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AEC RESEARCH AND
DEVELOPMENT REPORT

MASTER

Nuclear Materials & Propulsion Operation

710 REACTOR PROGRAM, PROGRESS REPORT No. 13

MARCH 31, 1965

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INTRODUCTION AND SUMMARY

Introduction

This report is the thirteenth 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 January 1 to February 28, 1965, 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

DEVELOPMENT OF REFRACTORY-METAL FUEL ELEMENTS

1. Five partial-length, 37-hole, Ta-clad fuel elements, 13 process-development Ta-clad capsules, and one single-hole W-Re-Mo-clad specimen were hot-gas pressure-bond cycled.
2. One full-length 37-hole fuel element was assembled and sealed.
3. A single-hole Ta-clad specimen was bonded and leak-tight after 511 hours at 3900°F and 53 thermal cycles from 3900°F to 500°F.
4. A single-hole W-Re-Mo-clad specimen was bonded and leak-tight after 100 hours at 3900°F and 30 thermal cycles from 3900°F to 500°F.
5. Two partial-length 37-hole fuel elements completed 100-hour tests at 3900°F and 30 cycles from 3900°F to 500°F. Dimensional stability was good and specimen weight change was negligible. Each of the specimens had minute helium-detected leaks, the majority of which are attributed to localized strain conditions between the header plates and the adjacent compacts.

CRITICAL EXPERIMENTS OF FAST-SPECTRUM REFRACTORY-METAL REACTORS

1. The Mockup No. 1A critical experiment achieved criticality.
2. Experiments were performed to determine the worth of the control system, the worth of fuel rods in various positions in the core, the Rossi-alpha constants, and the worth of poison segments in the reflector.
3. In the initial analysis of Mockup No. 1A, a better theoretical method was developed for representing the effects of the longitudinal gap between the core and the core shell in the one-dimensional reactor analysis.
4. Analysis of the 710 basic critical experiment continued.

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EXPERIMENTAL REACTOR PRELIMINARY DESIGN AND DEVELOPMENT

1. Thermal stress analyses were completed for the initial configuration, using preliminary heating rates. The revised heating rates are being computed by the Monte Carlo technique.
2. Detailed analysis and evaluation of specific areas of the initial configuration were performed in an effort to reduce complexities.
3. Preliminary initial assembly procedures and subsequent remote disassembly and re-assembly procedures were evolved.
4. Steady-state temperature, flow, and pressure drop calculations were completed for various components, also using preliminary heating rates.
5. Transient temperatures following reactor scram were determined for the pressure vessel.
6. Nuclear heating calculations were continued.
7. The simulated test loop was used to evaluate effects of certain component failures upon loop operating parameters, other loop components, and the reactor.
8. A three-stage simulation of the reactor thermal characteristics was operated simultaneously with a single-stage simulation to compare the dynamic responses.

TEST FACILITIES AND PILOT LOOP DESIGN AND EVALUATION

Advanced engineering studies were continued on the test equipment and the facility requirements for the 710 Experimental Reactor power test. Pilot loop procurement was continued; the specifications were issued for all major components of the pilot loop and sub-loop. The use of a commercial stud tensioning device to remotely make up the pressure vessel flange was evaluated.

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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 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, five partial-length 37-hole fuel elements, 13 process-development capsules (all tantalum-clad), and one advanced single-hole W-Re-Mo-clad¹ specimen were hot-gas pressure-bond cycled. With the exception of one partial-length 37-hole fuel element, which developed cracks in a header plate, all were bonded and leak-tight. A full-length 37-hole fuel element was assembled and sealed and is awaiting hot-gas pressure-bond cycling.

Matrix Compact Fabrication

During this report period, 116 hexagonal compacts and 46 circular compacts were processed through sintering; the results are shown in Table 1.1.

Studies were continued to establish the relationship between processing characteristics and the chemical and physical properties of different UO₂ powder lots. Shrinkage rates during sintering, observed with twin telemicroscopes, showed significant differences

TABLE 1.1
COMPACT FABRICATION DATA

Number Of Compacts	Composition, wt %			Average Density, % of theoretical
	W	UO ₂	ThO ₂	
<u>Hexagonal Compacts</u>				
12	54	46	-	95.7
64	54	43.3	2.7	94.4
40	54	41.4	4.6	94.1
<u>Circular Compacts</u>				
26	54	41.4	4.6	93.7
14	-	100	-	96.5
6	100	-	-	94.7

¹W - 30Re -

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among various powder lots and emphasized the need to fully characterize each lot in order to determine the best sintering cycle to produce the highest sintered densities.

Five partial-length 37-hole fuel elements were assembled using compacts that were given the wet (60°F inlet dewpoint) and dry (-50°F inlet dewpoint) hydrogen treatment.² Elements P-7, P-9, P-10, and P-2 were machined after treating in wet hydrogen, prior to the dry hydrogen treatment. The compacts used in element P-8 were treated in both wet and dry hydrogen prior to machining. Both wet and dry treatments were done at 2550°F for 4 hours. Free uranium analysis of W-UO₂ control samples after the wet - dry hydrogen treatment showed consistently less than 0.01 percent free uranium metal by the hydriding technique, and the oxygen-to-uranium ratios of UO₂ control samples were in the range of 2.000 to 2.003, determined by the ignition method, both within the limits of detection.

A series of six hexagonal compacts of 54W - 46UO₂ (wt. %) composition was processed through machining to further evaluate the effectiveness of the wet - dry hydrogen treatment and the Freon³ cleaning procedure. The results, given in Table 1.2, show that both the 3110°F and 2550°F wet hydrogen treatments reduced the percentage of free uranium metal from 0.25 to less than 0.01. The amount of carbon from the cores that were machined and cleaned averaged 40 ppm compared to 15 to 18 ppm in cores not machined. The oxygen-to-uranium ratio of UO₂ control samples was 2.002 after the wet - dry hydrogen treatment.

TABLE 1.2
WET - DRY HYDROGEN TREATMENT
AND FREON CLEANING EFFECTIVENESS DATA

Core Number	Treatment	Machined And Freon Cleaned	Free Uranium Content, %	Impurity Content, ppm		
				Carbon	Nitrogen	Hydrogen
H142	None	No	-	17	-	-
H143	None	Yes	0.25	40	101	3.2
H144	2550°F wet - dry H ₂	No	0.007	15	37	3.1
H145	2550°F wet - dry H ₂	Yes	0.01	39	19	5.6
H152	3110°F wet H ₂	No	0.005	18	19	1.3
	2550°F dry H ₂					
H153	3110°F wet H ₂	Yes	0.007	-	-	-
	2550°F dry H ₂					

Recent modifications to the high-pressure coolant transfer glands on the compact drilling machine have resulted in greatly improved gland operational life; 50-hour life is now indicated. With the ability to rebuild the glands in-house and change the glands in 15 minutes, the operational life is considered acceptable.

Cladding Fabrication

A torque-sensitive control unit, Figure 1.1, is being utilized for refractory-metal drilling. The unit continually senses the torque load on the cutting faces of the drill and adjusts drill speed and feed rate to compensate for variations in drill loading. Three 0.180-inch-thick Ta - 10W header plates were processed on this unit with excellent results, as shown in Table 1.3. These results are of greater significance when it is noted

²"710 Reactor Program, Progress Report No. 11," GE-NMPO, GEMP-323, November 30, 1964, pp.11-13.

³Registered trade name of E. I. du Pont de Nemours and Co., Inc.

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Fig. 1.1 - Torque-sensing control unit used to drill coolant tube holes in refractory-metal header plates (Neg. P65-1-24C)

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TABLE 1.3

HOLE DIAMETER DISTRIBUTION ON
THREE 37-HOLE Ta - 10W HEADER PLATES
DRILLED USING THE TORQUE-SENSITIVE CONTROL UNIT

OD Dimension, in.	Occurrence	
0.0800	0	Lower drawing limit
0.0801	0	
0.0802	0	
0.0803	0	
0.0804	0	
0.0805	1	
0.0806	15	
0.0807	22	
0.0808	42	
0.0809	10	
0.0810	18	Drawing nominal
0.0811	1	
0.0812	2	
0.0813	0	
0.0814	0	
0.0815	0	
0.0816	0	
0.0817	0	
0.0818	0	
0.0819	0	
0.0820	0	Upper drawing limit
Total 111		
Average 0.0808 in.		

that the drill was changed for each plate to determine drill tolerance effects. Drilling time averaged only 35 seconds per hole.

Initial tests were run to evaluate the expanding mandrel technique for forming a hexagonal can from a round tube. The tests were run on Type 310 stainless steel tubing, having a wall thickness of 0.015 inch. Figure 1.2 shows the setup before and after the application of pressure. Single-seam, electron-beam-welded tantalum tubing, having a wall thickness of 0.010 inch, is currently being fabricated for further evaluation of this technique.

Effort was initiated to develop comprehensive machinability data on the W-Re-Mo alloys; early results indicate that, while these materials are extremely abrasive, a technique can be developed to produce small holes to depths of 2 to 4 diameters. The alloys can be ground with relative ease to precision tolerances and excellent surface finishes. Initial tests of the electrical discharge machining characteristics of these materials show that cutting rates and electrode wear rates fall within ranges considered acceptable for the refractory metals.

Partial-Length Fuel Element Fabrication

During this reporting period, five partial-length 37-hole fuel element assemblies (P-2, -7, -8, -9 and -10) were assembled and sealed. Fabrication data pertinent to these assemblies were listed in the previous progress report.⁴ With the exception of P-10, all assemblies were successfully hot-gas pressure bonded. Table 1.4 lists matrix densification results for all partial-length 37-hole fuel elements pressure-bonded to date. Cracks in the outer skirt of a Ta - 10W header plate prevented bonding in the P-10 assembly.

⁴"710 Reactor Program, Progress Report No. 12," GE-NMPO, GEMP-333, January 29, 1965, Table 1.3, p. 13.

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Fig. 1.2 - Expanding mandrel for forming a hexagonal can from round tubing. An experimental stainless steel tube is shown on the mandrel before and after forming. (Negs. P65-3-3A, left, and P65-3-3B, right)

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TABLE 1.4

PARTIAL-LENGTH 37-HOLE FUEL ELEMENT MATRIX
DENSIFICATION IN HOT-GAS PRESSURE BONDING

Fuel Element	Matrix Density, % of theoretical		
	Pre-Bonding	Post-Bonding ^a	Increase
P-2	93.5	94.6	1.1
P-3 ^b	94.3	96.1	1.8
P-4 ^b	93.1	95.8	2.7
P-5 ^b	91.5	93.3	1.8
P-6	92.8	93.8	1.0
P-7	92.7	93.7	1.0
P-8	93.0	94.4	1.4
P-9	93.1	94.5	1.4

^aBonding cycle: 1 hour at 3000°F and 10,000 psi He gas pressure.

^bReported previously; included for completeness of data.

The defective header plate was removed and will be replaced with one machined from unalloyed tantalum. Header plate cracking also prevented successful bonding in assembly P-1. Hardness measurements of Ta - 10W header plates, both in the as-received condition and prior to final assembly, ranged from 160 to 260 DPH, which indicates the low ductility of this material.

Full-Length Fuel Element Fabrication

The first full-length 37-hole fuel element (F-1) was assembled and sealed successfully (see Figure 1.3). The fuel element is scheduled for hot-gas pressure bonding in the new pressure-bonding facility when final acceptance tests are completed.

Fig. 1.3 - Full-length 37-hole fuel element F-1 sealed for hot-gas pressure bonding (Neg. P65-2-1)

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Advanced Single-Hole Test Specimen Fabrication

The fabrication of the first three single-hole test specimens, consisting of a W - 41.4UO₂ - 4.6ThO₂ matrix with cladding of W - 30Re - 30Mo (at. %) alloy, was described in the previous report.⁵ An additional test specimen, ASH-4, was assembled and sealed for hot-gas pressure bonding.

Process-Development Capsule Fabrication

In addition to the 14 process-development capsules described in the previous reports,⁶ nine more were scheduled for fabrication to evaluate further specific process variables. The fabrication data for these capsules are listed in Table 1.5; the status of the capsules, categorized by process variable, is summarized below.

TABLE 1.5
FABRICATION DATA FOR X-TYPE PROCESS-DEVELOPMENT CAPSULES

Capsule	Matrix Composition, wt %				Process Variable	Matrix Density, % of theoretical ^a		
	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 ^b	94.0	96.0	2.0
X-32	54	46	-	-	High-density UO ₂ , hot-pressed ^b	86.2	92.3	6.1
X-33	54	46	-	-	Low-density UO ₂ , hot-pressed ^c	86.1	94.3	8.2
X-34	54	46	-	-	Low-density UO ₂ , hot-pressed ^c	83.9	91.7	7.8
X-35	54	46	-	-	High-density UO ₂ , HGPD ^d	97.6		
X-36	54	46	-	-	High-density UO ₂ , HGPD ^d			
X-47	30.9	52.7	-	16.4	Matrix composition, Mo-W-UO ₂	90.1	92.9	2.8
X-48	30.9	52.7	-	16.4	Matrix composition, Mo-W-UO ₂	90.2	93.7	3.5
X-49	54	41.4	4.6	-	Wet H ₂ + machining + dry H ₂ ^e	91.0	92.8	1.8
X-50	54	41.4	4.6	-	Wet H ₂ + machining + dry H ₂ ^e	92.6	94.1	1.5
X-51	54	41.4	4.6	-	Wet H ₂ + dry H ₂ + machining	91.8	93.9	2.1
X-52	54	41.4	4.6	-	Wet H ₂ + dry H ₂ + machining	91.8	93.6	1.8
X-53	100	-	-	-	Densification of pure W, HGPD	95.0	95.5	0.5
X-54	54	41.4	4.6	-	Time at temperature in wet and dry H ₂ ^f	92.7	93.7	1.0
X-55	54	41.4	4.6	-	Time at temperature in wet and dry H ₂ ^f	92.3	94.5	2.2
X-56	54	41.4	4.6	-	Time at temperature in wet and dry H ₂ ^f	92.6	94.3	1.7
X-61	54	46	-	-	Low-density UO ₂ , hot-pressed ^g			
X-62	54	46	-	-	Low-density UO ₂ , hot-pressed ^g			
X-63	-	100	-	-	Densification of UO ₂ , HGPD			

^aLack of data indicates that capsules have not yet reached that stage of fabrication.

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

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

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

^eRework of compacts changed scheduled treatment shown to include an additional machining operation followed by Freon cleaning.

^fHolding times in both wet and dry hydrogen at 2550°F increased from 4 to 18 hours.

^gIdentical to X-33 and X-34 except that matrix compacts will be given the wet - dry H₂ treatment.

Matrix Composition (Capsules X-29, -30, -47, and -48) - All capsules are bonded and ready for testing. The high matrix densities (>98% of theoretical) of the Mo-UO₂ compacts as compared to those of the Mo-W-UO₂ and W-UO₂ compacts (~93% of theoretical) reflect both the lower sinterability of the tungsten phase and its greater resistance to densification in hot-gas pressure bonding.

Hot-Pressed Matrix (Capsules X-31, -32, -33, -34, -61, and -62) - Capsules X-31 and -33 were thermal-cycled 17 times from 3900°F to 500°F with programmed cooling after 5-minute holding periods at temperature. Post-test inspection revealed severe cladding embrittlement and numerous leaks in the cans. Capsules X-31, -32, -33, and -34 were assembled and sealed prior to completion of metallographic and free uranium

⁵Ibid., p. 19.

⁶Ibid., pp. 19-21.

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analyses on sample sections of the matrix compacts. These analyses indicated that insufficient machining stock had been allowed in the matrix segments and that a high level of carbon and free uranium contamination was present. Post-test analyses of the cladding and fueled matrix in X-33 showed 620 ppm of carbon in the cladding and 8000 ppm of free uranium in the matrix. In view of these results, no significant information was likely to be gained by testing capsules X-32 and -34. Therefore, two new capsules were scheduled for fabrication, in which the size of the matrix compacts will be increased to allow additional machining stock to remove surface contamination from the graphite dies. Wet - dry hydrogen treatments will be used to reduce the free uranium content to the 100-ppm level.

Matrix Densification by Hot-Gas Isostatic Pressure (Capsules X-35, -36, -53, and -63) - Matrix compacts for capsules X-35 and -36 were densified by encapsulating the green compact in tantalum and isostatically pressing in the hot-gas pressure-bonding apparatus. After stripping the cladding from the X-35 compact, the irregular core was machined into a right circular cylinder for encapsulation as an X-type capsule. A densification of 97.6 percent of theoretical was achieved in this compact. As shown by the photomicrograph in Figure 1.4, the density of this tungsten matrix appears to be very close to theoretical; the 2.4-percent porosity appears to be confined primarily to the UO_2 particles.

Fig. 1.4 - Photomicrograph of capsule X-35 matrix compact densified to 97.6 percent of theoretical by hot-gas pressure bonding the blended powders at 3000°F for 1 hour in 10,000 psi He gas pressure (Neg. 5603, 100X)

Wet - Dry Hydrogen Treatment Sequence (Capsules X-49, -50, -51, and -52) - All four capsules were processed as scheduled except for X-50. A core chip after the dry hydrogen treatment in two of the matrix compacts necessitated a regrinding operation in which an ultrapure low-viscosity hydrocarbon was used as the coolant. These compacts are comparable to the 37-hole matrix compacts which are machined and cleaned by similar procedures.

The other three capsules contain cores treated similarly to those in single-hole test specimen SH-54 which was leak-tight after 53 cycles and 511 hours at 3900°F.

Capsules X-49 and -50 are now on test and capsules X-51 and -52 are awaiting test.

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Densification of 100 percent W and 100 percent UO₂ - Hot-Gas Pressure Bonding (Capsules X-53 and -63) - Due to the small amount of additional densification achieved in hot-gas pressure bonding the W-UO₂ matrices, two capsules, one containing only tungsten (X-53) and the other only UO₂ (X-63), were prepared so that the individual densification of these materials could be determined. As shown in Table 1.5, the density of the tungsten compact, previously sintered at 4000°F in dry hydrogen to 95.0 percent of theoretical density, increased by only 0.5 percent after 1 hour at 3000°F and 10,000 psi helium gas pressure. This indicates that there is not much closing of the internal voids in the as-sintered tungsten structure at the present hot-gas pressure bonding temperature and pressure levels. Capsule X-63, containing 100 percent UO₂, was not hot-gas pressure bonded during this reporting period.

Time at Temperature in Wet and Dry Hydrogen (Capsules X-54, -55, and -56) - Selection of the 4-hour holding period at 2550°F in the wet - dry hydrogen cycle was based primarily on furnace cycle time considerations. In addition to the time - temperature parameters, however, other variables such as sample size and sintered density may also affect the uranium oxidation mechanism. To evaluate longer holding periods in both wet and dry hydrogen, cylindrical compacts, 1-1/2 inches in OD by 3/8 inch thick and 5/8 inch in OD by 5/8 inch thick, were pressed from W, UO₂, 90UO₂ - 10ThO₂, and 54W - 41.4UO₂ - 4.6ThO₂ (wt %) blends, and sintered at 3990°F in dry hydrogen. Where applicable, free uranium and oxygen analyses were made on sintered, wet-treated, and dry-treated samples. The results (listed in Table 1.6) show that, for both sample sizes, (1) wet hydrogen (+32°F dewpoint) gas does not increase the oxygen content of tungsten significantly, and (2) the free uranium content in thoriated and nonthoriated UO₂ is reduced below limits of detection by present analysis techniques.

TABLE 1.6
URANIUM AND OXYGEN CONTENTS AND DENSITIES OF LARGE AND SMALL COMPACTS
AFTER 18-HOUR WET - DRY HYDROGEN TREATMENTS

Compact Composition, wt %			As-Sintered Density, % of theoretical		Content, ppm							
			Large Compact		Large Compact ^a				Small Compact ^b			
					As-Sintered		After Dry H ₂		As-Sintered		After Dry H ₂	
W	UO ₂	ThO ₂	Compact	Compact	U	O	U	O	U	O	U	O
100	-	-	94.8	94.2	-	4	-	6 ^c - 5.5 ^d	-	6.0	-	7.2 ^c - 7.1 ^d
-	100	-	95.5	95.0	3100	-	<100	-	6800	-	<100	-
-	90	10	93.0	92.0	1500	-	e	-	2700	-	<100	-
54	41.4	4.6	92.4	91.4	1700	-	<100	-	1400	-	<100	-

^aLarge compact size, 1-1/2-in. OD by 3/8-in. length.

^bSmall compact size, 5/8-in. OD by 5/8-in. length.

^cOxygen content after 18 hours in wet hydrogen.

^dOxygen content after 18 hours in dry hydrogen.

^eAnalysis not made.

These results are similar to those obtained for matrix compacts given the 4-hour holding periods and indicate that no appreciable benefit is to be gained by longer holding times at temperature. To substantiate this, three capsules (X-54, -55, and -56) containing matrix compacts treated for 18 hours in wet - dry hydrogen were assembled, sealed, bonded, and scheduled for cyclic testing.

Hot-Gas Pressure-Bonding Facility

The first complete system checkout run of the new hot-gas pressure-bonding facility revealed that an error had been made in calculating the required pumping capacity. An additional pumping unit was supplied by the vendor, installed, and a checkout run made using a 5-inch single-hole test specimen as the work piece. With the exception of the gas

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purifier furnace, all components performed satisfactorily and the test specimen was successfully bonded. The gas purifier furnace was returned to the vendor for rewiring necessary to maintain the specified temperature level for gas gettering. Pending return and installation of this accessory equipment, the facility will be operated under the procedures established for the original pressure-bonding facility.

FUEL ELEMENT DESIGN AND STRESS ANALYSIS

A keypunch error was detected in the input to the plane strain plastic-viscoelastic computer program calculation for the plot of tangential plastic strain at the maximum temperature point of the reactor scram cycle shown in the previous report.⁷ The corrected results produce a change in magnitude of the calculated plastic strains at the maximum temperature point but no change in direction. The conclusions based on these calculations remain unchanged. The corrected tangential plastic strain for the maximum temperature point is -0.000032 in./in. at the end of the first cycle and -0.000057 in./in. at the end of the sixth cycle.

A summary report covering all fuel element stress analyses accomplished to date was completed and is being prepared for publication.

FUEL ELEMENT CYCLIC TESTING

During this report period, three tantalum-clad 37-hole partial-length fuel elements, three tantalum-clad single-hole test specimens, and one W-Re-Mo-clad single-hole test specimen were cycle-tested from 3900°F to 500°F with long holding periods at 3900°F . Table 1.7 summarizes the test results. The test cycle was modified for the 37-hole fuel element tests and for the W-Re-Mo-clad single-hole test specimen to achieve 30 cycles and 100 hours simultaneously, with holding periods of equal length between cycles.

The significant nondestructive results of the 37-hole fuel element tests were (1) compact - cladding and cladding - cladding bonds remained good, (2) dimensional stability was well within tolerable limits (-0.04 to 0.17 percent radial expansion), (3) weight change was negligible (-5.4×10^{-3} to -1.02×10^{-1} percent), and (4) although each of the specimens had minute helium-detected leaks, most of these are attributed to localized straining conditions between the header plates and the adjacent cores.

As discussed in the previous report,⁸ three single-hole test specimens (SH-54, -56, and -58) containing wet - dry hydrogen treated compacts and 4.6, 0, and 2.7 weight percent ThO_2 additive, respectively, were to continue on long-time tests, after an original 30 cycles from 3900°F to 500°F , to evaluate the effect of the thorium additive. The specimen with no ThO_2 , SH-56, developed leaks after 30 additional hours at 3900°F ; SH-58 (2.7 wt % ThO_2) continued to a total of 306 hours at 3900°F and 44 cycles before a cladding leak was detected; and SH-54 (4.6 wt % ThO_2) continued for a total of 663 hours at 3900°F and 60 cycles before a leak was detected. The specimens were leak tight after the 10-, 200-, and 511-hour inspections, respectively. Diametral expansion was less with the nonthoriated specimen and increased with the amount of ThO_2 additive but was only 0.94 percent for SH-54 after 511 hours.

Advanced single-hole test specimen ASH-2, clad with W - 30Re - 30Mo, completed 100 hours at 3900°F and 30 cycles from 3900°F to 500°F using the same cycle described for the 37-hole partial-length specimens. The specimen was leak-tight and had a diametral expansion of only 0.02 percent and a weight increase of 1.6×10^{-3} percent. Visually, the specimen appeared to be in excellent condition with no significant change observed in the grain structure of the cladding.

⁷"710 Reactor Program, Progress Report No. 12," GE-NMPO, GEMP-333, January 29, 1965, Figure 1.8, p. 24.

⁸Ibid., p. 29.

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TABLE 1.7
CYCLE TEST RESULTS^a

Specimen	Compact Composition, ^b			Test Conditions ^c		Inspection Results ^e		
	wt %			Time At 3900°F, hr	Cycles ^d	Leak Test	Cumulative Change, %	
	W	UO ₂	ThO ₂				Radial Expansion	Weight
37-Hole Partial-Length Fuel Elements, Tantalum-Clad ^f (Wet - Dry H ₂ Treated)								
P-6	54	41.4	4.6	10.0	3	No leaks	-0.07	-2.1 x 10 ⁻²
				33.3	10	Leak	-0.04	-8.6 x 10 ⁻²
				66.7	20	Leaks	0.04	-1.0 x 10 ⁻¹
P-7	54	41.4	4.6	10.0	3	No leaks	-0.42	-1.0 x 10 ⁻²
				33.3	10	Leak	-0.27	-1.5 x 10 ⁻²
				100.0	30	Leaks	-0.04	-2.3 x 10 ⁻²
P-8	54	41.4	4.6	100.0	30	Leaks	0.17	-5.4 x 10 ⁻³
Single-Hole Test Specimens, Tantalum-Clad (Wet - Dry H ₂ Treated)								
SH-54	54	41.4	4.6	2.5	30	No leaks	0.33	-6.4 x 10 ⁻³
				10	31	No leaks	0.39	-2.1 x 10 ⁻²
				30	32	No leaks	0.47	-3.2 x 10 ⁻²
				60	33	No leaks	0.54	-3.5 x 10 ⁻²
				100	35	No leaks	0.59	-3.7 x 10 ⁻²
				110	36	No leaks	0.62	-4.1 x 10 ⁻²
				130	37	No leaks	0.66	-3.5 x 10 ⁻²
				160	38	No leaks	0.66	-2.8 x 10 ⁻²
				200	39	No leaks	0.73	-1.6 x 10 ⁻²
				306	44	No leaks	0.82	-4.4 x 10 ⁻²
				403	48	No leaks	0.84	-4.3 x 10 ⁻²
				511	53	No leaks	0.94	-3.5 x 10 ⁻²
SH-58	54	43.3	2.7	663	60	Leaks	1.18	-2.3 x 10 ⁻²
				2.5	30	No leaks	0.29	-1.4 x 10 ⁻⁴
				10	31	No leaks	0.32	-1.7 x 10 ⁻²
				30	32	No leaks	0.32	-3.1 x 10 ⁻²
				60	33	No leaks	0.36	-3.8 x 10 ⁻²
				100	35	No leaks	0.41	-3.8 x 10 ⁻²
				110	36	No leaks	0.41	-4.4 x 10 ⁻²
				130	37	No leaks	0.48	-3.7 x 10 ⁻²
				160	38	No leaks	0.50	-2.7 x 10 ⁻²
				200	39	No leaks	0.59	-1.8 x 10 ⁻²
				306	44	Leaks	0.62	-3.4 x 10 ⁻²
				SH-56	54	46	0	2.5
10	31	No leaks	0.30					-2.3 x 10 ⁻²
30	32	Leaks	0.27					-4.9 x 10 ⁻²
Advanced Single-Hole Test Specimen, W-Re-Mo-Clad (Wet - Dry H ₂ Treated)								
ASH-3 ^g	54	41.4	4.6	2.5	30	No leaks	0.07	+4.7 x 10 ⁻³
				5.0	60	No leaks	0.22	+1.4 x 10 ⁻²
				8.3	100	No leaks	0.60	+2.4 x 10 ⁻²
ASH-2	54	41.4	4.6	100	30	No leaks	0.02	+1.6 x 10 ⁻³
91-Hole Partial-Length Fuel Element								
9A ^g	54	41.4	4.6	0.8	10	No leaks	0.25	-1.7 x 10 ⁻²
				2.5	30	Leaks	0.68	-2.1 x 10 ⁻²

^aCompact sintering temperature, 3990°F; wet - dry H₂ treated where indicated.

^bWhen added, ThO₂ was in the agglomerated UO₂ with the exception of specimen 9A where it was added to the W.

^cTest atmosphere was bottled helium, essentially static at a pressure of 0.3 to 15 psig. Programmed cooling used for all specimens except 9A which was cooled exponentially by shutting off the furnace power.

^dNumber of cycles from 3900°F to 500°F.

^eBond checks made at each inspection; no evidence of bond deterioration. Leak tests by helium mass spectrometer.

^fTa-clad except for Ta - 10W header plates.

^gTested prior to the current period but not previously tabulated.

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EFFECTS OF OXYGEN PARTIAL PRESSURE ON THE Ta-CLAD W-UO₂ SYSTEM

Experiments conducted under the GE-NMPO high-temperature materials research program have shown that oxygen can permeate through tantalum cladding and change the composition of the UO₂ in the W-UO₂ matrix, according to the temperature and partial pressure of oxygen in the surrounding atmosphere. If the oxygen partial pressure is too low, the UO₂ phase becomes substoichiometric; however, if the partial pressure is too high, the corrosion rate of the tantalum cladding may become excessive due to the formation of a gaseous oxide, TaO.

To explore the feasibility of maintaining stoichiometry in the tantalum-clad W-UO₂ system, if this should be necessary for long-time operation, the behavior of the fuel element was studied in a flowing stream of inert gas containing oxygen at a low partial pressure. The objective of this analysis was to determine the oxygen partial pressure which is satisfactory for continuous operation. Table 1.8 summarizes the results of this work, showing the lowest oxygen partial pressure necessary to maintain the UO₂ at the stoichiometric value at four temperature levels.⁹ The total oxygen pressure is pertinent since oxygen is almost completely dissociated at these temperatures and pressures. The values shown in Table 1.8 are probably reliable to within a factor of 2.

TABLE 1.8
OXYGEN PARTIAL PRESSURE
NECESSARY TO MAINTAIN UO₂
AT STOICHIOMETRY COMPOSITION

Temperature, °F	P _O , atm	Equivalent P _{O₂} , atm
4350	2×10^{-5}	4×10^{-7}
3990	2×10^{-6}	3×10^{-8}
3630	6×10^{-8}	5×10^{-11}
3090	2×10^{-10}	1×10^{-13}

The maximum temperature of the 710 Experimental Reactor fuel element is 3850°F; therefore, the absolute pressure of atomic oxygen should be at least 6×10^{-7} atmosphere. Since the total pressure is approximately 50 atmospheres in the reactor, the O₂ content in the primary coolant neon should be at least $6/2 \times 10^{-7} \times 1/50 = 6 \times 10^{-3}$ ppm by volume.

To estimate the corrosion rate, it was assumed that the controlling step is diffusion of TaO gas through the boundary layer. The corrosion rate will be proportional to the partial pressure of TaO at the surface of the cladding. From the present best estimate of the free energy of formation of TaO gas, it is evident that over 90 percent of the oxygen which diffuses inward through the boundary layer will be converted into TaO gas. Thus, if the oxygen partial pressure is 6×10^{-7} atmosphere, the TaO pressure will be about 6×10^{-7} atmosphere at the surface of the cladding which is at a temperature of 3810°F. The estimated flux of TaO molecules escaping the boundary layer for a pressure drop of 6×10^{-7} atmosphere is about 2×10^{14} molecules/in.²-sec. If this flux persisted for 100 hours, the thickness of 0.008-inch-thick tantalum cladding would be reduced by about 1 percent at the hottest part of the fuel element.

GAS DYNAMIC TEST

Induction heating capability of the low-pressure, low-flow (4 scfm), circulating gas loop was increased from a specimen size of 1-inch diameter by 1.5-inch length to 1.5

⁹"High-Temperature Materials Program Progress Report No. 46, Part B," GE-NMPO, GEMP-46B, to be published.

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inches in diameter by 6 inches long. The maximum temperature obtained to date in the larger specimens was 3100°F at a power input of 50 kw, the maximum power available at the present location of the loop. A temperature of 4000°F should be attained when the loop is connected to the 300-kw, 3000-cycle equipment which will be available upon completion of repairs to the substation transformer.

A prime consideration in this loop is the maintenance of a low level of impurities in the gas stream. A hot (1000°F) calcium hydride scavenger operating in the loop was found to lower the impurity content of helium containing 757 ppm of oxygen and nitrogen to levels which had no observable effect on the appearance or hardness of niobium and molybdenum specimens heated to 3000°F.

A gas chromatograph is being installed in the gas loop to provide semicontinuous monitoring of the gas and to determine the analytical sensitivity of the chromatograph at the 1 to 0.1 ppm contamination level.

A current concentrator designed to match the partial-length 37-hole fuel element configuration was designed and is being fabricated.

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

On January 11, 1965, the Mockup No. 1A critical experiment achieved criticality. Experiments were performed to determine the worth of the control system, the worth of fuel rods in various positions in the core, the Rossi-alpha constants, and the worth of poison segments in the reflector. The initial analysis of the Mockup No. 1A was refined by developing a better theoretical method of representing the effects of the longitudinal gap between the core and the core shell in the one-dimensional reactor analyses. Further analysis work was performed on the basic critical experiment poison segment experiments.

MOCKUP NO. 1A CRITICAL EXPERIMENT

The Mockup No. 1A critical experiment achieved criticality with a uniformly loaded core as shown in Figure 2.1. The critical mass was 191.5 kilograms of Oralloy. Figure 2.2 shows the loading curve obtained during the approach to criticality. The curve indicates a smooth well-behaved function. Figure 2.3 shows the "Fat Man" checks made during the loading process. These data indicate that there is no criticality hazard associated with working between the separated tables.

After loading was completed, a noticeable acentricity existed in the reactor assembly. Also, the faces of the core at the midplane were not exactly parallel. The radial center of the core matrix was 0.3175 cm below the radial centerline of the core shell, which caused the top gap between the core and core shell to be wider by 0.6350 cm than the bottom gap. The core shell planes were made parallel to each other by placing 0.1778-cm-thick shims at the midplane between the core shell and beryllium reflector. Measurements made with this configuration indicated that the control bars were worth more at the bottom of the reactor than at the top because of the difference in the gaps between the core and the core shell at the top and at the bottom. Table 2.1 lists the results of these measurements.

Figure 2.4 is a typical plot of the worth of a four-bar cluster of shim rods as a function of the distance inserted into the configuration located in the movable table. Figure 2.5 is a typical plot of the worth of a four-bar cluster of shim rods as a function of the distance inserted into the configuration located in the fixed table. Both of these curves indicate that there is substantial worth remaining after the bars leave the core region.

The worth of the safety system was determined by interchanging two five-bar clusters from a safety actuator with the two upper four-bar clusters of the shim actuators. From the data obtained, a conservative calculation of worth was made for the safety system. The resultant 2.75 percent $\Delta k/k$ was sufficient to permit the removal of a safety actuator from each table.

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Fig. 2.1 - 710 critical experiment mockup No. 1A (Neg. IP 65-2-32)

Fig. 2.2 - Critical experiment Mockup No. 1A loading curve

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Fig. 2.3 – Critical experiment Mockup No. 1A loading curve "Fat Man" checks

TABLE 2.1
REFLECTOR BAR WORTH RESULTS

Shim No.	No. Of Bars	Location	Worth, % $\Delta k/k$
1	4	Top of movable table	0.2931
2	4	Top of fixed table	0.2854
3	4	Bottom of movable table	0.3257
4	4	Bottom of fixed table	0.3218
Total			1.2260

Because of the differences in the reflector bar worths at the top and at the bottom of the assembly, consideration was given to the possibility of aligning the centers of the core shell and the core structure matrix by increasing the thickness of the pads which support the core matrix. However, the decision was made to continue the experimental work in spite of the misalignment and to take advantage of the varying gap thicknesses. The misalignment will produce only second-order effects on the important integral measurements such as critical loading, neutron lifetime, and azimuthally symmetrical reflector effects. First-order differences may be encountered in the measurement of local parameters at different circumferential locations such as individual poison segment worth, reflector bar worth, and radial power distribution near the core surface, but the experimental determination of these differences as a function of gap thickness will provide valuable information for use in the nuclear design of the Experimental Reactor.

Fuel Worth Measurements

The worth of fuel was measured at six different radial locations by removing a full core-length rod and measuring the resultant period. The period measurement technique was used with all reflector bars fully inserted to eliminate the possibly uncertainty associated with repeating a reflector bar position. The results of the fuel worth measurements are listed in Table 2.2; the locations where measurements were made are shown in Figure 2.6. Measurements will be extended to the perimeter of the core where the power peak occurs when an outermost "homogeneous" cell is installed.

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Fig. 2.4 – Four-bar cluster normalized critical reflector bar worth,
No. 1 shim

Fig. 2.5 – Four-bar cluster normalized critical reflector bar worth,
No. 2 shim

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TABLE 2.2
FUEL ROD WORTH RESULTS^a

Cell No.	Distance From Center Of Core, cm	Rod Worth, $\% \Delta k/k \times 10^2$
0	0	5.10
110	4.83	4.79
120	8.00	4.17
130	12.50	3.27
140	16.31	2.38
150	20.14	1.77

^aThe fuel rod length is 37.49 cm; mass is 102.22 grams.

Fig. 2.6 - Approximate locations for fuel rod worth measurements

Rossi-alpha Measurements

A series of delayed critical runs was made at very low power to obtain neutron lifetime data utilizing the Rossi-alpha technique, in which the neutron chain decay is measured by making statistical observations of the neutron population. The experimental apparatus included a 256-channel analyzer, with a time-of-flight logic unit, and two He-3 chambers. To provide space for the chambers in the core, it was necessary to remove a number of rods of each material from the two cell locations used; the resultant fuel deficiency was overcome by distributing the 12 rods, each 22.25 cm in length, uniformly throughout the fixed table. The other material deficiencies were considered insignificant and no corrections were made.

Data were accumulated on the 256-channel analyzer for the following channel widths: 0.25, 1.0, 2.0, 4.0, 8.0, 16.0, and 64.0 microseconds. A least-squares curve-fit computer program was then used for either a single or double exponential decay and the resultant exponential decay constants, α , obtained. Analysis of these data is not complete;

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however, it appears, as was the case in the basic critical experiment,¹ that at least three distinguishable decay components are present. These are:

$$\begin{aligned}\alpha_1 &= (0.35 \pm 0.03) (\text{microseconds})^{-1} \\ \alpha_2 &= (0.26 \pm 0.04) 10^{-1} (\text{microseconds})^{-1} \\ \alpha_3 &= (0.90 \pm 0.05) 10^{-3} (\text{microseconds})^{-1}\end{aligned}$$

Poison Segment Experiments

A series of reactivity measurements was made in the radial reflector to determine the worth of boron as a function of thickness and B¹⁰ enrichment. The materials used in these experiments were as follows:

1. Borated stainless steel (B-SS) with a boron weight percent of 1.05, as determined by neutron chopper transmission measurements. This material is in the form of 0.0254-cm-thick sheet stock. The sample used in the poison experiment was composed of 26 layers with a total mass of 1080.75 grams (2.054 grams of B¹⁰).
2. Natural boron amorphous powder packed into four 0.6350-cm-thick trapezoidal copper cans. This powder weighed 140.146 grams including 9.5 percent impurities and contained 23.24 grams of B¹⁰.
3. Enriched boron powder (amorphous) packed into four 0.6350-cm-thick trapezoidal copper cans. This boron powder weighed 148.8 grams including 6 percent impurities and contained 98.2 grams of B¹⁰.

The measurements utilizing the three previously described samples were made at positions 29-S-F, M and 30-T-F, M (see Figure 2.7). A 0.762-cm-thick void was created in the reflector by replacing normal reflector pieces with specially milled beryllium pieces. The poison samples were then incrementally added. The results of the measurements are presented in Table 2.3 and Figure 2.8. Additional worth data, to aid in the evaluation of the asymmetry effects, were obtained by measuring the B-SS worth in positions 36-M-F, M and 37-L-F, M. These data, presented in Table 2.3, show the B-SS samples to be worth more at the bottom of the core than at the top, a factor which corroborates the results of the reflector bar worth measurements. The worth of B-SS as a function of thickness was measured in positions 34-Z-F, M and 36-Z-F, M and is presented in Figure 2.9.

TABLE 2.3

POISON SEGMENT MEASUREMENTS		
Material	Location ^a	Worth, % $\Delta k/k$
B-SS	29-S, 30-T	-0.1354
Natural boron	29-S, 30-T	-0.2778
B ¹⁰	29-S, 30-T	-0.4373
B-SS	36-M, 37-L	-0.1272
B-SS	34-Z, 36-Z	-0.1468
Void	29-S, 30-T	-0.0435
Void	36-M, 37-L	-0.0391
Void	34-Z, 36-Z	-0.0437

^aBoth fixed and movable tables.

ANALYSIS OF THE MOCKUP NO. 1A CRITICAL EXPERIMENT

Nuclear analysis of the Mockup No. 1A critical experiment by all methods currently used except the r, z transport code, TDC, requires some indirect method of handling the annular void between core and core shell. In radial diffusion calculations, the method of

¹"710 Reactor Program, Progress Report No. 7," GE-NMPO, GEMP-282, March 31, 1964, p. 25.

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Fig. 2.7 - Configuration of critical experiment reactor Mockup No. 1A

Garelis² should yield an adequate radial diffusion coefficient. The longitudinal leakage in the gap, a problem common to both diffusion and transport codes except TDC, can be approximated by using the leakage escape probability, F , from cylinders³ having the same height as the core and the same end areas as seen by neutrons entering the annulus (for small gaps, the leakage escape probability is nearly independent of the shape of the escape area). Letting r_o and r_i be, respectively, the outer and inner radii of the gap, then neutrons entering the gap from the inner surface see an escape area,

$$A_i = \frac{\pi r_o^2}{2} - \left(r_i \sqrt{r_o^2 - r_i^2} + r_o^2 \sin^{-1} \frac{r_i}{r_o} \right), \quad (2.1)$$

and have leakage escape probability F_i , while neutrons entering the gap from the outer surface see an escape area,

$$A_o = 2A_i + \left(r_i \sqrt{r_o^2 - r_i^2} - r_i^2 \cos^{-1} \frac{r_i}{r_o} \right), \quad (2.2)$$

²E. Garelis, "Treatment of Annular Voids in Diffusion Theory," *Nuclear Science and Engineering*, Volume 12, 1962, p. 547.

³J. R. Terrall, "Particle Escape from the Ends of Cylindrical Tubes," GE-ANPD, APEX-525, March 1958.

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Fig. 2.8 - B¹⁰ worth versus B¹⁰ areal density

Fig. 2.9 - B-SS integrated worth, with respect to void versus layers of B-SS

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and have a leakage escape probability F_0 . The transverse leakage cross section is given by

$$\Sigma_L = \frac{[r_0 F_0 / (2 - F_0)] + [r_1 F_1 / (2 - F_1)]}{r_0^2 - r_1^2}, \quad (2.3)$$

assuming a constant flux across the gap.

The ZIP diffusion analysis⁴ of the Mockup No. 1A critical experiment used the Garelis void diffusion coefficient. The built-in transverse leakage calculation based on normal mode buckling produced a value of $\Sigma_L = 0.00558 \text{ cm}^{-1}$, while the value based on cylindrical tube leakage is $\Sigma_L = 0.0185 \text{ cm}^{-1}$. The effect of these two transverse gap leakage cross sections was examined in a radial 16-group S-X calculation in S_4 angular detail, using the heterogeneous core cross sections⁵ for core materials derived in the basic critical experiment. The ZIP calculated leakage yields $k_f = 1.043$, while the cylindrical tube leakage gives $k_f = 1.003$. From these results, the cylindrical tube leakage appears to be a good approximation. The Program S-X calculated values of some of the other parameters which will subsequently be measured in the Mockup No. 1A critical experiment are shown in Table 2.4.

TABLE 2.4

16-GROUP S_4 RADIAL S-X CALCULATIONS OF MOCKUP NO. 1A CRITICAL EXPERIMENT

Gap Leakage	Fission Eigenvalue, k_f	Neutron Lifetime, microseconds	Radial Power Profile		Distance Of $P_{\min}$ From Core Surface, cm	$^{238}\text{U}/^{235}\text{U}$ Fission Rate Ratio At Center Of Core
			$P_{\max}/P_{\min}$	$P_{\text{center}}/P_{\min}$		
$\Sigma_L = 0.0185 \text{ cm}^{-1}$ (cylindrical tube)	1.003	5.18	4.74	1.65	1.74	0.0747

ANALYSIS OF THE BASIC CRITICAL EXPERIMENT

Poison Ring Correlation

The general approach used in calculating the reactivity worth of poison controls in the reflector has been to focus attention on radial and azimuthal effects and to either approximate or ignore any longitudinal effects. The purpose of the worth measurement of a complete ring of core-length poison in the reflector⁶ was to establish the adequacy of such an approach with a cleanly analyzable configuration in annular cylindrical geometry. Analyses of a two-layer ring (0.0508-cm-thick) and a five-layer ring (0.127-cm-thick) of 1.05 percent borated stainless steel were performed in a radial S-X transport calculation in S_4 angular detail using 16 energy groups and the heterogeneous core cross section set.⁷ The small effect of any change in longitudinal leakage in the poison ring itself was ignored; i. e., the longitudinal leakage in the poison was taken to be the same as in the surrounding, unperturbed reflector material. Results of calculations are compared with experiment in Table 2.5. The poison ring worth was underpredicted about 14 percent. The reactivity effect of the poison segments on core longitudinal leakage was then approximated by evaluating the reactivity worth of the change in core longitudinal reflector savings obtained from matched savings ZIP diffusion calculations of the reactor with no poison and poison rings of two layers and five layers, respectively. The excellent correlation obtained warrants the continued use of this approach and indicates the importance of the effect of change in longitudinal core leakage when relatively small reflector poison worths are involved.

⁴"710 Reactor Program, Progress Report No. 12," GE-NMPO, GEMP-333, January 29, 1965, p. 32.

⁵Ibid., pp. 33-34.

⁶"710 Reactor Program, Progress Report No. 11," GE-NMPO, GEMP-323, November 30, 1964, pp. 28-29.

⁷"710 Reactor Program, Progress Report No. 12," GE-NMPO, GEMP-333, January 29, 1965, pp. 33-34.

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TABLE 2.5

COMPARISON OF 16-GROUP S_4 RADIAL S-X CALCULATIONS OF POISON RING WORTH WITH EXPERIMENT

Poison	Experiment, % $\Delta k/k$	S-X With No Correction		ΔS_z Worth From		Corrected S-X Worth, % $\Delta k/k$	Error, % $\Delta k/k$
		Experiment, For ΔS_z (Core Longitudinal Savings Change), % $\Delta k/k$	Error, %	ZIP Calculations, % $\Delta k/k$			
Two layers, 1.05% B-SS	-1.09	-0.947	-13.1	-0.170		-1.12	+2.8
Five layers, 1.05% B-SS	-1.58	-1.347	-14.7	-0.225		-1.57	-0.6

Poison Segment Correlation

Results of the analysis of the control drum mockup experiment in the 710 basic critical experiment were summarized in a previous report.⁸ In those calculations, the effective core cross sections were not corrected for resonance self-shielding, and the 5-group core cross sections used in the 2DXY⁹ analysis were weighted with the flux spectrum averaged over a core region interior to the power minimum. Consequently, the spectrum was harder than that averaged over the entire core. Longitudinal leakage in the borated stainless steel absorber segment was also neglected. So that these effects could be taken into account, the two-dimensional transport theory Program 2DXY analysis of the control drum mockup experiment was repeated.

The 5-group core cross sections utilized in the new analysis were weighted with the core-averaged flux spectrum obtained from a radial 16-group, one-dimensional transport calculation of the basic critical experiment which utilized cross sections which were corrected for resonance self-shielding.¹⁰ Transverse leakage in the borated stainless steel poison segment was assumed to be the same as in the surrounding medium; i. e., the beryllium reflector. As discussed earlier in this section of the report, the boron content of the stainless steel was increased from the nominal 1 percent to the measured value of 1.05 percent.

Results of the 5-group 2DXY analysis of the unpoisoned 710 critical experiment are compared in Table 2.6 with results of one-dimensional Program S-X calculations and with measured values. The results indicate that a 5-group 2DXY calculation overpredicts the fission eigenvalue and underpredicts the peak-to-minimum power ratio. The center-to-minimum power ratio agrees with experiment. A 2DXY $1/v$ lifetime calculation, using the $\Delta k/k$ due to a uniform poisoning with 3×10^{17} atoms/cm³ of B¹⁰, underestimates the prompt neutron generation time by 12 percent.

TABLE 2.6

COMPARISON OF RESULTS FROM TRANSPORT THEORY CODES

Method	Fission Eigenvalue, k_f	Prompt Neutron Generation Time, ℓ , microseconds	Power Ratios		Distance Of Power Minimum From Core-Reflector Interface, cm
			Center-To- Minimum	Peak-To- Minimum	
Experiment	1.000	4.0	1.16 ^a	4.77 ^b	3.1
2DXY, 5-group, S_4	1.009	3.5	1.16 ^c	2.56 ^c	2.1 - 3.2 ^d
S-X, 16-group, S_4	1.017	4.2	1.17	4.55	3.1

^aFuel rod scanning data.^bHomogeneous cell catcher foil data.^cRatios of fission neutron production rate in the self-shielded core material.^dSpatial interval within which power minimum occurs.⁸"710 Reactor Program, Progress Report No. 11," GE-NMPO, GEMP-323, November 30, 1964, pp. 35-37.⁹J. Bengston, et al., "2DXY Two-Dimensional Cartesian Coordinate S_n Transport Calculation," Aerojet General Nucleonics, AGN TM-392, June 1961.¹⁰"710 Reactor Program, Progress Report No. 12," GE-NMPO, GEMP-333, January 29, 1965, pp. 33-36.~~CONFIDENTIAL~~

A comparison of calculated and measured reactivity worths of B-SS segments in the 710 basic critical experiment reflector at various radial positions is shown in Table 2.7. The comparison indicates that a 5-group 2DXY analysis underpredicts the worth of the B-SS segments. The underprediction ranges from about 1 percent for position "a" to 12 percent for position "c"; i. e., it increases with distance from the core-reflector interface. These results have not been corrected for the effect of changes in longitudinal core leakage. This correction should decrease the underprediction in all cases.

TABLE 2.7
REACTIVITY WORTH OF 1.05 WEIGHT PERCENT
BORATED STAINLESS STEEL SEGMENTS

B-SS Radial Position	Reactivity Worth, % $\Delta k/k$		% δ^a
	Measured	Calculated	
a	-0.346	-0.342	- 1.2
b	-0.337	-0.311	- 7.7
c	-0.204	-0.179	-12.3
a	-0.346	-0.279 ^b	-19.4 ^b

$$^a \% \delta = \left(\frac{\% \Delta k/k_{\text{calc}} - \% \Delta k/k_{\text{meas}}}{\% \Delta k/k_{\text{meas}}} \right) \times 100.$$

^b Transverse leakage neglected in the borated stainless steel absorber.

To assess the worth of the longitudinal leakage in the absorber segment, calculations were also performed in which axial leakage was neglected entirely in the absorber at radial position "a." This caused an 18 percent decrease in the calculated worth of the absorber segment. Clearly, longitudinal leakage in the poison cannot be neglected in control drum calculations and, when the absorber segments are not too large, the use of longitudinal leakage of the surrounding unperturbed medium seems to be justified as pointed out under "Poison Segment Correlation."

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3. EXPERIMENTAL REACTOR PRELIMINARY DESIGN AND DEVELOPMENT

Thermal and stress analyses of the initial configuration of the Experimental Reactor, using the preliminary heating rates, were completed. The revised heating rates are being computed, using the Monte Carlo techniques; these will be completed during the next report period and will be used to determine the temperature distributions throughout the Experimental Reactor. Simplifications in the analog representation of the Experimental Reactor were evaluated and applied.

INITIAL CONFIGURATION

Detailed analysis and evaluation of specific areas were performed on the initial configuration¹ of the Experimental Reactor to:

1. Investigate alternative designs that would reduce mechanical complexities.
2. Reduce the need for future development testing by evaluating potential deviations and measuring their perturbing effects.
3. Reduce development costs and improve system reliability through design simplifications.
4. Evaluate component design from a manufacturing cost and producibility standpoint.

This work is being done concurrently with the thermal, nuclear, and stress analyses of the initial configuration being performed for incorporation in a preliminary design.

Reflector Module

An example of the alternative designs being investigated is the circular reflector module shown in Figure 3.1. This design emphasizes mechanical simplicity and improves the effectiveness of the reflector structural material. Simplification of the coolant flow circuitry is also achieved. To achieve a reflector volume as large as possible within the constraints of the core diameter and the pressure vessel diameter, the drum diameter has been increased from 6.26 to 7.94 inches and the number of drums has been reduced from 12 to 10. This design would improve drum worth at a minimum cost in total reactivity by reducing the structural volume fraction to a minimum. Small diameter filler modules may be used to reduce neutron leakage between drums if necessary.

Pressure Vessel Main Flange

Previous stress analyses of the pressure vessel² considered an axisymmetric model in which the main flange was assumed to act as a unit ring. A more detailed analysis has now been performed to determine the rotation of each flange; this rotation results in decreased sealing pressure and increased secondary bending in the hub and adjacent shell section. The metallic O-ring seals (shown in Figure 3.2), deform plastically under preloading to obtain sufficient sealing pressure. When the vessel is pressurized, the flange

¹"710 Reactor Program, Progress Report No. 12," GE-NMPO, GEMP-333, January 29, 1965, pp. 37-38.

²"710 Reactor Program, Progress Report No. 11," GE-NMPO, GEMP-323, November 30, 1964, pp. 50-51.

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Fig. 3.1 – Alternate circular reflector module (Dwg. 221R340)

Fig. 3.2 - Pressure vessel main flange separation at inner gasket (due to flange rotation)

faces tend to separate by rotation due to eccentricity between the shell membrane and the bolt circle diameters. Sufficient elasticity is available in the O-rings to maintain sealing pressure if the net flange separation can be limited to not more than 0.004 inch.

Investigations were based on the fact that flange and hub temperature levels are independent of their axial length since virtually all heat is transferred radially. Figure 3.2 shows flange separation as a function of flange and hub geometry for the design pressure of 1040 psi, operating pressure of 840 psi, and no pressure preload condition.

Maximum hub stresses are shown in a similar manner in Figure 3.3. Figure 3.4 shows the overall height of flange and hub versus the net flange separation for the acceptable stress values that will be used in selecting a forging thickness and height. The relationship between bolt size, number of bolts, and bolt stress used in selecting the arrangement of the initial configuration is shown in Figure 3.5.

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Fig. 3.3 - Pressure vessel main flange hub stress

Deviation Study

A deviation study is being performed to evaluate the temperature and stress within various components as the controlling parameters depart from the nominal values used in the initial analyses. Deviations fall into one or more of the following categories:

1. Uncertainties - material properties, assumptions, correlations from available data, unmeasured quantities, etc.
2. Variations - thermally and mechanically produced geometry changes, properties that depend on time of exposure, etc.
3. Tolerances - dimensional, operational sensing and control, material chemistry, etc.

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Fig. 3.4 – Overall flange and hub height versus net flange separation

The results of this study will be used to evaluate off-design operation and to help in determining the need for physical testing of development hardware.

An example of this type of information is shown in Figure 3.6 which shows the vessel wall temperature variation at a selected axial point with changes in cooling gas bulk temperature, heat generation rate, and heat transfer coefficient. These curves may be used in estimating shell temperatures at any position on the cylindrical vessel having known local heat generation rate, bulk coolant temperature, and channel geometry. Conversely, for a calculated temperature at a point, a tolerance range can be estimated.

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Bolt material – SA-193B16, design stress intensity = $S_m = 28,850$ psi 750°F.

Reference: Section III Nuclear Vessel (ASME Code)
Table N-422, page 32.

Total bolt load – 1,659,000 pounds.

Bolt circle – 43.38-inch diameter.

⊕ Quantity and size selected – 44 1.50 – 8 UN bolts.

Fig. 3.5 – Pressure vessel flange bolt loading analysis

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Fig. 3.6 - Variation of pressure vessel temperature with deviations of heat transfer coefficient (h_i), cooling gas bulk temperature (T_b), and heat generation rate (q''')

Assembly Sequences

The mechanical design of the 710 Experimental Reactor is such to permit remote removal and replacement of major component assemblies while allowing uninterrupted aftercooling throughout the procedure. The proposed procedures to meet these requirements are shown in the sequences in Figures 3.7 through 3.9. The initial assembly procedure, done manually, is shown schematically in Figure 3.7. Remote disassembly and subsequent reassembly procedures are shown in Figures 3.8 and 3.9, respectively. The latter are following a planned shutdown for the replacement of a component after a period of normal reactor operation. Although preliminary, the sequences proposed here appear to satisfy all requirements and are consistent with the current design.

THERMAL DESIGN AND ANALYSIS

The steady-state temperature, flow, and pressure drop calculations for the components of the Experimental Reactor were completed to the extent appropriate with the preliminary heating rates. Transient temperatures following a reactor scram were determined for the pressure vessel.

Reactor Flow Analysis

An analytical model of the flow passages from the pressure shell coolant inlet to the core exit plenum is being constructed to facilitate reactor design point and off-design point operating condition studies.³ Single channel computational methods used previously⁴ are

³S. C. Skirvin, "Compressible Flow Network Computer Program," GE-NMPO, APEX-661, July 1961.

⁴"710 Reactor Program, Progress Report No. 12," GE-NMPO, GEMP 333, January 29, 1965, pp. 38-44.

STEP 1

1. Mount forward support structure on fixture having access above and below.
2. Install reflector modules using manual assembly tool.
3. Lock all drums to scram position.

STEP 2

1. Mount assembly of support structure and reflector modules on LPT fixture.
2. Mount actuators to support brackets.
3. Use manual fuel cartridge handling tool to raise and accurately position center cartridge.
4. Unlock drums for reactivity checks.
5. Install all cartridges using existing surfaces for orientation.
6. Lock drums in scram position.

STEP 3

1. Remove actuators from LPT fixture.
2. Mount pressure head on assembly fixture.
3. Install each actuator.
4. Install clamping rings.

STEP 4

1. Lower mixing device into pressure vessel

STEP 5

1. Mount elevator guide to top flange of vessel.
2. Lower reactor assembly into pressure vessel.

STEP 6

1. Unlock drums.
2. Install ring seals.
3. Adapt elevator guide to pressure vessel head and mount to top flange of shell.
4. Lower pressure vessel head.
5. Install studs and lower nuts.

STEP 7

1. Install upper nuts.
2. Pretension nuts.
3. Make external connections.

Fig. 3.7 – Preliminary manual assembly procedure for 710 Experimental Reactor

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STEP 1

1. Block main loop downstream of heat exchanger.
2. Open primary inlet to ambient air.
3. Exhaust through bypass inlet approximately 1 lb/sec outside air.
4. Remove main flange nuts using pretensioner.

STEP 2

1. Install elevator guide on main flange.
2. Remove head assembly.
3. Lock drums in scram position.

STEP 3

1. Install torus manifold on module pipe extensions.
2. Adapt elevator guide to forward support structure.
3. Remove reactor assembly.

STEP 4

1. Install reactor assembly in low pressure chamber.
2. Remove torus manifold.

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Fig. 3.8 - Preliminary remote disassembly procedure for 710 Experimental Reactor

STEP 5

1. Remove inner dome.
2. Install torus manifold if necessary.
3. Disengage fuel cartridge and lower into water cooled cask.
4. Install tube sheet plugs at vacancies.

STEP 6

1. Mount low pressure dome on support structure.
2. Plug module support tubes or remount torus manifold on module support extensions.
3. Reverse aftercooling flow by venting low pressure chamber and exhausting from low pressure dome, torus manifold or both.
4. Open access in low pressure chamber and remove fuel element cask.

STEP 7

1. Place reactor assembly on fixture.
2. Remove torus manifold.
3. Plug module support tubes.
4. Attach module removal tool and disconnect module from support structure.
5. Lower module into cask.
6. Plug vacant hole in support structure.

ACTUATOR

1. For manual replacement of electrical gear without system depressurization, remove 4 bolts and cover plate.
2. For remote replacement of actuator and upper drive shaft, break flange clamp and lift out actuator.

MIXING DEVICE

1. Lift out and replace entire assembly.

Fig. 3.8 (Cont.) - Preliminary remote disassembly procedure for 710 Experimental Reactor

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STEP 1

1. Place reactor assembly with low pressure dome in place on assembly stand.
2. Raise module through support structure using assembly tool.
3. Fasten module to support structure.
4. Install torus manifold if necessary.

STEP 2

1. Install reactor assembly in low pressure chamber.
2. Remove low pressure dome.
3. Remove tube sheet plugs and raise fuel cartridge in place using latching tool.
4. Install reactor inner dome.
5. Install torus manifold.

STEP 3

1. Mount elevator guide to main flange of pressure vessel.
2. Lower reactor assembly in place.
3. Unlock drums.

STEP 4

1. Adapt elevator guide for pressure head.
2. Lower pressure head in place.
3. Remove elevator guide, install nuts on studs, and pretension.

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Fig. 3.9 – Preliminary remote re-assembly procedure for 710 Experimental Reactor

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not adequate for detailed determination of the flow distribution and consequent aerothermal characteristics because of the multiplicity of parallel flow passages in the Experimental Reactor. The multiple flow passage calculation is designed to properly account for the effects of each plenum in the system as well as exit and entrance losses associated with each of the parallel channels which connect the plenums.

Active Core

Analysis Method for Temperature Variations Within a Fuel Element - The preparation of heat transfer computer code (TEMPER A), which utilizes linearized flow temperature relationships,⁵ was completed. This analysis technique will account for (1) internal heating and radial conduction within a fuel element, (2) thermal radiation and gas conduction resistance between adjacent fuel elements, (3) flow distribution as a function of channel heat addition, (4) heat losses from the surface of the core by conduction and radiation to surrounding radiation shields and/or cooled peripheral structure, and (5) fuel element gap axial variations at operating temperatures.

Core Support Structure

Calculations were made on the core support structure and nosepiece assembly using a three dimensional analytical model. The model consisted of a 30-degree angular sector and was divided into 10 axial, 3 radial, and 3 circumferential nodes. The preliminary volumetric heating rates were used in the study with an assumed linear axial variation and no radial variation.

A plot of axial average temperature distribution in the nosepiece assembly and in the core support structure is shown in Figure 3.10. The coolant temperature variation through the core support structure axial length is also presented. The maximum temperature in the core support structure material and in the nosepiece assembly occurs at the aft face of the structure due to the higher heating rates and to heat conducted away from the core.

The general shape of the temperature curves is the result of a combination of internal heat generation, heat conducted from the core, and the axial variation of heat transfer coefficient. It can be seen from Figure 3.10 that the flow area has a relatively large variation depending upon axial location.

The data presented in Figure 3.10 are based on the assumption that the coolant flow rates in the parallel channels of the nosepiece are proportional to flow area. Since the correct flow split is a function of the amount of heat added to each of the parallel channels as well as channel configuration, the temperatures calculated in the nosepiece and core support structure will be refined when the new heating rates in these materials become available.

Pressure Vessel

The preliminary design⁶ for the coolant passage geometry of the pressure head, between the actuator nozzles and the reactor centerline, was evaluated in terms of its effect on pressure vessel temperatures. Although the feasibility of maintaining a nearly constant value of heat transfer coefficient while bleeding off flow, as explained previously,⁶ was verified, pressure vessel temperatures increased gradually from approximately 700°F near the actuator nozzles to approximately 800°F near the reactor centerline. This coolant passage will be redesigned so that the heat transfer coefficient will increase as the coolant flow moves toward the reactor centerline, thereby maintaining essentially a constant pressure vessel surface temperature. This redesign will still employ the same

⁵Ibid., p. 44.

⁶Ibid., p. 45.

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Fig. 3.10 – Longitudinal temperature distribution along the core support structure

flow-bleedoff concept necessary to minimize pressure losses. However, both the local flow bleed rate and the coolant passage gap will be adjusted to provide the desired nearly constant pressure vessel surface temperature distribution over the pressure head. This redesign will be initiated when the new heating rates are available.

The pressure vessel temperature distribution was evaluated for a single transient condition: a step change from full to zero reactor power. In this evaluation, the pressure vessel coolant inlet temperature was maintained constant at 523°F^7 throughout the transient duration. The analysis verified the expected lag of the flange temperatures behind those of the shell in the initial part of the transient due to the relatively greater flange mass. The temperature differences between the outer flange surface and the pressure shell are maximum at both ends of the flange at approximately the same time, 120°F^8 at the forward end

⁷Estimated reactor inlet temperature at the current design point for steady state operation.

⁸These are differences in point values, not the differences between the mean flange and mean shell temperatures during transient which are roughly a factor of two lower.

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of the flange and 130°F^8 at the rear end, both near the end of a 10-minute time period in the transient. These values may be compared with a nominal 30°F^8 temperature difference during steady-state operation. The particular performance of the flange-shell region in this transient is shown on Figure 3.11.

The results of this transient temperature study will be employed in the evaluation of pressure vessel stresses during transient operations. It is not planned to evaluate other transient cases until the new heating rates become available.

Fig. 3.11 – Pressure vessel temperature distribution after step change from full to zero reactor power

Pressure Vessel Nozzles

Analytical models are being formulated to predict the three-dimensional temperature distributions in the pressure vessel nozzles at the coolant and actuator entry points. Three-dimensional detail is deemed necessary to provide the temperature distribution needed to accurately evaluate the thermal stresses, as well as absolute temperature levels, which are required to guarantee integrity. These nozzles are not symmetrical with the two-dimensional thermal analysis model for the overall pressure vessel. However, the two-dimensional model will be utilized to provide boundary temperatures for the three-dimensional nozzle models.

Analysis of the pressure-vessel-actuator entry nozzles is complicated by bleed from the primary flow for actuator cooling. In addition, in the region of the actuator entry nozzles, the pressure vessel proper crosses the nozzle at an acute angle, i. e., the

⁸Ibid.

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pressure vessel does not form a right angle with the nozzle centerline. These design characteristics dictate a three-dimensional analysis.

Analysis of the pressure-vessel-coolant entry nozzles is somewhat less complex. There is no flow bleed and the pressure vessel proper forms a right angle with the nozzle centerline. However, in the case of the primary flow inlet nozzle, the aft portion is partially cooled by the bypass flow. Again, a three-dimensional model is deemed the logical approach. The analysis of the bypass flow entry nozzles will be performed using the primary flow nozzle analytical model with slight modifications.

Mixing Device

An investigation of the general theory of mixing hot and cold streams of similar gases for various flow geometries has been completed. Prior to this study, work on the mixer design was concentrated on the general arrangement of two particular geometries without regard for mixing length and pressure loss effects. These two arrangements were an external mixing device (EMD) and an internal mixing device (IMD). The EMD would accept reactor discharge and bypass coolant flow and place them in parallel adjacent streams; the actual mixing would then be accomplished downstream. The IMD would mix the reactor discharge and bypass gas by introducing the bypass flow through perforated plates into the reactor discharge flow, thus performing some of the flow mixing totally within the mixing region.

It was found during the current investigation that:

1. For the EMD: (a) a downstream mixing length, measured from the mixer exit, of at least 20 reactor-discharge-flow duct diameters is required to mix the flows to a temperature variation of $\pm 200^{\circ}\text{F}$ and (b) the reactor-discharge-flow velocity should be appreciably greater than that of the co-flowing bypass flow in the mixing region aft of the mixer exit.
2. No method was identified or theory developed which would permit the rigorous analysis of the mixing process associated with a series of co-flowing hot and cold gas streams flowing side by side (characteristics of the EMD concept).
3. No method was identified or theory developed which would permit the rigorous analysis of the mixing process associated with many adjacent jets of a cold fluid injected into a deflecting hot fluid (characteristic of the IMD concept). A plausible starting point was identified for the analysis of this flow geometry.

Simple equations were also developed which will permit a general sizing of the EMD and IMD mixer concepts.

The design of the mixer is planned to be done experimentally using this information to guide the experimental work.

Impurities

Work was initiated on applying the data from the analytical study⁹ of nitrogen uptake by the tantalum cladding to evaluate design conditions and off-design points in the 710 Experimental Reactor. Major refinements which may be factored into the analysis are (1) generalization of the analysis to include any impurity uptake in any cladding material so long as the rate equations involved differ only by the magnitude of their coefficients, (2) the axial depletion of the impurity in the coolant along the channel as the impurity is removed by the cladding, (3) the effect of the axial variation in the coolant boundary layer diffusion thickness¹⁰ and the effect of this parameter on the diffusion rate from the coolant to the cladding surface, (4) the effect of previous impurity uptake in the cladding upon the

⁹"710 Reactor Program, Progress Report No. 12," GE-NMPO, GEMP-333, January 29, 1965, p. 61.

¹⁰T. K. Sherwood and R. L. Pigford, *Absorption and Extraction*, McGraw-Hill, New York, 1952, p. 58 et seq.

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time-dependent impurity concentration entering the reactor, and (5) the effect of small arbitrary leaks of impurities into the reactor coolant.

Pilot Test Loop Analysis

In support of the design and fabrication of the dynamic test facility, a study of two methods of electrically heating the test specimen was undertaken. The object of the study was to compare induction heating of the test specimen with resistance heating and to see how these methods of electrical heating differ from that of nuclear heating in the 710 Experimental Reactor.

To facilitate comparison, calculations were made for a single coolant channel with longitudinal heat-flux distributions that would simulate induction and resistance heating. Induction heating was simulated by imposing a constant heat flux along the length of the channel. Resistance heating was simulated by a temperature-dependent heat flux that was obtained from the electrical resistivity of the reactor core materials.

For this study, the mass flow, inlet and exit coolant temperatures, and hydraulic diameter were maintained at the present reactor design values.

A comparison between the longitudinal temperature distribution calculated for the constant heat flux (induction heating) and the temperature-dependent heat flux (resistance heating) is shown in Figure 3.12. Also shown is the present longitudinal distribution of the 710 Experimental Reactor. It can be seen from these curves that electrically heated specimens tend toward lower temperatures at the inlet of the specimen and higher temperatures near the rear of the specimen.

Additional work is being done to investigate methods of combining resistance heating with a heat source at the exterior of the specimen. This combination of heat sources will tend to increase front end temperatures and will more nearly simulate reactor design conditions.

NUCLEAR ANALYSIS AND DESIGN

Nuclear Heating Analysis

The detailed nuclear heating calculation for the Experimental Reactor reported previously^{11,12} was continued during this report period. The two-dimensional transport theory calculation resulted in a reactor multiplication factor of 1.0374 at operating temperature (excluding the Doppler coefficient, but including neutron temperature and estimated expansion effects). Using gamma production cross sections previously prepared, gamma sources for 448 annular zones, in 11 gamma groups, were computed and converted into a form which would be directly useable in Auxiliary Program 20-0.¹³ Table 3.1 shows the gamma sources, in Mev per fission, for the 11 gamma groups in the four source zones used in the Monte Carlo analysis. Figure 3.13 shows the four zones.

Program 20-0 was used to generate, and store on magnetic tape, the parameters for each source particle traced by Monte Carlo Program 18-1.^{14,15} Separate tapes were prepared for core and secondary gamma ray sources, and the 11 source energy groups were

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

¹²"710 Reactor Program, Progress Report No. 12," GE-NMPO, GEMP-333, January 29, 1965, pp. 50-53.

¹³J. E. MacDonald and J. T. Martin, "Shielding Computer Program 20-0," APEX-610, May 8, 1961.

¹⁴J. P. Yelch and J. E. MacDonald, "Addenda to GEMP-102 Describing Program 18-1," GE-NMPO, GEMP-272, January 1964.

¹⁵J. E. MacDonald, J. T. Martin and J. P. Yelch, "Specialized Reactor-Shield Monte Carlo Program 18-0," GE-NMPO, GEMP-102, October 1961.

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Fig. 3.12 - Effect of longitudinal heat flux distribution on coolant channel surface temperature

Fig. 3.13 - Gamma source zones used for Monte Carlo analysis

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TABLE 3.1

GAMMA SOURCES USED IN THE MONTE CARLO NUCLEAR HEATING ANALYSIS

Energy Group	Lower Energy, Mev	Upper Energy, Mev	Gamma Sources, ^a Mev per fission				
			Total Reactor	Core Zone	1st Front Zone	2nd Front Zone	Side Zone
1	0.01	0.40	2.5829	2.0234	0.1389	0.0523	0.3683
2	0.40	0.90	4.4183	4.2688	0.0321	0.0069	0.1105
3	0.90	1.35	3.7222	3.7013	0.0058	0.0029	0.0122
4	1.35	1.80	3.2279	3.1413	0.0257	0.0115	0.0494
5	1.8	2.2	2.2688	2.2404	0.0078	0.0022	0.0184
6	2.2	2.6	1.7030	1.6859	0.0058	0.0010	0.0103
7	2.6	3.0	1.1407	1.1140	0.0064	0.0015	0.0188
8	3.0	4.0	1.7857	1.6865	0.0343	0.0051	0.0598
9	4.0	5.0	0.8121	0.7648	0.0199	0.0024	0.0250
10	5.0	7.0	0.6997	0.5133	0.0487	0.0168	0.1208
11	7.0	10.0	0.5600	0.0268	0.1204	0.0597	0.3531
Total	-	-	22.9213	21.1665	0.4458	0.1624	1.1468

^aSee Figure 3.13 for location of source zones.

considered individually so that the Monte Carlo results can be normalized to any source energy spectrum if the source spatial distribution is the same.

The Experimental Reactor, including the top head of the pressure vessel, was included in Program 18-1 for the gamma heating and escape analysis. A series of Monte Carlo calculations was made of the core gamma ray sources in the 3- to 4-Mev energy range to establish an effective region weighting scheme. These weights were adjusted to maintain nearly uniform particle current counts throughout the assembly and to achieve reasonable particle absorption counts in all regions of particular interest. These weights effected splitting or Russian roulette at region boundary crossings. After an acceptable weighting scheme was obtained for the 3- to 4-Mev core source, it was used for the remaining groups of core sources. At least two runs of 5000 histories each were completed for each core source group.

Only straightforward Monte Carlo calculations were made for secondary sources outside the core. Previous experience indicates that this should yield acceptable results with considerably less effort than would be required if splitting were used. At least two runs of 5000 histories each in secondary zone A of Figure 3.13 and two runs of 2500 histories each in zones B and C were completed for each energy group. The results of all these calculations included the volume of each region and the energy deposited per unit volume in each region.

Monte Carlo Program 18-1 was also used for analysis of neutron fluxes and neutron energy deposition rates in each region, and neutron leakage from each region, of the Experimental Reactor. Source particle tapes were prepared for U²³⁵ fission neutrons in two energy ranges: 0.1 to 5 and 5 to 14 Mev. The spatial distributions were defined by the fission density distributions from TDC. Three runs of 5000 histories each were made for the low-energy sources, and two runs of 7500 histories each were made for the high-energy sources. In addition to the output described above for the gamma ray calculations, the results included fluxes in 23 energy groups for each region.

An auxiliary code was prepared to process the output from all the Monte Carlo runs into a form for direct use in the thermodynamic analyses. Specifically, this code will combine all the secondary sources of heat, i. e., gamma and neutron; perform an analysis of the variance of the runs to establish confidence limits; and convert the results to English units.

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Two-Dimensional Analysis of Control Drum Worth

The worth of the present control system of the Experimental Reactor was analyzed using a two-dimensional X-Y geometry transport theory code (2DXY) in five energy groups. The method used closely paralleled that used to analyze the poison absorber measurements made in the basic critical experiment.¹⁶ The calculated worth was 5.5 per-cent Δk . Figure 3.14 shows the quarter of the core containing two full reflector modules and two half modules, and shows the type of grid structure and compromises required by the geometry. The lines show the actual reactor layout, and the shaded areas show the grid structure needed by 2DXY. The control poison was composed of 65 volume percent fully enriched boron carbide and 10.6 volume percent Type 304 stainless steel.

Five-group, transverse-leakage corrected cross sections (with group energy boundaries set at 0.4 Mev, 0.1 Mev, 61.44 ev, 0.251 ev, and 0.0737 ev) were processed for use in the analysis.

Fig. 3.14 - Control drum analysis nuclear model

¹⁶"710 Reactor Program, Progress Report No. 11," GE-NMPO, GEMP-323, November 30, 1964, pp. 35-39.

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Control Drum Parametric Calculations

The control drum parametric studies mentioned previously¹⁷ were completed. One-dimensional diffusion theory calculations were carried out, to evaluate the control drum worth dependence on: (1) the number of drums, the drum diameter, and the radial reflector thickness; and (2) the structural material in the radial reflector and control module.

The nuclear model of the reactor and the calculational method were both the same as in previous analyses.^{18,19} A sector of the radial reflector consisting of a stationary beryllium section and a rotating drum with the absorber segment is shown in Figure 3.15. The radial dimensions of the reflector annular zones are given in Table 3.2. The rotating reflector control drum absorber segments were assumed to be boron carbide with 70-percent B¹⁰ enrichment.

Fig. 3.15 - A $\left(\frac{360}{N}\right)$ -degree radial reflector sector

¹⁷"710 Reactor Program, Progress Report No. 12," GE-NMPO, GEMP-333, January 29, 1965, p. 52.

¹⁸"710 Reactor Program, Progress Report No. 8," GE-NMPO, GEMP-287, May 29, 1964, pp. 47-49.

¹⁹"710 Reactor Program, Progress Report No. 9," GE-NMPO, GEMP-301, July 31, 1964, p. 60.

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TABLE 3.2
DIMENSIONS OF REFLECTOR AND HOMOGENIZED CONTROL ANNULAR ZONES

No. Of Drums (N)	Dimensions, in.									
	R_0^a	R_1	R_2	R_3	R_4	R_5	R_6	R_7	R_8	R_9
12	8.255	9.617	10.127	10.345	10.485	15.580	16.605	16.823	17.128	17.237
11	-	9.642	10.142	10.376	11.862	16.321	17.431	17.666 ^b	17.666	17.791
10	-	9.667	10.167	10.426	12.124	17.269	18.480	18.739	18.739	18.890
9	-	9.692	10.192	10.476	12.474	18.533	19.880	20.164	20.164	20.339
8	-	9.717	10.217	10.526	12.970	20.298	21.830	22.139	22.139	22.334

^a R_0 = core radius.

^bFor less than 12 drums, $R_7 = R_8$.

The control system worths as calculated for these parametric studies are used only to establish the relative significance of the various changes and are not intended as calculations of absolute worth. The determination of the absolute worth of the control system in any of the reflector configurations would be done by using the two-dimensional transport calculation described above.

The results of the calculations in which the number of drums, the drum diameter, and the radial reflector thickness were the variable parameters are shown in Table 3.3. These data indicate that the worth of the drums changes from 4.1 to 3.8 percent Δk , i. e., it decreases by about 7 percent when the number of drums is reduced from 12 to 8.

TABLE 3.3
DRUM WORTH VERSUS THE NUMBER OF DRUMS,
THE DRUM DIAMETER, AND
REFLECTOR THICKNESS

No. Of Drums	Drum Radius, in.	Be Reflector Thickness, in.	Drum Worth, -% Δk
12	3.130	7.501	4.12
11	3.528	8.024	4.18
10	4.027	9.082	4.07
9	4.702	10.472	3.93
8	5.652	12.422	3.81

Results of studies of the influence of reflector and control module structural materials on control drum worth are summarized in Table 3.4. In these calculations performed for a configuration of 12 drums, the Type 304 stainless steel structure, used in the present design of the 710 Experimental Reactor, was replaced in all radial reflector zones, except in the homogenized annular control regions, with Fe, Nb, Mo, and Zircaloy. Calculations were also carried out with the above materials replacing stainless steel in the radial reflector zones and the homogenized control annuli. To provide an estimate of the upper limit of drum worth, which occurs in the absence of neutron absorption in the reflector competing with the absorption in the drum poison segments, a calculation was performed with beryllium replacing the stainless steel structure.

The drum worth undergoes little change when either iron or niobium replaces Type 304 stainless steel; the worth decreases by about 25 to 28 percent when molybdenum replaces stainless steel. The substitution of Zircaloy for stainless steel as the structural material in the reflector causes about 30 percent increase in drum worth. Drum worth increases by approximately 45 percent when stainless steel structure is replaced with beryllium.

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TABLE 3.4
THE EFFECT OF REFLECTOR AND CONTROL DRUM MODULE
STRUCTURAL MATERIAL ON DRUM WORTH

Structural Material		k,	Drum Worth,	
Radial Reflector	Control Annuli	Drums Out	$-\% \Delta k^a$	$\% \delta^b$
304 SS	304 SS	1.0218	4.13	-
Iron	304 SS	1.0209	4.02	- 2.7
Iron	Iron	1.0209	4.05	- 1.9
Niobium	304 SS	1.0217	4.19	+ 1.5
Niobium	Niobium	1.0214	4.16	+ 0.7
Molybdenum	304 SS	1.0119	3.08	-25.4
Molybdenum	Molybdenum	1.0118	2.96	-28.3
Zircaloy	304 SS	1.0342	5.39	+30.5
Zircaloy	Zircaloy	1.0339	5.36	+29.8
Beryllium	304 SS	1.0423	5.97	+44.6

^aThe worth of 12 control drums.

$$b\% \delta = 100 \left(\frac{\% \Delta k_{S.M.} - \% \Delta k_{304 SS}}{\% \Delta k_{304 SS}} \right)$$

REACTOR CONTROLS

Test Loop Simulation

The analog simulation of the test loop is being used to determine the effects of certain component failures upon the loop operating parameters, the other loop components, and the reactor. The performance of the different control systems during the transient conditions which will be induced in the system by these perturbations will also be evaluated. Preliminary analysis has indicated the importance of the time relationship of these interactions due primarily to the time required for a pressure change to manifest itself upon the other loop components. The physical system is being analyzed to determine a mathematical model for the pressure coupling between components.

Reactor Simulation

A three-stage simulation of the reactor thermal characteristics was operated simultaneously with a single-stage simulation to provide a comparison of the dynamic responses. Although the accuracy of a three-stage simulation is somewhat better than single-stage simulation, the difference in dynamic behavior is so slight as to be insignificant. Henceforth, a single-stage simulation will be used in studying the overall test loop performance.

A stage representing the side and front reflector was added to both the single- and three-stage simulation and again the transient behavior of the two simulations was compared with negligible differences. The addition of the reflector stage did result in a considerable change in the effect of a change in reactor inlet temperature. The reflector acts as a filter to changes in inlet temperature due to the mass of the reflector, which creates a very long time constant (~140 seconds) for changes in inlet temperature. For changes in reactor power and coolant flow, both simulators had a time constant of approximately 22 seconds.

Because of the long time constant of the reflector, the core inlet temperature is fairly constant and changes very slowly. Therefore, the compensation for the core inlet temper-

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ature²⁰ was removed from the control system, thus permitting some simplification in the simulation of the system.

The primary purpose for simulating the reflector was to study the effects of temperature coefficients of reactivity. The most severe combination of temperature coefficients which could be practically postulated was then programmed on the computer and the performance of the control system analyzed. There were no control problems indicated. The effect of a positive Doppler coefficient associated with the fuel particle has yet to be analyzed. In the analysis the Doppler coefficient will be divided into a positive effect due to the Doppler broadening in the U^{235} and the negative contribution of the W, U^{238} , and Ta.

Reactor Safety and Indication

A preliminary safety philosophy was formulated for the 710 Experimental Reactor. Based upon this philosophy, a parameter and action list is being evolved and a block diagram prepared.

²⁰"710 Reactor Program, Progress Report No. 8," GE-NMPO, GEMP-287, May 29, 1964, pp. 49-51.

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4. TEST FACILITIES AND PILOT LOOP DESIGN AND EVALUATION

Advanced engineering studies were continued on the test equipment and the facility requirements for the 710 Experimental Reactor power test. Pilot loop procurement was continued; the specifications were issued for all major components of the pilot loop and sub-loop. The use of a commercial stud tensioning device to remotely make up the pressure vessel flange was evaluated.

FACILITIES

Work continued on the layout of the power test facility, to provide an economical structure with a functional arrangement. The facility, as presently planned, is shown in Figures 4.1 and 4.2. These drawings reflect a combination of the best features of plans No. 1 and 2 shown previously.¹ The lower cold equipment room contains most of the equipment for the auxiliary systems which are connected to the power test dolly through the lower coupling station. This arrangement results in the shortest piping runs between the equipment and the coupling station. The upper cold equipment room contains the equipment for the auxiliary systems which connect to the primary power test loop in the warm equipment area. Overhead crane service is available for the warm equipment room and the upper cold equipment room. For crane service in the lower cold equipment room, equipment will be moved to the hatch shown on the drawings.

The wall thickness of the test cell was increased from 30 inches to 42 inches of normal concrete to reduce the combined neutron and gamma dose from the reactor to less than 50 millirem during operation at design power levels. The crane hook clearances were increased from 35 feet to 40 feet to facilitate manual maintenance of the control rod actuators and to provide clearance over the shielding wall between the warm and cold equipment rooms.

The extent of electrical modifications necessary for the power test facility was defined. The power requirements exceed the present capacity by 2,000 kva, necessitating another substation. The present power lines from the TSF area appear to be adequate for the increased load, and the 750-kva emergency diesel - generator will require only wiring changes.

TEST EQUIPMENT

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

Primary Power Test Loop Cooling System

Further evaluations of a more detailed nature were made of the primary power test loop. The connections of the loop with secondary gas systems were given special consideration.

¹"710 Reactor Program, Progress Report No. 12," GE-NMPO, GEMP-333, January 29, 1965, pp. 58-59.

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Fig. 4.1 - Section view of the planned power test facility (Dwg. 692E374)

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Fig. 4.2—Plan view of the planned power test facility (Dwg. 692E376)

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The secondary systems consist of (1) a dolly cooling system, (2) an emergency gas cooling system, (3) a vacuum system, (4) a scavenging system, (5) a neon storage system, and (6) a purification system. Figure 4.3 is a schematic diagram of the primary power test loop cooling system showing tentative connection points with the secondary systems.

Dolly Cooling System - The dolly cooling system will be used to remove reactor after-heat during transit between the power test facility and the hot shop at the TSF area. As now planned, two 0.5-lb/sec oil-free compressors will be mounted on the dolly, each with a separate diesel - generator power supply. During normal transit operations, only one compressor will be required to maintain the fuel element temperatures below 500°F after 100 hours of power operation at 10 Mw and 24 hours of decay time. The second compressor will be a backup.

Fig. 4.3 - Schematic diagram of the primary power test loop cooling system (Dwg. 692E377)

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Emergency Cooling System - An emergency cooling system is necessary in the event of failure of the primary cooling system. Studies have resulted in an emergency system that incorporates inert gas blow-down, air blow-down, and air compression systems. This arrangement would be capable of providing emergency cooling in the event of an accidental rupture of the primary loop, a simultaneous failure of the normal and emergency power systems, inadvertent closure of a valve in the reactor flow line, or rupture of a system through which water would be introduced into the neon coolant.

Vacuum System - A vacuum system will be used to remove air from the loop piping and components. It will also be used to evacuate storage tanks to reduce the time required for salvaging neon coolant from the dolly loop during normal decoupling operations. The utility and desirability of this procedure will be evaluated during the pilot loop operational methods study.

Scavenging System - All valves used in the primary neon loop will be bellows-sealed to minimize neon leakage through the valve stems. Since valves with zero leakage across the seat are probably not available, two valves in series will be used in most connecting lines. The space between each pair of valves will be vented to a scavenging system to recover the neon. The system will contain a particulate filter, a booster compressor, and storage tanks. The neon can be stored as-recovered or purified and returned to the high-pressure storage system.

The scavenging system will also be used to recover purge neon from the dolly during decoupling operations.

Neon Storage System - The neon storage system will be used to store pure neon gas needed for filling the loop initially, for leakage makeup, and for purging during cleanup of the primary loop. It will consist of low- and high-pressure storage tanks and suitable booster compressors. Upon receipt from the vendor, the gas will first be purified and then stored in the high-pressure tanks.

Power Test Dolly

Work was continued throughout the reporting period on the equipment layouts for the test dolly. The piping arrangements were changed to conform with the latest reactor pressure vessel design, and piping stress analyses were completed. The basic dolly structure is 22 feet wide and 34 feet long. The shielding around the reactor will extend to a height of 32 feet, the maximum allowable as determined by the TSF hot shop doors.

Detailed layouts of the shielding for the reactor pressure vessel, primary heat exchanger, and fission product filter are being prepared.

Instrumentation

Design was started on the installation details for thermocouples and pressure sensors in the piping to meet ASA codes. These installations will be evaluated in the pilot loop.

PILOT LOOP

Purchase orders were placed for several major components of the pilot loop including the chromatograph, gas purifier, and moisture detector. Orders were also placed for the majority of the smaller components such as valves, flowmeters, and pressure transducers.

The gas chromatograph will be used to detect hydrogen, oxygen, nitrogen, carbon dioxide, and methane in the pilot loop system. Since it will be required to detect these impurities in the part-per-million range, it has several special features:

1. A radiation cross section detector using a tritium source.

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2. A helium purification system to supply carrier gas at the required purity level.
3. The detector, columns, and sample valve are contained in an oven whose temperature is electronically controlled. The oven operates in a helium atmosphere to minimize the leakage of contamination into the system.

The gas purifier will be used in the pilot loop to reduce the contamination level to less than 5 parts per million. It consists of two coils containing adsorptive materials operating in a liquid nitrogen bath. Two coils are provided to enable continuous operation on one coil while the other is being regenerated. A catalytic converter stage removes trace amounts of hydrogen.

The specifications for the diaphragm pump and the gas bearing circulator were revised and were ready to be released to vendors for bids. An evaluation was made of the transient conditions to which the gas bearing circulator could be subjected. The most severe transients appear to be those caused by a failure of the power supply used to heat a full-length, 37-hole fuel element operating at the design conditions of the Experimental Reactor. Table 4.1 contains the results of this study.

TABLE 4.1

LOOP CONDITIONS FOLLOWING POWER FAILURE

Time After Power Failure, sec	Circulator Inlet Temperature, °F	Circulator Inlet Pressure, psia
0	400	790
10	300	684
25	216	590
50	161	540
100	140	515

Since no bids were received to design and build the gas preheater, it will be done in-house; design work was started during this report period. It will be so designed that an existing electrical power supply and control system can be used.

The pilot loop schematic drawings shown in the last bimonthly report² were not changed. No additional design work was undertaken pending receipt of details of procured items.

Procurement, wiring, and assembly of the pilot loop data collection system continued throughout the period; this system is now approximately 90 percent complete.

REMOTE HANDLING

Work continued on conceptual design studies of remote handling equipment required for the 710 Experimental Reactor. An evaluation was begun of the feasibility of combining fixtures with similar functions and to improve the flexibility of the proposed devices. Two other possibilities under study are a common guide-rail ring for all the flange-servicing equipment and a single set of vertical rails to guide the pressure vessel head and reactor during assembly.

Pressure Vessel Flange Studs

The use of stud tensioners for remotely securing and removing the pressure vessel flange studs was investigated. A design for applying these tensioners to the 710 flange stud operations was proposed by Biach Industries, Inc. With some relatively minor modifications, the Biach design, which includes pneumatic drive motors for remote operation,

²Ibid., pp. 70-71.

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appears to be suitable for remote handling in the hot shop. Figure 4.4 shows how the stud tensioners might be used in this application. In this configuration, several tensioners would be guided and located over studs at specific locations by a fixture. The assembly would be rotated to tighten all nuts in the proper sequence. Air motors would be used to rotate the assembly on its guide rails, raise and lower the tensioners, and engage the studs and nuts at the proper time. Hydraulic pressure for the tightening and loosening operations would be furnished by the compressed-air-powered pumps which are standard equipment for commercial tensioners.

All operations would be monitored and controlled from a station located at one of the viewing windows. This concept has been used successfully for similar remotely controlled operations in the past.

Fig. 4.4 - Remotely operated stud tensioner for securing and removing pressure vessel flange studs

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Aftercooling Air Exhaust System

During remote disassembly and service operations in the hot shop, the 710 reactor will be cooled by means of an air exhaust system. Some of the equipment requirements for this system were studied, as well as the suitability of commercially available exhaust equipment. In the present concept, the existing hot shop airflow and exhaust system, including the exterior air disposal stack and blower, are used. A negative pressure exhauster provides the cooling airflow and the heated discharge is ducted to the hot shop disposal system intakes. Figure 4.5 illustrates a possible arrangement of this exhaust system.

Fig. 4.5 - Exhaust system concept for aftercooling the 710 reactor in the hot shop during servicing and disassembly operations

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