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AEC RESEARCH AND
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MASTER

Nuclear Materials & Propulsion Operation

710 REACTOR PROGRAM, PROGRESS REPORT No. 12

JANUARY 29, 1965

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INTRODUCTION AND SUMMARY

Introduction

This report is the twelfth in a series of progress reports on the development of refractory metal reactor technology and an experimental fast-spectrum, refractory-metal reactor (designated the 710 reactor). This report describes the work performed and progress made from November 1 to December 31, 1964, on the following current tasks in the 710 reactor program:

1. Development of Refractory-Metal Fuel Elements
2. Critical Experiment of Fast-Spectrum Refractory-Metal Reactors
3. Experimental Reactor Preliminary Design and Development
4. Test Facilities and Pilot Loop Design and Evaluation

Summary

Effort during the report period from November 1 through December 31, 1964, resulted in the following significant accomplishments.

DEVELOPMENT OF REFRACTORY-METAL FUEL ELEMENTS

1. During this report period, four partial-length tantalum-clad 37-hole fuel elements, three single-hole tantalum-clad test specimens, three process-development tantalum-clad capsules, and three advanced single-hole specimens clad with W-Re-Mo were hot-gas pressure-bond cycled.
2. Two single-hole tantalum-clad specimens and one single-hole W-Re-Mo-clad specimen were leak-tight and bonded after 100 thermal cycles from 3900 to 500°F.
3. One process development tantalum-clad capsule was leak-tight and bonded after 507 hours at 3990°F.
4. Improvement in dimensional stability of about 40 percent was gained by a modified preparation process of the matrix material prior to sealing the fuel element cladding. A reduction of 60 percent dimensional growth was observed when thermal cycling was performed with controlled cooling rates which result in thermal strains expected in the nuclear operation of the Experimental Reactor during a full-flow scram.
5. Checkout of the low-pressure circulating loop was begun.

CRITICAL EXPERIMENT OF FAST-SPECTRUM REFRACTORY-METAL REACTORS

1. Operation of the basic 710 critical experiment was completed and the assembly was dismantled.
2. Mockup No. 1A was assembled in the split table assembly and core loading was begun.
3. Preanalysis of Mockup No. 1A was refined, and analysis of the basic critical experiment was continued.

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EXPERIMENTAL REACTOR PRELIMINARY DESIGN AND DEVELOPMENT

1. The initial configuration was modified slightly, from that reported previously, in the core support structure and radial reflector areas. Preliminary analyses indicate that the mechanical and thermal stresses are within allowable limits, with the exception of two areas, which can be corrected with a minimum of change and that, at design point conditions, thermal deflections are all accommodated by the clearances provided. A full-scale model of the fuel element cartridge was fabricated to enable establishing manufacturing and assembly procedures.
2. Temperature and flow pressure drop calculations were made for the Experimental Reactor components. Use of analysis methods for determining fuel element temperature variations were validated. A preliminary analysis of the inner shell at the pressure vessel was completed, and specifications for gap spacing between the shell and the pressure vessel head, the shell hole pattern, and the geometry at the centerline of the reactor were determined. The temperature distribution in the control drum at a location near maximum heat generation indicates the rates are within the acceptable range.
3. Nuclear models were prepared for use in Monte Carlo analyses to determine (1) neutron energy deposition, fluxes, and leakage and (2) gamma ray energy deposition and leakage.
4. The overall test loop simulation was modified to incorporate the present core design configuration, to permit off-design evaluation of the system, and to permit analyses of temperature coefficients of reactivity. In addition, a heat transfer stage was added to the reactor heat transfer simulator to simulate the side and front reflectors.

TEST FACILITIES AND PILOT LOOP DESIGN AND EVALUATION

1. Studies of the test equipment and facilities were continued.
2. Studies to predict the magnitude of plateout of fission products in the primary test loop were initiated.
3. Advanced engineering work on the power test equipment was continued.
4. The control, presentation, and monitoring functions of the digital computer system were defined.
5. Specifications were completed and initial procurement action was undertaken for all major pilot loop components.
6. A study was made to determine preliminary remote handling equipment requirements.

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1. DEVELOPMENT OF REFRACTORY-METAL FUEL ELEMENTS

Progress in fabrication techniques, fuel element design, stress analysis, and non-nuclear testing is reported herein. The primary objective of this work is to translate the fuel element concept into practical hardware by developing and demonstrating product reproducibility within the limits necessary for reactor applications, and by proving physical acceptability of the fuel elements through analysis and testing.

Partial-length fuel elements of the 37-hole, single-hole, and process-development designs, respectively, are used for the basic investigations. This approach provides fuel elements suitable for test purposes without the expense of producing the number of compacts needed for full-length fuel elements.

FABRICATION TECHNOLOGY DEVELOPMENT

During this report period, four partial-length 37-hole fuel elements, three single-hole test specimens, three process-development capsules (all tantalum-clad), and three advanced single-hole W-Re-Mo-clad* specimens were hot-gas pressure-bond cycled. All were leak-tight and bonded with the exception of one partial-length 37-hole fuel element which developed cracks in a header plate.

Matrix Compact Fabrication

During this report period, 58 hexagonal compacts and 84 compacts, each 1.5 inches in diameter, were sintered. As shown in Table 1.1 and in previously reported data,[†] the present lot of UO₂ powder, even with increased hydrogen gas access to the compacts during the standard sintering cycle, did not produce compacts at the 95-percent of theoretical density level. The data in Table 1.1 indicate that slower heating rates in the temperature range of 2190° to 2750°F are required to maintain the average sintered density at a 94-percent of theoretical level for the larger hexagonal compacts.

Additional tests were initiated to provide a better understanding of the effect of heating rate on densification of fuel cores, with the ultimate goal of optimizing the sintering cycle to obtain maximum density. Dimensional changes during sintering will be observed with twin telemicroscopes as W, UO₂, and 54W - 46UO₂[‡] pressed compositions are heated at varying rates and conditions. The effect of powder from different lots and of various additives to the core material on rates of densification will be evaluated.

*40W - 30Re - 30Mo, at. %.

[†]"710 Reactor Program, Progress Report No. 11," GE-NMPO, GEMP-323, November 30, 1964, Table 1.1, p. 10.

[‡]All compositions are in weight percent unless otherwise noted.

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TABLE 1.1
COMPACT FABRICATION DATA

	No. Of Cores	Composition, wt %			Average Density, % of theoretical	
		W	UO ₂	ThO ₂	Sintered	Machined
Hexagonal Compacts^a						
Standard sintering cycle, 22°F per minute from 2190° to 2750°F	9	54	46	-	91.5	91.7
Modified sintering cycle, 9°F per minute from 2190° to 2750°F	8	54	46	-	94.4	93.0
Modified sintering cycle, 9°F per minute from 2190° to 2750°F	4	54	41.4	4.6	94.1	-
Modified sintering cycle, 90-minute hold at 2190°F, 9°F per minute from 2190° to 2750°F	37	54	41.4	4.6	94.6	-
Circular Compacts^b						
Standard sintering cycle, 22°F per minute from 2190° to 2750°F	21	54	41.4	4.6	92.5	-
Modified sintering cycle, 90-minute hold at 2190°F, 22°F per minute from 2190° to 2750°F	63	54	41.4	4.6	94.2	-

^aSintered in H₂ for 2 hours at 3990°F.

^bSintered in H₂ for 1 hour at 3990°F.

Changeover of the machining coolant to low-viscosity hydrocarbon was completed. Excellent results are being obtained with this lubricant. Freon* was substituted for water in the ultrasonic cleaning system to compensate for the change to the hydrocarbon coolant. Tests indicate that Freon will remove the hydrocarbon from the compacts.

Formal manufacturing instructions were issued for all fabrication, assembly, and inspection operations associated with fuel-bearing materials. The instructions were coordinated with health physics personnel to assure safe processing of enriched fuel.

A mechanical feed system on a core drilling machine was employed in initial machining tests. The first full-segment run on this machine was within drawing tolerance for hole location and hole diameter. Initial drilling time per hole was set at 2 minutes. Good test results were achieved with this equipment when non-thoriated compacts were used; however, use of thorium-bearing material resulted in drill breakage. The harder material surface, coupled with the continuous mechanical feed, led to an increase in torsional forces at the interface of the diamond crown and carbide shank, where the drills separated in shear.

With the exception of the coolant glands on the core drilling equipment, equipment operating reliability was excellent during this report period. Modifications and refinements are currently being made to the core drilling equipment which should reduce radial loads on the seals which have been causing uneven wear and seal failure.

The trial assembly of segments for the first full-length 37-hole fuel element (F-1) was completed. Each of the 27 segments has dimensional characteristics within drawing tolerances with the exception of one segment in which two hexagonal surfaces are 0.0003- and 0.0004-inch undersized, respectively. Table 1.2 presents the distribution of 999 hole diameters taken from the inspection of the 27 segments. All 999 holes fell within the 0.001 inch radius of true position.

*Registered trade name of E. I. du Pont de Nemours and Co., Inc.

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TABLE 1.2
DISTRIBUTION OF HOLE DIAMETERS
FOR THE 27 SEGMENTS OF FULL-
LENGTH FUEL ELEMENT F-1^a

Dimension, in.	Occurrences
0.0800 Drawing minimum	0
0.0801	0
0.0802	0
0.0803	0
0.0804	0
0.0805	2
0.0806	53
0.0807	65
0.0808	154
0.0809	153
0.0810 Drawing nominal	182
0.0811	98
0.0812	131
0.0813	28
0.0814	79
0.0815	37
0.0816	17
0.0817	0
0.0818	0
0.0819	0
0.0820 Drawing maximum	0

^aAverage diameter, 0.0810 inch.

The P-6 partial-length fuel element was assembled using cores containing 4.6 weight percent ThO₂ which were treated in wet hydrogen prior to machining and in dry hydrogen after machining. Core growth, ranging from 0.0004- to 0.0009-inch across flats, occurred during the dry hydrogen treatment after machining. The average growth across flats was 0.0007 inch.

Cladding Fabrication

Tooling was modified for the drilling of header plates and header-plate embossing electrodes, and a total of five Ta - 10W header plates and six electrodes was drilled within drawing tolerances.

Forming of tantalum hexagonal halves for fuel element outer cladding with a hardened punch and a urethane foam die was highly successful. Figure 1.1, showing a half-section being separated from the punch, illustrates this tooling. Excellent control and reproducibility of size and included angles were maintained using this technique. The isostatic sizing operation continues to bring the welded half-sections within the drawing envelope requirements.

Machinability evaluation of header-plate materials for the full-length fuel element configuration was begun. Studies on Mo-TZC (Mo - 1.25Ti - 0.30Zr - 0.10C) have shown a greater degree of brittle fracture in chip formation than was experienced with molybdenum. Cutting action is facilitated by the more brittle structure, with resultant tool wear being approximately equivalent to that experienced with tantalum and less than that experienced with Ta - 10W. Three 0.187-inch-thick Mo-TZC header plates were drilled to acceptable tolerances during this study.

Other machinability tests were run on Mo-TZC in an attempt to generate the required thread on the forward header. Brittle fracture of the material occurred, but not to a sig-

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Fig. 1.1—Hardened steel punch and urethane rubber die for forming tantalum half-hexagonal sections (Neg. P64-11-27C)

nificant degree. Results to date indicate that acceptable machining techniques can be developed for this operation.

Fabrication of tooling for forming tantalum hexagonal cans from round seamed or seamless stock is nearing completion. The hexagonal cladding will be formed by drawing a hexagonal mandrel through a round tube supported on the outside surface.

The ability to form tantalum to close tolerances by isostatically pressing has re-opened the possibility of using extruded seamless hexagonal tubing with relatively loose tolerances. A re-survey of vendor extrusion capabilities is currently being made to eliminate the need for in-house tube fabrication.

Partial-Length Fuel Element Fabrication

The assembly and sealing techniques for two partial-length fuel elements (P-1 and P-3) were described in a previous progress report.* In addition to these two assemblies, eight partial-length 37-hole fuel element assemblies, all with the P-3 header plate design,[†] are scheduled for fabrication. Table 1.3 presents matrix compact fabrication data for all ten partial-length fuel elements. Four of these partial-length assemblies (P-1, P-3, P-4, and P-5) were used to check out fabrication techniques used on the 91-hole elements.

As shown in Table 1.3, each of the eight new assemblies, with the exception of P-4 and P-5, will contain matrix compacts treated in wet and dry hydrogen intended to reduce the free uranium content. This treatment will be identical to that described in a previous progress

*"710 Reactor Program, Progress Report No. 11," GE-NMPO, GEMP-323, November 30, 1964, p. 11.

[†]Ibid., Figure 1.3, p. 12.

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TABLE 1.3
PARTIAL-LENGTH FUEL ELEMENT FABRICATION DATA

Fuel Element	Matrix Composition, wt %			Fuel Mesh Size	Post-Sinter Treatment	Status
	W	UO ₂	ThO ₂			
P-1	54	46	0	-100/+200	None	^a
P-2	54	41.4	4.6	-100/+250	Wet H ₂ - Dry H ₂	In process
P-3	54	46	0	-200/+325	None	Bonded & leak tight
P-4	54	46	0	-200/+325	None	Bonded & leak tight
P-5	54	41.4	4.6	-100/+250	None	Bonded & leak tight
P-6	54	41.4	4.6	-100/+250	Wet H ₂ - Dry H ₂	Bonded & leak tight
P-7	54	41.4	4.6	-100/+250	Wet H ₂ - Dry H ₂	In process
P-8	54	41.4	4.6	-100/+250	Wet H ₂ - Dry H ₂	In process
P-9	54	41.4	4.6	-100/+250	Wet H ₂ - Dry H ₂	In process
P-10	54	41.4	4.6	-100/+250	Wet H ₂ - Dry H ₂	In process

^a Assembled, sealed, and hot-gas pressure-bond cycled; cracks in header plate prevented good bonding.

report,* except that the dry hydrogen cycle conditions will be the same as for the wet hydrogen cycle; i. e., 4 hours at 2550°F. All compacts, except those in the P-8 assembly, will be treated in wet hydrogen prior to machining and in dry hydrogen prior to assembly. The compacts in the P-8 assembly will be treated in both wet and dry hydrogen prior to machining and ultrasonically cleaned in Freon after machining.

Two assemblies, P-3 and P-4, contain matrix compacts with -200/+325 mesh fuel. As previously reported,[†] experimental capsules containing such fuel compacts showed less dimensional growth during thermal cycling than capsules containing -100/+200 mesh fuel particles in the matrix; however, because subsequent metallographic analysis showed fuel clustering, especially in the large hexagonal compacts, to be more pronounced with the -200/+325 mesh particles, compacts will continue to be fabricated using the larger fuel particles. Subsequent evaluation of fuel particle size ranges revealed that extending the mesh size range from +200 to +250 significantly reduced the amount of fuel requiring re-processing. Therefore, -100/+250 mesh size was incorporated into the fuel agglomeration process.

The outer hexagonal cladding for all ten partial-length fuel elements is 0.010-inch-thick tantalum. The tubes are 0.078-inch-OD tantalum with 0.008-inch-thick walls. Header plates for all ten assemblies were electrical-discharge-machined from Ta - 10W alloy stock.

Partial-length fuel elements P-1, P-3, P-4, and P-5 were hot-gas pressure-bonded for 1 hour in helium at 10,000 psi and 3000°F. With the exception of P-1, all assemblies were completely bonded and were leak-tight to helium gas at 500 psi. Assembly P-4, shown after bonding in Figure 1.2, is typical of the other bonded assemblies. Leaks developed in the header plates of assembly P-1 at the bases of the tube extensions and prevented successful bonding in the pressure-bonding cycle.

The increase in matrix densification during hot-gas pressure-bonding was calculated for each assembly; results are shown in Table 1.4.

At the close of this reporting period, partial-length fuel element P-6 was assembled; all other assemblies were in various stages of processing.

* Ibid., p. 11.

† Ibid., p. 26.

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Fig. 1.2 - Partial-length fuel element P-4 after hot-gas pressure bonding at 10,000 psi and 3000°F for 1 hour (Neg. P64-11-37)

Full-Length Fuel Element Fabrication

Fabrication of the first full-length, 37-hole fuel element assembly, F-1, was begun during this reporting period. To obtain the required 15 inches of fueled compact length, 27 compacts of 54W - 46UO₂ were processed. All hardware components for the fuel element were nearing completion at the close of this reporting period.

The design of the full-length fuel elements incorporates an electrical-discharge-machined header plate of Mo-TZC alloy (Mo - 1.25Ti - 0.30Zr - 0.10C). In addition to metallographic analysis, a bend test in the temperature range from 3000° to 3750°F was conducted on extruded material, in the stress-relieved (1 hour at 2300°F) condition, to observe effects of recrystallization on ductility. Bend test results are given in Table 1.5. Photomicrographs of the material in the as-received condition and after each heating are shown in Figure 1.3. The metallographic results indicate that a temperature of 3600°F for 1 hour is required to attain complete recrystallization. All test pieces exhibited brittle fractures after heat treat-

TABLE 1.4

PARTIAL-LENGTH FUEL ELEMENT MATRIX DENSIFICATION IN HOT-GAS PRESSURE-BONDING

Fuel Element	Matrix Density, % of theoretical		
	Pre-Bonding	Post-Bonding	Increase
P-3	94.3	96.1	1.8
P-4	93.1	95.8	2.7
P-5	91.5	93.3	1.8

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TABLE 1.5
Mo-TZC RECRYSTALLIZATION STUDY^a

Heat Treatment ^b		Estimated Percent Of Recrystallization	Microhardness ^c DPH No.	Bend Test ^d Results
Temperature, °F	Time, hr			
As-received	1	0	431	Brittle
3000	1	10	305	Brittle
3200	1	40	282	Brittle
3400	1	75	262	Brittle
3600	1	100	243	Brittle - slight deformation at fracture
3750	1	100	226	Brittle

^aAll samples 0.060 by 0.50 by 0.50 inch cut from extruded rod 1.75 inches in diameter and 11 inches long.

^bAll heat treatments in helium atmosphere.

^c100-gram load; 50X objective.

^dSpecimen supported in "V" block, load applied in hand press. Load applied parallel to direction of extrusion.

ment. Some deformation at fracture was observed in the sample tested at 3600°F. In all samples, the major axis of the elongated grains was parallel to the direction of bending-load application which may have contributed to the brittle behavior. Additional samples are being machined with the major axis of the grains perpendicular to the bending-load direction.

Single-Hole Test Specimen Fabrication

The fabrication of eight single-hole test specimens containing compacts specially treated to reduce the amount of free uranium was discussed in the previous progress report.* At that time, three specimens (SH-51, SH-53, and SH-55) had been tested; two (SH-52 and SH-56) were ready for test, and three others (SH-54, SH-57, and SH-58) were being assembled. The latter three specimens were sealed and hot-gas pressure-bonded during this report period.

Complete fabrication data on these eight single-hole test specimens, including compact treatment, uranium content analysis, and composite matrix density before and after hot-gas pressure-bonding, are given in Table 1.6. Originally, an 8-hour hold at 2200°F in dry hydrogen, to insure elimination of excess oxygen, was planned;* however, this was changed to a 4-hour hold at 2550°F. As shown in Table 1.6, the uranium content of all specimens was less than 0.01 weight percent, after the dry hydrogen treatment, which approaches the lower limit of detection by hydriding under the present analytical procedure. Small dimensional changes were observed in the machined compacts after the wet hydrogen cycle, with little or no additional changes occurring in the dry hydrogen cycle. While the dimensional changes for the single-hole compacts were slight and did not interfere with the assembly operation, further studies of the wet - dry hydrogen treatment cycle are required to determine the sequence of operations to be used for the 37-hole hexagonal compacts. The extent of matrix densification during the hot-gas pressure-bonding cycle is shown in Table 1.6. The increases in density are similar to those observed both in 37-hole fuel elements

*Ibid., p. 13.

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As-received, stress-relieved 1 hour at 2300°F. Grains elongated in the direction of extrusion. (Neg. 5321)

3000°F, start of recrystallization (Neg. 5322)

3200°F, partial recrystallization (Neg. 5323)

Fig. 1.3—Structure of Mo-TZC 1.75-inch-diameter extruded bar after heat treating for 1 hour at various temperatures (Etched, 100X)

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◁ 3400°F, additional recrystallization and beginning of grain growth (Neg. 5324)

3600°F, complete recrystallization and grain growth. Structure is mixture of some equi-axed grains and others oriented in original extrusion direction with scattered grain boundary carbide formation. (Neg. 5325)

◁ 3750°F, additional grain growth tending to the formation of an equi-axed grain structure with carbide networks at grain boundaries. (Neg. 5326)

Fig. 1.3 (Cont.) – Structure of Mo-TZC 1.75-inch-diameter extruded bar after heat treating for 1 hour at various temperatures (Etched, 100X)

TABLE 1.6
FABRICATION DATA OF SINGLE-HOLE TEST SPECIMENS

Specimen	Composition, wt %			Post-Sinter Treatment ^a			Uranium Content, ^b wt %		Total Wet H ₂ - Dry H ₂ Change In OD, %	Fueled Matrix Density, % of theoretical		
	W	UO ₂	ThO ₂	Atmosphere	Temperature, °F	Time, min	Pre-Treatment Wet H ₂	Post-Treatment Dry H ₂		Pre-Bond	Post-Bond	Increase
SH-51	54	41.4	4.6	Vacuum (1 x 10 ⁻⁴ Torr)	2200	30	0.09	0.03	0.0	92.5	93.8	1.3
SH-52	54	41.4	4.6	Vacuum (1 x 10 ⁻⁴ Torr)	2200	30	0.07	0.07	0.0	90.8	92.4	1.6
SH-53	54	41.4	4.6	Wet H ₂	2550	240	0.10	<0.01	+0.4	90.7	93.2	2.5
				Dry H ₂	2550	240						
SH-54	54	41.4	4.6	Wet H ₂	2550	240	0.10	0.01	+0.8	91.4	93.2	1.8
				Dry H ₂	2200	480						
SH-55	54	46	0	Wet H ₂	2550	240	0.13	<0.01	+1.9	94.8	96.4	1.6
				Dry H ₂ ^d	2200	480						
				Dry H ₂	2200	480						
SH-56	54	46	0	Wet H ₂	2550	240	-	<0.01	+1.6	94.5	97.6	3.1
				Dry H ₂	2200	240						
SH-57	54	43.3	2.7	Wet H ₂	2550	240	0.18	-	+1.3	93.3	94.9	1.6
				Dry H ₂	2550	240						
SH-58	54	43.3	2.7	Wet H ₂	2550	240	0.18	-	+1.5	93.5	95.1	1.6
				Dry H ₂	2550	240						

^aFor all wet hydrogen runs, +60°F inlet dewpoint hydrogen was used; for all dry hydrogen runs, -50°F inlet dewpoint hydrogen was used.

^bUranium content is reported on total sample weight.

^cAll specimens were hot-gas pressure-bonded at 10,000 psi in helium for 1 hour at 3000°F.

^dMatrix compacts were unloaded from furnace prematurely and surface oxidation was observed; compacts were then re-cycled in dry hydrogen.

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and X-type, process-development capsules. Additional studies are planned to determine the factors preventing greater matrix densification during hot-gas pressure-bonding.

Advanced Single-Hole Test Specimen Fabrication

Alloy development and fabrication work on W-Re and W-Re-Mo conducted under the High-Temperature Materials Program* has shown that the composition, W - 30Re - 30Mo (at. %), exhibits excellent fabricability to sheet of suitable quality for test specimen evaluation. Three advanced single-hole test specimens (ASH-1, ASH-2, and ASH-3) were fabricated with cladding of W - 30Re - 30Mo (at. %) alloy over the W - 41.4UO₂ - 4.6ThO₂ matrix composition.

Prior to assembly, all finish-machined matrix compacts were treated for 4 hours at 2550°F in wet and dry hydrogen, respectively.

The inner and outer tubes were fabricated from 0.015-inch-thick powder-metallurgy sheet stock. Electron-beam-welded inner and outer tubes are shown in Figure 1.4 prior to machining to proper test specimen length.

All three test specimens were hot-gas pressure-bonded at 10,000 psi helium pressure for 2 hours at 3000°F. The standard 1-hour holding period was increased to 2 hours to insure creep of the stronger cladding and to assure cladding-to-matrix bonding. Specimens ASH-2 and ASH-3 (Figure 1.5) were bonded successfully in one cycle. A crack in the inner seam weld of ASH-1 prevented bonding in the first cycle. The crack was repaired and ASH-1 was recycled and successfully bonded.

Matrix densities were calculated before and after hot-gas pressure-bonding with the results shown in Table 1.7. The amount of matrix densification was similar to that observed for tantalum-clad specimens, indicating that with the 2-hour holding period the creep rate of the W-Mo-Re cladding was not limiting the matrix densification.

TABLE 1.7

ADVANCED SINGLE-HOLE TEST SPECIMEN MATRIX DENSIFICATION

Specimen	Density, % of theoretical		
	Pre-Bonding	Post-Bonding	Increase
ASH-1	91.8	92.9	1.1
ASH-2	90.3	92.1	1.8
ASH-3	91.3	93.0	1.7

Process-Development Capsule Fabrication

Fourteen additional experimental process-development capsules (X-type) were scheduled for fabrication; these capsules will be used to investigate variations in core composition, method of core densification, and sequence of the wet and dry hydrogen treatments.

Tantalum tubing, with a 1.030-inch OD, 0.015-inch-thick wall, and 0.100-inch-thick tantalum end caps, is used for cladding all capsules. Fabrication data pertinent to this series of capsules are listed in Table 1.8.

As shown in Table 1.8, three capsules were hot-gas pressure-bonded and are ready for test. The remaining 11 capsules are in various stages of preparation. Data obtained from capsules X-49 through X-52 will aid in establishing the wet - dry hydrogen treatment for the 37-hole fuel segments where slight dimensional changes in the wet - dry hydrogen

*"High-Temperature Materials Program Progress Report No. 41, Part A," GE-NMPO, GEMP-41A, November 30, 1964, pp. 34-38.

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Fig. 1.4 – Electron-beam welded inner and outer W-Mo-Re tubes prior to machining to proper specimen length (Neg. P64-11-12)

Fig. 1.5 – Advanced single-hole W-Mo-Re-clad test specimen ASH-3, bonded and leak tight (Neg. P64-12-23)

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TABLE 1.8

CAPSULE FABRICATION DATA FOR X-TYPE PROCESS DEVELOPMENT SPECIMENS

Capsule	Matrix Composition, wt %				Process Variable	Matrix Density, % of theoretical		
	W	UO ₂	ThO ₂	Mo		Pre-Bonding	Post-Bonding	Increase
X-29	-	61.7	-	38.3	Matrix composition, Mo-UO ₂	96.0	98.3	2.3
X-30	-	61.7	-	38.3	Matrix composition, Mo-UO ₂	96.1	98.6	2.5
X-31	54	46	-	-	High-density UO ₂ , hot-pressed ^a	94.0	96.0	2.0
X-32	54	46	-	-	High-density UO ₂ , hot-pressed ^a	b		
X-33	54	46	-	-	Low-density UO ₂ , hot-pressed ^c			
X-34	54	46	-	-	Low-density UO ₂ , hot-pressed ^c			
X-35	54	46	-	-	High-density UO ₂ , HGPB ^d			
X-36	54	46	-	-	High-density UO ₂ , HGPB ^d			
X-47	30.9	52.7	-	16.4	Matrix composition, Mo-W-UO ₂			
X-48	30.9	52.7	-	16.4	Matrix composition, Mo-W-UO ₂			
X-49	54	41.4	4.6	-	Wet H ₂ + machining + dry H ₂			
X-50	54	41.4	4.6	-	Wet H ₂ + machining + dry H ₂			
X-51	54	41.4	4.6	-	Wet H ₂ + dry H ₂ + machining			
X-52	54	41.4	4.6	-	Wet H ₂ + dry H ₂ + machining			

^aVacuum hot-pressing of tungsten and high-density nonsinterable UO₂ spherical particles.

^bCapsules X-32 to X-52 inclusive, in fabrication.

^cVacuum hot-pressing of tungsten and low-density sinterable UO₂ -100/+250 fuel particles.

^dHot-gas pressure bonding of tungsten and high-density, nonsinterable UO₂ spherical particles followed by clad stripping and machining into segments for encapsulation.

cycle can interfere with compact assembly. Hydrogen treatment of the fueled compacts prior to machining, if proven as effective as the post-machining treatment, would result in better dimensional control for assembly of the fuel element.

Hot-Gas Pressure-Bonding

Calculations of fueled matrix densities before and after hot-gas pressure-bonding of fuel element assemblies show that a relatively uniform degree of densification for a given pressure level is obtained regardless of assembly size, geometry, type of cladding, matrix initial density or composition, fuel particle size, matrix cleaning method, or special treatments. From the data accumulated, it appears that increasing the holding time at the 10,000 psi level from 1 to 7 hours has no significant effect on the percentage of matrix densification. Increasing the pressure level from 5,000 to 10,000 psi does appear to improve matrix densification significantly. The data also indicate that materials of initially low densities do not achieve higher density increases than matrixes of much higher initial densities.

The status of the new hot-gas pressure-bonding facility was described in a previous report.* After reviewing the furnace element failures, the vendor decided to procure individual isolating 440/110-volt transformers to reduce voltages between windings and between windings and ground. With this change, the cross-sectional area of the heater windings was increased and a grooved coarse-grain alumina core secured for the winding. The isolating transformers were installed and all components required for installation of the new heating element were received except the alumina-cored heating element.

MATERIALS INSPECTION

Inspection was completed on the replacement order of tantalum tubing with 0.078-inch OD and 0.008-inch-thick wall.* An 85-percent yield, based on acceptable 3-inch lengths, was realized. Inspection of an order of tantalum tubing with 1.001-inch ID and 0.015-inch-

*Ibid.

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thick wall for single-hole test specimen fabrication yielded only 25 percent of tubing acceptable for 3-inch test specimens. These results indicate that the quality of tubing is not so much reliant on diameter or wall thickness as on inherent material quality and processing procedures.

FUEL ELEMENT DESIGN AND STRESS ANALYSIS

Analysis of fuel element stress - strain relationships in the Experimental Reactor and in non-nuclear tests was continued during this report period using the equivalent single-hole model in conjunction with the plane strain plastic-viscoelastic computer program. Results of computations* showed the effect of thermal cycling, under non-nuclear conditions, on the fuel element as a result of plastic strains caused by differential thermal expansion between the cladding material and the fueled matrix at elevated temperatures. This analysis was extended to include the effects of a differential temperature developing during the heating and cooling portions of the cycle for: (1) the non-nuclear cycle test† and (2) the nuclear case at design conditions in the Experimental Reactor during a full-flow scram.

Figure 1.6 compares the outer surface temperatures as a function of time (equivalent single-hole model) for: (1) the nuclear case at the maximum temperature point, and at the maximum power generation point, and (2) the non-nuclear cycle test when a fast exponential cooling from the high-temperature isothermal hold period is employed by shutting off the furnace power, and when the specimen is cooled at a linear rate by reducing the furnace power over a 1-hour period from 3900° to 500°F. Only that part of the time cycle when the test specimen is above 2500°F is included in the analysis, since essentially all of the plastic creep of the matrix should occur above this temperature. The resulting radial temperature distributions for three of the four cases are shown in Figure 1.7, where a radial temperature differential of 100°F is shown to exist for a short period of time at the

Fig. 1.6 - Comparison of temperature cycles in non-nuclear cyclic testing and in full-flow scram of the Experimental Reactor

*Ibid., pp. 13-16.

†"710 Reactor Program Progress Report No. 10," GE-NMPO, GEMP-310, September 30, 1964, p. 18.

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beginning of cooling in the non-nuclear, fast exponential cooling case (as opposed to a maximum radial temperature differential of only 20°F in the programmed cool cycle) and in the opposite direction to the differentials existing in the nuclear case since heat is extracted from the outer surface of the specimen during the non-nuclear cooling cycle as compared to the inner surface in the scram cycle.

Fig. 1.7 - Comparison of cycling test radial temperature differential with Experimental Reactor scram radial temperature differentials

The tangential plastic strains resulting from the combined effect of the radial temperature gradients and the differential thermal expansion between the matrix and the cladding are shown in Figure 1.8, where the exponential cooling test cycle is seen to result in a ratcheting mechanism causing radial expansion of the fuel element with repeated cycling. In the nuclear case at the maximum power point, the scram cycle results in an insignificant shrinkage ratchet mechanism. In the nuclear case at the maximum temperature point, the first cycle results in a slight shrinkage of dimensions; however, no significant ratcheting mechanism results from additional cycling. In addition, the analysis shows that by adjusting the cooling cycle in the non-nuclear test to a 1-hour power-reduction cooling from 3900°F to 500°F, the plastic strain at the single-hole model OD closely approximates that of the nuclear case at the maximum power extraction point for the fuel elements of the Experimental Reactor. Subsequent non-nuclear cycling tests of single-hole test specimens tends to substantiate this analysis; diametral expansion of the specimens when cooled over a 1-hour period was reduced by about 60 percent.

Figure 1.9 presents more detailed results of the analysis of the exponential cooling non-nuclear case, in which the temperature, ID-to-OD temperature differential, tangential stress at the OD, and the tangential strains at the ID and OD of the equivalent single-hole model are given as a function of time. This analysis utilized materials property data presently available. Additionally, external gas pressure forces were not taken into account

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Fig. 1.8—Comparison of tangential plastic strains resulting from non-nuclear cyclic testing with those resulting from a full-flow scram of the Experimental Reactor

in the stress calculation for the Experimental Reactor fuel element. The exponential cooling curve is simulated in Figure 1.9 and in the analysis by two linear approximations.

FUEL ELEMENT DEVELOPMENT CYCLIC TESTING

During this report period, 14 fuel element specimens were cycle-tested: nine tantalum-clad single-hole test specimens, two partial-length 91-hole fuel elements, two process development samples, and one advanced single-hole test specimen clad in W-Mo-Re. One other process development sample completed 507 hours at 3990°F with cycling only for inspection.

Table 1.9 presents the test results for single-hole specimens tested during this report period. Table 1.10 presents test results for all of the previously tested 91-hole fuel elements containing non-thoriated cores sintered at 3990°F, and includes fuel elements 7-C

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Fig. 1.9 - Stress analysis of single-hole test specimens during non-nuclear cycling test over the temperature range 3900° to 2500°F

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TABLE 1.9

CYCLIC TEST RESULTS OF SINGLE-HOLE TEST SPECIMENS^a

Specimen	Core Composition, wt %			Test Conditions				Inspection Results ^b	
	W	UO ₂	ThO ₂	Temperature Range, °F	Pressure, ^c psig	Cumulative		Leak Test ^d	Cumulative Radial Expansion, %
						Time, hr	Cycles		
SH-53 ^e	54	41.4	4.6	3900 - 500	0.3	30 ^f	30	No leaks	0.76
						60	60	No leaks	1.85
						100	100	No leaks	3.59
SH-55 ^e	54	46	0	3900 - 500	0.3	30 ^f	30	Leaks	1.64
						60	60	Leaks	3.36
						100	100	Leaks	5.65
SH-21	54	41.4	4.6	3900 - 500	0.3	45 ^g	30	No leaks	0.52
						90	60	No leaks	1.04
						150	100	One leak	1.79
SH-16	54	41.4	4.6	3900 - 500	0.3	90 ^h	30	No leaks	0.44
						180	60	No leaks	0.87
						300	100	No leaks	1.32
SH-17 ^k	54	41.4	4.6 ^h	3900 - 500	0.3	45 ^g	30	Leaks ^j	0.58
SH-18 ^k	54	41.4	4.6	3900 - 500	0.3	45 ^g	30	No leaks	0.92
						90	60	Leaks	1.72
SH-54 ^e	54	41.4	4.6	3900 - 500	0.3	45 ^g	30	No leaks	0.33
SH-56 ^e	54	46	0	3900 - 500	0.3	45 ^g	30	No leaks	0.30
SH-58 ^e	54	43.3	2.7	3900 - 500	0.3	45 ^g	30	No leaks	0.29

^aCore sintering temperature, 3990°F.^bBond checks made at each inspection; no evidence of bond deterioration.^cTest atmosphere was bottled helium, essentially static. Pressure shown is minimum helium pressure. Pressure increased on heating to 12 to 15 psig.^dLeak test by mass spectrometer. Leak locations determined by pressurizing in helium at 1000 psi and observing bubbles under methanol.^eSpecial sintering or cleaning procedures for these specimens are presented in Table 1.6.^fTest cycle: 20-minute heat, 5-minute hold at 3900°F, 20-minute power-off cooling to 500°F; total cycle time, 60 minutes.^gTest cycle: Same as "f" except for 60-minute programed cooling, total cycle time, 90 minutes.^hTest cycle: Same as "f" except for 150-minute programed cooling, total cycle, 180 minutes.^jOne core did not contain ThO₂; all of the cladding leaks observed in this specimen at the 30-cycle inspection were over this core.^kThese specimens were vacuum outgassed as an open assembly at 3000°F prior to final sealing of the cladding by EB welding.

and 7-D which were tested during this period. Table 1.11 is a complete compilation of long-time tests of engineering test specimens. Process-development capsule X-18, tested during this report period, is also included in Table 1.11.

Two major conclusions were drawn from the results of fuel element development testing during this period:

1. The original cooling cycle (rapid exponential cooling from 3900° to 500°F over a 20-minute period after furnace shutdown) was responsible for a significant amount of the dimensional expansion observed during the cyclic tests. The dimensional expansion appears to be due to a cycling ratchet mechanism caused by an unrealistic radial temperature distribution. The detailed analysis of this effect was presented in the previous section, "Fuel Element Design and Stress Analysis." Analysis indicates that an actual reactor scram from an established power level would not produce this ratcheting mechanism. Test results demonstrated that a slower power cooling; i.e., cooling from 3900° to

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TABLE 1.10

CYCLIC TEST RESULTS OF 91-HOLE PARTIAL-LENGTH FUEL ELEMENTS WITH W-UO₂ CORES SINTERED AT 3990°F

Specimen	Core Composition, wt %		Test Conditions				Inspection Results			
	W	UO ₂	Temperature Range, °F	Pressure, psig	Cumulative Time, hr	Cumulative Cycles	Outer Cladding - Matrix Bond	Leak Test ^a	Cumulative Radial Expansion, %	Weight Change, %
6-C	54	46	3900 - 800	0.3	6.0	10	Bonded	No leaks	0.25	1 x 10 ⁻³ loss
			3000 - 70	800 ^b	7.0	11	Bonded	No leaks	0.17	-
6-F	54	46	3900 - 800	0.3	18.0	30	Bonded	No leaks	1.2	-
			3900 - 800	0.3	36.0	60	Bonded	No leaks	4.1	1 x 10 ⁻² gain
			3000 - 70	10,000 ^c	37.0	61	Bonded	No leaks	1.3	-
7-A	54	46	3940 - 800	200	22.50	30	Bonded	No leaks	0.5	-
			3940 - 800	200	45.00	60	Bonded	One bore, two cladding leaks	1.1	2 x 10 ⁻³ gain
7-B	54	46	3900 - 70	0.3	0.75	1	Bonded	No leaks	0.1	-
			3900 - 500	0.3	7.50	10	Bonded	Two bore leaks ^d	0.3	3 x 10 ⁻⁴ loss
8-E	54	46	70 - 3900	0.3	2.00					
			3900 (Hold)	0.3	4.00					
			3900 - 70	0.3	6.00	1	Bonded	One tube leak ^e	0.0	1 x 10 ⁻² loss
7-C ^f	54	46	3900 - 500	0.3	90.0	30	Bonded	Leaks	-0.3	-
					180.0	60	Bonded	Leaks	-0.2	1 x 10 ⁻² gain
7-D ^g	54	46	3900 - 500	0.3	90.0	30	Bonded	No leaks	-0.8	-
					180.0	60	Bonded	Two bore leaks	-0.3	1 x 10 ⁻² gain

^aNo leaks determined by mass spectrometer. Leak locations determined by pressurizing with helium at 1000 psi and observing small bubbles under methanol.

^bCycled in hot-gas pressure-bonding furnace at 800 psi for 1 hour at 3000°F.

^cCycled in hot-gas pressure-bonding furnace at 10,000 psi for 1 hour at 3000°F.

^dElement given special hot-gas pressure-bonding cycle after original standard cycle and prior to isothermal cycling test.

^eForeign material melted from one tube (tube that leaked) during test.

^fElement vacuum outgassed in EB welder at red heat prior to final closure. Coolant tubes were fine-grain Ta - 10W.

^gCoolant tubes were fine-grain Ta - 10W.

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TABLE 1.11
LONG-TIME, STEADY-STATE TESTS OF ENGINEERING TEST SPECIMENS^a

Specimen	ThO ₂ , wt %	Test Conditions			Inspection Results	
		Temperature, °F	Cumulative Time, hr	Cycles	Leak Test	Radial Expansion, %
91-Hole Partial-Length Specimen						
6-H	0	3900	100	1	No leaks	<0.1
Single-Hole Specimens						
SH-38	0	3900	100	2	No leaks	<0.1
SH-35	4.6	3900	100	1	No leaks	0.5
SH-47	4.6	3900	100	1	No leaks	0.4
SH-46	4.6	3990	1027	7	Leaked at 141-hr, 4-cycle inspection	0.5
X-Type Capsules						
X-18	0	3990	507	2	No leaks	0.2

^aAll specimens tantalum-clad W - 46UO₂ or W - (41.4UO₂ + 4.6ThO₂), wt %.

500°F over a 1-hour period by reducing furnace power at a linear rate, accounted for a significant reduction in dimensional expansion. Stress analysis indicated that the tangential strain produced in the fuel element during a reactor scram and in the 1-hour cooling cycle in the non-nuclear cycling test should be in good agreement.

- The wet and dry hydrogen treatments of the fueled cores, which were intended to adjust the oxygen content of the fuel to UO₂.00, resulted in considerably less dimensional expansion in subsequent cycling tests than that observed from cores sintered only in dry hydrogen.

Table 1.12 is a complete summation of radial expansion data for the various single-hole test specimens, 91-hole partial-length fuel elements, and process-development capsules, all of which were tantalum-clad and cycled 30 times from 3900° to 500°F. Data are presented for specimens having fueled cores made with no post-sintering treatment (dry-hydrogen sintering only) and those made with post-sintering treatment (wet - dry hydrogen treatment to adjust the oxygen-to-uranium ratio after initial sintering). Included also are

TABLE 1.12
EVALUATION OF DIMENSIONAL STABILITY AFTER 30 CYCLES 3900° TO 500°F
FOR TANTALUM-CLAD 710 ENGINEERING TEST SPECIMENS

Sample Type	Expansion, %			
	No Post-Sinter Treatment		Post-Sinter Treated ^a	
	Test Cycle A ^b	Test Cycle B ^c	Test Cycle A ^b	Test Cycle B ^c
91-hole partial-length	1.16 (5) ^d	-	-	-
Single-hole test specimen	1.22 (7)	0.51 (3)	0.76 (1)	0.30 (3)
X-capsules	1.36 (5)	0.25 (1)	-	-
Average all types	1.24 (17)	0.44 (4)	0.76 (1)	0.30 (3)
Post-sinter treatment improvement, %		-	39	32
Test Cycle B improvement, %		64	-	61

^aWet - dry hydrogen treatment.

^bExponential cooling over a 20-minute period after furnace shutdown.

^cCooling over a 1-hour period by reducing furnace power at a linear rate.

^dNumber of tests shown in parenthesis.

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data on those tested with the exponential cooling cycle (test cycle A) and with the programmed cooling cycle (test cycle B). Although limited for the programmed cooling cycle and post-sintering treatment, the data indicate that an improvement of 30 to 40 percent in dimensional stability results from the post-sinter treatment and an improvement of about 60 percent results from the programmed cooling cycle.

In addition to the improvements noted in dimensional stability, it appears that the ability of the cladding to withstand the cycling effects without developing leaks has been enhanced both by the programmed cooling cycle and by the post-sinter treatment of the fuel. Table 1.13 is a summary of leak test results of specimens tested based upon the ability of the cladding to withstand 30 cycles from 3900° to 500°F without leaking. Again, the data are limited for the post-sintering treated cores and the programmed cooling cycle, but early test results show positive trends.

TABLE 1.13

EVALUATION OF LEAK OCCURRENCE IN TANTALUM-CLAD ENGINEERING TEST SPECIMENS
CYCLED FROM 3900° TO 500°F

Sample	No Post-Sinter Treatment				Post-Sinter Treated ^a			
	Test Cycle A ^b		Test Cycle B ^c		Test Cycle A ^b		Test Cycle B ^c	
	Leaks ^d	No Leaks ^e	Leaks ^d	No leaks ^e	Leaks ^d	No Leaks ^e	Leaks ^d	No leaks ^e
91-hole partial length	9 ^f	3	1	1	-	-	-	-
Single-hole test specimens	7	3	1	3	0	1	0	3
X-type capsules	4	3	-	-	-	-	-	-
All types	20	9	2	4	0	1	0	3
Percent of specimens in which no leaks occurred upon completion of 30 cycles		31	-	67	-	100	-	100

^aWet - dry hydrogen treatment.

^bExponential cooling over a 20-minute period after furnace shutdown.

^cCooling over a 1-hour period by reducing furnace power at a linear rate.

^dLeaks occurred before completion of 30 cycles.

^eNo leaks upon completion of 30 cycles.

^fColumn numbers refer to number of specimens tested.

The data in Table 1.9 reveal (1) that one single-hole specimen, SH-53, fabricated using the post-sintering treatment, was leak-tight after 100 thermal cycles even with the exponential cooling cycle, (2) specimen SH-16, fabricated by the former process (no post-sintering treatment), was leak-tight after 100 cycles with the programmed cooling cycle, and (3) specimen SH-21, fabricated by the former process, was leak-tight after 60 programmed cooling cycles and at 100 cycles had only one small cladding leak.

Single-hole specimens SH-54, SH-56, and SH-58 are being tested simultaneously in a single furnace. All had the wet - dry hydrogen treatment, and the cores contain 4.6, 0, and 2.7 weight percent ThO₂, respectively, which was blended with the fuel prior to agglomeration. The effect of the ThO₂ additive on stoichiometry adjustment is being evaluated.

As indicated by the inspection results (Table 1.9), radial expansion in these three specimens is almost identical. All three were leak-tight after 30 cycles. Long-time tests at 3900°F will be conducted on these specimens to further evaluate the ThO₂ additive.

Advanced single-hole test specimen ASH-3, clad with 40W - 30Mo - 30Re (at. %),* was cycled 100 times from 3900° to 500°F using the programmed cooling cycle. The element

*"High-Temperature Materials Program Progress Report No. 41, Part A," GE-NMPO, GEMP-41A, November 30, 1964, pp. 35-36.

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was leak-tight and remained bonded after 100 cycles; its radial expansion of 0.60 percent is the lowest of any specimen tested to date. The cores in this element had received the wet - dry hydrogen treatment.

GAS DYNAMIC TEST

Assembly of the low-pressure, inductively heated circulating gas loop for investigation of the effect of flowing gas on fuel element cladding was completed and checkout of the loop was begun. The maximum specimen temperature reached was 3000°F with a helium flow of 4 scfm. Preliminary tests indicate that O₂ and N₂ impurities in the as-received gas can be significantly reduced by the use of a CaH₂ getter in the loop.

Design modifications to convert the loop to a high-pressure (500 psi) assembly are nearing completion and fabrication of components has begun. The modified loop will have the capability for testing a single-hole (0.065-inch hydraulic diameter), 15-inch-long specimen at the mass flow conditions of the Experimental Reactor.

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2. CRITICAL EXPERIMENT OF FAST-SPECTRUM REFRACTORY-METAL REACTORS

On November 4, 1964, operation of the basic 710 critical experiment was completed. It was dismantled and the new experiment, Mockup No. 1A (the non-power-flattened version of the Mockup No. 1 critical experiment) was assembled. Core loading, which began on December 22, 1964, was approximately 25 percent completed by the end of the reporting period. The preanalysis of the Mockup No. 1A critical experiment was refined and analysis of the basic critical experiment was continued, primarily to investigate the effects of refinements in the core cross sections.

MOCKUP NO. 1A CRITICAL EXPERIMENT

The Mockup No. 1A critical experiment core* was inserted in the split table assembly at the Idaho Test Station. Some of the hexagonal aluminum cans which make up the split table assembly were removed to make room for the new core. Figure 2.1 shows the axially reflected half of the core installed in the split table assembly as viewed from the midplane. The radial reflector consists of beryllium bars inserted in the aluminum cans. The unfilled hexagonal cans near the core will contain the beryllium bars which provide shim and safety control. The pattern of shim and safety control bars was revised from that previously described,[†] and their worth is higher with this new pattern.

All of the reactor core materials have been received at the Idaho Test Station. Inspection of the fuel rods revealed the presence of chips on the surface of about 20 percent of the first 38.3 kilograms of rods received. The chips were removed by tumbling in a 7.62-cm diameter pipe. This process was followed by inspection and coating. This system, while time consuming, has proven to be adequate.

The materials are loaded in each of the hexagonal cells of the core as shown in Figure 2.2. Tantalum rods, 0.396 cm in diameter, are used to simulate the tantalum in the coolant tubes of the 710 fuel elements, rather than wire as originally planned.

Fuel loading was initiated on December 22, 1964. A total of 51.7 kilograms of fuel was loaded into the assembly by December 31, the end of the report period.

The critical loading of the Mockup No. 1A critical experiment was re-evaluated using a more consistent set of Oralloy cross sections[‡] as well as better data on the actual configuration of the critical experiment. The principal changes from the nuclear model previously described[§] are:

1. The tantalum end plate simulation (region 205) has been omitted for the initial experiments. The effect of these plates will be measured later in the program.

*"710 Reactor Program, Progress Report No. 11," GE-NMPO, GEMP-323, November 30, 1964, pp. 37-42.

[†]Ibid., p. 44.

[‡]"710 Reactor Program, Progress Report No. 5," GE-NMPO, GEMP-247, October 31, 1963, p. 44.

[§]"710 Reactor Program, Progress Report No. 11," GE NMPO, GEMP 323, November 30, 1964, pp. 40-44.

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Fig. 2.1-710 critical experiment Mockup No. 1A (Neg. IP 64-12-30)

2. The density of the aluminum shim pieces in the region between the pressure shell and the split table matrix (region 104) is approximately 95 percent of theoretical, not 41 percent as previously calculated.
3. Each core cell contains 3 Ta rods, 11 W rods, and 25 $W O_3$ rods to approximate the desired W, Ta, and O content.

The results of a Program ZIP, matched savings, diffusion calculation indicate that using 17 rods of Oralloy per cell (which yields a total loading of 189 kg) should provide about 1 percent excess reactivity, and that the critical mass should be 185.3 kilograms of Oralloy under these conditions.

A report on the test program and procedures for the Mockup No. 1A critical experiment was issued. It outlines a test series intended to provide data related to the critical loading and power distribution. Supplementary measurements will also be performed to determine control system worth, reactivity worth, fission rate, neutron lifetime, and poison segment worth.

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Fig. 2.2 - Mockup No. 1A cell loading

ANALYSIS OF THE BASIC CRITICAL EXPERIMENT

The method of calculating effective core multigroup cross sections was improved. The 16-group set of isotopic core cross sections* presently used was obtained by fission spectrum weighting the top 5 energy groups (above 0.1 Mev) and by $1/E$ flux weighting groups 6 thru 15 (0.1 Mev to thermal cutoff). A homogeneous core slowing-down calculation (Program C-Fine) in fine energy detail (4968 levels) shows that the fission spectrum weighting is satisfactory, but the $1/E$ flux weighting overpredicts the absorption cross sections in every group because the absorption cross section increases with decreasing energy and, in the resolved resonance region, also experiences resonance self-shielding. Therefore, a set of homogeneous core effective 16-group cross sections was obtained by applying to the standard 16-group isotopic core cross sections correction factors for each isotope in each energy group determined from the C-Fine calculation as the ratio of the flux-weighted cross section value to the $1/E$ weighted value.

The significant parameters from radial S-X transport calculations using the standard 16-group cross section set (which may be termed infinitely dilute) and the effective homogeneous core set are compared with experimental values in Tables 2.1 and 2.2. Although improvement was achieved in all parameters except fission eigenvalue, further improvement is needed, especially in the radial power profile. Since resonance self-shielding in the homogeneous core calculation underestimates the effect of spatial self-shielding, which becomes important near the core - reflector interface, the effective

*H. L. Bermanis, W. B. Henderson, and C. S. Roberts, "Nine and Sixteen Group Cross Sections for Reactor Analysis, Supplement No. 1," GE-NMPO, GEMP-173, to be published.

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TABLE 2.1
COMPARISON OF 16-GROUP S_4 RADIAL S-X CALCULATIONS USING EFFECTIVE CORE CROSS SECTIONS

Effective Core Cross Section Set	Fission Eigenvalue, (k_f)	Radial Power Profile			Prompt Neutron Generation Time, μ sec	$^{238}\text{U}/^{235}\text{U}$ Fission Rate Ratio At Core Center
		$P_{\text{max}}/P_{\text{min}}$	$P_{\text{center}}/P_{\text{min}}$	Distance Of Minimum From Core - Reflector Interface, cm		
Infinitely dilute ^a (standard 16-group set)	1.0114	4.0 ^b	1.22	2.5	3.50	0.0855
Homogeneous core	1.0170	4.5 ^b	1.19	2.9	3.98	0.0856
Heterogeneous core	1.0166	4.63	1.17	3.1	4.25	0.0856
(homogeneous core with 50% reduction of resolved resonance values)	1.0051 ^c	4.38 ^c	1.18 ^c	3.1 ^c	3.98 ^c	0.0862 ^c
Experimental	1.000	4.7	1.15	3.1	4.0	0.085

^aPreviously reported results of this case have been slightly in error because the beryllium density in the outermost 0.89 cm of the radial reflector was inadvertently set to zero. The only significant effect was to lower k_f by 0.003 and to lower prompt neutron generation time by 0.3 μ sec. Since the error was consistently present in all cases where comparisons have been made as to the effect of parameter changes, all previous conclusions are unchanged.

^bApproximate values scaled from $\nu\Sigma_f$ traverse to effective and infinitely dilute Σ_f traverses.

^c S_8 angular detail.

isotopic cross sections in those groups containing resolved resonances were arbitrarily reduced by a factor of 2 to observe the effect. The S-X parameters using these heterogeneous core cross sections are also shown in Tables 2.1 and 2.2 in both S_4 and S_8 angular detail.

The center-to-minimum power ratio is within the experimental measurement error (about $\pm 3\%$) using the heterogeneous core cross sections. It would be in even better agreement with the nominal value of the ratio if the 2-percent reduction* is applied which represents the difference between results of a power traverse on the average core radius and on the short core dimension along which the power traverses were taken.

The method of computing the peak-to-minimum power ratio was refined by making use of the "sensor activation traverses" which are an optional output from the S-X transport calculation. An Oralloy Σ_f activation traverse gives about a 3-percent larger peak-to-minimum power ratio than the $\nu\Sigma_f$ fission production traverse[†] which has been used in

TABLE 2.2
RATIO OF TOTAL ACTIVITY TO EPICADMIUM ACTIVITY (CdR) IN
THE RADIAL REFLECTOR 4.5 CENTIMETERS FROM THE
CORE - REFLECTOR INTERFACE

Effective Core Cross Section Set	Effective Cutoff Energy, ev	Oralloy CdR	Manganese CdR
Infinitely dilute ^a (Standard 16-group set)	0.414	4.61	2.34
Infinitely dilute ^a (Multi-thermal-group, 5 x 5 transfer matrix)	0.414	4.83	2.42
Homogeneous core	0.414	4.78	2.42
Heterogeneous core	0.414	5.04 4.98 ^b	2.50
Experimental	-	5.51	3.70

^aThese cases are missing the outermost 0.89 cm of radial reflector mentioned in footnote "a" of Table 2.1.

^b S_8 angular detail. All other results are in S_4 angular detail.

*"710 Reactor Program, Progress Report No. 10," GE-NMPO, GEMP-310, September 30, 1964, p. 34.

[†]H. L. Bermanis, W. B. Henderson, and C. S. Robertson, "710 Critical Experiment Correlation," GE-NMPO, GEMP-297, June 30, 1964, p. 110.

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previous calculations. The power traverse from core center through the power minimum is assumed to be given correctly by the heterogeneous Σ_f traverse, while the value at the interface, although not strictly correct, is taken from the infinitely dilute Σ_f traverse (some flux dip at resonant energies would be present even at the core surface).

The cadmium ratios of Oralloy in Table 2.2 can be used to deduce the value that would be obtained if the heterogeneous core cross section set had been used in the multi-thermal-group calculation. For a cutoff energy of 0.414 ev in S4 angular detail, the cadmium ratio would be about 5.26 and in S8 angular detail, about 5.20, compared to a measured value of 5.51. Until a more accurate effective cutoff energy for 20-mil cadmium is used in a multi-thermal-group heterogeneous core calculation, there is no reason to suspect that the reflector spectrum is appreciably in error, especially since a good correlation of neutron lifetime has already been obtained, as shown in Table 2.1. The poor correlation of the manganese cadmium ratio may be the result of cadmium resonance-shielding of the manganese resonance at 337 ev.

While the use of a single set of effective multigroup constants throughout the core in this correlation is a crude approximation, it may be adequate until the critical experiment, and the measurements therein, are refined to more closely represent the essentially homogeneous core of the Experimental Reactor. Complications arise with the present critical experiments due to the heterogeneous core structure which makes it very difficult to achieve better correlation of measured and calculated quantities than is now obtained with the heterogeneous set. The neutron spectrum in the reflector, even in the vicinity of the homogeneous cell, is dependent on the resonance absorption of the heterogeneous core. In the interface region, the catcher foil data are responses to an approximately homogeneous medium flux which is predominately dependent on the reflector spectrum and hence somewhat dependent on the heterogeneity of the core. The catcher foil data in the interior of the core are responses to the interstitial flux in the few microns of Oralloy from which fission fragments escape and these data may not be truly representative of either a homogeneous or a heterogeneous medium flux. The rod scanning data in the core interior are probably more nearly representative of the heterogeneous core power distribution. The planned use of homogeneous fuel compacts of the 710 Reactor Experiment type in a portion of the core of the Mockup No. 1B critical experiment should indicate the importance of the above effects.

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3. EXPERIMENTAL REACTOR PRELIMINARY DESIGN AND DEVELOPMENT

Revisions to the initial configuration of the Experimental Reactor reported previously* were completed. Evaluation of the resulting design indicates that, with minor modifications, all major structural components are within allowable stress limits at steady-state operating conditions. Thermodynamic and nuclear analysis is being continued on this model to provide a consistent preliminary design.

INITIAL CONFIGURATION

The Experimental Reactor assembly, as now defined, is shown in Figures 3.1 and 3.2. Modified slightly in the areas of core support structure and radial reflector, the arrangement conforms to the initial configuration characteristics, assembly procedures, and other data previously reported.[†] All mathematical models in use for nuclear, thermal, and stress analyses are based on this design.

Forward Support Structure

The modified forward support structure is shown in Figure 3.3. The built-up outer ring spans the gap between the core and pressure vessel and supports the reflector modules while accommodating differential radial expansions. The ring section is cooled from the inside by return coolant from the reflector modules. The coolant enters internal plenums from which it impinges over all surfaces of the irregular ring structure. This coolant method permits relatively uniform distribution of coolant while maintaining a high convective heat transfer coefficient. Resulting temperature variations around the structure are expected to be under 50°F with a pressure drop of about 0.2 psi.

Figures 3.4 and 3.5 show the calculated mechanical and thermal stresses, respectively, at major stress points in the assembly. The temperatures indicated, and therefore thermal stresses, are based on preliminary heating rate distribution and thermal analysis. Figure 3.6 shows the maximum combined stress intensities at these points and the allowable stress limits. A change in allowable stress for the primary plus secondary stress intensity from $3 S_m$ as previously used[‡] to $2 S_y$ reflects a recent opinion by the ASME Committee on Section III of the Vessel Code. For strain hardening materials (Austenitic steels) S_m may go to 90 percent of S_y instead of the former 67 percent. However, to prevent ratcheting under cyclic straining, the combined allowable intensity must still be limited to $2 S_y$. Two areas have been identified where the combined stresses exceed the allowable limits; however, these can be corrected with a minimum of change because of the low value of mechanical stresses at these areas. Although the structure is designed to withstand mechanical loading, it has sufficient flexibility to

*"710 Reactor Program, Progress Report No. 11," GE-NMPO, GEMP-323, November 30, 1964, pp. 45-48.

[†]Ibid., p. 48.

[‡]Ibid., p. 52.

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accommodate thermal loads. Resulting displacements and rotations indicated in Figures 3.7 and 3.8 are within acceptable limits as determined by their effects on other components.

Reflector Module

The reflector module initial configuration, shown in Figure 3.9, is a completely self-contained-unit assembly. Inside the 0.109-inch-thick Type 304 stainless steel shell are the stationary beryllium reflector and rotating control drum with boron carbide poison, bearings, and coolant distribution system. Extending from the forward face of the module is a tube which houses the control drum drive shaft and spring system for positioning the drum when not being driven by the actuator. The assembly is bolted on the outer ring of the forward support structure forming a gas seal.

Primary coolant gas enters through the forward tube extension, cooling the shaft, springs and forward bearing. It then flows through the drum and poison and over their external surfaces emerging at the aft collection plenum from where it passes through the rear bearing. The return coolant passage is through channels in the stationary beryllium and along the structural wall as well as a gap which forms the thermal barrier between the core and beryllium reflector. At the forward end of these channels, coolant is again collected and directed into the support structure. Coolant channels in both the stationary beryllium and structure are spaced to accommodate variations in heat distribution. In the rotating drum, a uniform cooling pattern is used. At design point conditions, thermal deflections are all accommodated by the clearances provided. Thermal radiation from the core is reduced by a niobium alloy barrier designed to limit the maximum structural temperature to approximately 800°F. Because of a pressure differential of approximately 42 psi across the module wall, the outer vessel and the inner structural cylinder are joined together to limit outward deflection of the vessel to less than 0.020 inch. The structural cylinder also provides support for the rear bearing.

Fuel Element

A full-scale model of the Experimental Reactor fuel element cartridge was fabricated to aid in establishing manufacturing and assembly procedures and techniques for the forward reflector and attachment hardware. The components of the assembly and the assembled fuel element are shown in Figure 3.10.

The fuel element section of the fuel cartridge is faced and threaded on the forward end, after hot-gas pressure-bonding, and joined to a molybdenum nosepiece, which also serves as a forward reflector. The nosepiece is threaded and has an internally machined plenum chamber to evenly distribute coolant gas to the fueled section. These two units are butt-welded together with a nonstructural gas-tight seal. The fuel element is held in the core support structure with a spring-loaded latch; the spring is pre-loaded to seal the fuel element to the back face of the support structure. The fuel element is inserted and removed by a special tool which withdraws the keeper permitting the fingers to deflect sufficiently to allow passage of the nosepiece through the support structure. Removal of the tool locks the fuel element in position.

THERMAL DESIGN AND ANALYSIS

Temperature flow and pressure drop calculations were made for the components of the Experimental Reactor. Although these calculations will be refined by the use of more accurate secondary heat calculations, the present data serve to define extra core component cooling requirements with particular emphasis placed on temperature discontinuities and gradients.

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Fig. 3.1 - Longitudinal arrangement of the Experimental Reactor (Dwg. 221R331)

Fig. 3.2 – Radial arrangement of the Experimental Reactor (Dwg. 221R331)

Radial Dimensions, inches	
A	8.255 Active core
B	8.824 Maximum core
	0.438 Clearance gap
C	9.262 Inside insulation
	0.125 Insulation
D	9.387 Inside module shell
E	13.474 Control module pitch circle
F	17.224 Outside module shell
	0.200 Gap
H	17.424 Inside mixer support
	0.094 Mixer support structure
	17.518 Outside mixer support
	0.162 Gap
J	17.680 Inside coolant passage shell
	0.300 Shell amplitude including thickness
K	17.980 Inside coolant passage
	0.210 Coolant passage minimum
L	18.190 Inside pressure shell
	0.830 Pressure shell thickness
M	19.020 Outside pressure shell

109 Fuel cells
12 Control drums

Fig. 3.3 - Forward support structure (Dwg. 221R326)

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Fig. 3.4 - Forward support structure mechanical stress distribution

Fig. 3.5 - Forward support structure thermal stress distribution

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Fig. 3.6 – Forward support structure combined stress intensities

Fig. 3.7 – Forward support structure displacements and rotations (mechanical case)

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Fig. 3.8 - Forward support structure displacements and rotations (thermal case)

Active Core

Single Channel - Temperature distributions for the analytical model of the single coolant channel described in the last report* were obtained. Figure 3.11 is a comparison of cladding surface temperatures with and without axial heat conduction. The curve labeled "with axial heat conduction" was obtained from the single coolant channel model for an average channel. At the front end of the core, axial heat conduction will have an effect to a fractional length (X/L) of about 0.075 and then will be negligible until X/L equals 0.65. At the aft end, a decrease of 20°F in surface temperature was calculated due to heat conduction away from the location of maximum surface temperature. All of the above calculations have assumed that no heat is lost from the forward and aft faces of the end caps.

Analysis Methods for Temperature Variations Within a Fuel Element - The use of analysis methods using linearized flow - temperature relations described previously† was validated. The maximum discrepancy with the General Flow Passage (GFP) Program‡ computations of the wall temperature was only 5°F.

Core Radial Temperature Flattening - Preparations for determining radial power tailoring criteria for radially temperature-flattened cores were continued by expanding the analysis methods utilized previously to predict radial temperature variations within single fuel elements.§ Although those methods can be used to simultaneously compute the temperatures in more than one fuel element, the boundary conditions are limited to one of equality of the heat flux into the inner surface of the innermost element and out of the

*"710 Reactor Program, Progress Report No. 11," GE-NMPO, GEMP-323, November 30, 1964, p. 55.

†Ibid., p. 57.

‡Ibid.

§"710 Reactor Program, Progress Report No. 10," GE-NMPO, GEMP-310, September 30, 1964, pp. 44-46.

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Fig. 3.9 - Initial configuration of the reflector module (Dwg. 221R332)

outer surface of the outermost fuel element. The expanded analysis is capable of simultaneously computing temperatures in all fuel elements in a radial row with heat transfer by radiation and conduction from the outermost fuel element to an outer cooled structure having specified outer surface temperatures.

Pressure Vessel

The coolant passage geometry* on the pressure head, between the actuator nozzles and the reactor centerline, caused a high pressure loss, over 2.0 psia, resulting from the flow turning and dumping near the reactor centerline. However, continuous bleeding off of the flow in this region, through the inner shell, will reduce the pressure loss while maintaining high heat transfer coefficients on the pressure head.

A preliminary analysis of the inner shell was completed. This determined preliminary specifications for gap spacing between the shell and the pressure head, the hole pattern

* "710 Reactor Program, Progress Report No. 11," GE-NMPO, GEMP-323, November 30, 1964, p. 54.

Fig. 3.10 - Model of Experimental Reactor fuel cartridge components
(Neg. P64-12-25B, P64-12-25A)

Fig. 3.11 - Effect of axial heat conduction upon longitudinal coolant channel temperatures

in the shell, and the geometry at the centerline of the reactor. The conduit formed by the pressure head and the inner shell was sized as a constant Mach number passage. A gap spacing of 0.375 inch was selected to provide heat transfer coefficients consistent with pressure head surface temperatures of about 700°F. Ninety percent of the total reactor flow enters the head coolant passage peripherally at the actuator circle; approximately 8 percent of the total reactor flow then leaves this coolant passage at the centerline of the reactor. The difference in these flows is uniformly bled off through approximately 1400 holes uniformly spaced over the shell. The diameter of these holes increases from 0.113 inch near the actuators to a maximum of 0.246 inch near the reactor centerline. Eleven different hole sizes are used to achieve a nearly uniform distribution on the pressure vessel surface. The 8 percent of the total reactor flow which finally exits at the reactor centerline is turned through 90 degrees and diffused slightly over a distance of approximately 1.4 inches. Then it is finally dumped into the plenum upstream of the reflector modules. The estimated total pressure loss for all the flow passing through this system is 0.7 psia.*†

Bolt and Flange - Thermal stress calculations indicate that the mean temperatures of the flange and the bolt shank should be held within about 50°F of each other. The steady-state temperature distributions in these components were evaluated with nominal heating rates, without separate bolt cooling. The results of this analysis indicated mean temperatures of 766°F in the bolt shank and 730°F in the flange. The resultant 36°F temperature difference is within the maximum allowable of 50°F for the case evaluated.

* A. L. Loeffler, Jr., and M. Perlmuter, "Turbulent Flow Through Porous Resistances Slightly Inclined to the Flow Direction," NACA, TN 4221, February 1958.

† P. A. Kolodzie, Jr., and M. Van Winkel, "Discharge Coefficients Through Perforated Plates," *AIChE Journal*, Vol. 3, No. 3, September 1957.

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Core Support Structure

The method presently being considered for cooling the core support structure near the reflector module is multiple jet impingement.* The gas stream is broken up into jets by a porous plate located inside the structure. Design data are given below:

1. Orifice size - 0.094 in.
2. Porosity of orifice plate - 2.52 orifices/in.²
3. Orifice pitch - 0.63 in.
4. Structure thickness - 0.25 in.
5. Orifice plate pressure drop - 0.18 psi

These data are based on a temperature drop from structure to coolant of 30°F which tends to maintain a radially constant temperature in the core support structure.*

Stationary Side Reflector

The coolant flow on an upward pass provides cooling to the stationary portion of the reflector module. The flow splits into three general branches: (1) 51 percent to the inner channel adjacent to the core, (2) 10 percent to the beryllium reflector material, and (3) 39 percent to the structure (see Figure 3.9).

The beryllium coolant channel diameters are the same as in previous module calculations (0.207 in.) and the inner channel and structural channels were sized to produce the same pressure drop with the chosen flow split.

The coolant channels in the structure were located by allowing a maximum temperature gradient in the structure of 50°F when the upper heating rates were used.† Figures 3.12, 3.13, and 3.14 show the estimated temperature distribution of the channel surfaces and of the coolant fluid. In Figure 3.13 the distribution shown for the structure is for an average channel, i. e., average heat addition.

Fig. 3.12 - Module inner coolant channel temperature distribution of a 4.4375- by 0.1331-inch channel

*G. C. Huang, "Investigations of Heat Transfer Coefficient for Air Flow Through Round Jets Impinging Normal to a Heat Transfer Surface, *Journal of Heat Transfer*, August 1963.

†"710 Reactor Program, Progress Report No. 7," GE-NMPO, GEMP-282, March 31, 1964, p. 45.

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Fig. 3.13 - Structure and coolant channel temperature distribution

Fig. 3.14 - Beryllium coolant channel temperature distribution

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Reflector Drum

The reflector drum design is unchanged from that previously reported. It was found that the pressure drop was excessive when the total coolant flow was passed in series through the drum and stationary reflector. To reduce the pressure drop, 22 percent of the flow was bled through the annular space between the drum and stationary reflector. Figure 3.15 shows the channel surface and coolant temperature distribution through average channels in the drum and in the poison as calculated by the THT-C Program.* Figure 3.16 shows the temperature distribution in the control drum at an axial location near maximum heat generation ($X/L \approx 0.35$). These results are based on the upper heating rates and lie within the acceptable range.

Fig. 3.15 – Drum and poison average coolant channel wall and fluid temperature distribution with a 22 percent bleed flow

NUCLEAR ANALYSIS AND DESIGN

Nuclear Heating Analysis

The detailed nuclear heating calculation mentioned previously[†] was continued during the reporting period. The nuclear model was modified to reflect the most recent me-

*"710 Reactor Program, Progress Report No. 4," GE-NMPO, GEMP-226, July 31, 1963, p. 41.

[†]"710 Reactor Program, Progress Report No. 11," GE-NMPO, GEMP-323, November 30, 1964, p. 57.

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Fig. 3.16 – THT-C-derived temperature distribution in control drum at an axial location near maximum heat generation (upper heating rates)

chanical design. The dimensions of the model were changed to simulate assumed operating temperatures. This analysis is nearly complete and will provide the gamma source distribution for the gamma Monte Carlo (FMC-G) Program and the neutron source distribution for the neutron Monte Carlo (FMC-N) Program. Preliminary results verify the reactor multiplication factor calculated by simpler methods.

Gamma ray production cross sections for 15 neutron lethargy groups (750°F Maxwellian) and 11 photon energy groups were prepared for use in two-dimensional transport theory (TDC) calculations of gamma ray source distributions. Cross sections had to be prepared for niobium, titanium, and manganese. These cross sections account for the production of gamma rays by radiative neutron capture, neutron inelastic scattering, neutron-induced fissioning, and (n, α) reactions.

Nuclear models were prepared for use in Monte Carlo analyses to determine (1) neutron energy deposition, fluxes, and leakage and (2) gamma ray energy deposition and leakage. These

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models include the top head of the pressure vessel. The neutron cross section data for the Monte Carlo analysis were also completed. The form of these data is basically different from that of the cross section data required by the reactor analysis codes; the Monte Carlo code requires four types of neutron data: (1) the total neutron cross section for each material, (2) the partial cross sections for up to six nuclear events for each type of nucleus, (3) the angular distribution following an elastic scattering for each type of nucleus, and (4) the energy of neutrons inelastically scattered from each type of nucleus.

These data were processed in a 23-energy-group format. This format was chosen in a manner that allows fluxes calculated in the 23 group structure to be condensed by summation of groups to the 15-energy-group format used in reactor analysis.

Control Drum Worth

In a previous report,* preliminary calculations of control drum worth indicated that about 6 percent $\Delta k/k$ could be obtained with highly enriched boron-10 in the control drums; this value is close to the 5 percent $\Delta k/k$ needed. A more rigorous analysis using a two-dimensional transport calculation of the drum worth in the Experimental Reactor was started. The methods used will closely parallel those used to correlate the poison segment measurements made in the 710 basic critical experiment.† This calculation should give a more reliable indication of the expected worth of the Experimental Reactor control system. To supplement this single-point calculation, a parametric study is also being made.

Control Drum Parametric Calculations - For a given core composition and size and a given reflector material, the worth of rotating reflector control drums is influenced by a number of parameters of design-limited flexibility:

1. Thickness and density of the reflector
2. The absorbing material
3. Thickness and poison content of the drum absorber segment
4. Diameter of the control drums
5. Distance of control drums from the active core
6. Thickness and nuclear properties of materials located between the core and the drums

Some of these parameters, e. g., the relative effectiveness of various absorber materials and the influence of boron-10 enrichment in the boron carbide absorber segments upon the drum worth, were previously investigated and reported.‡

To determine, for the power-tailored 710 Experimental Reactor, the dependence of the worth of 12 control drums upon the thickness of the beryllium reflector separating the drums from the core shell, diffusion theory calculations were made using the same nuclear model of the reactor and the calculational method as in the previous analyses.§ The drum absorber segments were assumed to be 70-percent enriched boron carbide.

The first results of the calculations, for up to 0.79 inch beryllium thickness separating the drums from the core shell, are given in Table 3.1. These results indicate that the worth of the drums increases slightly as the beryllium thickness decreases. Its maximum is obtained when the drums are adjacent to the core shell. However, the cell correction factors calculated for the core shell - drum separation of 0.50 inch were used for

*"710 Reactor Program, Progress Report No. 9," GE-NMPO, GEMP-301, July 31, 1964, pp. 59-61.

†"710 Reactor Program, Progress Report No. 11," GE-NMPO, GEMP-323, November 30, 1964, pp. 35-37.

‡"710 Reactor Program, Progress Report No. 7," GE-NMPO, GEMP-282, March 31, 1964, pp. 36-38.

§Ibid., p. 59.

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TABLE 3.1
DRUM WORTH VERSUS BERYLLIUM
THICKNESS SEPARATING THE DRUMS
FROM THE CORE SHELL

Beryllium Thickness, in.	Drum Worth, - Δk
0.0	0.0592
0.25	0.0558
0.50	0.0526
0.79	0.0488

the other three beryllium thicknesses. This may be a contributing factor to the calculated drum worth increase.

Studies are continuing to evaluate the control drum worth dependence on the number of drums, the drum diameter, and the radial reflector thickness.

REACTOR CONTROLS

Test Loop Simulation

The overall test loop simulation was modified to incorporate the present core design configuration, to permit off-design evaluation of the system, and to permit the analysis of temperature coefficients of reactivity.

The original simulation was based upon the 91-hole fuel element and pure neon as the coolant. The present simulation of the reactor heat transfer characteristics was modified to the 37-hole fuel element and a Ne - 10He coolant.

The circulator simulation is now based on standard atmospheric conditions and then corrected for loop pressure and temperature conditions. The average loop pressure is calculated as a function of the average loop temperature. The simulated system can now be operated at off-design conditions with a fair degree of accuracy. However, some linearization is required to get the system simulated on the available analog equipment.

A heat transfer stage was added to the reactor heat transfer simulator to simulate the side and front reflectors. The reflector stage will be used to evaluate the effects of temperature on reactivity since the reflector will operate at a different temperature and will have a different temperature coefficient of reactivity and time constant than the core.

The modified simulation was programed on the computer, checked out and is now available for evaluation of the Experimental Reactor. The present simulation will be used to evaluate the effects of various component failures and malfunctions in the loop, the effects of temperature coefficients of reactivity, and the effect of different circulator characteristics on the overall loop performance.

The revised loop simulation was used to generate the data shown in Figure 3.17. The weight flow through the reactor core was step-reduced to 80 percent of design with reactor power maintained constant. When the core discharge temperature had increased by 50°F (to 3550°F), the reactor was scrammed with -3 percent Δk . The results are shown for core discharge temperature, circulator inlet pressure, pressure drop across the reactor section, pressure drop across the heat exchanger section, weight flow in the reactor loop, and weight flow through the bypass circulators.

The core temperature, prior to reactor scram, increased because of the decrease in weight flow. After scram, the reactor discharge temperature decreased as the reactor power decayed and as the stored heat was removed from the core. As the core discharge

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Fig. 3.17 - Response to a -3 percent Δk scram due to an overtemperature in reactor discharge which was initiated by a decrease in weight flow (not corrected for recorder nonlinearities)

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temperature decreased, the average loop temperature also decreased which then resulted in a reduction in the loop pressure. The reduction of loop pressure results in a decrease in weight flow through the circulators under the assumption that the rotor speed is held constant. This drop in weight flow also permits a decrease in the pressure drop as seen by the circulators, and thus the flow reaches equilibrium at some intermediate value.

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4. TEST FACILITIES AND PILOT LOOP DESIGN AND EVALUATION

Studies were continued on the requirements for the facilities and test equipment needed for power testing at the Initial Engineering Test facility at NRTS. Advanced engineering on the test equipment is in process and facilities design criteria are being assembled.

FACILITIES

Work continued on the layout of the test facility to reflect the latest space requirements and to provide a functional arrangement. Two possible layouts, Figures 4.1 and 4.2, for arranging the equipment rooms in relation to the test cell were evolved during this report period. These layouts, as well as the conceptual layouts shown in a previous report,* will be used to develop an optimum piping and equipment arrangement and to determine which plan results in the lowest overall project cost. For the two plans shown, the shield wall between the test cell and the warm equipment area has been located near the front of the coupling station to shorten the piping runs and to simplify the coupling station. With this arrangement, the existing periscopes cannot be used and visual monitoring of the test cell would be accomplished by using a portable TV camera mounted on the overhead bridge crane for plan No. 1; a shielded window would be used for plan No. 2. The existing periscope tunnels are used for routing instrument lines from the coupling station to the control room. Plan No. 1 results in shorter piping runs from the test cell to the cold equipment area and an easier access from the control room to the cold equipment area. Plan No. 2 allows easier placement of equipment in the cold equipment area and requires less excavation.

A study to determine the feasibility of supplying electrical power to the test dolly by electrifying the rail system rather than by mounting diesel generator sets on the dolly was completed. The study showed that an electrical supply system would be more complex and expensive than dolly mounted diesel generators and is therefore not warranted.

Stress analyses were completed on the primary-loop piping in the warm equipment area. Figure 4.3 shows the piping layout; the required flexibility is provided by pipe loops rather than expansion joints. This layout has a system volume of 250 cubic feet and a surface area, in contact with the neon, of 1500 square feet. These volumes and areas do not include the dolly mounted piping and equipment.

Fission Product Effects

In support of the safety analysis, studies to predict the magnitude of plateout of fission products in the primary test loop were initiated. Because of the differences in release rates of different fission products, it is not certain which fission products will be of principal concern. However, recent in-pile tests in the ETR and at ORNL using Ta-clad W-UO₂ fuel

*"Conceptual Studies and Preliminary Design Criteria for 710 Reactor Program Test Equipment and Facilities,"
GE-NMPO, GEMP-291, June 1, 1964.

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Fig. 4.1 - 710 facility floor plan No. 1 (Dwg. 692E370)

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Fig. 4.2- 710 facility floor plan No. 2 (Dwg. 692E371)

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Notes:

1. This drawing was made to show the approximate configuration in the warm equipment room if expansion joints are not used.
2. Insulation, anchors, supports and guides were omitted for clarity.

Fig. 4.3 - Piping layout of the primary loop in the warm equipment room (Dwg. 848D481)

samples indicate that the principal fission products released (excluding releases from cladding failures) will be the noble gases. If pinhole leaks in the fuel cladding develop during periods of sustained testing, particulate fission products could be released. However, a large percentage of these particulate fission products would be trapped in the fission product filter.

Very little is known about the deposition behavior of most fission products under the turbulent flow conditions which will exist in the primary test loop (Reynolds' Number about 10^6). While some results correlating deposition as a function of temperature and materials have been reported in the literature, these data are principally on once-through, swept-gas systems and little work has been reported on recirculatory systems with turbulent flow. Some insight into the problems can be gained from data taken during out-of-pile experiments which simulate gas-cooled reactor flow and temperature conditions. Such a series of experiments was conducted at Battelle Memorial Institute.* Various theoretical

* A. Abriss, BMI, Private Communication.

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models of fission-product deposition from gas-cooled reactor systems have been explored as an aid in correlating and understanding the experimental data from model loop experiments. The approach which appears to be the most appropriate of the various models studied is described by Raines.* This model is an exact analogue of the transient cooling of a cylinder with a corrective coefficient at the surface.

The model described by Raines was used to give a preliminary estimate of the deposition of the daughter products of the noble gases. A release rate (defined as the amount released per unit time as a fraction of the amount formed per unit time) of 10^{-7} [†] was used to calculate the equilibrium xenon and krypton concentration in the primary loop. Preliminary calculations of the dose rate resulting from daughter product deposition indicate less than 5 mr/hr in the primary circuit with the deposited activity located at the entrance to the downstream piping from the heat exchanger and at bends in the piping. The calculated equilibrium noble gas activity in the loop was 0.528 curies/cm³.

Effects of Impurities

An analytical study of the nitrogen uptake by the tantalum tubing in the 710 Reactor Experiment was completed. Three aspects were considered: (1) the capacity of tantalum for nitrogen as a function of X/L for various partial pressures, (2) the rate of uptake as a function of X/L, and (3) the sources of nitrogen in the loop. The calculations were based on equilibrium data and reaction rate data obtained from the literature.^{‡§} The principal mechanism controlling the nitrogen uptake was the diffusion of nitrogen through a stagnant film present because of turbulent flow. For a continuous background pressure of 10^{-6} atm, the nitrogen reached equilibrium saturation at $X/L > 0.5$ after 30 hours. In the low-temperature region, the nitrogen accumulated primarily in a Ta₂N layer. Maximum nitrogen uptake occurred at $X/L = 0.2$. No consideration was given to possible cross interference of oxides in calculating the nitrogen uptake. To provide a continuous background pressure of 10^{-6} atm, it was found that the nitrogen present as a contaminate (at an assumed level of 1 ppm) in the neon supply was insufficient. Virtual and real leaks are the major potential sources. If an average contamination level of 2 atomic percent nitrogen is considered tolerable, then a leak rate of 0.1 cm³/sec at NTP is tolerable in the loop for 100 hours under the operating conditions considered in this study.

A more detailed calculation of the nitrogen uptake in the Reactor Experiment will be performed for the design condition and several off-design operating points.

TEST EQUIPMENT

Advanced engineering work on the power test equipment was continued throughout the reporting period.

Instrumentation

An instrumentation schematic was prepared to depict the tentative location and quantity of measured parameters desired on the test loop. This schematic, Figure 4.4, incorporates both the demineralized water and treated water loops, but does not include all the auxiliary systems.

A summary was made of the transducers needed for the various temperature, pressure, and flow measurements (listed in Tables 4.1, 4.2, and 4.3, respectively). For the listed temperature measurements, a standard commercial-mineral-insulated thermocouple is

*G. E. Raines, "Mathematical Models of Fission-Product Deposition and the Correlation of Experimental Data," BMI, BMI-X-278, March 17, 1964.

[†]"High-Temperature Materials Program Progress Report No. 39, Part B," GE-NMPO, GEMP-39B, September 30, 1964.

[‡]E. Gebhardt, H. Seghezzi, E. Fromm, *Z. Metallkunde*, Vol. 52, 1961, p. 464.

[§]K. Östhaugen and P. Kofstad, *Journal of Less Common Metals*, Vol. 5, 1963, pp. 7-25.

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TABLE 4.1

710 REACTOR EXPERIMENT TEMPERATURE MEASUREMENTS (EXCLUDING REACTOR)

Name	Symbol	Temperature Range, °F	Media	Pipe Size, in.	Location
1. Main circulator No. 1 inlet	TE-1	0 - 500	Gas	6	Facility
2. Main circulator No. 1, outlet	TE-2	0 - 500	Gas	6	Facility
3. Main circulator No. 2, inlet	TE-3	0 - 500	Gas	6	Facility
4. Main circulator No. 2, outlet	TE-4	0 - 500	Gas	6	Facility
5. Main loop preheater, inlet	TE-5	0 - 500	Gas	6	Facility
6. Main loop preheater, outlet	TE-6	0 - 700	Gas	6	Facility
7. Main loop reactor, inlet	TE-7	0 - 700	Gas	6	Dolly
8. Main loop heat exchanger (primary), inlet	TE-8	0 - 2000	Gas	Chamber	Dolly
9. Main loop heat exchanger (primary), outlet	TE-9	0 - 500	Gas		Dolly
10. Main loop heat exchanger (secondary), inlet	TE-10	0 - 250	Water	-	Dolly
11. Main loop heat exchanger (secondary), outlet	TE-11	0 - 450	Water	-	Dolly
12. Main loop heat exchanger temperature control	TE-27	0 - 500	Gas	10	Dolly
13. Bypass circulator No. 1, inlet	TE-12	0 - 500	Gas	6	Facility
14. Bypass circulator No. 1, outlet	TE-13	0 - 500	Gas	6	Facility
15. Bypass circulator No. 2, inlet	TE-14	0 - 500	Gas	6	Facility
16. Bypass circulator No. 2, outlet	TE-15	0 - 500	Gas	6	Facility
17. Test leg flowmeter, inlet	TE-16	0 - 700	Gas	8	Facility
18. Bypass circulator heat exchanger (primary), inlet	TE-17	0 - 500	Gas	6	Facility
19. Bypass circulator heat exchanger (primary), outlet	TE-18	0 - 500	Gas	6	Facility
20. Bypass circulator heat exchanger (secondary), inlet	TE-19	0 - 250	Water	-	Facility
21. Bypass circulator heat exchanger (secondary), outlet	TE-20	0 - 450	Water	-	Facility
22. Test leg heat exchanger (primary), inlet	TE-21	0 - 700	Gas	6	Facility
23. Test leg heat exchanger (primary), outlet	TE-22	0 - 500	Gas	6	Facility
24. Test leg heat exchanger (secondary), inlet	TE-23	0 - 250	Water	-	Facility
25. Test leg heat exchanger (secondary), outlet	TE-24	0 - 450	Water	-	Facility
26. Demineralizer water heat exchanger (primary), inlet	TE-25	0 - 250	Water	-	Facility
27. Demineralizer water heat exchanger (primary), outlet	TE-26	0 - 450	Water	-	Facility
28. Demineralizer water heat exchanger (secondary), inlet	TE-28	0 - 250	Water	-	Facility
29. Demineralizer water heat exchanger (secondary), outlet	TE-29	0 - 450	Water	-	Facility
30. Dolly air compressor No. 1, inlet	TE-31	(-50) - 150	Air	-	Dolly
31. Dolly air compressor No. 2, inlet	TE-32	(-50) - 150	Air	-	Facility
32. Facility air compressor, inlet	TE-33	(-50) - 150	Air	-	Facility
33. Demineralizer water heat exchanger control	TE-30	0 - 450	Water	-	Facility
34. Dolly shield water, inlet	TE-34	0 - 450	Water	-	Dolly
35. Dolly shield water, outlet	TE-35	0 - 450	Water	-	Dolly
36. Dolly bypass flowmeter, inlet	TE-36	0 - 500	Gas	8	Dolly
37. Primary loop heat exchanger (primary) flow, inlet	TE-37	0 - 450	Water	-	Facility
38. Dolly shield-water flow, inlet	TE-38	0 - 450	Water	-	Facility
39. Test leg heat exchanger (primary) flow, inlet	TE-39	0 - 450	Water	-	Facility
40. Bypass circulator heat exchanger (primary) flow, inlet	TE-40	0 - 450	Water	-	Facility

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Fig. 4.4 - Schematic drawing of instrumentation for reactor test equipment (Dwg. 692E369)

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TABLE 4. 2
710 REACTOR EXPERIMENT PRESSURE MEASUREMENTS

Name	Symbol	Pressure Range, psi	Media	Location
1. Primary circulator No. 1, inlet	PE-1	100 - 1000	Gas	Facility
2. Primary circulator No. 1, ΔP	PdE-1	0 - 100	Gas	Facility
3. Primary circulator No. 2, inlet	PE-2	100 - 1000	Gas	Facility
4. Primary circulator No. 2, ΔP	PdE-2	0 - 100	Gas	Facility
5. Primary loop reactor flow, inlet	PE-3	100 - 1000	Gas	Dolly
6. Primary loop reactor flow, ΔP	PdE-3	0 - 20	Gas	Dolly
7. Primary loop heat exchanger (primary), inlet	PE-4	100 - 1000	Gas	Dolly
8. Primary loop heat exchanger (primary), outlet	PE-5	100 - 1000	Gas	Dolly
9. Fission product filter, ΔP	PdE-4	0 - 20	Gas	Dolly
10. Primary loop heat exchanger, helium (monitor gas)	PE-6	0 - 600	Gas	Facility
11. Primary loop micron filter, ΔP	PdE-5	0 - 20	Gas	Facility
12. Bypass circulator No. 1, ΔP	PdE-6	0 - 100	Gas	Facility
13. Bypass circulator No. 1, inlet	PE-7	100 - 1000	Gas	Facility
14. Bypass circulator No. 2, ΔP	PdE-7	0 - 100	Gas	Facility
15. Bypass circulator No. 2, inlet	PE-8	100 - 1000	Gas	Facility
16. Bypass loop flowmeter, inlet	PE-9	100 - 1000	Gas	Facility
17. Bypass loop flowmeter, ΔP	PdE-8	0 - 20	Gas	Facility
18. Bypass loop micron filter No. 1, ΔP	PdE-9	0 - 20	Gas	Facility
19. Bypass loop micron filter No. 2, ΔP	PdE-10	0 - 20	Gas	Facility
20. Dolly air compressor filter, ΔP	PdE-11	0 - 20	Air	Dolly
21. Dolly air compressor No. 1, ΔP	PdE-12	0 - 100	Air	Dolly
22. Dolly air compressor No. 2, ΔP	PdE-13	0 - 100	Air	Dolly
23. Facility air dust filter, ΔP	PdE-14	0 - 20	Air	Facility
24. Facility air compressor, ΔP	PdE-15	0 - 2000	Air	Facility
25. Dolly bypass flowmeter, ΔP	PdE-16	0 - 20	Gas	Dolly
26. Dolly bypass flowmeter, inlet	PE-10	100 - 1000	Gas	Dolly
27. Dolly shield water flowmeter, inlet	PE-11	0 - 250	Water	Facility
28. Dolly shield water flowmeter, ΔP	PdE-17	0 - 20	Water	Facility
29. Primary loop heat exchanger (secondary) flow, inlet	PE-12	0 - 250	Water	Facility
30. Primary loop heat exchanger (secondary) flow, ΔP	PdE-18	0 - 20	Water	Facility
31. Test leg heat exchanger (secondary) flow, inlet	PE-13	0 - 250	Water	Facility
32. Test leg heat exchanger (secondary) flow, ΔP	PdE-19	0 - 20	Water	Facility
33. Bypass circulator heat exchanger (secondary) flow, inlet	PE-14	0 - 250	Water	Facility
34. Bypass circulator heat exchanger (secondary) flow, ΔP	PdE-20	0 - 20	Water	Facility

TABLE 4. 3
710 REACTOR EXPERIMENT FLOW MEASUREMENTS

Name	Symbol	Flow Range	Media	Location
Primary loop flow (reactor-inlet), lb/sec	FE-1	1.5 - 15	Gas	Dolly
Test leg loop flow, lb/sec	FE-2	3 - 30	Gas	Facility
Bypass circulator heat exchanger (secondary) water, gpm	FE-3	1 - 10	Water	Facility
Primary loop heat exchanger (secondary) water, gpm	FE-4	50 - 500	Water	Facility
Test leg heat exchanger (secondary) water, gpm	FE-5	2 - 20	Water	Facility
Demineralizer water heat exchanger (secondary) water, gpm	FE-6	250 - 2500	Water	Facility
Dolly bypass loop flow, lb/sec	FE-7	3 - 30	Gas	Dolly
Dolly shield water flow	FE-8	Unknown	Water	Facility

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being considered for use with an ungrounded junction. The sheath material of this unit is expected to be compatible with the environmental conditions and would be mounted into the piping by welding a seal fitting. For the ranges shown, the accuracy of the thermocouples should be ± 3 percent. The pressure transducers and flow primary elements are still being evaluated.

Dolly Piping

A stress analysis of the dolly piping was completed. Figure 4.5 shows the equipment arrangement and piping layout. This layout also uses pipe loops in place of expansion joints to provide the required flexibility. The system, exclusive of the reactor, has a volume of 250 cubic feet and a surface area (in contact with the neon) of 2500 square feet.

There are eight individual inlets - four primary inlet lines and four bypass coolant inlet lines - sized and located to equalize pressure at all inlet ports.

To facilitate shield removal, the shield is penetrated by the neon lines at one level only.

Valves

The requirements and expected reliability for valves in the primary test loop are being determined. These valves will require bellows seals to eliminate, or at least minimize, leakage from the primary loop and back-diffusion of contaminants. Valve specifications from users of commercial bellows valves are being evaluated for application to the 710 test loop environment. Documents containing experimental data regarding leakage rates and reliability of bellows valves were ordered.

Dolly Assembly

Work was initiated on the configuration of the dolly shielding and on the development of a detailed assembly procedure for the dolly mounted piping and equipment. The assembly procedure will be used at this time to establish delivery requirements for the dolly mounted components.

Circulator Control

A nondimensional calculation of circulator inlet temperature following a reactor scram was made to obtain an estimate of the maximum rate of temperature change that could be expected. The assumptions made to simplify the mathematical model of the system were (1) incompressible coolant gas, (2) constant pressures and heat transfer coefficients, and (3) core latent heat introduced the only time lag.

The Untermeyer-Weills approximation was used to describe the afterheat decay following a 100-hour operation at 10 megawatts. The loop parameters that were assumed to be constant are presented in Table 4.4.

The results of the calculation are presented in Figure 4.6. Although the simplifying assumptions made for calculational purposes greatly reduce the approximation of the physical process, the results are indicative of the magnitude of the environmental change that could be experienced by the circulators. The temperature transient shown in the figure is considered to be worse than that which would be expected if the heat exchanger latent heat transfer and flow characteristics were included in the calculation.

CONTROL AND DATA SYSTEM

The control, presentation, and monitoring functions of the digital computer system were defined.

Reactor System Control

Computer programs will provide the forward and feedback controlling system elements for the reactor coolant loop components. Brief descriptions of the anticipated control loops are presented below. Specific transfer functions and interaction restraints for these loops will be established later.

Fig. 4.5 - Equipment and neon piping layout of the 710 dolly (Dwg. 849D419)

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TABLE 4.4

ASSUMED CONSTANTS USED TO CALCULATE
AFTERHEAT DECAY FOLLOWING A 100-HOUR OPERATION
AT 10 Mw OF 710 EXPERIMENTAL REACTOR

Core inlet gas temperature, °F	500
Bypass circulator temperature rise, °F	20
Heat exchanger water temperature, °F	200
Primary loop flow rate, lb/sec	12.8
Bypass loop flow rate, lb/sec	28.2
Core heat capacity, Btu/°F	62
<u>Initial Conditions Assumed</u>	
Core exit gas temperature, °F	3500
Core surface temperature, °F	3800
Heat exchanger inlet gas temperature, °F	1460
Heat exchanger exit gas temperature, °F	440

Fig. 4.6 - Circulator inlet temperature following reactor scram

Primary Circulators - The indirectly controlled parameter in the primary circulators will be reactor coolant mass flow rate; the directly controlled parameter will be circulator speed. Speed control for the circulators will be effected by controlling the frequency of the input voltage to the compressor drive motors. The frequency change will be produced by varying the coupling between a generator and its driver.

The demand to the controlling system will be a mass flow rate. This demand will be a digital value entered manually from the circulator operator's console. Since a single circulator speed does not produce a unique flow rate, two primary feedback loops will be needed. One loop will be a calculation of measured mass flow rate for comparing to the demand. The second will be a calculated modification to the actuating signal to account for changes in the pressure rise across the compressor. Calculations of actuating signals will require the solutions to mathematical or tabular descriptions of the controlled system characteristics.

The primary requirement of the control system will be to maintain the demand mass flow rate. It may be desirable to program the mass flow rate during reactor startup and

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shutdown to maintain a particular system parameter, e.g., reactor or primary heat exchanger exit gas temperature. This choice of control would not require an additional demand input but would require an expanded control calculation.

Bypass Circulators - The indirectly controlled parameter in the bypass circulators will be heat exchanger inlet gas temperature; the directly controlled parameter will be circulator speed. The method of speed control will be the same as for the primary circulators.

The demand to the controlling system will be the heat exchanger inlet gas temperature. This temperature will be specified in the computer programs and will not require a manual input. Control calculations will be similar to those performed for the primary circulators with an additional calculation to describe the mixing characteristics of the primary and bypass gas streams.

The primary objective of the control loop will be to maintain a heat exchanger inlet gas temperature which is less than a fixed maximum allowable temperature. Control of the bypass circulators would afford the most effective means of limiting heat exchanger discharge temperature transients. This control mode would result in a noticeable increase in calculational requirements, since a solution of the heat exchanger process would be necessary.

Gas Preheater - The indirectly controlled parameter in the gas preheater will be reactor inlet gas temperature; the directly controlled parameter will be current to the control winding of a saturable reactor.

The demand to the controlling system will be the reactor inlet gas temperature. This temperature will be a digital value entered manually from an operator's console. Control calculations will require solution of models describing the reluctance of the saturable reactor and the heat transfer to the gas stream.

The primary requirement of the heater control will be to maintain a constant gas temperature. Limitations on the heater element temperatures will be monitored.

Primary Heat Exchanger - Modulation of the water flow rate through the secondary side of the heat exchanger could be used as an auxiliary means of limiting exit-gas-temperature variations. Control would be effected by a single flow-control valve.

Operation Monitor

The function placing the greatest demands on the computer will be that of monitoring the operation of the reactor system. The monitor programs will be designed to minimize the amount of test data that must be reviewed by operating personnel while conducting the reactor tests. These programs will provide evaluation of data integrity, analysis of loop component performance, and specification of control and data acquisition requirements.

Data Integrity

Measured and calculated system parameters will be compared to expected values and to fixed limits. The existence of an abnormal value will initiate an investigation of the cause. This investigation could include a check of the data system calibration, a review of the history of a particular sensor, and/or an analysis of the component with which the parameter is associated. The results of all diagnostic surveys will be recorded for review by the system operators.

Component Performance

All loop components and associated subsystems will be monitored for proper operation. Temperature distributions, heat balances, operating voltages and currents, etc., will be evaluated to detect any trends in operating characteristics. Statistical variations in meas-

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ured parameters will be tested for a measure of component performance. A particularly significant calculation will be comparing the observed operating point of each circulator with expected conditions and determining the safety margin from the surge line.

The occurrence of an alarm condition will be signalled by a contact sensor input. The computer will be programmed to perform a diagnostic analysis of the possible cause of such an occurrence and present the results of the diagnosis to the test operators.

Control and Data Acquisition

Requests for data acquisition and control will be generated within the computer programs. The requirements will be determined from prevailing rates of parameter change or from calculated predictions of operating conditions, e.g., afterheat removal, which would necessitate a different mode of control or data acquisition. The measurement or calculation of an abnormal or questionable value could require a repetition of a data sampling sequence or an increase in control frequency.

Information Presentation and Storage

Two general types of data presentation and storage will be required. The first type will be temporary and will concern information needed to guide the operation of the current testing program. The second type will be permanent, involving data which will be used for later analysis and reporting.

Presentation - Information describing the existing operating point, e.g., reactor coolant mass flow rate, reactor power level, and excess reactivity, will be displayed on pre-assigned decimal registers on operators' control consoles. These registers will be updated as new values are calculated. During transient operations, transmission of information will be scheduled so that display changes are consistent with operator reaction times.

Storage - The major portion of acquired data will be recorded on magnetic tape. The availability of computer time for data reduction will influence the permanency and formats of recorded data. All information recorded for permanent file will have been converted to engineering units. Transient data acquired for future analysis will be recorded in raw form. Steady-state data recordings will be stored as an average value and a variance for each measured parameter.

PILOT LOOP

All technical specifications were completed and the procurement of all major pilot loop components was initiated. Vendor proposals were received and are being evaluated on several of these items including the diaphragm compressor, gas purification system, and moisture detector. In addition, specifications were generated and procurement initiated on smaller components including valves, flowmeters, pressure transducers, and temperature sensors.

More detailed schematic drawings were made of the pilot loop, sub-loop, and impurity measurement system (Figures 4.7, 4.8, and 4.9, respectively). Design changes include the addition of the sub-loop test station, a greater loop volume and storage capacity, more detailed information on loop instrumentation, and expanded details on the impurity measurement system.

The design and fabrication of an automatic data collection system for use with the pilot loop were initiated during this reporting period. The system, Figure 4.10, is based on the use of existing equipment and is designed to accommodate 10 digital inputs including such items as date, run number, time, total flow, etc.; 90 voltage inputs for thermocouples, pressure transducers, and flowrate instrumentation. Turbine-type flowmeters

Fig. 4.7 – Schematic drawing of the pilot loop (Dwg. 849D423)

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Fig. 4.8 – Schematic drawing of the sub-loop (Dwg. 849D416)

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Fig. 4.9 – Schematic drawing of the pilot loop impurity measurement system
(Dwg. 849D418)

Fig. 4.10 – Data system of the 710 pilot loop

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connected to rate meters and scalars will be used to measure total flow and flow rate. Equipment which was retained from the original IET data system includes a 100-point scanner, digital voltmeter, a digital clock, scalars, tape punch, typewriter, and the necessary power supplies. Only minor items, such as relays and selector switches, need be procured. The circuit diagrams were completed and circuit wiring and component checkout were initiated.

710 Remote Handling

Remote handling equipment requirements of the 710 Reactor Experiment were studied. Manual access, for servicing all test-dolly-mounted equipment except those items within the reactor - shield assembly and the filter shield is planned. In general, service operations will be done manually wherever possible; remote operations are planned only where necessitated by safety or radiation requirements. Maximum use will be made of existing remote handling facilities and equipment at the ITS-TAN Building 607 hot shop and associated services. Some remote, semi-remote, or manually-assisted operations will require special handling equipment, jigs, and fixtures. This equipment is designed to be compatible with and to complement the general purpose equipment of the existing facility.

Routine remote service operations are expected during the course of the 710 Reactor Experiment test program. In particular, removal and inspection of fuel elements and other reactor and control system components are planned at various stages of the power tests.

The planned remote handling operations can be categorized as follows:

1. Reactor shields
 - a. Remove and replace top shield.
 - b. Remove actuator augmentation shields.
 - c. Remove reactor side shields.
2. Instrumentation
 - a. Remove and replace reactor section jumpers and disconnects.
 - b. Remove mixer section jumpers and disconnects.
 - c. Remove and replace sensors and connections.
3. Pressure vessel flange
 - a. Release and secure flange fasteners.
 - b. Separate flange faces and remove seals.
 - c. Replace studs and nuts.
 - d. Replace seals.
 - e. Inspect and prepare flanges for closure.
 - f. Remove and re-assemble pressure vessel head assembly.
4. Reactor assembly
 - a. Remove and re-assemble inner dome,
 - b. Install and remove refueling aftercooling systems.
 - c. Remove and replace reactor assembly in pressure vessel.
 - d. Replace fuel elements in reactor assembly.
 - e. Inspect and store fuel elements.
 - f. Remove and re-assemble reflector module.
5. Mixer - heat exchanger assembly
 - a. Remove, inspect, and replace primary coolant mixing device.
 - b. Remotely inspect accessible portions of pressure vessel and main heat exchanger.
6. Filter
 - a. Remove and re-install filter assembly shields.
 - b. Remove and replace filter assembly.
 - c. Remove filter unit instrumentation.

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7. Semi-remote or manually assisted operations
 - a. Replace control drum actuators.
 - b. Remove and re-install filter loop piping.

General purpose manipulating equipment of the TAN building 607 hot shop which will be used for the 710 Reactor Experiment remote service operations includes the following:

1. Whiting overhead bridge crane; 100-ton main hoist; 10-ton auxiliary hoist; 51-ft lift.
2. Heavy duty G-E overhead manipulator (O-Man); 500-lb capacity; 33-ft working height.
3. G. M. wall-mounted manipulators (3); models C and 300; 25-lb capacity; 27-ft working height.
4. G. M. Gradall boom wall-mounted manipulator; model C; 25-lb capacity; 50-ft + working height.
5. CRL master slave manipulators (1 pair), model F; light capacity work.

Figure 4.11 shows the relative size of the test dolly with the height limits of hot shop manipulating equipment. While none of the planned remote work areas are beyond the reach limits of the manipulators, some operations with the special equipment will require manipulating very close to these limits. Viewing requirements are being considered in shield design as well as in the design of handling jigs and fixtures.

Fig. 4.11 - Hot shop section of TAN building 607 at ITS (Dwg. 119C1133)

The preliminary investigation of remote handling operations and equipment for hot service of the 710 Reactor Experiment program have disclosed a number of areas requiring investigation and study. There are, however, no apparent problem areas which would require extensive development of equipment or techniques. All remote operations presently contemplated for the 710 Reactor Experiment are based on similar operations which have been performed successfully in the 607 hot shop facility.

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