reactor utilizes on-line refueling and has the ability to compensate over a wide range for the flux requirements of the individual loops. This allows the reactor and the loops to run constantly at their maximum power for the duration of an experiment, even though fresh fuel may have been inserted into one of the other loops during the interim. Reference 5 describes the NRU after the 1964 modification.

2. The U-3 Loop

The U-3 test facility is an organic cooled loop with its test section in the E-12 core position (see Figure 11) of the NRU. This loop was initially constructed for the purpose of irradiation-testing the WR-1 (Whiteshell Reactor) fuel bundles and pressure tubes. In 1965, the U-3 facility and operational services were contracted on a rental basis by the USAEC for the purposes of providing a suitable irradiation facility for the HWOOCR program experiments. The loop is described in References 6 and 7 in detail, and the following description is to orient the reactor for the operational phases of the U-305 experiment to be described.

The loop is of carbon steel construction and, with the exception of the Zircaloy 4 pressure tube, has a maximum rating of a test section outlet temperature of 800°F (427°C) and a pressure of 600 psia. The primary organic coolant rejects heat to the secondary river water coolant through a refluxing-type heat exchanger with a nominal rating of 2 Mw. Table 3 lists the nominal operating conditions of the U-3 loop.

TABLE 3
U-3 LOOP NOMINAL OPERATING CONDITIONS

Heat Rejection	1.66 Mw
Flow	300 USgpm (16.5 kg/sec)
Pump Inlet Pressure	100 psi
Pump Discharge Pressure	300 psi
Test Section Inlet Temperature	575°F
Test Section Outlet Temperature	655°F
Test Section Outlet Pressure	200 psi
Coolant Type	Santowax OM + 15% HB

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Figure 11. U-3 Loop Position in NRU Core

The U-3 loop is illustrated schematically in Figure 12. The loop consists of three primary subsystems. These are the main loop, the degassifier system, and the purification system. The characteristics and purposes of each of these subsystems are as subsequently described.

a. Main Loop

The main loop provides the circulating coolant at controlled temperature and flow rate for use in the test section. The operation of the main loop circuit under normal U-305 conditions is described in the following paragraphs.

The organic coolant is continuously circulated by two of the three main pumps. The third pump remains on standby and is automatically started when another pump fails or a loop trip occurs. The coolant discharged from the pumps passes directly to the test section where it is used as a heat transfer and transport medium for the thermal energy generated by the fuel string.

Coolant flow rate control is provided in the test section outlet piping by an orifice plate sensing element in conjunction with a throttling valve. The coolant temperature regulation system is next in line. A flow diverter valve is used to partially bypass coolant around the main loop cooler, as required to maintain the test section inlet temperature constant. The loop cooler is a refluxing shell-type cooler in which the organic stream loses heat by boiling the water contained in the shell, which in turn gives up heat by condensing on tubes containing the secondary river water coolant.

The surge tank is next in line, and is used to provide a constant inlet pressure to the pumps and for coolant level control. Pressurization is provided by a bottled nitrogen supply and a vent system. The surge tank liquid level is controlled by using the dump tank as a liquid source or sink as required.

Electrical resistance heaters are incorporated in the main loop to maintain the coolant at desired temperatures during non-nuclear operation. They are positioned between the flow orifice and the flow control valve, and are not bypassed during nuclear operation. Another special component associated with the main loop is the water addition system. This system provides a means of maintaining the desired water concentration in the coolant up to a maximum content of approximately 900 ppm. The system consists of a distilled water source and a variable capacity positive displacement pump.

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Figure 12. Simplified U-3 Loop Schematic

The portion of piping that passes through the reactor is a special pressure vessel that completely isolates the loop from the reactor, and is termed the pressure tube. The present pressure tube is constructed of Zircaloy 4 with a 3.25-in. ID and a length of approximately 30 ft. The U-3 pressure tube is separated from the NRU D₂O moderator by a CO₂ annulus and a Zircaloy 2 calandria tube. The design operating pressure for this pressure tube is 335 psi at 750°F.

b. Degasification System

A degasification system was installed on the U-3 loop to remove the non-condensable gases from the main loop coolant. This is necessary to maintain the coolant vapor pressure below the saturation point at the test section discharge. The degasification system also provides a flow of coolant to the dump tank which is used for level control in the surge tank.

The degasification system is operated in the following manner. A side-stream is withdrawn from the main loop downstream of the flow control valve and fed to the degas tank. The temperature and pressure are adjusted so that about 10% of the feed stream is flashed to vapor. This vapor stream, composed of noncondensable and low boiling vapors flows into two equilibrium condensers, where reusable coolant is condensed and returned to the main loop. The resulting vapor stream then enters two cold traps where water and undesirable light ends are collected and discharged into the waste system. The resulting noncondensable gases are then vented to atmosphere. The degased coolant and the condensed coolant are fed to the dump tank and returned to the main loop.

The nominal operating conditions of the degasifier system during U-305 operation were as follows.

Degas feed flow	5 USgpm
Feed pressure	110 psia
Feed temperature	650°F
Degas tank pressure	50 psia
Degas tank temperature	565°F
Percent of feed flashed	10%
Noncondensable discharge flow	1 ft ³ /min at 45 psi and 212°F

Figure 13. One of Five U-3 Loop Control Panels
(Photograph supplied by AECL)

C. PURIFICATION SYSTEM

The intended purpose of the purification system is to remove gross particulate matter and fouling producing species from the organic coolant. During normal operation, a 10-USgpm side stream from the main loop pump discharge is fed into two adsorption columns which may be connected either in parallel or in series. These columns use 210 lb of activated attapulgite adsorbent material per column. The purification stream then enters two glass fiber filters with 1- μ porosity. The clean organic stream is then returned to the main loop.

A heater is provided on the inlet to the system to maintain the purification stream at about 650°F. A chlorine removal column is also available (though not normally used) in this system. A pelletized 0.5-wt % palladium on alumina catalyst which is capable of removing various chemical forms of chlorine is used.

1. Experiment Data System

The U-305 experiment data system includes the experiment instrumentation and control system, the data collection system, and the data reduction system.

a. Experiment Instrumentation

The experiment instrumentation and control system is connected with two primary sources — the loop and the fuel string. One of five control panels for the loop is shown in Figure 13. Figure 12 shows some of the more important main loop instrumentation to be discussed. Resistance temperature detectors (RTD's) are provided for accurate inlet and outlet test section temperature measurements. Test section inlet and outlet pressures are monitored as well as other pressures around the loop. A differential pressure monitor, designated dP-2, is positioned across the test section to detect changes in the hydraulic performance of the fuel string. The main loop flow is measured with an orifice plate utilizing three differential pressure (D. P.) monitors in parallel to determine the pressure drop across the orifice. The signal from one D. P. cell is used to control the flow, the signal from the second is sent to the data logger, and the third instrument is an unrecorded backup. Nuclear instrumentation on the U-3 loop includes a coolant gamma activity monitor which is positioned downstream of the test section, and flux monitors positioned on the pressure tube. These flux monitors are of the vanadium-beta emitter type and are

approximately 8-ft long. Each flux monitor is wrapped around the pressure tube, and a suitable millivolt signal is obtained by measuring the voltage drop across a 10-k Ω resistor in the circuit.

The U-305 fuel strong was instrumented with 12 thermocouples. The type and placement of these TC's are described in Section II.A.3a. These thermocouples were operated against a 100°C reference junction, and were monitored on both the data logger and recorders.

The loop instrumentation also provided alarm and trip functions in the event of loop operation outside prescribed limits. The instrumentation as designed is capable of initiating two types of trips — a loop-only trip and a reactor trip. The loop-only trip results in full cooling by diverting all coolant into the loop cooler, and maximum flow by starting the third main pump and opening the flow control valve. A reactor trip initiation occurs only in the event a loop trip will not provide a safe condition. Selected instruments are used to initiate these trips under conditions of low flow, low pressure, low liquid level, high temperature, main pump failure and fire. During the operation of the U-305 experiment, however, the limits were adjusted so that the reactor trip would be initiated prior to the loop trip.

b. The Data Collection System

The data collection system consisted of the usual manual data logged from meters, gauges, and strip chart recorders, in conjunction with a sophisticated data logger and computer system. The manual logging of data was done by the loop operators, either once a shift or once an hour, depending on the importance of the data. These "loop logs" were an important source of information to assist in analyzing experiment behavior.

The automatic data collection system used for the NRU and NRX loops was also used to log data for the U-305 experiment. It is called REDACE, which stands for REactors DAta CEnter. The system utilizes a multiplexing input, analog to digital conversion, magnetic core memory, computerized conversion, with a punched paper tape and typewriter output. Simply, this system obtains the input signals either on demand or at a preset interval, converts them to digital engineering units, and presents them on a typewriter listing. The unit performs all conversion during the data scan and can be called an on-line

computer in this sense. The typewriter output was used as the primary source of immediate data for the U-305 experiment operation. The punched paper tape generated by REDACE was run on the G-20 computer at ORNL so that the data pertaining to only the U-3 loop were extracted and placed on punched cards. These cards were then shipped to AI where further data reduction was performed.

The complete data acquisition system is schematically illustrated in Figure 14. The two types of data procurement (REDACE and manual) were used to assemble the operational data.

The process of utilizing the punched card data from REDACE is illustrated in Figure 15. As previously mentioned, REDACE provides a data logging function for all the loops in the NRU and NRX reactors. The punched paper tape from REDACE therefore contains much more information than just that from the U-3 loop. The desired U-3 loop data were extracted from the punched paper tape by the G-20 computer at ORNL. The output was punched cards which contained all the U-3 loop data for each scan on six cards. During normal operation, a data scan was taken every half-hour which resulted in about 13,000 data scans being taken during the U-305 irradiation. This many data scans resulted in the accumulation of about 78,000 punched cards or thirty-five 14-in. boxes of data cards. After being shipped to AI at periodic intervals, the data were computer-processed so that they could be utilized. The first step was a pre-processing operation necessary to provide data compatible with the subsequent computer codes. The first raw data from the experiment contained an assortment of random errors of the type that precluded the economic use of a computer to correct. The preprocessing philosophy adopted was that the bad data cards were listed with comments about the errors,⁽⁸⁾ and then manual editing (removal or correction) of the cards was performed to eliminate incompatible data scans. Later in the experiment, the REDACE - G-20 combination was improved so that very little manual correction was necessary. The data cards were then processed with a computer code⁽⁹⁾ to perform conversions, calculations, listings, and plotting of the data. The raw data were in units commonly used in Canada, and conversion to U.S. engineering units was necessary. Typical conversion was °C to °F, and bars to psi. The REDACE computer calculated channel temperature rise, mass flow, and experiment power, so that

Figure 14. The U-305 Data Acquisition System

Figure 15. Schematic of the Computerized U-305 Data System

performance of these calculations at AI was not necessary. However, calculations for burnup and beta parameters were performed. Because of the quantity of data and thermocouple calibration changes, a "run" was limited to one month's worth of data. Examples of both the listing and the plots are contained in Figure 16.

D. EXPERIMENTAL TEST PLAN AND EXPECTED RESULTS

The U-305 Test Plan consisted of three phases: Startup, Long Term Operations, and Postirradiation Examination. The first phase consisted of inserting the U-305 fuel string into the NRU U-3 facility, and establishing system design operating conditions as listed in Table 4. Except for the neutron flux and the test section outlet pressure, the conditions in Table 4 were to be independently established by appropriate operation of the irradiation facility process equipment. The neutron flux was to be the primary control parameter and was to be adjusted as necessary to establish the design power output of the fuel string. The listed outlet pressure was an estimate based on the design inlet pressure and calculated test section ΔP .

After satisfactory completion of the startup phase, the long-term operation phase was to be initiated. This phase would then be continued until an accumulated fuel string peak burnup of 10,000 Mwd/MTU was reached. The plan for this phase was to operate the experiment at continuous steady-state design condition except for periodic NRU cycle shutdowns. During these periods, the U-3 loop would be maintained at design flow and temperature with the aid of the loop heaters. In addition, the fuel string and coolant performance were to be monitored continuously and compared with the expected performance as listed in Table 5 and Figure 17. The coolant chemistry parameters, MST and SPFT, indicated in Table 5, are both measurements of the fouling potential of the coolant.*

* The first test, MST, is performed by filtering a measured quantity of coolant through a small filter paper and measuring the weight of filtrate and the amount of light adsorption of the dirty filter paper. The second test, SPFT, is conducted by circulating a small main loop coolant side-stream past a heated wire held at constant temperature, and measuring the weight of film buildup on the wire over a specific period of time.

U-3 LOOP PARAMETERS FEBRUARY 1966

DAY-HR-MIN	INLET PRESSURE (PSI)	PRESSURE DROP (PSI)	INLET TEMP. (DEG. F)	OUTLET TEMP. (DEG. F)	LOOP DELTA T (DEG. F)	COOLANT FLOWRATE (LBS/HR)	COOLANT ACTIVITY (CPS)	LOOP POWER (KW)	POWER RATIO (ACTL./DESN.)
22-11- 0	322.57	120.91	567.0	645.4	78.44F	134827.	19.	1736.F	1.046F
22-11-30	322.86	119.75	566.8	645.3	78.53F	134732.	18.	1737.F	1.046F
22-11-50	322.28	120.18	566.8	645.1	78.26F	134930.	16.	1734.F	1.044F
22-12-30	322.86	120.33	566.8	645.1	78.44F	134906.	20.	1731.F	1.046F
22-13- 0	322.86	120.62	566.8	645.6	78.80F	134525.	18.	1740.F	1.048F
22-13-30	323.00	120.76	566.4	645.4	79.16F	134898.	18.	1753.F	1.056F
22-13-40	324.31	119.89	567.0	645.3	78.26F	134740.	19.	1731.F	1.043F
22-14-33	322.57	119.89	566.8	644.9	78.25F	134478.	17.	1728.F	1.041F
22-15- 0	323.29	120.62	566.2	645.4	79.25F	134827.	20.	1754.F	1.056F
22-15-36	321.99	119.31	566.6	645.3	78.71F	134470.	18.	1738.F	1.047F
22-16- 0	321.99	119.75	566.4	644.9	78.52F	134390.	19.	1732.F	1.044F
22-16-30	323.29	120.76	566.4	645.3	78.89F	135089.	17.	1749.F	1.054F
22-17- 0	323.58	120.33	566.4	645.3	78.98F	134644.	18.	1745.F	1.051F
22-17-30	322.86	120.91	566.2	645.4	79.25F	134819.	19.	1754.F	1.056F
22-18- 0	322.86	121.05	566.4	645.3	78.89F	135041.	20.	1749.F	1.053F
22-18-30	322.86	120.33	567.0	645.3	78.44F	134898.	18.	1737.F	1.047F
22-19- 0	322.57	120.91	566.6	645.3	78.71F	135073.	19.	1745.F	1.051F
22-19-30	323.29	120.04	566.8	641.5	74.72F	134589.	17.	1650.F	0.994F
22-20- 0	322.42	120.91	566.8	643.3	76.54F	134359.	16.	1688.F	1.017F
22-20-30	323.29	120.62	567.0	643.1	76.18F	134295.	19.	1679.F	1.011F
22-21- 0	323.15	120.76	566.4	643.6	77.35F	134779.	17.	1711.F	1.030F
22-21-30	323.29	121.20	566.6	637.7	71.17F	134383.	18.	1568.F	0.945F
22-22- 0	323.29	120.91	567.5	640.9	73.46F	135121.	17.	1629.F	0.981F
22-22-30	323.44	119.89	566.8	644.9	78.16F	134367.	17.	1724.F	1.039F
22-23- 0	324.02	120.91	566.4	641.3	74.99F	134986.	18.	1660.F	1.000F
22-23-30	323.58	120.18	566.6	642.4	75.71F	134954.	17.	1677.F	1.010F
23- 0-30	323.00	120.91	566.6	641.3	74.81F	135335.	18.	1661.F	1.000F
23- 1- 0	323.29	120.47	566.8	640.8	74.09F	134208.	17.	1631.F	0.983F
23- 1-30	323.87	120.04	566.6	646.3	79.70F	134240.	16.	1757.F	1.058F
23- 2- 0	322.86	120.76	566.6	646.2	79.70F	134803.	16.	1764.F	1.063F

a. Data Listing

Figure 16. Samples of Computerized Data Output

TABLE 4
EXPERIMENT DESIGN OPERATING CONDITIONS

Nuclear	
Peak unperturbed thermal neutron flux (n/cm ² sec)	2.4×10^{14}
Peak UC outer ring flux	9.9×10^{13}
Peak UC inner ring flux	8.8×10^{13}
Peak UO ₂ outer ring flux	9.9×10^{13}
Peak UO ₂ inner ring flux	7.0×10^{13}
Coolant Thermal and Hydraulic Conditions	
Inlet temperature (°F)	620
Outlet temperature (°F)	702
Flow rate (gpm)	306
(lb/hr)	134,300
(ft/sec)	30
Inlet pressure (psi)	300
Heat load (Mw)	1.66
Coolant Chemistry Control	
Water content (ppm)	100 ± 50
Iron content (ppm)	<2
Ash content (ppm)	<5
Chlorine content (ppm)	<2
High boiler content (wt %)	20
Noncondensables (cc/kg)	50

TABLE 5
U-305 DESIGN PERFORMANCE PREDICTIONS

Nuclear and Thermal Performance	
Initial peak specific power (Mwd/MTU)	
UC	46.8
UO ₂	39.5
Initial peak fission rate (fissions/cc-sec)	
UC, outer ring	1.94×10^{13}
UC, inner ring	1.24×10^{13}
UO ₂ , outer ring	1.31×10^{13}
UO ₂ , inner ring	0.92×10^{13}
Initial peak linear heat rating (kw/ft-rod)	
UC	20.8
UO ₂	13.5
Initial peak temperature (°F)	
UC fuel centerline	2000
UC cladding OD	850
UO ₂ fuel centerline	3000
UO ₂ cladding OD	815
Initial peak cladding heat flux (Btu/ft ² -hr)	
UC	326,000
UO ₂	304,000
Power output (Mwt)	
Total	1.66
UC	0.92
UO ₂	0.74
Coolant Hydraulic and Chemistry	
Fuel string pressure drop (psi)	85
Membrane strain test (MST) (Adsorption units/mg x 10 ⁵)	10
Small probe fouling test (SPFT) (μg/cm ² -hr)	2
Coolant ash content (ppm)	5
High boiler formation rate (lb/Mwd)	10

Figure 17. U-305 Axial Temperature Profiles

The final test phase, termination and postirradiation examination, was to be initiated following cladding defect. The plan for this phase was to have the reactor shut down and the fuel string kept in-pile for 24 hr with full coolant circulation. After this initial cooldown, the string was to be removed from the reactor and placed vertically in a storage block containing HB-40 coolant. When the radioactive and thermal activity had decayed sufficiently, the string would then be transported to a hot cell for detailed examination and measurement of the cladding fouling films, fuel swelling, fission gas release, and burnup.

III. EXPERIMENT OPERATION HISTORY

A. PREIRRADIATION EVENTS

The U-305 fuel string was assembled at Chalk River from the components supplied by both Atomics International and Combustion Engineering. The three instrumented SAP-UC fuel bundles and the fuel string hardware were fabricated at AI and shipped to CRNL. Of the two Zircaloy- UO_2 bundles, JRA (centermost Zircaloy- UO_2 bundle) was fabricated at CE, and bundle JPA (bottom fuel string bundle) was fabricated for CE by Canadian Westinghouse at their Peterborough facility. Assembly of the fuel string proceeded by installing the instrumented SAP-UC bundles on the hanger rod, which were then followed by the two Zircaloy- UO_2 bundles. The lower end hardware was positioned, and a load of 300-lb force applied to the fuel bundles by compressing the string with the adjusting nut. The instrumentation leads were passed up through the hanger tube, which was then welded to the upper end hardware and hanger rod. The final step was to make the instrumentation lead splices and attach them to the connector.

Site representatives, who supervised the conduct of the HWO CR U-3 irradiations, arrived at CRNL in mid-May 1965. At this time, the U-302 experiment (an AECL SAP and Zr - 2.5 Nb clad UC fuel string) was being irradiated in the U-3 loop, and its operation provided an opportunity to familiarize the site representatives with the U-3 loop. Operation of the U-302 ended about the 1st of July. This fuel string was removed, as was the existing U-3 pressure tube. A new Zircaloy 4 pressure tube, designated EXP-NRU-306 (described in Section II-B-2), which had five vanadium beta emitter-type flux monitors positioned to coincide with each of the U-305 fuel bundles, was installed in the U-3 loop at this time. A dummy fuel string was inserted in the loop and the reactor operated for about two weeks before shutting down in late July. At this point in time, the U-3 loop was ready for operation, the coolant was being purified, the U-305 fuel string was assembled, and the HWO CR irradiation program was about to commence.

B. INITIAL OPERATION

The U-305 experiment was installed in the U-3 loop in the NRU reactor on July 28, 1965. Insertion of the fuel string and hookup of the experiment instrumentation were performed routinely. Following insertion, the coolant flow rate

was brought to the design value of 16.9 kg/sec (306 USgpm), and the bulk coolant temperature increased with the loop heaters. During this temperature increase, TC 24 failed. After the coolant had become isothermal at 650°F, the fuel string thermocouples were calibrated with the inlet and outlet coolant temperature sensors. Agreement within $\pm 1^\circ\text{F}$ (0.5°C) was indicated.

The hydraulic characteristics of the fuel string were then determined by measuring the pressure drop at three coolant flow rates. The results of this test are shown in Figure 18. For further hydraulic analysis, see Section IV.A.

Figure 18. U-305 Initial
Hydraulic Test —
July 28, 1965

Nuclear operation of the U-305 experiment began at 1158 hr on July 29, 1965, with the bulk coolant temperature reduced to 510°F (265°C) as a precaution against possible excessive cladding temperatures. Initial operation lasted only 4 hr and was halted when a fuel failure in the U-1 loop at 40% reactor power forced a shutdown. Extrapolation of data obtained during this low-power operating period indicated that, at full reactor power, the U-305 string would operate at approximately 140% of design power without remedial action. While no definitive cause could be assigned to this discrepancy between design and actual operating conditions, the most probable cause appeared to be a greater than expected thermal neutron flux near the U-3 loop position. This overpower

situation was eventually compensated for by the placement of lower reactivity fuel elements near the U-3 loop core site during the normal on-line refueling procedures. By August 6, these fuel movements had allowed both the NRU and the U-305 to mutually obtain their desired power levels.

On July 30, at 1500 hr, the coolant inlet temperature was raised to about 620°F (325°C) for a brief period to observe operation at design conditions. Table 6 is a comparison of the measured values with the design values. In

TABLE 6
COMPARISON OF U-305 DATA WITH DESIGN VALUES

Parameter	Design	Measured
Inlet Coolant Temperature (°F)	620	616
Coolant ΔT (°F)	82	80
Coolant Flow Rate (kg/sec)	16.9	17.0
Experiment Power (kw)	1660*	1800 [†]
Fuel String Pressure Drop (psi)	85 [§]	90
Maximum Cladding Temperature (°F)	850	860**

* Based on coolant heat capacity of 0.515 Btu/lb-°F

[†] Based on coolant heat capacity of 0.567 Btu/lb-°F

[§] This value is based on extrapolation of ex-reactor hydraulic data⁽¹⁰⁾

** This value was estimated from the TC 21 reading of 817°F

general, the experiment matched the design values quite well. The deviation in the maximum cladding temperature should be tempered by the fact that the axial flux profile as indicated by the pressure tube flux monitors was not the ideal chopped cosine curve used in the design calculations. The actual flux profile (see Figure 19) was such that the maximum cladding temperature was elevated to bundle LSA (the center SAP-UC bundle). The maximum cladding temperature for the data set presented in Table 6 was estimated from the TC 21 value, since TC 23 (located on Bundle LSA) was not operative at the time. The maximum cladding temperature was arrived at by estimating a TC 23 temperature from the TC 21 value based on subsequent history, and then calculating the outermost surface temperature by adding the across-the-rod ΔT obtained from TC pairs 17-18 and 20-21 (see Figure 8 for TC location). The value of 860°F obtained

Figure 19. U-305 Axial Flux Profile

for this data scan can be corrected for the slight deviation from design power (8%) and inlet temperature (4°F) resulting in the maximum "measured" cladding temperature at design conditions for initial U-305 operation of 850°F.

The close agreement between the calculated values and measured values at design conditions is one of the more important facts generated by the U-305 experiment. The thermal-hydraulic and nuclear and analytical models used in the design of the U-305 are now shown to be valid.

C. LONG-TERM OPERATION

The chronological operating history for the U-305 experiment is described in Figures 20 through 25 and will be discussed in three parts: thermal history, hydraulic history, and coolant chemistry.

1. Thermal History

After the brief operation at design coolant temperature, the experiment was returned to the 510°F inlet temperature condition as planned. This low temperature operation was to provide a safe condition while the U-3 flux was being adjusted so that the NRU could operate at full power. As soon as full reactor power was reached, the experiment was to be returned to design coolant temperature. The subsequent thermal history for the U-305 experiment is depicted in Figure 20. Figure 20a is for Bundle LTA and shows the test section inlet temperature, the average temperature indicated by the four coolant TC's located at the bottom of LTA, and the temperature of cladding TC 21. Figure 20b is for Bundle LSA, and shows the test section inlet temperature, the temperature of

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Figure 20. U-305 Temperature History

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the coolant entering LSA as measured by TC 22, and the cladding temperature as measured by TC 23. Since there was no direct measure of the hot-spot cladding temperature (because of thermocouple location), it was estimated using the TC 23 readings. Analysis of the initial data indicated that 30°F added to the TC 23 reading would result in a reasonable estimate of the hot-spot temperature.

As shown, the experiment was operated from start to November 19, at a 510°F inlet temperature and then raised to 560°F for the remainder of the test.

Between August 1 and August 4, there was a marked increase in cladding temperatures. This increase was on the order of 60 to 70°F, and is the reason that, while the initial data were satisfactory, continuous operation at design bulk coolant temperature conditions was not possible. As seen from early August data, the hot-spot temperature at design power would have been in excess of 900°F if operated at the 620°F design inlet temperature conditions. Materials limitations make it undesirable to operate continuously with cladding temperatures above 870°F. As a result of the increase in cladding temperature, the U-305 experiment was not operated for an extended period of time at the design bulk inlet temperature of 620°F.

A similar temperature increase occurred between September 5 and September 9, when the rate of increase was about 10°F per day. These temperature increases did not appear to be consistent with the coolant fouling potential indices being measured. Since startup, the MST, SPFT, etc., had been within the desired limits, as shown in Table 7. These limits were established at OMRE, Piqua, and X-7, as sufficient for operation without fouling. One coolant parameter, however, did deviate from the intended value. This was the water content of the coolant which is illustrated in Figure 21. On startup, the water content was quite low (29 ppm), and was not brought to the proposed value until August 4. During the early September shutdown, the water content again dropped (17 ppm), and was not corrected until September 9. In each case, as soon as the water content was increased, the temperature increase was halted. These low-water periods were not intentional, but were due to equipment malfunctions. During the rest of the test, the water content was generally between 150 and 300 ppm, and no other significant fouling periods were noted. Specific tests to study this interesting phenomenon were incorporated into test plan for the U-305-A1 experiment.

TABLE 7
U-305 COOLANT QUALITY

Index	Desired	Range	Typical
HB (wt %)	25 ± 5	12.3 to 22.0	18
MST (A/mg × 10 ⁵)	—	6 to 142	10
SPFT (μg/cm ² -hr)	2	0.5 to 2.7	1.0
Conductivity (picomhos/cm)	—	6.0 to 1170	20 to 60
Water (ppm)	100 ± 50	17 to 1000	150 to 300
Carbonyl Oxygen (ppm)	—	1 to 174	25
Ash (ppm)	<5	22 (maximum)	2
Fe (ppm)	<2	2	2
Cl (ppm)	<2	0.5 to 5	0.5
Main Loop Gamma Activity (gpm)	—	1000 to 46,000*	1500

* This high activity occurred only during postdefect operation.

While observing the cladding temperature increases, it was apparent that a method to separate temperature changes due to power changes and fouling effects was necessary. The experiment power fluctuations were frequent and over a range of about 20%, due to the on-line refueling feature of the NRU. These power fluctuations make it difficult to determine fouling rates on a day-to-day basis by observing only the cladding temperatures. A power normalizing parameter previously used at OMRE was put into use and was defined as:

$$\beta = \frac{T_{\text{clad}} - T_{\text{cool}}}{\Delta T_{\text{UC}}} , \quad \dots(1)$$

where

T_{clad} = the cladding temperature at the point in interest,

T_{cool} = the bulk coolant temperature at the axial position in interest, and

ΔT_{UC} = the coolant temperature increase for the three UC bundles.

This β parameter is simply a measure of the driving force across the cladding-to-coolant resistance normalized for fluctuations in power generation. A more detailed derivation and discussion of limitations of the β parameter are presented in Section IV-B.

Figure 21. U-305 Coolant Water Content History

The β parameters for the five continuously operating cladding surface thermocouples are shown in Figure 22. The two periods of rapid increase are readily apparent, as are other periods of gradual decrease and increase.

The cladding temperature measurements were compromised by the failure of all cladding thermocouples caused predominantly by open circuits in the Alumel leg. Thermocouple 24 was inoperative from the start; TC 17 failed on August 3, and was never recovered. The remaining cladding thermocouples operated intermittently during the first several weeks. When they finally became inoperative during mid-August, substitute signals were obtained from the Chromel-to-sheath circuit. This type of junction had to be calibrated frequently, and many of the step-changes in the β parameters can probably be attributed to changes in these hybrid TC calibrations. All coolant TC operated normally for the duration of the experiment.

The TC failures were attributed to a separation of the Alumel wire during thermal cycles, due to differential thermal expansion between the stainless steel sheath and the TC wires. An apparatus to repair the broken leg by capacitance discharge welding was available; however, the repaired TC was very susceptible to thermal cycles. The failure rate of the TC and the inability to make lasting repairs was a serious problem because of the need for cladding temperature information. At the time of these failures, the experiment had just undergone a severe fouling period during initial operation, and if adequate measurements of cladding temperature could not be made, then the experiment would have to be terminated. This action did not become necessary when it was shown that reasonable temperatures could be obtained using the hybrid TC junctions. However, it should be noted that the 0.020-in.-OD TC used to measure surface temperature contained conductors which were only 0.003 in. in diameter. These TC, available only on special order, could not be expected to have the reliability associated with larger size TC covered by Instrument Society Standards. The size of the TC was dictated by the need to implant them within the 0.035-in.-wide cladding fins.

The ability of the NRU operators to maintain the desired nuclear conditions for U-3 is shown in Figure 23. This illustrates the chronological history of the five U-3 flux monitors. The flux at U-3 remained almost constant for the duration of the U-305 experiment. Flux Monitor 4 began to degrade in February, as

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Figure 22. U-305 Beta Parameter History

Figure 23. U-305 Flux Monitor History

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indicated by lead wire resistance, and subsequent data should be ignored. Test section coolant ΔT produced by the fuel string operating at this constant flux is shown in Figure 24. Due to burnup effects, the ΔT decreased by about 11% during the 9-mo irradiation.

2. Hydraulic History

The hydraulic history of the U-305 experiment is shown in Figure 25. The coolant flow rate was controlled very closely to the desired value of 16.9 kg/sec, and no significant problems in maintaining the desired flow were encountered. The data presented in Figure 25 are the most probable estimate of what the test section pressure drop was during the test. The initial test section ΔP was 93 psi, and increased to 108 psi at termination. A complete story of the pressure drop history is included in Section IV.A.

3. Coolant Chemistry

The philosophy adopted for coolant quality during the U-305 test was to operate with a relatively clean system. Maximum use was made of the purification system, and there were no major problems encountered in obtaining and maintaining the desired values for the coolant quality indices during most of the irradiation. Table 7 contains the range of values encountered during the test, and a value that was typical of the coolant quality.

In early July, the U-3 coolant was circulated through the adsorber columns to achieve maximum purification prior to insertion of the fuel string. The U-3 coolant initially exhibited MST values of about $6000 (A/mg) \times 10^5$, Mg^{56} activities from 20,000 to 44,000 dpm/gm, and Na^{24} activities greater than 2000 dpm/gm. Following the insertion of the fuel string and initial power operation on July 29, the MST values were about 40 to 80 $(A/mg) \times 10^5$, the Mn^{56} activity was 610 dpm/gm, and Na^{24} was not detected. The two adsorption columns were recharged on August 4, when the indices began to increase slightly. The MST dropped to 30 to 35, and the Na^{24} activity again decreased to nondetectable levels. One adsorber was recharged on August 10, two more on August 17, and by August 20 the MST values were about $10 (A/mg) \times 10^5$, and the Mn^{56} and Na^{24} nuclides were not detectable. During this initial cleanup, the coolant quality indices were brought within the normal operating ranges shown in Table 7 and maintained therein throughout the remainder of the experiment.

Figure 25. U-305 Hydraulic History

D. TEST TERMINATION AND INITIAL POSTIRRADIATION EXAMINATION

The irradiation of U-305 was terminated on April 16, 1966, because of a cladding failure. The rupture was initially suspected on April 8, and was confirmed by a sustained increase in the coolant activity due to the presence of fission products.

The initial activity release was detected by a sharp increase in coolant activity. Samples of coolant and cover gas were analyzed by pulse height techniques and the presence of gaseous fission products with associated daughter products in quantities much larger than normal was confirmed. The initial activity spike rose from a normal coolant activity of about 2000 to 35,000 cpm in about 45 min, and then decayed normally. The reactor power was decreased by 10% after the first spike, and later on two power cycles of about 20% in magnitude were performed in an attempt to release additional activity. These attempts were unsuccessful and the reactor was returned to full power on April 9. The coolant activity remained at the normal 2000 cpm until April 12, even though there had been several power cycles associated with the normal refueling. As shown in Figure 26, the activity spikes that occurred on April 12, 13, and 14, all decayed

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Figure 26. U-3 Loop Coolant Activity Between U-305 Defect and Removal

to normal levels. On April 15, the reactor was shut down for reasons not associated with the U-3 loop, and upon startup 30 min later a sharp increase in coolant activity was noted. This again decayed to a normal level until 2210 hr, when the coolant activity increased to 45,000 cpm. For the next 12 hr, the coolant activity remained above 13,200 cpm with several spikes toward 45,000 cpm. These later activity spikes were different from those prior to the reactor outage, in that the activity rose to the maximum level within 1 to 5 min, as compared with 1/2 to 1 hr for the previous bursts. It was now apparent that a cladding rupture had indeed occurred, and that continued operation might compromise the fuel string materials information or be harmful to the loop. On this basis, the U-305 irradiation was terminated by shutting down the reactor at 1832 hr on April 16.

During operation of the fuel string in the defected state, the following were observed.

- 1) The initial fission gas release occurred during steady-state operation.
- 2) A correlation between reactor power cycles and coolant activity spikes was nonexistent; i. e., cycling the reactor did not necessarily produce an increase in the coolant activity.
- 3) There were no deviations from normal operation; test section pressure drop and cladding temperatures remained at normal levels.
- 4) The isotopes identified were $\text{Kr}^{85\text{M}}$, Kr^{87} , Kr^{88} , Rb^{88} , Xe^{133} , $\text{Xe}^{133\text{M}}$, Xe^{135} , and Cs^{138} . No iodine was observed, and only fission gases which have half-lives long enough for the noble gas to diffuse out of the element were observed.

The fuel string remained in the U-3 loop for a 40-hr cooling period following shutdown. It was then transferred to a HB-40 filled storage tube in the NRU storage block where it resided until June 7. The experiment was then removed to the universal cells at ORNL where the initial visual examination was conducted. Figure 27 is a composite of photographs taken after washing the excess coolant from the fuel string. The initial visual examination revealed no obvious dimensional distortions, splits in the cladding, or buildup of "coke" deposits. All of the stainless steel components (end hardware, spring and spacer rings) were very clean and free of fouling film. The bottom fuel bundle (JPA) exhibited the least fouling of the fueled surfaces. This fouling was a thin powder-like black film, and was easily removed. The next higher bundle (JRA) exhibited an apparently thicker black film which was more adherent, but which loosened in

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Figure 27. Post-Irradiation Condition
of U-305 Fuel String

flakes when scraped. The upper three SAP-UC bundles appeared to be covered with two types of film: one was a thin brownish translucent film, and the other was the thick black film most usually associated with fouling films. On all of the SAP-UC bundles, portions of this film had flaked off, exposing bare cladding.

The hot-cell examination was briefly interrupted during which the fuel string was temporarily stored in a large fuel transfer cask. Upon reinsertion into the hot cell, a drastic change in the amount of adherent fouling film was noted. The film was now very loose and large strips were detached and lying on and about the element. It was later concluded that this flaking and spalling were due to an overnight exposure to air while the element was in the handling cask, prior to the second insertion into the hot cell. It was also concluded that the temperature of the cladding surface could have reached 320°F while in the closed fuel handling cask.

The next step in the U-305 examination was to remove the bottom two bundles, fabricated of Zircaloy 4 clad UO_2 , from the hanger rod. Due to the clean condition of the hardware, the disassembly was not difficult. The upper of these two bundles (JRA) was then examined under the stereomicroscope at 20X, and a number of cracks and defects were revealed. Rod H, which was on the side oriented toward core center, appeared to be the most severely affected, and the largest defect was found 4 to 5 in. from the top of the rod. In this area, severe cracking of the cladding extending approximately 1/2 in. along the rod between and including two adjacent fins was noted. In the center of this area, a piece of cladding about 0.1 to 0.2 in. square was missing, and bare fuel was exposed.

IV. ANALYTICAL RESULTS

In order to provide a meaningful interpretation of the U-305 test data, several areas of the experiment behavior required extensive analysis. These areas included the test section hydraulic characteristics and the in-pile observation of fouling on the heat transfer surfaces.

A. HYDRAULICS

The raw data depicting the hydraulic behavior of the U-305 experiment appear to indicate that, during the operation of the test, an increase in fuel element pressure drop in excess of 35% had occurred. Throughout the operation of the U-305, certain inconsistencies indicated that this pessimistic observation was not correct.

An extensive study was therefore performed to determine the most probable pressure drop history, and it was found that the increase in pressure drop at constant flow was in reality only about 12%. This 12% increase is a reasonable number, and might be expected because of changes in the surface condition of the fuel element.

The raw data from the experiment, illustrated in Figure 28, indicated that although the coolant flow rate was essentially constant during the irradiation, the test section pressure drop had experienced several step-changes in addition to periods of gradual increase. The apparent step-changes will be discussed first.

Concurrent with the first apparent step-change in early December, the test section pressure drop measuring instrument, dP-2, was removed and calibrated. This was the first known maintenance of dP-2 during the U-305 test. At this time, the dP cell was removed from the system, taken to the shop, cleaned, assembled, and calibrated. It was then returned to the system and a zero offset of 400 in. of water (15 psi) was applied to compensate for the difference in elevation between the test section inlet and outlet pressure taps. All maintenance was supposedly performed to existing U-3 loop procedures. The reasoning behind the static head correction was questioned, and as a result the 15-psi zero offset was intentionally removed on January 10, 1966.

Figure 28. Indicated U-305 Hydraulic History

The 15-psi step decrease in ΔP in January, as shown in Figure 28, was indeed due to this zero offset removal. The data between December 1 and January 10 should therefore be 15 psi lower; however, for the period from startup to December 1, it was not known for certain whether or not the dP-2 readings included the 15-psi head correction.

When the early December step-change occurred, the experiment had been operating smoothly and no other step-changes in performance were noted. If the 15-psi increase was due to an increase in flow, then a 7% or 1.2 kg/sec increase would have had to occur. The coolant flow rate as measured by the flow instrumentation did not change significantly during this period.

A separate measurement of the flow, independent of the flow orifice, was also examined. This was obtained by dividing the flux monitor readings by the test section ΔT . Since flux (ϕ) is proportional to the experiment power (Q), then:

$$\phi \cong Q = wC_p \Delta T \quad ,$$

where

w = coolant flow rate

C_p = coolant specific heat.

Upon dividing by ΔT :

$$\phi / \Delta T \cong \frac{Q}{\Delta T} = wC_p \quad .$$

The parameter $\phi / \Delta T$ was calculated for the entire experiment irradiation and it was found to substantiate the flow meter indications, particularly during the time period presently in question. (The assumption that C_p remains relatively constant, particularly over short periods of time, is valid.) It was therefore concluded that the ΔP step increase in early December was not due to a flow increase.

Another possible explanation of the dP-2 behavior would be that the test section hydraulic resistance underwent a real step increase as indicated by dP-2. To evaluate this possibility, two additional test section ΔP sources were examined to determine if they had experienced any change. These were absolute pressure gauges PIA-6 minus PIA-3A, and PIA-4 minus PIA-3B, located on the inlet and outlet sides of the test section as shown in Figure 12. The ΔP derived from both sets of gauges indicated that the test section pressure drop had not changed during early December.

By this point in the hydraulic analysis, it was concluded that the raw ΔP data for the periods of startup to December 1, and January 10 to termination were without the zero offset of 15 psi. However, for the period between December 10 and January 10, the 15-psi static head correction must have been included. The chronological test section pressure drop data obtained from dP-2 were then adjusted as indicated in Figure 29.

The hydraulic history of the U-305 experiment now appeared to be more reasonable, and it was decided to concentrate on the absolute value of the pressure drop.

While investigating the head correction problem, it was also found that one leg of the dP system was at a temperature elevated sufficiently to cause an appreciable difference in the density of the liquid in the two pressure lines. The net effect of this density difference is that 2 psi should be added to the raw ΔP data to obtain a more correct test section ΔP . This type of correction is characteristic of this particular installation, and remained constant for the duration of the irradiation.

Figure 29. Adjusted U-305 Pressure Drop Data

At the time of the December 1965 dP-2 calibration, it was noted that the dP cell bypass valve (normally closed) was leaking. If this condition had existed previously, then the data prior to December could have been low. It was decided to check the dP-2 readings against the ΔP derived from other pressure gauges in the U-3 loop. PIA-6 minus PIA-3A, and PIA-4 minus PIA-3B data were used for this comparison.

In order to proceed with the analysis, it becomes necessary to digress into the operation of the U-3 loop after the U-305 fuel string was removed and the U-305-A1 experiment was in operation. Due to the continued unsatisfactory behavior of the pressure drop instrumentation, the test section hydraulics measuring system of the U-3 loop was rebuilt in October of 1966. During this overhaul, two new dP cells were installed, the flow measuring system rebuilt, additional absolute pressure gauges added to the main loop, and the entire system calibrated. After this modification, complete hydraulic characteristics of the U-3 loop were studied in some detail. Part of the findings were useful in analyzing the U-305 data. These were the relationships of (PIA-6) - (PIA-3A) and (PIA-4) - (PIA-3B) with the true test section pressure loss. The installation of these gauges was not disturbed during the October 1966 modification, and these

TABLE 8
COMPARISON OF PRESSURE GAUGE READINGS WITH dP-2 READINGS

Month	(PIA-6) - (PIA-3A)	(PIA-4) - (PIA-3B) + 10 psi	Raw dP-2 Data	Deviation of (3) from Average of Pressure Gauge Derived ΔP	Corrected dP-2 Readings	Deviation of (5) from Average of Pressure Gauge Derived ΔP
	(1)	(2)	(3)	(4)	(5)	(6)
August	—*	—*	86	—	93	+3
September	104	104	91	+13	98	+6
October	109	110	97	+13	104	+6
November	112	111	99	+12	106	+6
December	110	111	101 [†]	+10	103	+8
January	112	111	103	+9	105	+7
February	109	108	104	+5	106	+3
March	112	110	105	+6	107	+4
April	114	112	106	+7	108	+5

*Instrument readings not recorded during August.

[†]This reading includes correction for the 15-psi head compensation used during this month.

relationships should be reasonably correct for the U-305 experiment. These relationships are:

$$\text{PIA-6} - \text{PIA-3A} + 0 \text{ psi} = \text{true } \Delta P$$

$$\text{PIA-4} - \text{PIA-3B} + 10 \text{ psi} = \text{true } \Delta P .$$

Using this information, a comparison between the pressure-gauge-derived ΔP and the dP-2 readings was made for the U-305 experiment. Table 8 lists the results in the form of monthly averages. Column 4 of Table 8 is the deviation of dP-2 raw data from the estimated true test section ΔP . From this, it was concluded that dP-2 was apparently reading low, especially during the portion of the test from startup to early December.

A separate indication that the dP-2 readings were low during the initial operation of U-305 is available from the initial flow- ΔP test performed on July 29, 1965. These data were presented in Figure 18 and are shown plotted on a log-log scale in Figure 30. The data should theoretically fall on a straight line of slope 2.0; however, these data fit a curve that is concave toward the flow scale and has a slope greater than 2.0.

This type of flow- ΔP plot has been used during the U-305-A1 operation with a great deal of success to determine if the hydraulic instrumentation was calibrated correctly. Based on this experience, it was determined that dP-2 was reading 5 psi low at the time of the flow- ΔP test on U-305 in July 1965. If 5 psi is added to the dP-2 readings, then the curve becomes a straight line with a slope of 2.0. (This 5 psi also includes the 2-psi system offset.)

Another method of predicting the absolute value of the initial pressure drop is to extrapolate the pressure gauge data back to the beginning of the test. The readings for PIA-4, PIA-6, and PIA-3 were not logged until September 1965, and these early data are missing. If one, however, assumes that the shape of the ΔP curve obtained from the pressure gauges and dP-2 is identical, then the pressure-gauge-derived ΔP can be extrapolated back to late July 1965. This is illustrated in Figure 31 and shows that the initial pressure-gauge-derived ΔP would be on the order of 96 psi. This indicates that the early dP-2 value of 86 psi might be 10 psi low.

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Figure 30. Initial U-305 Flow- ΔP Test Data

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Figure 31. Comparison of Gauge and dP Cell Data

The discussion of the absolute ΔP value for the U-305 test may be summarized as follows.

It was first determined that the system installation was such that the dP cell indicated 2 psi low during the entire U-305 test, and therefore 2 psi should be added to all dP-2 data. Then, by using data from other pressure gauges, it was also determined that dP-2 was reading particularly low for the period of startup to early December. This led to checking the hydraulic data obtained from the initial flow- ΔP test, from which it was determined that the initial readings from dP-2 were at least 5 psi low. This was supported by an extrapolation of pressure-gauge-derived ΔP (two sources) which indicated that the initial dP-2 readings might be 10 psi low. A correction of +7 psi (including +2 psi correction noted above) to the raw dP-2 readings is concluded to give a more probable estimate of the initial test section ΔP ; and in addition, if applied to the data up to the dP cell calibration in December 1965, would make the relationship with the pressure-gauge-derived ΔP more uniform. After December, only the 2-psi system offset needed to be added to the raw data. The dP-2 values corrected in this manner and the deviation from the pressure-gauge-derived ΔP are shown in Columns 5 and 6 of Table 8.

There is an additional aspect that was analyzed in some detail and will complete the picture on the hydraulic performance of the U-305 experiment. In any discussion of the pressure drop, one must be certain that the flow rate is known quite accurately, since the pressure drop varies proportionally to the square of the flow. The indicated flow rate during the U-305 experiment was verified by a method independent of the flow measuring system. This method is a comparison between indications of the nuclear power. An approximation of this is the flux monitor readings (nuclear power) divided by the test section temperature increase (thermal power) or $\phi/\Delta T$ which was previously used to independently support the flowmeter indications. This type of a calculation may be performed in more detail by working with the basic relationship between neutron flux ϕ and experiment power Q as defined by the following equation.

$$Q = \phi \left(\alpha - \tau \int Q dt \right) \quad \dots(2)$$

In this equation, α is a scale factor and $\tau \int Q dt$ is a measure of the decrease in neutron fission cross-section with exposure. The net effect is that, as burnup increases, a decrease in power occurs at constant flux resulting from the depletion of the fissile species in the fuel. The amount of this power decrease at constant flux should be on the order of a few percent per month for the U-305 fuel string.

In order to facilitate analysis of the experimental data, Equation 2 was rearranged and integrated, resulting in:

$$\ln \left(\alpha - \tau \int Q dt \right) = \ln \alpha - 1/\tau \int \phi dt \quad ; \quad \dots(3)$$

and upon further rearranging:

$$\ln Q/\phi = \ln \alpha - 1/\tau \int \phi dt \quad . \quad \dots(4)$$

Equation 4 makes it possible to plot the power decrease at constant flux (Q/ϕ) against the integral of ϕdt which is a measure of exposure. The usefulness of this plot is that the relationship should follow a straight line. Since Q is calculated using a measured value of flow, any deviation from the straight line was interpreted as an error in test section flow because of greater confidence in the indicated relative neutron flux. A plot of Equation 4, using the U-305 experimental data, is presented in Figure 32. The line drawn through these data is a least-squares fit and has a standard deviation of $\approx \pm 2\%$. The Q/ϕ decrease with exposure from this analysis was 12% for the duration or about 1.5 % per month.

The fitted line was used to calculate flow from the following equation, and the results are compared with the measured flow in Table 9.

$$W = \left(\frac{\alpha \phi}{\Delta TC_p} \right) \exp \left(-1/\tau \int \phi dt \right) \quad \dots(5)$$

This comparison indicates only minor deviations between calculated and measured flow rates that are within the accuracy of the data. Based on this independent analysis, it is concluded that the U-305 experiment had a relatively constant flow rate, as was indicated by the flow data obtained from REDACE.

Figure 32. U-305 Experiment
Power Decrease as a
Function of Exposure

TABLE 9
COMPARISON OF MEASURED AND CALCULATED
U-305 FLOW RATES

Month	Calculated Flow (kg/sec)	Measured Flow (kg/sec)
August	17.0	17.0
September	16.8	16.8
October	17.0	17.0
November	17.2	17.1
December	16.7	17.0
January	16.6	17.0
February	16.8	17.1
March	17.2	17.1
April	17.2	17.2

At this point in the hydraulic analysis, the shape of the chronological ΔP curve has been defined, the most probable value for ΔP chosen, and the validity of the flow rate data verified. Figure 25 illustrates this most probable chronological hydraulic history for the U-305 experiment.

Because there were slight flow and significant bulk coolant temperature variations between the start and finish of the experiment, the chronological ΔP values should be normalized to the experiment design conditions, to allow a more precise evaluation of the pressure drop. The theoretical relationship between flow and pressure drop is:

$$\Delta P = \frac{Kw^2}{\rho} \text{ or } K = \frac{\Delta P \rho}{w^2} ,$$

where

P = pressure drop

w = flow rate

ρ = coolant density

K = proportionality constant.

The constant K is a characteristic test section resistance at any point in time. The monthly averaged values for K along with the flow and corrected ΔP are listed in Table 10. The ΔP was then normalized to design conditions from

TABLE 10
SUMMARY OF U-305 HYDRAULIC PARAMETERS

Month	Flow (kg/sec)	ΔP (psi)	K $\left(\frac{\text{lb-sec}^2}{\text{in}^2\text{-kg-l}} \right)$	ΔP^* (psi)
August	17.0	93	0.287	96
September	16.8	98	0.312	104
October	17.0	104	0.319	106
November	17.1	106	0.323	108
December	17.0	103	0.310	103
January	17.0	105	0.318	106
February	17.1	106	0.320	107
March	17.1	107	0.322	107
April	17.2	108	0.321	107

* ΔP normalized to design flow of 16.9 kg/sec and an inlet temperature of 620°F.

the K values, using a flow of 16.9 kg/sec and an average density computed for the design inlet coolant temperature of 620°F. On this basis, the initial clean ΔP for the U-305 was 96 psi, and during the course of the irradiation increased 11 psi (12%) to a final value of 107 psi. It is interesting to note that the large portion of this increase was incurred during the first month of operation. By September, the ΔP had gained 8 psi or 73% of the total 11-psi increase. This was during the same period when the fuel element surface obtained most of its fouling film, as will be discussed in the next section.

From the hydraulic mockup tests,⁽¹⁰⁾ a value of 85 psi was calculated for only the fuel string. The pressure taps for the test section ΔP measuring system are, by necessity, located so that there are other pressure loss contributing factors. Piping bends, test section entrance expansion and exit contraction,

and the zirconium test samples are estimated to produce 5- to 7-psi pressure loss for a total of 91 psi. Comparing this with the initial normalized ΔP of 96 psi, it is therefore concluded that data obtained from the hydraulic mockup tests were successfully extrapolated to the actual situation.

B. IN-PILE FOULING DETERMINATIONS

The beta parameter (β) previously mentioned in the experiment operations section may be used as an indicator of the amount of fouling on the heat transfer surfaces of a fuel element. This is particularly useful in an irradiation experiment such as the U-305, where varied operating conditions were being studied.

The beta parameter may be derived analytically and is specifically related to the total heat transfer resistance R . The general equation for heat transfer through a combination coolant laminar film and a surface fouling film is as follows.

$$R = \frac{1}{U} = \left(\frac{1}{h} + r \right) , \quad \dots(6)$$

where

R = total heat transfer resistance

U = overall heat transfer coefficient

h = coolant film heat transfer coefficient

r = fouling film heat transfer resistance.

By relating U to the total U-305 UC fueled section (because only this portion of the fuel string was instrumented) power and the heat flux at any given location, the following equation may be derived.

$$R = \frac{A_T \Delta T_L}{P_L W_T C_P \Delta T_{UC}} , \quad \dots(7)$$

where

A_T = total UC heat transfer area

ΔT_L = local cladding to bulk coolant ΔT

W_T = coolant mass flow rate through experiment

ΔT_{UC} = temperature increase of coolant due to passing through UC fueled portion

P_L = ratio of local heat flux to average UC heat flux.

The term $\Delta T_L / \Delta T_{UC}$ in Equation 7 is in fact the beta parameter as defined in Section III. C. By rearranging the equation and solving for β , the following equation is obtained.

$$\beta = R \left[\frac{P_L W_T C_p}{A_T} \right] \quad \dots(8)$$

If the terms enclosed in brackets in Equation 8 are assumed to remain constant with time, then beta is proportional to R, the total heat transfer resistance. The A_T term is geometry-dependent and does not change, the flow (W_T) and specific heat (C_p) terms were known to have remained relatively constant. The P_L term does change with burnup; however, for the purposes of this analysis the change is relatively small where compared to uncertainties in the temperature.

Proceeding with the assumption that β is proportional to R, then Equation 8 is solved at time "zero" and the result divided into Equation 8 with the resulting relationship.

$$\beta / \beta_o = R / R_o \quad \dots(9)$$

A plot of these β ratios for five U-305 cladding thermocouples is shown in Figure 33. The baseline (1.0) for these parameters ($\beta = \beta_o$) is equivalent to an R of about 0.0005 (Btu/ft²-hr-°F). By using the average monthly values of β / β_o , Table 11 was constructed which contains values of R and the corresponding coolant to cladding ΔT (°F) at a local design heat flux of 300,000 Btu/ft²-hr.

The total heat transfer resistance (R) may be separated into its components for the fluid film and the fouling film, $1/h$ and r respectively. If the empirical relationships for h and r appropriate to U-305 operating conditions are substituted into Equation 9, the following equation can be derived.

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Figure 33. U-305 Normalized Beta Parameter History

TABLE 11
MONTHLY AVERAGES FOR R AND COOLANT TO CLADDING ΔT

Month	R_{18} (Btu/ft ² /hr/°F) ⁻¹	ΔT_{18} (°F)	R_{19} (Btu/ft ² /hr/°F) ⁻¹	ΔT_{19} (°F)	R_{20} (Btu/ft ² /hr/°F) ⁻¹	ΔT_{20} (°F)	R_{21} (Btu/ft ² /hr/°F) ⁻¹	ΔT_{21} (°F)	R_{23} (Btu/ft ² /hr/°F) ⁻¹	ΔT_{23} (°F)
August	0.00071	212	0.00065	194	0.00077	234	0.00065	194	0.00056	168
September	0.00082	246	0.00088	264	0.00081	242	0.00072	216	0.00061	183
October	0.00069	206	0.00075	225	0.00065	194	0.00065	194	0.00049	147
November	0.00067	200	0.00075	225	0.00058	174	0.00053	159	0.00046	138
December	0.00069	206	0.00081	242	0.00065	194	0.00059	177	0.00049	147
January	0.00073	218	0.00085	255	0.00067	200	0.00057	171	0.00051	153
February	0.00077	230	0.00089	267	0.00069	206	0.00060	180	0.00053	159
March	0.00083	249	0.00088	264	0.00073	219	0.00063	189	0.00059	177
April	0.00077	230	0.00083	249	0.00073	219	0.00059	177	0.00051	153

$$\beta/\beta_0 = \frac{T_0}{T} + \alpha t_f \quad \dots(10)$$

where

T = average bulk coolant temperature in UC section at any point in time

T_0 = initial coolant temperature

$\alpha = 0.368$, a constant defined as h_0/k_f

h_0 = initial coolant h

k_f = fouling film thermal conductivity

t_f = fouling film thickness (mils).

Here, it is assumed that the flow rate and the coolant physical properties remained constant, which is subject to the same limitations as the previous assumption. For purposes of these calculations, it was also assumed that $h_0 = 2200 \text{ Btu/ft}^2\text{-hr-}^\circ\text{F}$ and $k_f = 0.5 \text{ Btu/ft-hr-}^\circ\text{F}$.* Table 12 is a listing of film

TABLE 12
MONTHLY AVERAGED CALCULATED
FOULING FILM THICKNESSES

Month	t_{18} (mils)	t_{19} (mils)	t_{20} (mils)	t_{21} (mils)	t_{23} (mils)
August	1.1	0.8	1.5	0.8	0.3
September	1.7	2.0	1.7	1.2	0.6
October	1.0	1.4	0.8	0.8	0.0
November	0.9	1.4	0.5	0.2	0.0
December	1.3	1.9	1.0	0.7	0.2
January	1.5	2.2	1.2	0.6	0.3
February	1.7	2.4	1.3	0.8	0.4
March	2.1	2.3	1.5	0.9	0.7
April	1.7	2.1	1.5	0.7	0.3

* h_0 is based on analysis of initial U-305 heat transfer data. k_f was obtained from postirradiation measurements of OMRE fouling films.

thicknesses in mils evaluated at locations on fuel rods adjacent to thermocouples. It is recognized that the accuracy of the values presented is subject to question due to the assumptions made and to the accuracy of the experimental data; however, two significant observations can be made. These are: (1) the largest portion of the fouling film accumulated was incurred during the first month of operation — this corresponds with pressure drop increase observations made in Section IV.B; and (2) the film thickness computed at the end of the irradiation compares favorably with the postirradiation measurements of the actual film thickness as indicated in Table 13.

TABLE 13
COMPARISON OF CALCULATED AND MEASURED
FOULING FILM THICKNESSES

TC Number	Location (Bundle)	Rod	Calculated t_f (mils)	Measured t_f (mils)
18	LTA	J	1.7	1.1 to 2.3
19	LTA	H	2.1	1.6
20, 21	LTA	F	0.7 to 1.5	1.3 to 1.9
23	LSA	J	0.3	*

*No data available.

As added information, a calculation was performed to determine the amount of pressure drop increase that could be expected from the accumulation of fouling film. It was assumed that the total heat transfer surface was covered with a film 1 mil thick with 0.5 mil roughness. This film would increase the pressure drop by 8 psi, which is the same order of magnitude experienced in the U-305 irradiation.

Based on the knowledge gained from the U-305 experiment, a fouling parameter of the type described here is a worthwhile tool to use for in-pile observation of fouling films. The accuracy of the U-305 β parameter was compromised due to the necessity of making assumptions; however, if a concentrated effort were made to define some of the input parameters as a function of time, the beta parameter concept could provide invaluable information from in-pile measurements.

V. CONCLUSIONS

The irradiation of the U-305 experiment at its design operating conditions, for its designed lifetime, successfully fulfilled its stated objectives.

Additional conclusions may be drawn from the operational history and analysis of the U-305 experiment that are pertinent to the HWOCR concept. The first of these is that the excellent agreement between design predictions and actual fuel string performance indicates a high degree of knowledge of the material properties and design techniques necessary for the implementation of the HWOCR fuel element concepts. Secondly, the general absence of significant fuel rod fouling, except during periods of low water content in the coolant, indicates a highly probable correlation between low water content and fouling. Except for the fouling periods accompanying the unintentional low water incidents, the exhibition of satisfactory vapor pressure and heat transfer coefficients by the coolant, and the capability of maintaining satisfactory coolant quality with present state-of-the-art techniques indicates the technical suitability of Santowax OM as a HWOCR reactor coolant. Finally, the moderate increase in test section ΔP , apparently associated primarily with the early fouling, together with the predominantly constant hydraulic resistance, indicates that fuel string hydraulic performance is within the acceptable limits of the HWOCR concept.

In closing, the U-305 experiment is considered to have made a significant contribution toward verifying and extending the state-of-the-art of the HWOCR fuels systems technology.

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