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of the
University of California
LOS ALAMOS • NEW MEXICO

Quarterly Status Report on
Ultra High Temperature Reactor Experiment (UHTREX)
for Period Ending April 30, 1968

UNITED STATES
ATOMIC ENERGY COMMISSION
CONTRACT W-7405-ENG. 36

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This LA...MS report presents the status of the LASL Ultra High Temperature Reactor Experiment (UHTREX). These activities were formerly included in the quarterly reports of the LASL Advanced Reactor Technology (ART) program. Previous Quarterly Status Reports (all unclassified) in this series, now discontinued, are:

LA-3244-MS	LA-3482-MS	LA-3665-MS
LA-3316-MS	LA-3524-MS	LA-3708-MS
LA-3370-MS	LA-3582-MS	LA-3760-MS
LA-3431-MS	LA-3625-MS	

Previous Quarterly Status Reports (all unclassified) in the UHTREX series are:

LA-3835-MS
LA-3952-MS

This report, like other special-purpose documents in the LA...MS series, has not been reviewed or verified for accuracy in the interest of prompt distribution.

Printed in the United States of America. Available from
Clearinghouse for Federal Scientific and Technical Information
National Bureau of Standards, U. S. Department of Commerce
Springfield, Virginia 22151

Price: Printed Copy \$3.00; Microfiche \$0.65

LOS ALAMOS SCIENTIFIC LABORATORY
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University of California
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Report written: June 1968

Report distributed: July 1968

Quarterly Status Report on
Ultra High Temperature Reactor Experiment (UHTREX)
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ULTRA HIGH TEMPERATURE REACTOR EXPERIMENT
(UHTREX)

During this reporting period, experiments were performed to measure the difference in reactivity between air and helium in the reactor and to confirm that mechanical effects such as core position, and loader ram positions have negligible effect on reactivity.

Analytical studies have continued to improve agreement between UCX-I, UCX-II, and the calculations. Revised calculations of critical mass at the intermediate operating temperature of 914°K and full operating temperature of 1714°K have been made.

The effort on checking and improving the reliability of instrumentation and control systems has continued. Repairs have been made to the core indexing mechanism. The gas sampling system was modified and two diffusion tube probes have been installed.

A high temperature $^{238}\text{PuBe}_{13}$ neutron source has been fabricated, tested, and installed on the end of a control rod for use during startup.

Two techniques are being developed for measuring fuel element temperature during operation. One technique depends on observation of the melting of small wire inserts and the other on the measurement of the thermal expansion of a metal insert.

REACTOR OPERATIONS

During an irradiation run that was made on January 24, 1968, it was determined that the reactor was critical when the final control rod to be removed was withdrawn to 31.3 inches instead of the 40-inch level which had been the critical position

during the last previous run made on September 1, 1967. The apparent reactivity gain between runs was 65¢. No significant change had occurred in the reactor configuration between runs except for the gas which filled the primary coolant loop. In September, the loop contained air at atmospheric pressure, but in January the gas was helium at one atmosphere. Although a reactivity gain was expected to occur when helium was substituted for air in the core, the observed gain was greater than anticipated. To measure the value of the nitrogen reactivity coefficient, and to verify that other previously unrecognized factors were not involved in the reactivity change, two experimental plans were prepared and executed.

The first set of experiments were performed to indicate whether there were sources of excess reactivity other than the helium-air substitution. These consisted of:

1. A redetermination of the critical position of rod no. 4 plug rod.
2. A determination of the critical position of each of the other seven control rods.
3. Position vs reactivity calibrations of rods no. 4 and no. 5.
4. Determination of the reactivity worth of each of the plug rods over the interval between its nominal "out-of-core" position (55 inches) and the fully retracted position (75 inches).
5. Determination of any activity effects associated with core rotation and movement of the fuel loading rams.

As anticipated, no single one of these mechanical variations in reactor configuration produced any significant changes in reactivity. It appeared that the only reason for a reactivity shift was the substitution of He for air. The second set of experiments provided precise measurements of the reactivity worth of N_2 , which has a 1.8 barn capture cross section.

The measurement of the nitrogen reactivity coefficient was made by comparison of critical configurations of the reactor during two reactor runs. In preparation for the first reactor run, the primary loop was evacuated to a pressure of about 1000 μ . After evacuation, the loop was filled with pure nitrogen at about 13 psia, the reactor was started on rod no. 4, and the residual reactivity was measured with the rod withdrawn to 55 in. For the second run, the primary loop was again evacuated and refilled, this time with helium; the reactor was made critical, and the reactivity measurements were repeated. Results of the experiment verified that the poison effect of nitrogen is sufficient to explain the gain in reactivity observed on January 24.

Measurement of Mechanical Sources of Changes in Reactivity

The investigation was undertaken to check the validity of the control rod position indications, and to measure the effects of core rotation and load ram motion on reactivity. Intercalibrations of neutron detectors were also repeated. The results established that reactivities associated with uncertainties in rod position indication, with rotational position of the core, and with fuel loading ram position are all small. Detailed experimental results are presented below.

Figure 1 is a plot of inverse count rates vs rod no. 4 position for both BF_3 startup channels, as obtained in a typical approach to critical during these investigations.

Data collected during the cross calibration of the nuclear instrumentation are presented in Table I.

Reproducibility of the critical configuration of the reactor is indicated by the data in Table II. The central startup source ($^{239}PuBe$, 10^6 n/sec) was withdrawn after the power level was raised to 20 W.

Fig. 1. Inverse count rate vs rod no. 4 position.

From these results and a subsequent calibration of rod no. 4, the worth of the source was found to be 6.8 cents.

With the reactor power level at 100W, plug rod exchanges were carried out and the critical position of each plug rod was measured. Results are presented in Table III. Rod no. 4 critical position was obtained at both the beginning and the end of this set of measurements to determine the effect of the temperature rise produced at the 100-watt level.

Rods no. 4 and no. 5 were calibrated over part of their range by rod displacements of up to 30 cents positive and 40 cents negative. Positive reactivities were inferred from doubling-time measurements and also calculated from the linear power channel no. 1 signal by the analog computer reactivity calculator. Negative reactivities were obtained only from the reactivity calculator. The results are plotted in Fig. 2. For comparison, data taken from a similar set of measurements made in August 1967 with rod no. 4 are also plotted in Fig. 2. Figure 3

TABLE I
COUNT RATE AND CHAMBER CURRENT MEASUREMENTS
FROM NUCLEAR INSTRUMENTATION

Channel	Measurement
<u>Reactor Power: 1.7 W</u>	
Startup #1	9.72×10^5 cpm
Startup #2	9.82×10^5 cpm
Linear #1	0.72×10^{-11} A
Linear #2	0.84×10^{-11} A
<u>Reactor Power: 21 W</u>	
Linear #1	0.87×10^{-10} A
Linear #2	1.0×10^{-10} A
Log #1	0.58×10^{-10} A
Log #2	0.54×10^{-10} A
<u>Reactor Power: 2 kW</u>	
Linear #1	0.87×10^{-8} A
Linear #2	0.96×10^{-8} A
Log #1	0.57×10^{-8} A
Log #2	0.55×10^{-8} A
Safety #1	0.51×10^{-8} A
Safety #2	0.51×10^{-8} A
Safety #3	0.59×10^{-8} A
Safety #4	0.53×10^{-8} A

Fig. 2 Reactivity vs Rod Position.

TABLE II
ROD NO. 4 CRITICAL POSITIONS

Run No.	Date	Critical Positions(Inches)		Avg. Core Temp.
		Source In	Source Out	
23	3/11/68	36.85	37.9	73°F
24	3/12/68	37.0	38.07	73°F
25	3/14/68	37.0	38.08	73°F

TABLE III
CRITICAL POSITIONS OF ALL PLUG RODS

Rod No.	Critical Position (Inches)
4	38.32
5	38.67
1	39.29
2	38.94
3	38.89
6	38.68
7	39.01
8	39.11
4	38.48

Fig. 3 Derivative of ρ with respect to rod position.

shows the derivatives of the reactivity plotted against indicated rod no. 4 position. It can be inferred from Fig. 3 that in the vicinity of the 40-inch position, an offset of about one inch between the true and the indicated position of the rod occurred between August 1967 and March 1968. Electrical connection of the digital computer's analog input system to the rod position potentiometers was completed during the time interval; interaction between these two circuits, which cannot be completely isolated from one another, could account for the effect noted here.

For measurements of the reactivity effect of core movements, reactor power was leveled at 100 W and the core was rotated through 360°. At each index position, the reactivity change due to the preceding 15° rotation was obtained from the reactivity calculator, and then the power level was returned to 100 W by adjusting the position of rod no. 5. The results are presented in Table IV. Deviation through the entire 360° rotation was approximately 2 cents.

Reactivity vs. rod position was determined for each of the 12 control rods near the outer limits of its travel. Reactor power was leveled at 100 W and each plug rod, in turn, was inserted 10 inches from the 55-inch out-limit position. Each reactivity change was obtained from the reactivity calculator and rod no. 5 was then displaced to return the power level to 100 W. The results are shown in Table V. Similar measurements were made on plug rod withdrawals from 55 inches to the fully retracted 73-inch position and on core rod insertions from 96 inches to 45 inches. Reactivity associated with the plug rod withdrawals from the nominal "OUT" position to that of maximum retraction were in each case, less than 1 cent. Results of the core rod insertions are listed in Table VI.

TABLE VI

CORE ROD INSERTION (96 to 45 inches)

Rod No.	Rod No. 5 Critical Position (inches)		Calculated Change in Reactivity (cents)
	Before Insertion	After Insertion	
9	38.64	40.53	-8.3
10	38.67	40.43	-7.1
11	38.65	40.20	-7.3
12	38.68	40.70	-9.5

TABLE IV

REACTIVITY DUE TO CORE ROTATION

Core Position	Rod No. 5 Position, (inches)	Calculated change in Reactivity (cents)
24	38.6	0
1	38.6	0
2	38.6	0
3	38.6	0
4	38.6	0
5	38.34	+0.75¢
6	38.30	small
7	38.30	0
8	38.38	small
9	38.38	0.1¢
10	38.38	-0.1¢
11	38.49	-0.5¢
12	38.49	-0.3¢
13	38.49	-0.2¢
14	38.49	-0.25¢
15	38.50	-0.4¢ (peak -0.6¢)
16	38.50	+0.4¢
17	38.3	+0.8¢
18	38.3	+0.4¢ (peak +0.5¢)
19	38.24	+0.3¢
20	38.26	-0.25¢
21	38.28	-0.25¢
22	38.35	-0.4¢
23	38.53	-0.6¢
24	38.57	-0.4¢

TABLE V

PLUG ROD INSERTIONS (55-45 inches)

Rod No.	Rod No. 5 Critical Position (inches)		Calculated change in Reactivity (cents)
	Before Insertion	After Insertion	
1	38.66	40.54	-8.6
2	38.71	40.47	-7.8
3	38.73	40.29	-7.3
4	38.61	39.87	-6.1
6	38.72	40.06	-6.3
7	38.59	40.27	-7.8
8	38.62	40.50	-8.1

Reactivity change associated with motion of load ram no. 7 from the reset position (22.3 inches) to the full "IN" position (0.0 inch) was 0.5 cents.

As a final part of these experiments, the nuclear safety system was tested by increasing the power level until the trip level setting was reached. The test resulted in a premature dropping of only one of the three scram rods, which precluded continuation of the test. Circuit adjustments were later made to increase sensitivity and eliminate spurious noise assuring simultaneous dropping of all three scram rods in later tests.

Measurement of the Nitrogen Reactivity Coefficient

Neutronic calculations had indicated that at least some of the reactivity gain between successive cold critical experiments could be attributed to a difference in the gas atmosphere in the reactor. In the experiments described above with air in the reactor, the apparent reactivity observed in January was found to be 17¢ above that typical of the earlier experiments run with air in the core. Of this amount, 7.2¢ was due to a nonlinearity in the rod position indication introduced since the first air experiment, and hence not a true reactivity gain; and 6.8¢ was due to the 10^6 n/sec ^{239}Pu neutron source which had been placed in the reactor after the January run. The remaining 3¢ was probably due to a difference in moderator temperatures. The experiments described below were done to evaluate the poison effect of nitrogen (which accounts for 99.8% of neutron absorptions in dry air), and to determine, in the absence of any other identified sources of reactivity, whether the air-He interchange was solely responsible for the effect observed.

To prepare for a measurement of the critical rod position with an atmosphere of N_2 in the core, the primary loop was evacuated and then filled with N_2 to 12.9 psia. A 10^6 neut/sec ^{239}Pu source remained at the core centerline during the entire experiment. When the reactor was brought critical, the rod no. 4 critical position was found to be 43.05 inches. Next the residual reactivity of rod no. 4 was measured by withdrawing the rod to 55 in. The resulting value of 12.5¢ was obtained from the

reactivity calculator and doubling time measurements. The run was terminated by a test of the nuclear safety system in which the 3 scram rods dropped when the reactor power level reached the nominal flux trip settings.

For the second reactor run, the primary loop was evacuated, to remove the N_2 , and then was filled with helium to a pressure of 12.9 psia. Mass spectrographic analysis of a gas sample from the primary loop showed a nitrogen content of 1.2 volume percent, equivalent to a partial pressure of 0.16 psia. On reactor startup, the rod no. 4 critical position was 30.0 inches. A reactivity calibration of rod no. 4 in the 30-inch region was obtained by rod displacements. Results of these and earlier measurements are combined in Fig. 4.

Fig. 4 Rod no. 4 reactivity calibration.

Table VII summarizes the results of this and previous reactivity effect experiments.

TABLE VII
INFLUENCE OF N₂ ON REACTIVITY

Reactor Atmosphere and Pressure	N ₂ Partial Pressure (psia)	Rod No. 4 Critical Position (Inches)	Reactivity Change (Cents)	N ₂ Pressure Coefficient (Cents/psi)
Air, 11.5 psia	9.0	37.0	-36.5	-6.2 -6.6* } -6.6*
N ₂ , 12.9 psia	12.9	43.0	-12.4	
He, 12.9 psia	0.16	30.0	-14.4	

*Corrected for estimated 0.8¢ worth of helium.

The above experiments were undertaken to explain the reactivity gain observed on January 24, 1968. The rod no. 4 critical position at that time, recorded at a power level of about 6 kW that had probably effected some small temperature rise in the fuel, was 31.3 inches. Analysis of a reactor gas sample taken January 29, 1968 showed 8% N₂ by volume, or a partial pressure of 0.92 psia of N₂. The value of 0.92 psia of N₂ is an upper limit because fuel unloading operations between January 24 and January 29 may have allowed some air to enter the reactor.

Two conditions existed during the reactor run of January 24, but not during the second reactor run described above:

1. The ²³⁹Pu source, worth +6.8 cents, was absent from the reactor, and
2. An unknown amount of N₂, worth between 0 and -5 cents, was present inside the reactor.

From the 30.0-inch rod critical position in the last run, and from the rod calibration curve of Fig. 1, it is concluded that the zero-power critical position, on January 24 must have been between 30.6 inches and 31.2 inches. This neglects any possible difference in moderator temperature due to seasonal temperature changes. This conclusion is consistent with observations on the 24th, and indicates that within the limitations imposed by available instrumentation, the poison effect of nitrogen is sufficient to explain the gain in reactivity observed on January 24.

REACTOR PHYSICS

Analytical studies were completed of the data collected for critical mass and power profile determinations during the first and second UHTREX Critical Experiments (UCX-I and UCX-II). Detailed investigations were made of calculational methods for better treatment of the effects of three characteristics of the reactor: the unusual heterogeneity and nonuniform moderator/fuel ratio of the core, the thermal neutron spectrum across the core and the reflector regions, and the presence of low concentrations of contaminants. After the investigations were finished, new estimates were made of the critical mass as a function of temperature.

Homogeneity Effects

The detailed study of the heterogeneity effects in the core confirmed that the approximate methods used previously for treating the fuel heterogeneities are adequate. However, it was found that a more accurate specification of the amount and location of the voids in the core led to better agreement between the calculated and observed values of the eigenvalues (k_{eff} 's) and power profiles.

The refinements in the calculational model have led to close agreement between calculations and measurements in UCX-I and UCX-II. The computer eigenvalues (k_{eff}) for the two reactor assemblies are given in Table VIII below.

In order to perform complicated neutronic calculations in a reactor system such as UHTREX; much

TABLE VIII
UCX CALCULATED k_{eff} 's

Case	Measured Critical Mass (Kg - highly enriched U)	k_{eff}
UCX-I	7.00	1.000
UCX-II	6.28	1.007

The error of 0.7% in k_{eff} for UCX-II corresponds to an error of only 2% in critical mass.

of the detail must be treated by averaging. In particular, before the final two-dimensional calculations can be run for the system as a whole, constants must be derived which represent a limited number of simplified, homogenized fuel regions. A series of cell calculations gives weighting functions which are subsequently used in averaging the neutronic constants. Cells are chosen to represent a basic unit of a particular region of the reactor, the entire region being approximated by an infinite array of these units. The effects of identical neighbors on a unit cell are included in the calculation by using reflective boundary conditions at all exterior surfaces. In UHTREX, these calculations are made more difficult by the fact that the reactor is radially loaded. A wedge-shaped "cell", shown in Fig. 5, that contains a central cylindrical fuel element is derived from an examination of the regular array of radially directed fuel channels. In most calculations to date, this cell has been approximated by a one-dimensional circular cell as depicted in Fig. 6.

Fig. 5 Three-dimensional UHTREX cell.

Fig. 6 One-dimensional circular cell.

A more accurate treatment of the cell was accomplished recently by combining the results of two-dimensional calculations. The first calculation was an X-Y geometry as illustrated in Fig. 7. The principal advantage of this geometry is that the differences in the vertical and horizontal distances between fuel elements are considered. The second calculation was performed in the R- θ geometry, illustrated in Fig. 8, a horizontal section through a representative fuel channel adapted to the R- θ coordinate system. Here, the advantage is that the variation in the horizontal fuel element spacing as a function of radial position in the core is taken into account.

Fig. 7 Two-dimensional X-Y cell.

Results of the two calculations were combined to obtain estimates of the disadvantage factors, d_F and d_M , of the fuel and moderator, which are defined below in terms of the average fluxes $\bar{\phi}_F$, $\bar{\phi}_M$, $\bar{\phi}_{cell}$ in the fuel, moderator, and cell.

Fig. 8 UCX-I horizontal section.

$$d_F = \frac{\phi_F}{\bar{\phi}_{\text{cell}}} ; d_M = \frac{\bar{\phi}_M}{\bar{\phi}_{\text{cell}}}$$

First, partial disadvantage factors d_F' and d_M' were obtained from the R- θ calculation. To compensate for the flux variations in the vertical direction, which are not taken into account in the R- θ problem, computations were made of the ratios of the total disadvantage factors to the horizontal, linear disadvantage factors along the Y axis in the X-Y calculation:

$$C_F = \frac{\bar{\phi}_{F, \text{total}} \bar{\phi}_{\text{cell, total}}}{\bar{\phi}_{F, Y} \bar{\phi}_{\text{cell, Y}}}$$

$$C_M = \frac{\bar{\phi}_{M, \text{total}} \bar{\phi}_{\text{cell, total}}}{\bar{\phi}_{M, Y} \bar{\phi}_{\text{cell, Y}}}$$

The net factors were obtained from

$$d_F = C_F d_F' ; d_M = C_M d_M'$$

These factors, multiplied by the fuel and moderator volume ratios, were applied to the macroscopic cross sections in the standard fashion to complete the effective homogenization of the core.

The procedure was applied to each of the neutron groups used in the calculation. The outcome was that the power profiles and k_{eff} were little different from those obtained from the simple one-dimensional circular cell approximation. The reason is that the variations in the thermal flux, from which most of the heterogeneity effects are derived, are relatively small.

Thermal Neutron Spectrum

The aim in the spectrum study was to find a suitable method for spectrum averaging of the thermal cross sections for use in finite geometry, two-dimensional calculations. For these calculations, the use of more than one thermal group is not, in general, feasible because of the long running times of such problems on computers. In the averaging process, the core and reflector are divided into a number of regions; an infinite medium

flux calculation is performed in a homogenized mixture of the materials in each region; and the fluxes so derived are used as weighting factors to average the thermal cross sections.

Previously, the core was divided into four radial regions corresponding to the four fuel element radial positions. Thus, from the infinite medium calculations, five different sets of thermal cross sections were generated; one set for each of the core regions and one set for the reflector regions. Partly to simplify the procedure, a calculation was done recently in which the entire core was treated simply as a single homogenized region in the spectrum calculation. It was found that this simpler approach provided better agreement between calculated and experimentally observed fluxes than the five-region treatment.

To investigate the effects of core subdivision, four cases were studied as listed in Table IX below:

TABLE IX
DEPENDENCE OF k_{eff} ON REGION SELECTIONS

Case	Calculated k_{eff}	
	25 Group	6 Group
Four core regions	0.99181	0.99149
One core region	0.99054	0.99374
One region combining 5 cm reflector with core	0.99079	1.00165
One region combining 30 cm of reflector with core	0.99181	1.02337

These test calculations were one-dimensional. In the 25-group cases, 13 of the groups were thermal. The results show the eigenvalue does not depend strongly on the method used to derive the 13 thermal groups. More variation is seen in the six-group cases which contained only one thermal group.

Results for flux profiles were similar in that the profiles for all four cases using 25 groups were almost identical, but significant profile differences are obtained when six groups are used. For six-group calculations the best agreement with 25-group results and with experimental results is obtained for the one-region core. Results for this case are plotted in Fig. 9. The wire activation data are from an irradiation experiment in UCX-II.

Fig. 9 Power profile, Case 2.

Power profiles from two-dimensional calculations in which the new spectrum averaging procedure is used, are plotted with UCX-II irradiation data in Fig. 10.

The experimental profile data were described earlier in the Quarterly Status Report on UHTREX for Period Ending October 31, 1967, LA-3835-MS.

Effect of Contaminants

In calculations of the effects of low concentrations of contaminants inside the reactor components, it was found that air, present at 11 psia (one atmosphere at Los Alamos) in the voids, was worth $-0.0034 \Delta k$ in either the UCX-I or UCX-II reactor configuration. Experimental results from UCX-II confirmed this value. The concentrations of other contaminants present in the porous carbon insulation during UCX-II were estimated by chemical analysis, and their worth was calculated to be $-0.006 \Delta k$.

Absorbed water in the UHTREX graphite was evaluated as a possible contributor to contaminant

Fig. 10 Two-dimensional power distribution for UHTREX at room temperature.

worths. From a six-group calculation (5 fast, 1 thermal) the worth of water was determined to be -62¢ per mg of H₂O homogeneously dispersed in each cm³ of UHTREX graphite. The calculation included the effects of water on fast neutron moderation and thermal adsorption. A further calculation, including 25 groups (12 fast, 13 thermal), was done to evaluate the effects of water on the thermal neutron spectrum. The result for the worth of water was -56¢ per mg/cm³. The thermal spectrum in UHTREX is sufficiently soft that the water acts more strongly as an absorber than as a moderator. Based upon the above water worths and reasonable assumptions concerning the possible amounts of water absorbed, it is concluded that for neutronic calculations in UHTREX, water is negligible as a contaminant.

Critical Mass Estimates

Based upon the results of the investigations discussed above, new estimates of the UHTREX critical mass as a function of temperature were made.

These included the design temperature case (1714°K); an intermediate temperature case (914°K), and the ambient temperature case. A summary of the results is given in Table X.

TABLE X
SUMMARY OF NEW UHTREX CALCULATIONS

Average Core Temperature (°K)	Critical Mass M_c (kg U 93.3% enriched)	$\frac{dM}{M} / \frac{dT}{dT}$ at $M=M_c$	$\frac{dM_c}{dT}$ (g/°K)
293	6.277	3.704	2.288
914	8.009	3.507	3.290
1714	11.158	3.659	4.582

The critical mass data versus temperature, together with a polynomial fit to the three points, are plotted in Fig. 11.

Fig. 11 UHTREX critical mass versus average core temperature.

An estimate of the critical mass at an average core temperature of 914°K is needed during the UHTREX startup. The design temperature (1714°K) critical mass is given in Table X does not include the effects of xenon poisoning. Previous estimates indicate that about 1 kg of additional fuel will be required to compensate for the xenon equilibrium poisoning at 3 MW. More precise calculations of the poison worth are in progress.

ENGINEERING RESEARCH AND DEVELOPMENT

Fuel used during startup and the first full power demonstration runs will be inspected visually for physical change and by gamma spectral analysis, metallography, and radiochemical analysis of selected elements. The fission product behavior in the coolant will be determined from analysis of gas samples, diffusion tube probes, wire samples inserted into the coolant, and by surveys of the equipment in the primary loop where possible. During this reporting period emphasis has been placed on installation of diffusion tubes and on the gas sampling system and on design of gamma spectral analysis equipment.

For determining the temperature of fuel elements during operation, two techniques are being developed. One technique depends on observation of the melting of wire inserts and the other on the measurement of thermal expansion. Experiments are being conducted with the two techniques, which will be used during startup to collect data for comparison with calculated temperature profiles.

To facilitate hot restart of the reactor, a high temperature $^{238}\text{PuBe}_{13}$ neutron source has been fabricated, tested, and installed on the lower end of a control rod.

Diffusion Tubes

Two diffusion tube assemblies have been installed in the UHTREX primary loop. One assembly is located in a spare core rod position, sampling gas directly from the reactor core. The other samples gas from the primary blower discharge, Fig. 12. The gas from both diffusion tube assemblies is returned to the primary blower inlet. Each tube assembly consists of three diffusion tubes in parallel. These 0.035-in. wall tubes are each 3/16-in. od x 18-in. long. The material is 304

Fig. 12 Diffusion tube layout.

stainless steel. A filter membrane has been placed in each loop downstream of the diffusion tube assembly. The wiring to the control station in Room 211 and to the process computer is complete and work is proceeding on techniques for removal and handling of the diffusion tubes, gamma scanning and data analysis. A second set of diffusion tubes is on order.

Gas Sampling

The gas sampling system has been modified so that sampling over a range of pressures, rapid detection of a valve failure, and continued use of the system during reactor operation even with the failure of one of several critical valves is possible. To achieve this, three additional valves and a Heise pressure gauge have been added to the system. The new valves are interlocked so that two or more valves isolate the primary coolant loop from the sample point or vacuum pump at all times. When the system is not in use, it is isolated from the primary

coolant loop by four valves.

Gamma Spectral Analysis of UHTREX Elements

Provision is being made to take a gamma spectral analysis of each fuel element shortly after it is removed from the reactor. A solid state Ge(Li) detector in anti-coincidence with a large NaI detector will be installed directly above the fuel cell in Room 104 of the UHTREX facility. A fuel transfer mechanism will transport the element up through the overhead shield from the fuel cell and position it within the shield cave for spectral analysis. It will thus be possible to observe gamma spectra within a few minutes after fuel element discharge and at repeated intervals before transferring the elements for additional observations or long-term storage. With the large number of elements that will be analyzed and the careful control of time interval between irradiation and observation it is anticipated that relatively small differences between elements can be detected.

The mechanism for transferring the element, the detector shielding, cryogenic cooling system, and detector cart for positioning the detector relative to the element have been designed and fabrication of some parts has started. The penetration of the shield slab over the fuel cell is complete and a conveyor tube from the drybox into Room 104 has been installed.

Temperature Inference

Temperature measurement of fuel elements in UHTREX cannot be implemented readily by thermocouples, optical pyrometry, acoustical, or other devices requiring signal transmission. Techniques for inferring the temperature of selected fuel elements are therefore being investigated with the principal effort directed toward fusible wires and thermal expansion devices.

Commercially prepared thermal paints were tested and were completely unsatisfactory for this application. The soft, powder-like lacquers were rejected because they would probably be lost before the fuel elements could be loaded into the reactor. Only a few of the paints responded to temperature, as advertised, when they were applied to graphite and heated in a helium environment and of those, some melts were brittle and nonadherent and would

likely have been lost from the fuel element during unloading.

Fusible, pointed, metal pins, encased within graphite cups, have been tested with very satisfactory results. Holes 0.070-in. in diam were drilled in dummy fuel elements (graphite cylinders) through the cylinder wall parallel to the axis. The graphite cups containing the various melt wires were inserted into the holes and the ends of the holes were closed with a small plug of graphite cement. After loading the melt wires, the element was radiographed so that the shape of the pins and the pointed ends, in particular, could be recorded for later comparison with radiographs after heating. The dummy elements were heated in a helium-filled induction furnace to prescheduled temperatures. The dummy elements were again radiographed and comparisons of shape were made to determine if a melt had occurred.

Generally, radiography has been a satisfactory means to determine whether or not the pins have melted. However, some metals such as aluminum and lead have not indicated melt at the expected temperatures and it is postulated that this is due to an oxide film that helps to retain the point on the wire even though the metal is molten. The investigation will continue on all metals and alloys of interest to determine if radiography is a sufficiently sensitive detector of melt or if additional visual or metallographic methods will have to be employed.

Table XI summarizes the results for each metal or metal combination tested to date. The column, "Maximum No-Melt Temperature," represents the highest recorded temperature which did not induce melting in a wire. The next column, "Minimum Melt Temperature", represents the lowest recorded temperature at which melting was observed.

The second technique being investigated for temperature inference utilizes the thermal expansion properties of a pair of metals. This method shows more promise than the melt wires as a sensitive temperature detector, but remains unproved. In the one device tested, a nickel wire is tightly stretched within a stainless steel tube which upon heating expands at a greater rate than the wire. On cooling, the nickel, which has been stretched beyond its elastic limit, remains expanded. The percent increase of length is a direct measure of the temperature attained.

TABLE XI

Metal or Metal Combination	Maximum No-Melt Temperature, °F	Minimum Melt Temperature, °F
Bi-Pb	260	280
Sn	450	470
Bi	520	530
Pb	650	655
Ag-Sb	1030	1045
Ag-Ge	1610	1645
Al	1280	1290
Ag-Cu	1520	1540
Ag	1800	1800
Cu	2000	Undetermined
60 Pd-40 Cu	2140	2150
77 Pd-23 Cu	2275	2420
Ni	2390	2410
Co	2400	2420
88 Pd-12 Cu	Undetermined	2476
Pd	2480	2670
50 Pt-50 Pd	2690	2940
Rh	2984	3000
88 Pt-12 Ir	3137	3272

The first experimental results (Fig. 13) show the few points attained which, although they form a smooth curve, are too few to be conclusive. It can also be seen that very careful measure of the pin length is essential; any flats produced by micrometer measurements, or temperature variation of the pin during measurement, negate the results. However, continuing developmental work is expected to lead to a reliable and reproducible temperature sensor.

High Temperature Neutron Source

A $^{238}\text{PuBe}_{13}$ neutron source with an intensity of 1.3×10^7 n/sec has been fabricated and proof tested for use in the reactor. The source is doubly contained in niobium and is designed to operate in the reactor helium environment at 500 psia and 2400°F. The source consists of 0.83 g of PuBe_{13} (80% ^{238}Pu - 20% ^{239}Pu). The PuBe_{13} was prepared with a small excess of beryllium present to minimize the presence of free plutonium. The source material

Fig. 13 Temperature estimation through thermal stretching nickel wire.

is held in a beryllium oxide crucible surrounded by granular beryllium oxide filler material to prevent movement of the source relative to its duplex niobium container. Adequate void volume is provided to prevent excessive pressure buildup from the alpha decay of the plutonium. The source container configuration is shown in Fig. 14.

Tests have been run to demonstrate the adequacy of the container to withstand the reactor core environment and to demonstrate that the PuBe_{13} is stable and that no free plutonium is present. The doubly contained source was subjected to a 100 hr proof test in helium at 500 psia and 2000°F before being installed in the reactor.

The source is attached to the end of one of the plug control rods as shown in Fig. 15. The source will be withdrawn from the core during operation at significant power levels. This method of operation will minimize the burnup of the source material and limit high temperature exposure.

Analog Computers

The two TR-48 analog computers have been converted to electronic mode control, making them compatible for use with the 680 analog computer. The current problems of interest, the UHTREX simulation model and the reactivity calculator, were run and debugged following the conversion.

Fig. 14 UHTREX High Temperature Source.

Process Computer

The new PBC (primary blower control) and SBC (secondary blower control) programs have been rewritten and checked extensively. They work well with the new digital control hardware, which has performed faultlessly thus far.

The binaview program has been rewritten to accommodate a second display unit which is on order. A number of improvements have been made in the fuel inventory program and Op-console program.

PLANT ENGINEERING

Effort continued on improving the performance of control systems. The installation of new circuitry and the rearrangement of existing equipment made the nuclear instrumentation more reliable. A new system for indicating control rod position was developed and tested on a mockup of a control rod drive system.

Fig. 15 Source/control rod assembly.

Movement of a drive shaft coupling caused a malfunction of the core indexing drive mechanism. The mechanism was repaired and restored to service. A new power cable feedthrough for the coolant loop blowers was developed to replace one that failed under load.

Instrumentation and Control

Nuclear Instrumentation

Two log N period amplifiers were installed in Rack 18 to replace the two original log power channel chassis, which had not performed satisfactorily. The log N chassis were modified to permit the use of a single test signal input for checking the positive and the negative period readouts, and a chassis was designed and built to produce the test signal. Auxiliary chassis were built for conditioning the log N chassis outputs to furnish an isolated power level signal to the computer, and for operation the < 10W power interlock on source and startup detector withdrawal.

The locations of the ion chambers for the log and linear power channels were rearranged to put the chambers for No. 1 log and No. 1 linear channels in the same well, and the chambers for both No. 2 channels in another well. Thus, in each of these two wells are 2 safety chambers, a log power chamber, and a linear power chamber.

A fission counter was installed in the vacant third slot of the startup detector housing. The fission counter is not retractable, like the BF₃ detectors, because its power and signal wires are enclosed in rigid mineral-insulated high temperature cable. Connected to startup channel No. 3, the fission counter produces a count rate that is 2 decades lower than those of the BF₃ detectors. Eventually, the fission counter will burn out in the neutron flux from the reactor, but during the initial stages of reactor operation the fission counter will make control on startups more certain.

To bring the nuclear safety system wiring into closer conformity with IEEE standards, new coaxial cables were run to connect the ion chambers for nuclear safety system channels 3 and 4 to the terminal strip at the penetration plate cabinet on the west wall of the reactor room. Previously, all four channels used cables that passed across the floor, inside a duct made of aluminum channel bolted to the floor slabs. Now, the two new sets of cables go from the detector well to a cabinet on the south wall, then through large ducts mounted high on the south and west walls, and finally to the west cabinet and terminal strip.

Control Rod Position Indication

Work started on the development of a new position indicator system for the control rods. The new system will be designed to replace the mechanically driven digital meters that indicate rod position and the associated circuitry that receives the position signal from the selsyn transmitter in each control rod drive.

The present selsyns (transmitter in the rod drive, and slave at the control room chassis) will be used in the new system to drive an indicator system that is more direct, less subject to drift, and less difficult to calibrate than the present system.

In the first test of an indicator system, commercial units that are used for very precise indication of angular movement in servo systems were tested with the output signal from the No. 1 plug rod selsyn. The test was unsuccessful; the readout units would not operate fast enough to follow the rod on a scram descent.

For later tests the spare rod drive was set up in a test rig. The second design to be tested used a mechanical counter directly coupled to the slave selsyn through a gear train. Although the design was successful in a breadboard assembly, the slave selsyn did not develop enough torque to drive the counter during a rod scram.

A third design uses a shaft encoder driven by the slave selsyn. The shaft encoder system performed with great ease and precision in a breadboard setup, and followed the rod scram in a test run on the rod drive rig. However, on the scram test, inertia caused the encoder to overrun at the end of the stroke. A stop is being built to correct this fault. Hysteresis in the encoder system alone was 0.02 - 0.03 in. on slow approaches from both directions to a single position of the rod drive. When the computer input potentiometer was added to the gear train, the hysteresis increased to about 0.04 in., well within the desired limit of 0.1 in. The associated electronic circuits are being investigated in (1) a mockup, using plug-in logic circuit cards of the nixie tube readout display, and (2) a breadboard version of an ultrastable power supply for use with the computer input potentiometer.

Safety Systems

Mercury batteries, formerly used as sources of reference voltages, have been replaced with Zener diode-regulated reference voltages from power supplies in each of the following systems:

- Primary loop pressure relief,
- Primary loop coolant flow monitor,
- Heat exchanger tube temperature monitor,
- Reactor inlet pipe temperature monitor,
- Secondary loop pressure relief, and
- Secondary loop coolant flow monitor.

As originally designed, in each of the independent trip circuits, a battery supplied a standard voltage to balance input signals from the transducers. If a battery were removed or were to fail, the associated circuit was disabled and could not function if a trip signal were generated by the transducer. To correct this defect, the batteries were removed from the circuitry and replaced with the output of a stable power supply already installed in the chassis. Now, if the power supply fails, the trip circuits can function.

Signal Conditioning

In recent tests of the temperature and pressure instrumentation channels, 8 of the 15 amplifiers that produce signals for temperature meters and annunciator trip circuits were not operating. Bench tests of a new design in a breadboard chassis showed that plug-in circuits, made from stock parts and readily replaceable with spare cards, could be substituted for the original chassis with a considerable improvement in reliability and ease of maintenance. The new circuits are being built and will be substituted for the original chassis when operating plans permit.

Reactor

Internal Neutron Source

After proof tests were finished on the internal neutron source, (see section Engineering Research and Development) it was installed on the end of the spare plug control rod and the combination rod and source was substituted for the original plug rod no. 6 in the reactor. No changes were made in the rod drive.

Core Indexing Mechanism

During fuel loading operations which required the rotation of the core, the core drive stopped operating as the core was moving counter clockwise past one index position to the next. Thorough checks were made of the computer program that was executing the operation and of the electrical circuits associated with the core drive, but no faults were found. The overload protectors for the electric drive motor had opened. Meters to monitor the motor current and voltage were installed and the core drive was operated under manual control. The drive operated spasmodically with motor current surges and stops during operations in both the clockwise and counter directions. While operating, the drive made unusual noises. Finally, the core stopped moving altogether and the decision was made to remove and examine the indexing mechanism.

After the seal weld was cut with a grinding wheel, the pressure case was removed from the indexing mechanism, and inspection at the working parts revealed substantial damage. The two cam rollers that engage the star wheel (see Fig. 16) were broken off the driver assembly, and all four camming surfaces on the star wheel and the adjacent flat surfaces were deeply eroded. The rear surface of the coupling half on the shaft (see Fig. 17) that is driven by the star wheel was severely galled by contact with the housing through which the shaft protrudes.

Apparently, when the indexing mechanism was connected to the core drive shaft, during assembly the coupling was driven back along the shaft into contact with the housing. Subsequently as the core was rotated, the contacting surfaces galled until the output torque required to drive the shaft was enough to deform and finally break the driver cam rollers.

To repair the mechanism, a new star wheel was machined, new cam rollers were bought, and the drive gear teeth were dressed to remove marks. As a result of reassessments of temperature gradients along the core drive shaft support tube, a decision was made to provide more room for expansion at the outer end of the tube, which is restrained by a steel spacer with a resilient facing made of Garlock gasketing material. A 1/16-in. cut was taken off of the spacer and replaced with Garlock. To prevent a recurrence of contact between the drive

Fig. 16 Star wheel indexing mechanism.

shaft coupling and the drive housing, 3/8-in. of steel was machined off of the rear face of the coupling half and the coupling half was pinned to the shaft with an axial set screw. The short drive on which the coupling half is mounted was shortened 1/8-in. to improve clearances in the assembled coupling.

In addition to the damage which had occurred in the coupling and star wheel assembly, during disassembly the ceramic insulators were broken on two of the pins in the power and instrument feedthrough connector. To replace these pins, a two-conductor Ceramaseal feedthrough for the position indication circuit was mounted on the flange of the pressure case. Three of the pins in the original feedthrough were left in use for power conductors, and the two defective pins were sealed off.

After the star wheel and rollers were replaced, the mechanism was reassembled and tested, and the

Fig. 17 Core indexing mechanism.

pressure case reinstalled and seal welded. The entire mechanism was reinstalled on the reactor and performed smoothly in operational tests.

Coolant Systems

Diffusion Tubes

After preliminary sketches were made, alterations to the core diffusion tube and methods for removing it from the core were discussed. Present plans are to make the diffusion tube section removable from the upper structure; for disassembly, the diffusion tube section will be withdrawn into a helium-filled enclosure, and a collet will hold the top of niobium tube just above the conoseal joint at the top of the core rod tube. The diffusion tube section then will be disconnected and replaced with a fresh section inside the enclosure. Finally, the entire assembly will be replaced in the core. Provision will also be made for removal of the entire assembly and capping of the core rod tube.