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630A MARITIME NUCLEAR STEAM GENERATOR PROGRESS REPORT NO. 2

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630A MARITIME NUCLEAR STEAM GENERATOR PROGRESS REPORT NO. 2

SEPTEMBER 28,1962

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FLIGHT PROPULSION LABORATORY DEPARTMENT
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TASK CODE INDEX

Code numbers have been assigned by GE-NMPO to each major work element of the 630A Maritime Nuclear Steam Generator program. These code numbers remain constant and will appear in parentheses as part of section and subsection titles. This device should simplify identification of tasks from report-to-report, as the work progresses.

Work elements will be reported only as related significant events and data develop in each reporting period.

- 100 Propulsion System Investigations
  - 101 Cycle Analysis
  - 102 Steam Generator Assembly
  - 103 Ship Installation
  - 104 Ship Service
- 110 Reactor
  - 111 Mechanical Design
  - 112 Nuclear Design
  - 113 Aerothermal Characteristics
- 120 Shield Plug
- 130 Side Shield
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- 190 Controls
  - 191 System Design and Checkout
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  - 205 Air Supply System

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210 Auxiliary Cooling System
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- 212 Component Cooling System

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290 C. E. Test Operations

300 LPT Test Operations

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INTRODUCTION AND SUMMARY

This report summarizes the progress made on the 630A Maritime Nuclear Steam Generator program from June 30, 1962 to August 31, 1962. Although the temporary 630A program funding limit in effect for the first quarter of FY 63 curtailed the work planned in some areas, the availability of additional funds in the latter part of the report period permitted reinstatement of virtually all work originally planned.

Highlights of the period include:

1. A layout of a reduced-height 630A assembly (23 feet versus the original 34 feet) was prepared and is presently being evaluated for use in a warehouse ship design.
2. Evaluation of a relatively simple, low-cost ground test arrangement of components was begun. All indications are that it will be a satisfactory mockup for shipboard operating conditions.
3. While shielding studies indicate the need for some rearrangement of the shield materials, the desired radiation constraint can be obtained without an increase in shielding weight.
4. A preliminary stress analysis of the pressure vessel, flow path analysis and insulation evaluation has been completed. No major problems were encountered.
5. Bids on the boiler study contract were requested and are expected before October 1, 1962.
6. Evaluation of the total containment design uncovered an error in the volume noted in the Scoping Study (GEMP-108). Since the actual volume is larger, the value of the design pressure is reduced from 115 psig to 45 psig.
7. Procurement and fabrication of the Critical Experiment (CE) mockup is about 80 percent completed. The major parts will be in Idaho by the middle of September.
8. The work on the CE tank and dolly is about 50 percent completed. All materials are on hand or ordered. This equipment and the CE mockup components will be assembled in the Lower Power Test Facility (LPT) by the end of September.
9. The rough draft of the CE hazards report has been reviewed by ORO and IDO. No major problems were encountered. The final draft will be issued before the middle of September. Hazards approval is expected to be the limiting item for startup of the critical experiment. The rough draft of the Test Program and Procedures document is 75 percent complete.
10. The LPT control room modifications were made, and the rough draft of the Standard Operating Procedures was completed. The Facility Operating Manual is approximately 30 percent complete.
11. The 630A system performance requirements, safety, cost and schedule continued. The Scoping Study estimates still appear to be achievable.
12. The CE fuel was inspected and found to be in very poor condition. About 60 percent requires recoating. A significant portion of the fuel is not usable and additional fuel has been requested.
13. Creep and oxidation test time on some of the fuel sheet has exceeded 3000 hours with no significant oxidation or elongation on any of the samples.

14. The Ni-Cr samples have been irradiated and are in Idaho awaiting stress testing.
15. The Ni-Cr-sheet high-temperature (1750°F) stress and oxidating testing has exceeded 5000 hours with elongations below 0.8 percent except for one sample which has shown an elongation of about 2.3 percent.
16. Experimental fuel sheet samples have been prepared and comparative property studies have been initiated.
17. A new ORR specimen-capsule assembly (ORF-3) is being prepared. Start of this test is now expected in October.
18. Fabrication of the ring test assembly, 3-F-1, for test in the MTR is essentially complete. Start of this test is now expected late in October.
19. Facilities studies for the power test phase have indicated that a significant saving in cost over a separate facility could be achieved by using one of the pools in the Shield Test Facility.
20. Bids were received for the remaining parts required for the two tube boiler test and orders were placed. Testing is now scheduled to begin in December.
21. The design of the test rig for seal evaluations was initiated. Testing is scheduled to begin in December.
22. A detailed schedule for the work in FY 63 has been prepared and issued for comments and concurrence.

1. PROPULSION SYSTEM INVESTIGATION

(100)

A preliminary heat balance for the prototype test cycle has been prepared that demonstrates the feasibility of the selected components in accomplishing the desired thermodynamic conditions of the cycle. The cartridge flow test and scale model flow test work items are now proceeding after a temporary delay caused by funding limitations. A nuclear steam generator configuration study for a warehouse ship application is being carried out. This configuration consists of a boiler which is outside and concentric to the reactor. The over-all height is 23 feet.

1.1 CYCLE ANALYSIS (101)

1.1.1 GROUND TEST COMPONENT ARRANGEMENT

A preliminary schematic and full-load heat balance diagram (Figure 1.1) was prepared to depict ground-test operating conditions and component arrangement. Calculations are based on approximations of component performance for the 30,000 shaft horsepower condition, with steam conditions and shipboard component arrangement as shown on the heat balance diagram presented in the previous report.* In Figure 1.1, the first valve downstream from the nuclear steam generator simulates the propulsion turbine throttle valves and reduces pressure, by means of a throttling process, to 425 psia. Steam flows from the boiler through this valve and is then desuperheated to 800°F to provide the required conditions for auxiliary equipment, such as the blower turbine and boiler feed pump turbine drives. Additional throttling and desuperheating of the main steam flow to 305 psia pressure and 418°F temperature respectively occur prior to the steam admission to the main condenser. Desuperheating to this temperature is accomplished with condensate from the auxiliary condenser. Feedwater is pumped from the storage tank to the boiler thus completing its cycle. A portion of the feedwater, about 6 percent, is bypassed to the desuperheater located just after the first throttling valve.

1.2 STEAM GENERATOR ASSEMBLY (102)

For reference purposes, Figure 1.4 of Progress Report No. 1 (GEMP 130) is reported here as Figure 1.2.

1.2.1 CARTRIDGE FLOW TEST

The XMA-1 fueled cartridges have been received from Oak Ridge storage for low flow testing. Two test points were obtained prior to testing curtailment during the early part

*"630A Maritime Nuclear Steam Generator Progress Report No. 1," GE-NMPO, GEMP-130, July 31, 1962, p. 12.

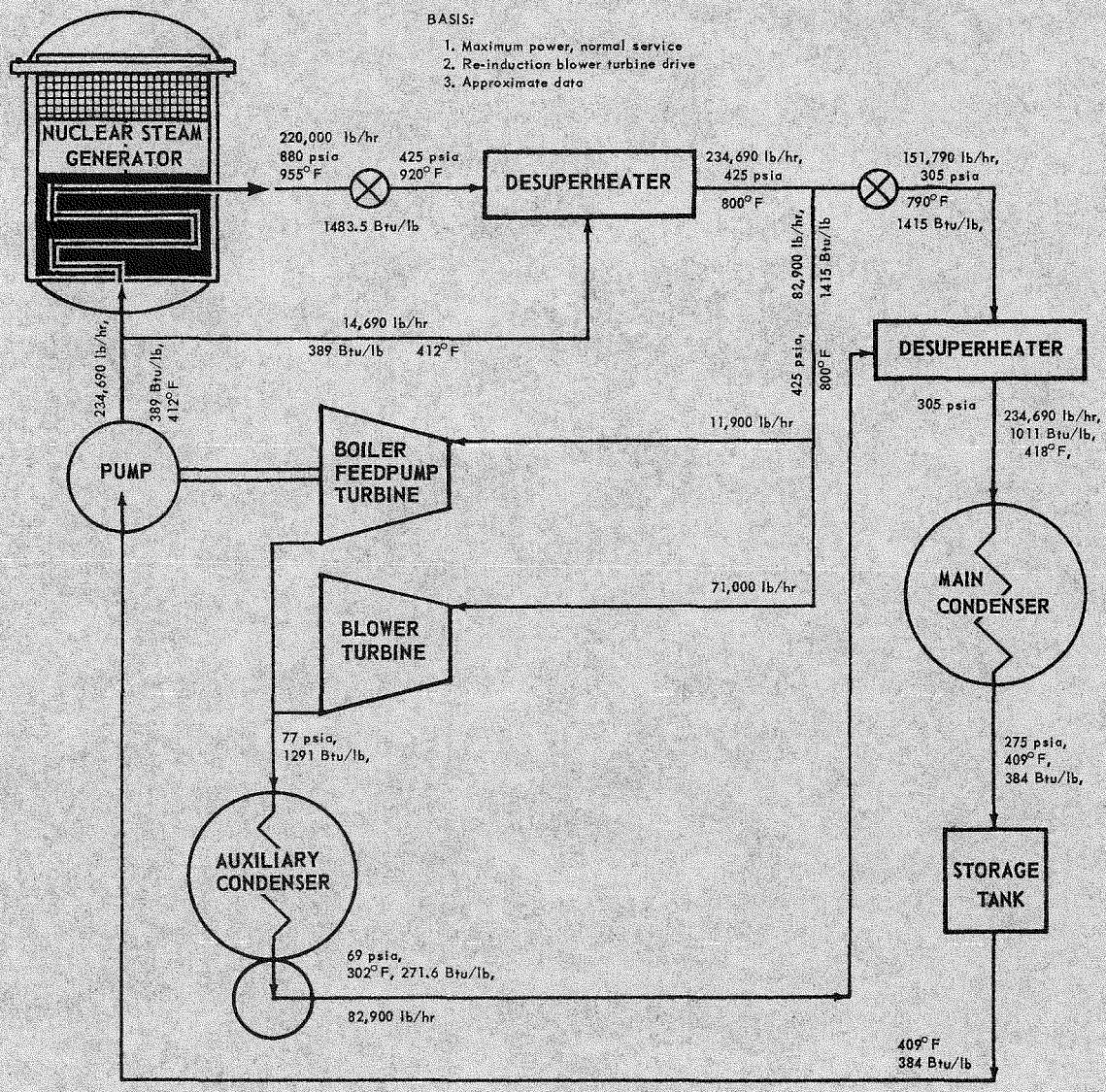

Fig. 1.1 – 630A prototype cycle preliminary performance

of August because of funding limitations. The full testing program has since been reactivated and will proceed as planned.

1.2.2 SCALE MODEL FLOW TEST

Formulation of test procedures for the scale model ducting experiments has continued during this period. Pressure losses and flow distributions will be measured at a variety of flow conditions to simulate load range operations of the 630A Nuclear Steam Generator. Duct pressure will be maintained close to atmospheric pressure to simplify fabrication techniques and to permit use of a transparent ducting material. This will be a "cold-flow" test utilizing a maximum airflow rate of about 8 lb/sec.

The broad objective in testing a scale model of the 630A ducting is to obtain estimates of prototype duct pressure losses and flow distribution. If pressure losses and/or flow distributions attained with the basic duct configuration are unacceptable, several flow path modifications will be made to establish their effects upon possible problem areas.

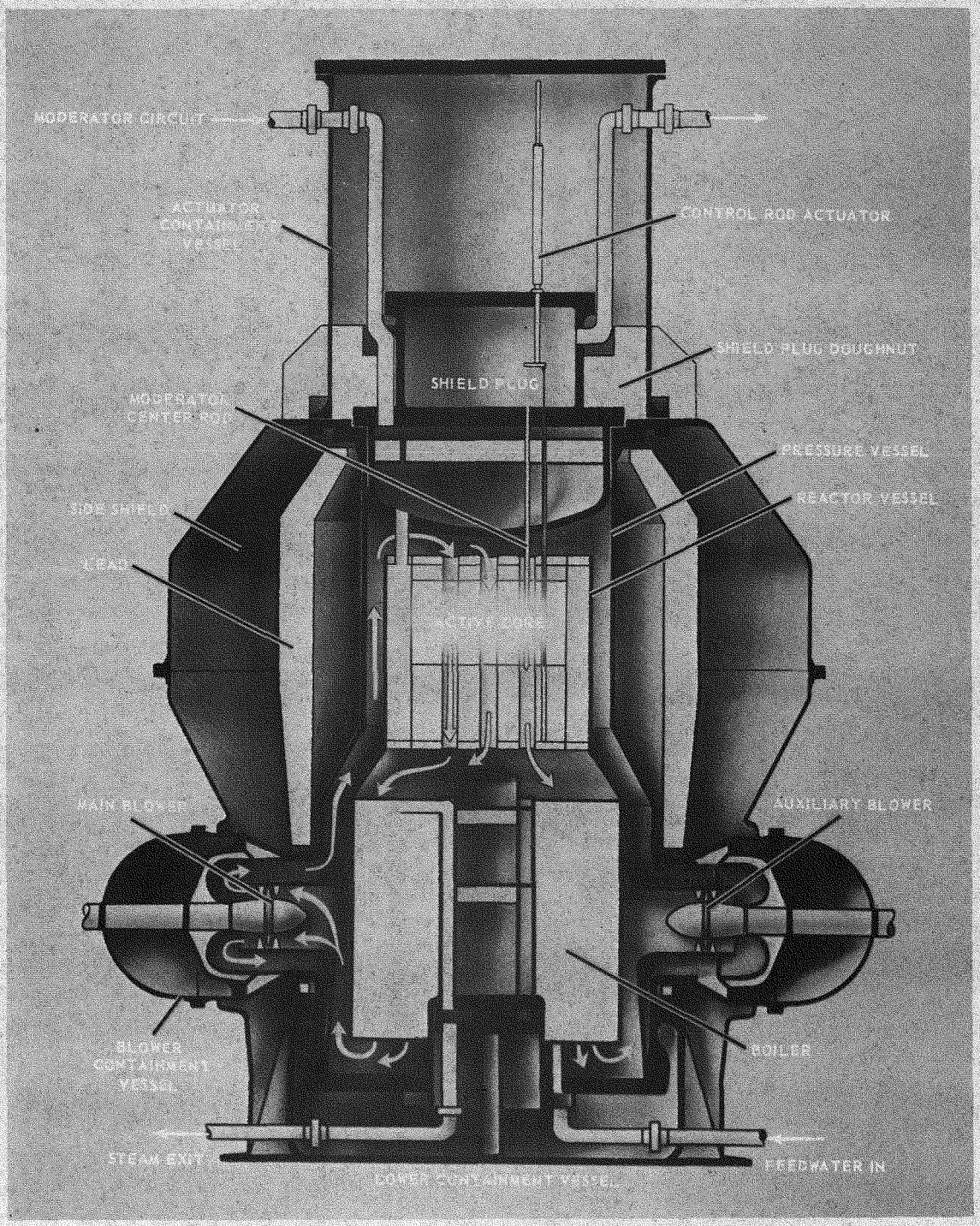

Fig. 1.2 – Artist's concept of the 630A nuclear steam generator (AS-71)

The scale model will simulate that portion of the prototype flow path from boiler discharge to reactor inlet plenum. Blower passage diffusers and turns will be mocked-up but the blower blades and their effect upon pressure losses will not be simulated.

The model will be built to 1/4 scale. Manufacturing considerations will dictate whether the model will be a 180 degree or 360 degree mockup; a full 360 degree mockup is not required, since flow symmetry about a vertical plane through the blowers' centerlines is expected. Eccentricity and ovality of the boiler shell and/or pressure vessel will not be simulated during model testing, since these conditions are not expected to present flow problems in the prototype. The auxiliary blower, the reactor and the boiler flow passages will not be mocked-up, but orifice type resistances are to be used in the annular duct just upstream of the dump to the reactor inlet plenum and in the region of air discharge from the boiler.

Air having approximately standard conditions of pressure and temperature will pass downward through the central core of the model. It will then expand into the boiler discharge plenum and pass upward along the annulus leading to the blower entrance. Guide vane performance at the annulus entrance and along the annulus will be tested if pressure losses or flow distribution are unacceptable in this region.

A removable plug will be provided at the blower-cavity entrance to evaluate two basic types of inlets. Another blower inlet passage variation will allow for additional flow area at the top of the blower inlet passage. Simulation of blower eccentricity caused by differential thermal expansion is not planned at this time. The estimate of this eccentricity for the prototype is 1/8-inch maximum.

Diffusion and turning losses within the blower passages will be evaluated. Diffuser design and turning vane adjustments will be evaluated.

Flow distribution at the reactor inlet plenum, will be determined by means of dynamic pressure measurements. Air will be discharged through orifices at the top of the annular duct leading to the reactor inlet plenum.

1.2.3 ALTERNATE CONFIGURATION STUDIES

In conjunction with George G. Sharp, Inc., alternate configurations of the 630A Nuclear Steam Generator, for application to a warehouse ship, were studied. Such an application requires that the height of the nuclear steam generator be limited to 23 feet, including the containment vessel.

The estimated total weight of the configuration is 395 long tons.

The only access would be to the bottom of the assembly for inspections, requiring a minimum height of 18 inches under the power plant. The configuration developed to meet these restrictions (Figure 1.3) is quite similar to the original 630A design. The 23-foot height was achieved by placing the boiler concentric with the reactor-shield plug assembly. Because access to the bottom of the power plant, except for inspection, is not permissible as in the 630A Nuclear Steam Generator, the boiler headers are located concentric with the shield plug at the top of the assembly, to provide access to the headers for maintenance.

As in the 630A Nuclear Steam Generator, the primary-loop working fluid, air, is contained entirely within the pressure vessel and is circulated around the closed loop by the main axial-flow blower which takes its suction from the plenum below the boiler. The air, compressed by the blower, flows through the annular passage around the blower, down the passage formed by the pressure vessel and flow divider and into the plenum formed by the lower pressure vessel head and the rear face of the reactor. Heated as it flows

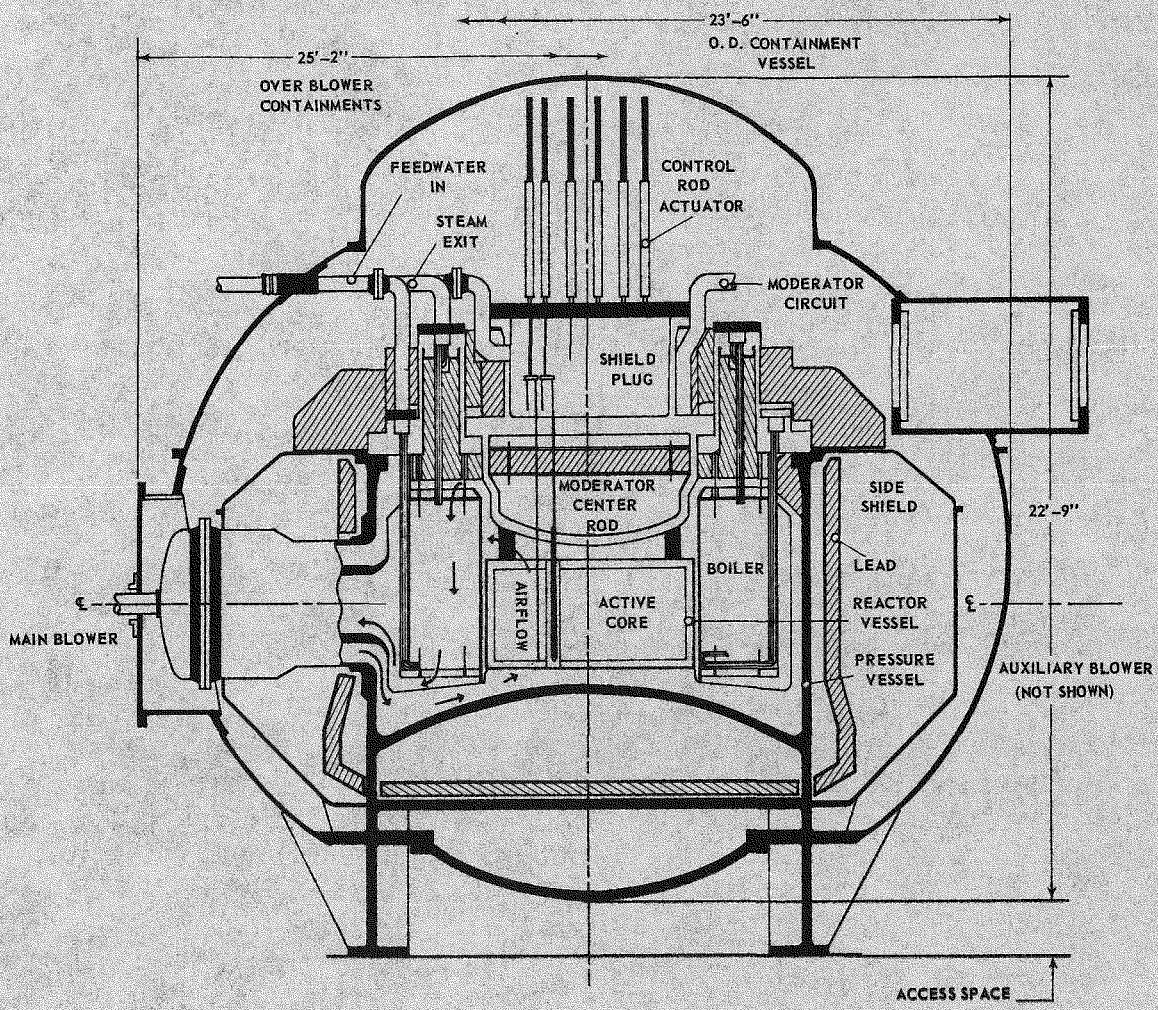

Fig. 1.3 – Concentric boiler arrangement (219R865)

through the reactor, the air is discharged into the plenum between the reactor and the shield plug. From this plenum the flow is upward between the shield plug and the inner structure of the boiler to the top of the boiler. The cycle is completed as the heat is extracted during downward flow through the boiler. The surfaces of the pressure vessel and the shield-plug that are exposed to the air stream are insulated to minimize heat losses to the shield water and moderator water.

Preliminary investigation of the stresses in the boiler header induced by pressure loads and differential thermal expansions indicate that the stresses exceed the stress allowables of the ASME Boiler and Pressure Vessel Code. Investigation of these stresses will be continued and the configuration changed, as required, to satisfy the requirements of the ASME Boiler and Pressure Vessel Code.

1.3 WORK PLANNED FOR NEXT PERIOD

Completion of the fuel cartridge low flow test is scheduled for the middle of September.

Detailed specifications, test procedures, and drawings required for the scale model flow test will be prepared.

Investigation of the stresses in the boiler header for the warehouse ship design will be continued.

2. REACTOR

(110)

During this report period, work was performed in the following areas of reactor design: (1) selection of a configuration for the first critical experiment, (2) optimization of the gross radial power flattening, both at startup and end of life, (3) determination of the fine radial power profiles for each of the cell types for start and end of life, (4) determination of the fuel loadings for the critical experiment and, (5) analysis of fuel plate temperature perturbation.

2.1 NUCLEAR DESIGN (112)

2.1.1 ESTABLISHMENT OF FIRST CRITICAL EXPERIMENT ARRANGEMENT

Fuel Loading and Distribution for Fine Radial Power Shaping

During this reporting period, the fuel loadings and distributions for the critical experiment were determined. The fuel loading per stage of the critical experiment is roughly the same as for the reactor fuel element. There is no longitudinal variation. In the "measurements" fuel cartridges, the fuel and 80Ni - 20Cr inventory are as closely maintained as practical, consistent with preserving the optical thickness of materials in order to check the adequacy of the theory used for predicting the fine radial power distribution.

The three 80Ni - 20Cr alloy cylinders of the fuel cartridge are wrapped with successive layers of fuel and 80Ni - 20Cr alloy sheet. Several layers of 80Ni - 20Cr alloy are wrapped around the innermost cylinder, and outside of the fuel on the outermost cylinder to mock up the inner and outer dead rings, respectively. The thickness of the 80Ni - 20Cr alloy per sheet of fuel is varied to mock up the lower fuel loading per unit of heat transfer area near the moderator and the higher fuel loading per unit of heat transfer area in the "middle" of stage. The fuel distribution is based on obtaining initial heat fluxes per unit of heat transfer area which vary from approximately 1.10 to approximately 0.91 times the cell average. These extremes exist on the outermost and the inner rings of the fuel element, respectively. The fuel/80Ni - 20Cr alloy ratio is also reduced in the fine radial direction in the regions next to the moderator, to mock up the fuel dilution which will be made to reduce the percent of atomic burnup in the rings presently exhibiting more than average burnup. There is no longitudinal variation in this factor.

Moderator Displacement for Gross Radial Power Shaping

During this reporting period, analyses were continued to trim the gross radial power profile for acceptable distribution both at start of life and end of life. As discussed in the previous progress report,* aluminum was employed to displace moderator at the outside

*Ibid., p. 22.

of the individual cells and to effect an artificial pitch variation into the analyses. The aluminum, which is a poor moderator and has a relatively low neutron absorption cross section, displaces the water moderator and reduces the moderator fraction per unit cell without significantly increasing the neutron absorption. Variations in the aluminum distribution, which may be necessary for fine trimming of the gross radial power profile, can be accomplished rather easily in the critical experiment, whereas changes in pitch variation are difficult to achieve for a particular set of tube sheets. The configuration to be mocked up in the initial critical experiments is described in Table 2.1.

TABLE 2.1

CORE CONFIGURATION FOR THE INITIAL CRITICAL EXPERIMENT

Fuel Cell Location	No. Of Cells In Reactor	Number Of Rings	Aluminum Volume Fraction, %
Central Element	1	13	12.641
First Concentric Annulus	6	13	12.075
Second Annulus	12	13	9.791
Third Annulus	18	13	6.311
Fourth Annulus	24	12	1.567
Fifth Annulus	24	11	0.515
Average Aluminum Volume Fraction			4.307

The core described in Table 2.1 was also used as a reference to compare the performance of cores with pitch variation to those with aluminum insertion. Two cases of pitch variation were investigated. In the first case, the active diameter was held fixed at 48 inches. The pitch was varied so that the change in moderator fraction from cell-to-cell was approximately the same as when the aluminum was in the constant pitch core; proportionately more water was added to the outer cells than to the inner cells. A cross section of the core which resulted is shown in Figure 2.1.

In the second case, the voids caused by the removal of the aluminum shims were squeezed out by reducing the pitch between cells; this volume was not distributed to the remaining cells and the over-all active core diameter was reduced to 47.05 inches. The gross radial power profiles for these cases are shown in Figure 2.2 with a profile for the case with aluminum in the core.

The 48-inch pitch-variation case shows a decrease in the effect of the beryllium reflector as a result of the higher moderator fraction in the outermost cells and a reduction in the power generation of these cells compared to the average. The maximum power occurs in the second concentric annulus of the cells where the power is 5 percent greater than the core average. For the case of the 47.05-inch core, the power generation at the center of the reactor increased approximately 4 percent over that which was obtained with aluminum in the core, thus making the center region, at the start of life, the highest power region.

Reactivity Analysis

Analyses were made to determine the effect of core diameter on reactivity at the end of life condition. The original 48-inch-diameter core had been selected on the basis of cold-clean reactivity at the start of life. For the end of life analysis, each reactor core consisted of 85 identical cells; no fuel or moderator variations for gross radial power flattening were included. The changes in cell size that accompany the over-all reactor diameter changes were effected by varying the amount of moderator outside of the fuel

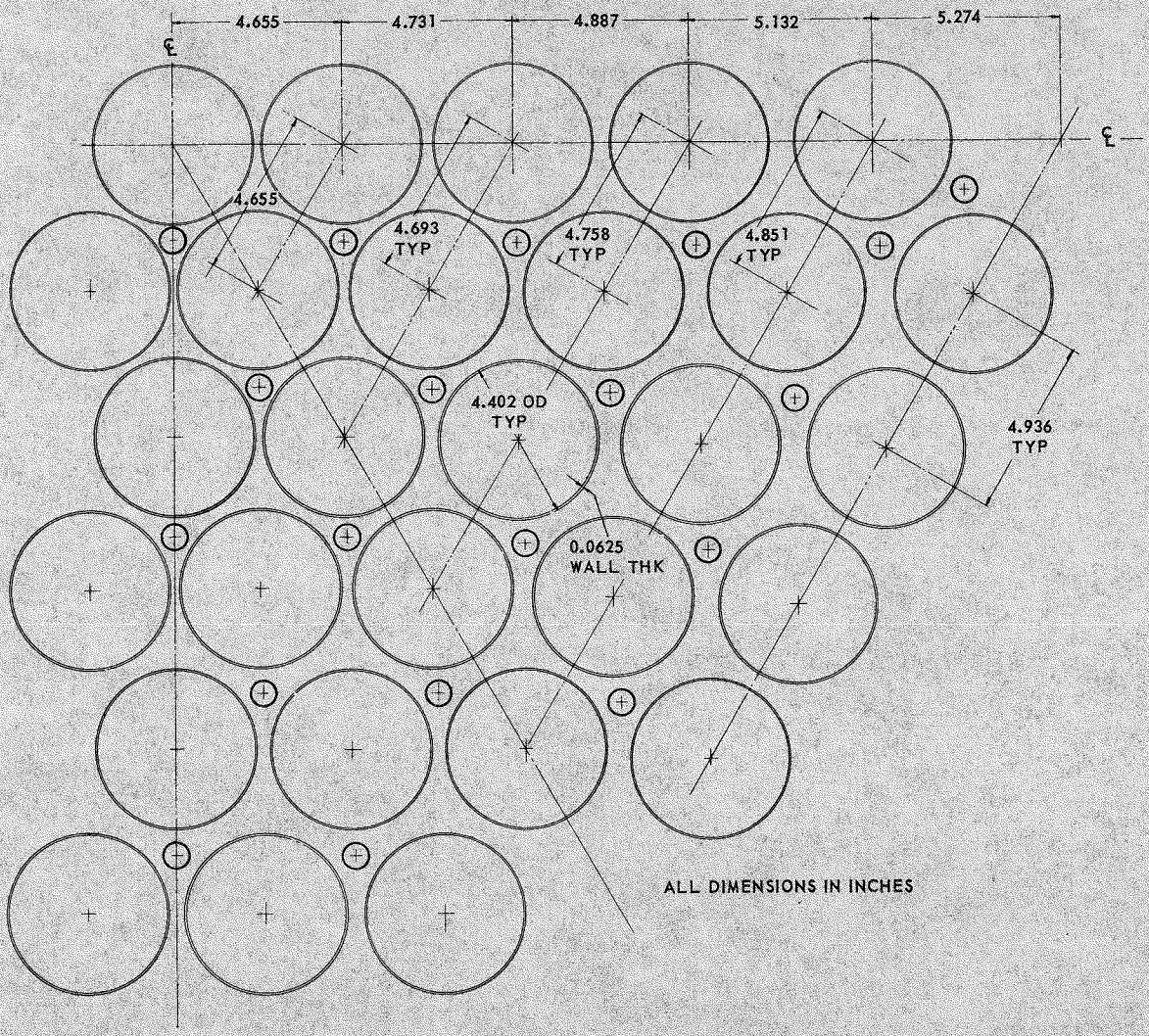

Fig. 2.1 – Cross section of core with pitch variation (48 inch diameter) (Dwg. 692E618)

tube. For each cell size, flux disadvantage factors were calculated for neutron energies extending from the thermal group to 8.315 ev. The ratio of fission to non-fission capture in the U^{235} and the fission product inventory was assumed to be the same as that in the 48-inch core. The results of these analyses are shown in Figure 2.3. The reactivity of the system divided by that of the 48-inch-diameter core is plotted against diameter. This curve indicates that the 48-inch-diameter core is near optimum and should become optimum with power flattening. As the core diameter and the moderator fraction increase, the reflector has less effect on the gross radial power distribution, thus increasing the required cell variation for power flattening. As the fuel-moderator ratio inside of the core is varied, a reactivity reduction is effected. Therefore, although a 49-inch core has about 0.07 percent more reactivity than a 48-inch core on the basis of the unpower flattened condition, the 48-inch core will have more reactivity after the core is power flattened.

Predictions of Power Distribution

Fuel Burnup Effects on Fine Radial Power Distribution - As discussed in the previous progress report,* the heat flux from each ring of the fuel element will change during

*Ibid., pp. 24–25.

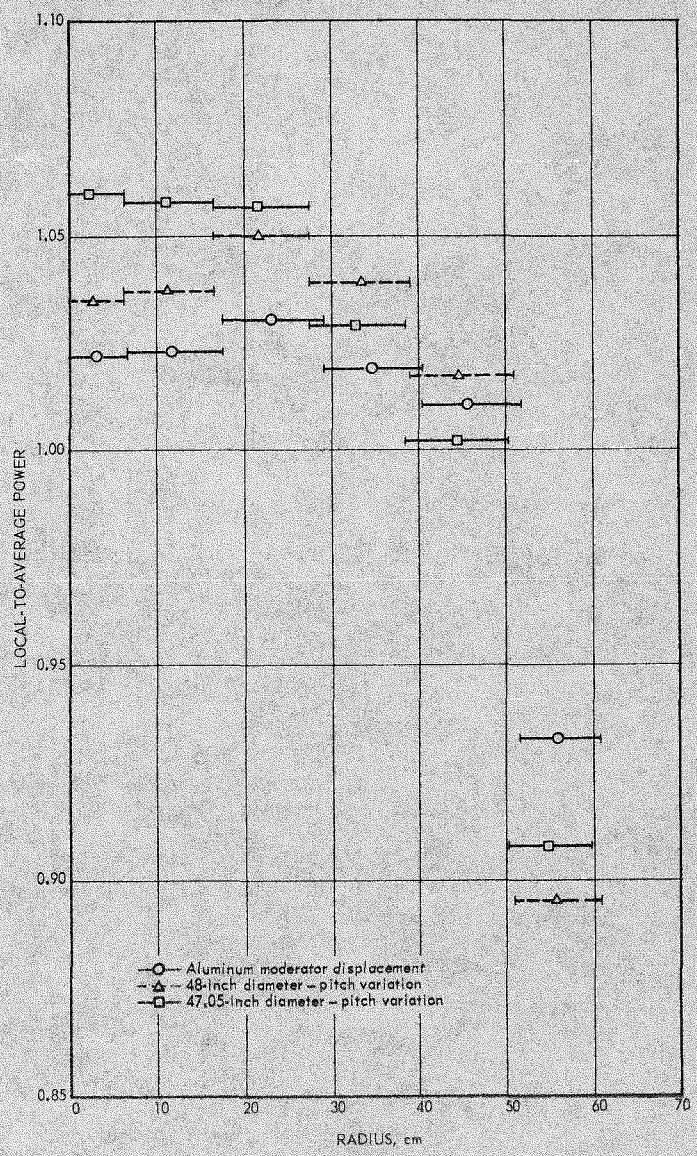

Fig. 2.2 – Comparison of gross radial power distributions

operation because of the unequal rate of fuel burnup in each ring. In Table 2.2, the heat flux on each ring of a 12-ring element is shown at the beginning and at the end of operating life. The initial fuel inventory per unit of heat transfer on each ring is adjusted to yield the initial heat flux, Q/A. The parameter used to determine the fission probability in a particular ring was the percentage of optical thickness of U^{235} between that ring and the outside of the fuel region.

The fission probability in each ring changes slightly during the reactor's operating period, as does the relative fuel inventory per ring. Therefore, in order to determine the fuel inventory of a ring at the end of life, the average of the initial and final fission probability was used in the equation for maximum atomic burnup, and the average was assumed to be constant over the operating history. The equation appears later in this section.

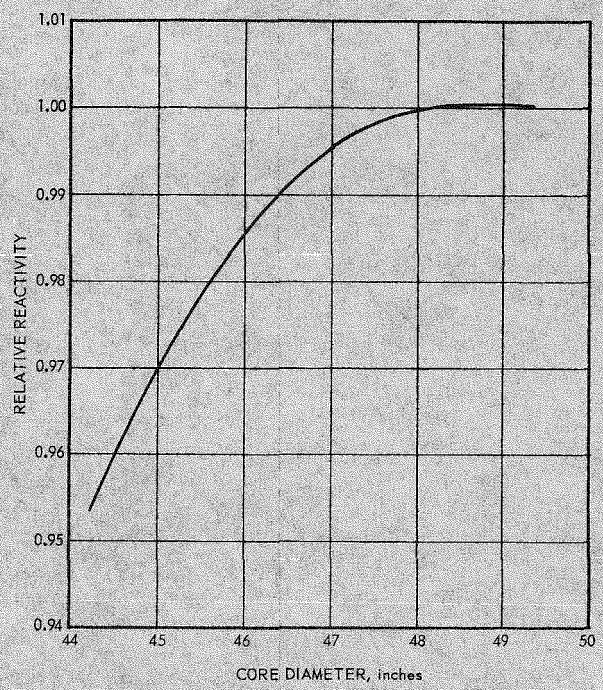

Fig. 2.3 – End of life reactivity relative to 48-inch core as a function of core diameter

TABLE 2.2

BURNUP EFFECTS ON HEAT FLUX AND FISSION PROBABILITY

	Initial		Final		Percent Fuel Left
Ring No.	Fission Probability	Heat Flux Ratio $Q/A / \overline{Q/A}$	Fission Probability	Heat Flux Ratio $Q/A / \overline{Q/A}$	Divided By Percent Fuel Left In Average Ring x 100, %
12	1.612	1.10	1.651	0.899	79.83
11	1.379	1.07	1.4665	0.983	86.36
10	1.179	1.04	1.2795	1.040	92.19
9	1.00	1.0	1.1015	1.081	98.17
8	0.871	0.96	0.9465	1.078	103.32
7	0.7845	0.92	0.825	1.037	107.21
6	0.7425	0.915	0.7555	1.018	109.37
5	0.747	0.92	0.7425	1.002	109.56
4	0.8055	0.95	0.7785	0.989	107.73
3	0.912	0.98	0.8675	0.969	104.0
2	1.067	1.03	0.968	0.929	99.38
1	1.2885	1.07	1.1165	0.862	93.01

Upon examining Table 2.2, it is apparent that ring 12, with the lowest final heat flux, has the highest burnup. Since the heat flux in rings 8 to 11 was less than anticipated, (an increase with time was found on rings 8 and 9) the initial heat fluxes should be reduced 2 to 5 percent. Similarly, rings 2 through 6 should have initial heat fluxes which are about 2 to 4 percent higher than those chosen. With such changes, the final heat fluxes (or ring temperatures) and fuel burnup, would be more compatible.

Maximum Atomic Burnup - The average atomic burnup percentage listed in the Scoping Study (GEMP-108) for the 630A system was 2.0 percent. The maximum atomic burnup is somewhat larger due to the distributions of specific power generation in a reactor core

which occur as a consequence of the configuration. The principal factors which increase the atomic burnup are the fine radial power distribution in the individual cells and the reactor gross longitudinal power distribution. In addition to these factors, there are smaller effects due to gross radial power perturbations and perhaps to circumferential power scalloping which will be determined in the critical experiment. The effect of burnup on the fine radial power distribution was discussed previously and is shown in Table 2.2 as the ratio of the percent of fuel remaining in a particular ring to the percent of fuel remaining in the average ring. This effect is discussed in more detail below.

The average fraction of the U^{235} which fissions or is converted to U^{236} is 30 percent. This corresponds to the 2 atomic-percent fissioning in the meat of the reactor as described in the Scoping Study. In the original burnup estimate, the assumption was made that the U^{235} inventory reduction in regions of other than average flux (or fission probability) equalled the average reduction multiplied by the relative fission probability. Upon closer examination, it became apparent that this led to erroneous results. The fraction of the U^{235} left at the end of life in particular location was found to be the average fuel left (70 percent of initial) raised to a power equal to the local relative fission probability. The derivation of this equation is shown below. A sample case for the region with the highest probable fuel burnup rate is included.

Derivation of Local Burnout Expression

Terms

N = number of atoms of U^{235}/cm^3 in the whole cell
n = number of atoms of U^{235} at a particular location
R = particular location
ϕ = average flux in whole cell
B = local/average burnout ratio (ϕ_R/ϕ)
σ_a = per atom total capture cross-section in U^{235}
σ_f = per atom fission capture cross-section in U^{235}
t = time (% of total operating time at constant power level)
a = average fuel burnup (fission and non-fission capture) fraction at the end of life
0 = subscript referring to initial values
1 = subscript referring to end of operation

The assumptions were made that (1) the ratio of fission to non-fission capture in the fuel is constant and (2) that the power level is constant.

$\overline{Q}''' \sim \sigma_f \phi N$ = constant (heat generation rate/unit volume = constant)

$\therefore \phi N = \phi_0 N_0$ or $\phi = \phi_0 N_0/N$

$dN/dt = -\sigma_a \phi N = -\sigma_a \phi_0 N_0$

$N = N_0 (1 - \sigma_a \phi_0 t) = N_0 (1 - at)$

(Note: $\sigma_a \phi_0 = a$)

$\therefore \phi = \phi_0/(1 - at)$

$\phi_R = B\phi$ (definition) $= B\phi_0/(1 - at)$

$dn/dt = -\sigma_a \phi_R n = -\sigma_a \phi_0 Bn/(1 - at)$

$dn/n = -\sigma_a \phi_0 B\, dt/(1 - at)$

LET $C = 1 - at \quad dC = -a\,dt$

Substituting C, and a for $\sigma_a \phi_0$

$dn/n = -aB\,(dC/-a) \times 1/C = B\,dC/C$

@ $t = 0$, $C = 1$ @ $t = 1$ (end of life), $C = 1 - a$

$\log_e n/n_0 = B \log_e C/C_0$

$\therefore n/n_0 = (1 - a/1)^B = (1 - a)^B$

or $n = n_0\,(1 - \text{average burnout})^{\text{local/average burnout ratio}}$

Note: In order to justify the form of the solution of the differential equation, B must be assumed to be constant over life. In the fine radial direction, B will change as much as 10 percent during life (see Table 2.2). By using the average value of B over life, the error introduced is assumed to be minor.

Sample Calculation, Worst Expected Local Factors for Burnup - Assuming that a fine radial fission probability factor of 1.65, a gross longitudinal factor of 1.27, and a gross radial factor of 1.03 exist at the same place in the region where fuel burnup is at a maximum, $B = 1.65 \times 1.27 \times 1.03 = 2.16$. Since the average fission and non-fission fuel capture factor is 30 percent in this region, $n/n_0 = (1 - 0.3)^{2.16} = 0.4629$, therefore, fission and non-fission capture = 53.71 percent which is 1.79 x average capture.

The average atomic (fission)burnup of the configuration as presented in the Scoping Study (uniform weight fraction of fuel in the meat) is 2.0 percent. Therefore, the maximum atomic (fission) burnup is 3.58 percent (2% x 1.79 = 3.58%).

A point to consider in examining the atomic burnup is that the rings with higher-than-average fuel burnup will be reduced in temperature at the end of life when the maximum burnup is present. Conversely, the hottest rings at the end of life will have lower than average fuel burnup. As an example, the hottest point of the longitudinal temperature profile is at the trailing edge of the seventh stage where the power density at start of life is about 99.4 percent of the longitudinal average. Therefore, using the factors 1.65, 0.994, and 1.03 (as in the example), the atomic burnup percentage at this point would be about 3.02 percent for an undiluted outer ring and the ring's temperature would be well below the original temperature for this location.

It is expected that some dilution, by Ni-Cr addition, will be desirable in the rings that experience high fuel burnups. For example, the ring in the preceding sample problem has 1.79 times the average burnup, and a reduction of the UO_2 weight fraction to 25 percent of the meat reduces the percent of atomic burnup to 2.23 percent. Note that this location will be operating at lower-than-average temperatures when the high burnup condition is reached. As previously mentioned, the initial critical experiment 80Ni - 20Cr loadings include the extra 80Ni - 20Cr that such dilution entails. This additional 80Ni - 20Cr is expected to reduce the system reactivity by one percent or less, as compared to a system fabricated of meat with a constant weight fraction (38+%) of UO_2.

Life Effects on Gross Radial Power Distribution - The configuration for the reactor core which was chosen to be mocked up in the initial critical experiments is described in Table 2.1. The gross radial power distributions for start of life and end of life, assuming uniform burnup, obtained for this configuration are shown in Figure 2.4 as the relative average power generation for each region. The power distributions are normalized to unit heat transfer area. At the start of life, the maximum power occurs in the second annular region (12 cells) in which the power is 3 percent greater than the average, while the outside region operates at approximately 93 percent of the average. The power distribution at the end of life shows the depression in relative power at the center of the core and increase in the outer regions and is caused by the boron and fuel burnup which increase the worth of the moderator volume displacement. The assumption of uniform burnup and

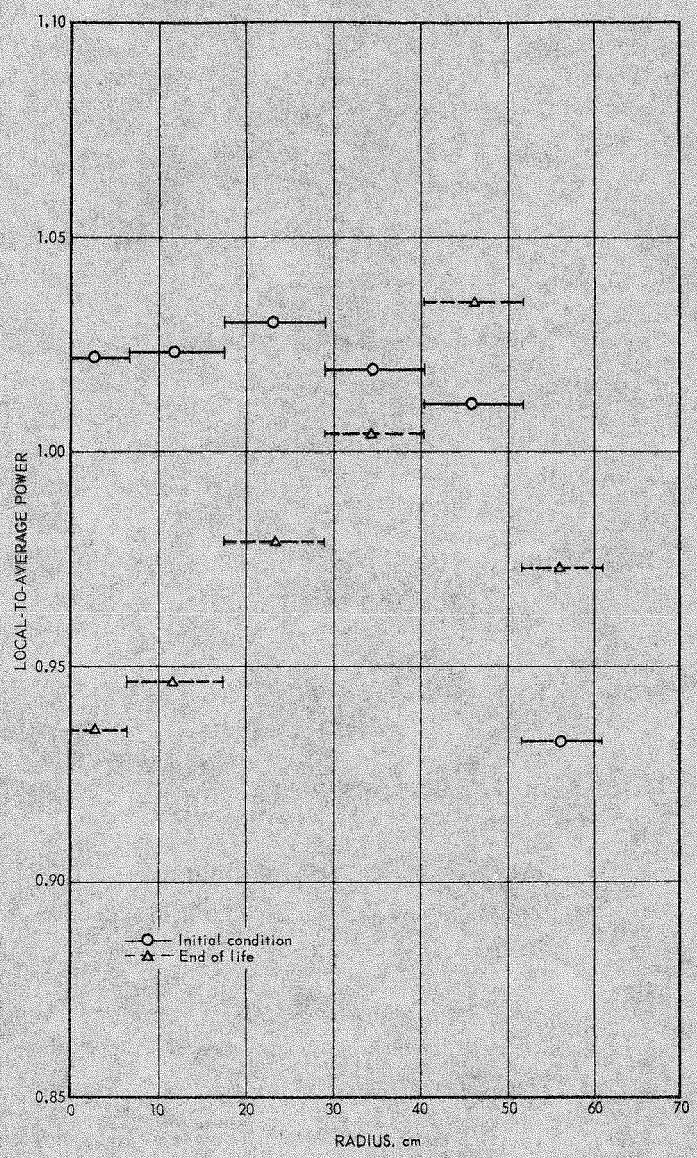

Fig. 2.4 – Predicted gross radial power distributions for critical experiment with aluminum moderator displacement

fission product accumulation in each cell results in a power swing which is somewhat greater than would occur in the actual case. The outside cells, for instance, increase from 93.3 percent of the core average power to 97.3 percent of the core average power with this assumption. Obviously, these cells would be expected to increase somewhat more since the burnup would be less than average throughout the life with the lower than average power.

2.2 AEROTHERMAL ANALYSIS AND DESIGN (113)

2.2.1 CALCULATION OF REACTOR CHARACTERISTICS

Fuel Region

Three IBM 704 computer programs have been reactivated and are being checked out for use in aerothermal analysis and design computations. These programs are the General Flow Passage, Heated Annuli, and Reactor Transient Programs.

2.2.2 PERTURBATION ANALYSIS

Fuel

Listed in Table 2.3 are the calculated average operating temperatures and the power profile upon which calculations have been based for the normal power condition requiring 67.4 mw total reactor power. The power profile ratios shown are stage trailing edge power divided by reactor average power, P_x/P_{avg}. T_b is bulk air temperature, ^oF and T_p is plate temperature, ^oF. Both of these temperatures are at stage exit location.

TABLE 2.3

FUEL ELEMENT OPERATING CONDITIONS

Stage	P_x/P_{avg}	T_b	T_p
1	0.766	616	810
2	1.012	681	938
3	1.19	760	1057
4	1.28	848	1180
5	1.28	938	1260
6	1.18	1023	1308
7	0.994	1098	1335
8	0.744	1158	1334
9	0.54	1200	1326

Bulk air temperatures shown are calculated on the basis of stage average power with a total mass flow of 352 lb/sec, an air inlet temperature of 569^oF and air exit temperature of 1200^oF. A total of 5.75×10^4 Btu/sec are transferred to the air from the fuel elements.

Plate temperatures are calculated using the expression $h = (44)\, G^{.8}/D_h^{.2}\; T_b^{.8}/T_f^{.56}$ Btu/hr-ft^2 ^oF for the heat transfer coefficient wherein G, the mass velocity, equals 352 lb/sec-592 in.^{2} = 0.595 lb/sec in.^{2}. The hydraulic diameter, D_h, is 0.12 inch. T_f is the film temperature and equals $T_b + T_p/2$. Free flow area is 592 in.^{2}. The film temperature difference, ΔT_f, is evaluated by the expression $\Delta T_f = (Q)\,(P_x/P_{avg})/(A)\,(h)$. Q/A is the average heat flux and h is the local film coefficient defined above. Q is 5.75×10^4 Btu/sec and A is 3480 ft^2 heat transfer area.

The maximum average plate temperature is seen to be 1335^oF in Table 2.3 and is obtained in the following manner:

$$h = \frac{(44)\,(.595)^{.8}}{(.12)^{.2}}\,\frac{T_b^{.8}}{T_f^{.56}}\ \text{Btu/hr-ft}^2,\ ^oR$$

$$h = 44.4\,\frac{T_b^{.8}}{T_f^{.56}}\ \text{Btu/hr-ft}^2,\ ^oR$$

$$\Delta T_f = (T_p - T_b) = \frac{(5.75 \times 10^4\ \text{Btu/sec})\left(3600\ \frac{\text{sec}}{\text{hr}}\right)(.994)}{(3480\ \text{ft}^2)\left(44.4\,\frac{T_b^{.8}}{T_f^{.56}}\right)\ \text{Btu/hr-ft}^2,\ ^oR}$$

$$\Delta T_f = 1332\,\frac{T_f^{.56}}{T_b^{.8}},\ ^oF$$

By iterative calculations ΔT_f is found to be 237^oF and $T_p = 1098 + 237 = 1335^oF$.

Plate temperatures will deviate from the maximum-average value of 1335^oF by virtue of gross radial, fine radial, and circumferential scallop power perturbations.

For an outer ring the estimated power perturbations at the beginning of core life could be as much as: gross radial, + 3 percent; fine radial, + 10 percent; circumferential scallops, + 5 percent. (See Section 2. 2). This 18 percent power perturbation causes about a 23 percent temperature deviation on bulk air temperature rise and on film temperature rise. The plate temperature for the outer ring during initial operation would then be 1510°F as a result of the three described power perturbations.

For operation near the end of core life, the same outer ring will experience a reduction in heat flux relative to its initial value of about 20 percent (see ring 12, Table 2. 2). The effect of fuel burnup in the outer ring is to reduce the plate temperature by about 220°F from the maximum perturbed temperature of 1510°F to 1290°F.

At the end of core life, central rings, such as ring number 9 in Table 2. 2, experience temperatures higher than those at the beginning of core life. However, no appreciable circumferential power scallop is present. The fine radial power perturbation is only about 8 percent and the gross radial power perturbation is about 3 percent. Consequently, plate temperatures of the central rings at the end of life could be as high as 1440°F considering the power perturbations presented here. Therefore, maximum hot spot temperatures at the end of life are of relatively low magnitude because of their location within centrally located rings.

2.3 WORK PLANNED FOR NEXT PERIOD

A single cell, heterogeneous fuel region analysis will be made to determine the effects of fuel, fission product, and 80Ni - 20Cr distribution on the fine radial power distribution. The initial fuel inventory will be distributed to give about the same initial heat fluxes as shown in Table 2. 2.

The effect of moderator temperatures on the gross radial power distribution of the initial critical experiment configuration will be determined. Figure 2. 4 shows that the change in gross radial power distribution as a function of time is small when uniform burnup is assumed. The effects of non-uniform boron burnout, U^{236} production, and fission product accumulation on the gross radial power distribution will be obtained.

Prediction of component heating rates will be made. Reactivity and reactor kinetics analyses will be performed. Mechanical design study layouts will be made and stress analyses of the reactor components will be initiated.

3. SIDE SHIELD

(130)

During this report period, analysis of the side shield design was concentrated on elimination of the radiation peak in the region of the plastic shield plug discussed in the previous report.* Total dose rates at the surface of the shield were determined for the cases of power operation and shutdown.

3.1 SHIELD ANALYSIS (131)

Nuclear design of the 630A shield is progressing under a tentative system of radiation constraints. These constraints limit the surface total dose rates on the shield, from the actuator cover to the middle of the side shield, to 1.2 millirem per hour. From the middle of the side shield to a point opposite the middle of the boiler, the dose rate will be limited to 12.5 millirem per hour. From the middle of the boiler to the base of the power plant, the dose rate will be limited to approximately 50 millirem per hour. Allowable levels have not yet been determined for the region below the power plant; it is expected, however, that the radiation dose rate in this area will be greater than in the areas discussed above.

The constraints will provide an area of radiation protection above the midplane of the reactor in which the dose rate 20 feet from the reactor through a steel bulkhead will be equal to or less than the allowable 0.5 rem per year general-populace dose rate. Below the midplane of the reactor, the dose rate 20 feet from the reactor will equal the laboratory tolerance of 100 millirem per week for a 40-hour week. The area below the midplane in the immediate vicinity of the shield will be a limited-access area with access times limited to between 1/2-hour and 1 hour per week.

3.1.2 DOSE RATES

Dose rates from radiation originating in the reactor have been reduced to acceptable levels above the reactor horizontal midplane by moving the lead slab in the shield plug closer to the reactor and eliminating a weak path in the upper part of the main shield. All the lead has been removed from the doughnut and the lead inside the main shield has been changed to provide adequate attenuation of core gamma rays. The effects of these changes on secondary gamma ray production have not yet been fully evaluated. The changes have resulted in no appreciable weight difference.

Initial evaluation of the shield below the mid-plane of the reactor indicates that some reduction in shield thickness can be made in the upper part of this region. However, shielding will have to be added in the region adjacent to the boiler where the shield thins down rapidly. The net effect of the changes has not yet been determined.

*"630A Maritime Nuclear Steam Generator Progress Report No. 1," GE-NMPO, GEMP-130, July 31, 1962, p. 28.

Calculations of secondary gamma dose rates at the side surface of the shield indicate an excessive contribution from the outer clad of the lead slab. One obvious solution is to reduce the clad thickness; this approach is being investigated, as is the effect of splitting the lead into two slabs and moving the second slab further out into the water.

3.1.3 NUCLEAR HEATING RATES

Gamma heating rates have been determined for some of the critical regions of the shield. The highest heating rate in the lead slabs has been calculated to be 2.4×10^{-3} watts/gram at normal power.

3.1.4 ACTIVATION EFFECTS

Analysis has been initiated to determine the activation of the components of the assembly. Dose rates due to activation of the boiler have been estimated to be 20 mr/hr, 50 centimeters below the boiler, 10 hours after shutdown.

3.2 WORK PLANNED FOR NEXT PERIOD

Work will continue on shaping of the shield to reduce the dose rates from radiation originating in the core. Analysis of secondary gamma production will also be continued, with the objective of reducing dose rates to a negligible level by internal changes in the shield which will not affect the envelope. Further effort will also be directed toward determining and reducing dose rate levels in the region below the boiler. Calculation of material activations and drawings of the shield configuration showing local heating rates within the slabs are scheduled for completion.

4. PRESSURE VESSEL

(140)

Blower inlet and exit flow path areas, insulation requirements and wall thicknesses have been tentatively selected.

4.1 DESIGN

A preliminary design study and stress analysis of the pressure vessel has been completed. Figure 4.1 is a preliminary drawing delineating the pressure vessel. The inside diameters of the pressure vessel and nozzle include a 1/2-inch allowance for the insulation on these surfaces. This insulation will be 1/4-inch thick and consist of a 1/16-inch lining of stainless steel in a waffle pattern attached to the inside of the pressure vessel to produce a 3/16-inch stagnant air barrier to reduce the heat loss from the air stream through the pressure vessel wall to the shield water.

With allowance made for the insulation, the inside diameters of the pressure vessel and nozzles of this configuration and hence the primary coolant flow areas are the same as those reported in the previous progress report.*

4.2 WORK PLANNED FOR NEXT PERIOD

During the next reporting period, the insulation on the inside surfaces of the pressure vessel will be defined so that a development program to establish the method of attachment and structural integrity can be outlined.

*Ibid, p. 32.

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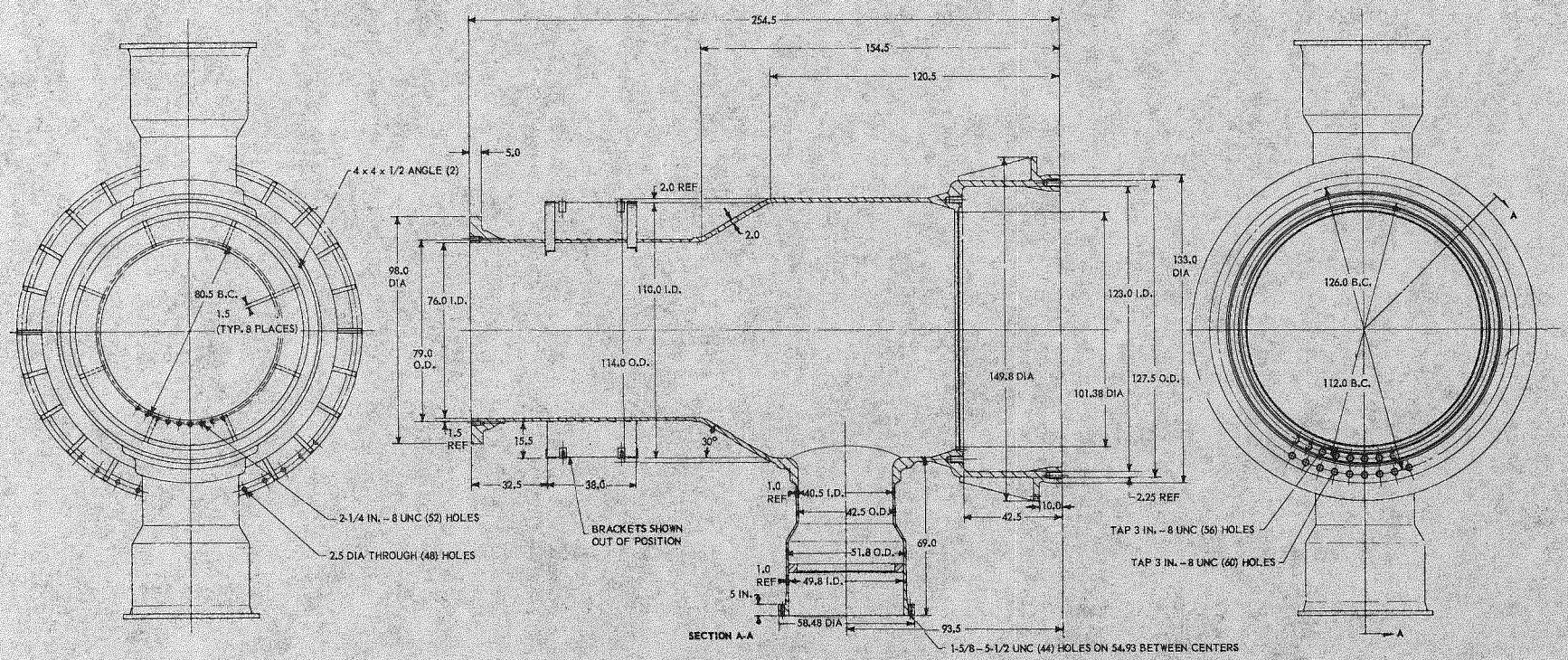

Fig. 4.1 – Pressure vessel – 630A (Dwg. 219R849)

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5. BOILER

(150)

During this report period, the due dates for proposals and the engineering study were re-established as a result of discussions with several boiler vendors.

5.1 DESIGN (151)

Study Contract

The request for proposal on the engineering study of the boiler was sent to four vendors on July 12, 1962. The four vendors are Combustion Engineering, Inc., Baldwin-Lima-Hamilton Corporation, The Babcock and Wilcox Company, and the Foster-Wheeler Corporation. Originally, proposals were to be submitted by August 12, 1962 so that the selected vendor could perform the work in the period from August 1962 to December 1962. However, at the request of the vendors, the due date for proposals has been extended to October 1, 1962 and will be performed in the period from November 1, 1962 through April 1963. Three of the four vendors have stated they will submit a proposal in accordance with this revised time schedule.

5.2 WORK PLANNED FOR NEXT PERIOD

During the next reporting period, proposals will be evaluated, the vendor selected, and the subcontract established with the vendor so that the engineering study may start on November 1, 1962.

6. CONTAINMENT

(160)

During this report period, it was determined that the total containment method provides more than eight times the expansion volume provided by the individual containment method. Studies to data indicate that total containment will permit simplified handling and maintenance and minor actuator adjustments, without power plant shutdown.

6.1 CONTAINMENT VESSEL DESIGN STUDY (161)

During this report period, the study to evaluate the merits of the three possible methods of containment discussed in the previous progress report* - Individual Containment, Modified Individual Containment and Total Containment - was continued.

As stated in the previous report, the correct expansion volume in the total containment vessel is 6125 cubic feet, which reduces the pressure postulated from a conceivable accident to approximately 45 psig, and the required wall thickness of the containment vessel to one-half inch.

Some of the comparisons made to date show that the total containment provides more than eight times the expansion volume than does the individual or modified individual containment. This comparison assumes that the four individual containment vessels are piped together. This multiple piping could be a restriction for handling and maintenance accessibility of the individual containment methods.

The total containment method would allow servicing of the actuators for minor adjustments and repairs by entrance through the air lock without shutting down or disturbing the power plant and containment. In the individual containment methods, actuator servicing would require a shutdown of the system since the containment would have to be violated.

6.2 WORK PLANNED FOR NEXT PERIOD

Further comparisons of the merits of the three methods of containment will be made.

*Ibid, p. 37.

7. CONTROLS

(190)

During this reporting period, bids were received for the remainder of the two-tube boiler parts and orders were placed for all major items except the boiler shell and boiler tubes. Work was begun on the safety system engineering study.

7.1 SAFETY SYSTEM (194)

The safety system of a power plant is required to maintain surveillance and cognizance of the power plant and its operating conditions. It is also required to protect the power plant and its environs from damage due to erratic power plant operation. The possible sources and causes of such erratic behavior are: power plant failures, control system failure, facility and auxiliary failures, and operator error.

The safety philosophy for a maritime application of nuclear power will probably differ, to some degree, from that for the airborne installation toward which ANP system designs were formerly directed. The control and operating system and procedures will also differ and thus effect the safety system requirements. The environmental requirements are quite different for the two applications. The less severe nuclear radiation and temperature environment of the maritime application allows a wider latitude in the choice of the mechanization method used.

A safety system engineering study was begun this reporting period by making a literature search: (1) to gain a better feel for shipboard safety system philosophy, (2) to determine the types of mechanization that have been used, and (3) to determine the state of the art with respect to the application of transistors and static switches to safety circuitry. Static switches may provide improved reliability over the mechanical relays used in the ANP-developed systems. The findings to date indicate that solid-state devices have been successfully applied to several safety systems and they should be seriously considered for the 630A system.

7.2 TWO-TUBE BOILER TEST RIG (198)

Bids were received from vendors for the remainder of the major parts required for building the test rig. The only bid received for the boiler shell assembly quoted 15-weeks delivery, which means that the test cannot begin for about 18 weeks. The only other major hardware for which placement of an order was delayed were the boiler coils, for which delivery was quoted at 8 weeks.

Instructions were written and orders were placed in the shops for fabrication of nonstandard components, for the buildup of subassemblies, and for installation of the complete test rig in the test facility.

Writing of detailed test instructions, and test operating procedures were begun. Interconnecting instrumentation and control wiring diagrams required for installation of the test rig in the test facility were completed.

7.3 WORK PLANNED FOR NEXT PERIOD

The literature search and evaluation of safety systems will be continued. A study of the safety system functional requirements and methods of mechanizations will be begun. Procurement of two-tube boiler test rig parts will be continued, and fabrication work will be started.

8. PRIMARY COOLANT SYSTEM

(200)

During this report period, work was begun on specifications for the blower final design, fabrication and testing, and for the design of a test rig for evaluating blower shaft air seals.

8.1 SEAL TEST

The only component in the blower which may require development in order to attain the minimum desired life of 15,000 hours is the shaft air seal. Therefore, a seal evaluation program will be conducted this fiscal year in order to better assure the availability of a seal design which will meet the requirements for performance, life and reliability for the final design of the blower. All seals described in the previous progress report* may be evaluated during this phase. The designs and the test results may be included with the detail specifications for the blower which will be made available to the vendors from whom bids will be solicited. The vendors will be given the option of using the design recommended or of proposing a different seal design for which they will be required to show evidence of satisfactory performance, life and reliability.

Work was begun on the design of a test rig which could simultaneously test three different seal configurations on a common shaft. A preliminary layout is shown in Figure 8.1. In addition to the presumed advantages of testing three seals at the same time, the requirement for a thrust bearing was eliminated since the seals operated back-to-back. By eliminating the thrust bearing, the power requirements for the drive was only 6 hp. However, after discussing the design with other engineers who had attempted to use a multiple seal tester, and after consideration of the problems inherent in isolating the seals to obtain good usable test data, the multiple-seal test rig was abandoned in favor of a single-seal test rig.

A layout of a test rig capable of testing any one of the three seals under consideration is shown in Figure 8.2. In order to simulate blower operation, one side of the seal must be pressurized up to 400 psia. This produces a thrust load of approximately 11,000 pounds on the shaft. The following four methods of overcoming this thrust load have been considered:

1. A hydrodynamic thrust bearing of a commercially available type which would operate with low pressure oil supplied from an external source. This bearing would have a friction loss of approximately 60 hp.
2. Tapered-roller or ball-thrust bearings were investigated but no standard model could be found to meet the requirements.
3. A GE-NMPO designed hydrostatic thrust bearing supplied with 10 gpm of oil at 800 psig. The friction hp loss in this would also be under 20 hp.
4. A GE-NMPO designed hydrostatic thrust bearing supplied with about 700 scfm of air at 400 psig. The friction loss would be only about one hp but the power required to supply the air would be about 200 hp.

*Ibid., pp. 47–50.

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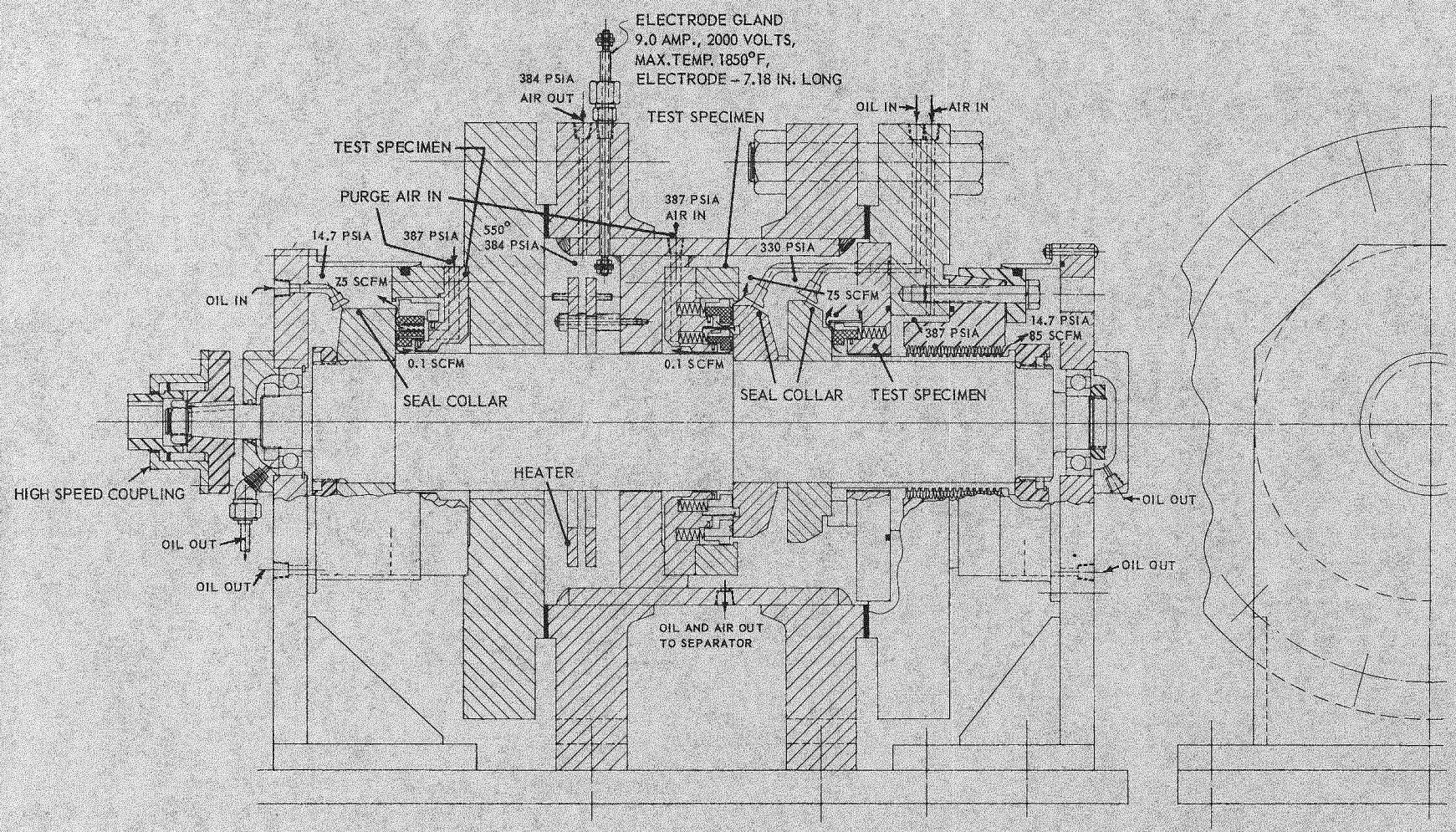

Fig. 8.1 – Study seal test chamber (Dwg. 692E612)

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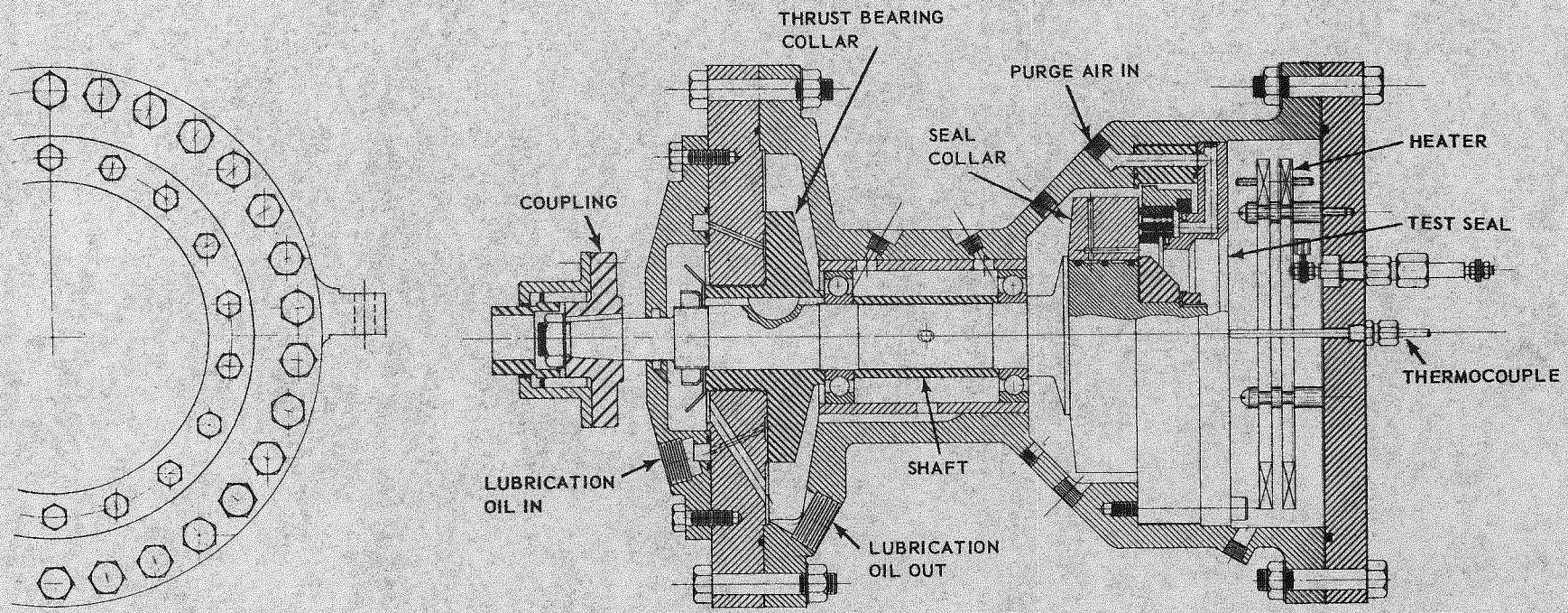

Fig. 8.2 – Study No. 2 seal test chamber (Dwg. 692E615)

After reviewing the merits of each of these systems, the hydrostatic bearing was chosen as best for this application.

Two types of drives are being considered. One is an air turbine, directly coupled to the test chamber drive shaft, and the other is an electric motor driving through a step-up gear box. At the present time, an attempt is being made to obtain a used air turbine drive on a temporary loan basis. An electric motor of the required capacity is available but a gear box would have to be procured.

An electric heater is to be mounted inside the test chamber to heat the air to the temperatures expected to exist in the actual blower. Removal of the cover plate which contains the heater allows direct access to the seal being tested.

A schematic diagram of the test setup is shown in Figure 8.3. Oil for the bearings and for cooling the shaft-mounted sealing collar is provided by an existing hydraulic test stand. The air supply for pressurizing the seal is provided by an existing high-pressure boost compressor. The flow to the seal is regulated to maintain a fixed drop across the seal as in the actual blower. Pressure in the test chamber is maintained at the desired constant value by a back-pressure regulating valve.

8.2 BLOWER REQUIREMENTS

Writing of a detail specification for the design, fabrication and qualification testing of both the main and auxiliary blowers was begun. The blower study described in the previous report* is serving as the primary basis for the specifications.

8.3 WORK PLANNED FOR NEXT PERIOD

The preliminary detail specification for the blower will be completed. Design of the seal test rig will be completed through the check print stage. Completion of the drawings will be delayed pending results of developmental testing to be done at the seal vendor's plant. An order will be placed with a seal vendor for development models of blower shaft air seals and for their development testing.

*Ibid., pp. 45–50.

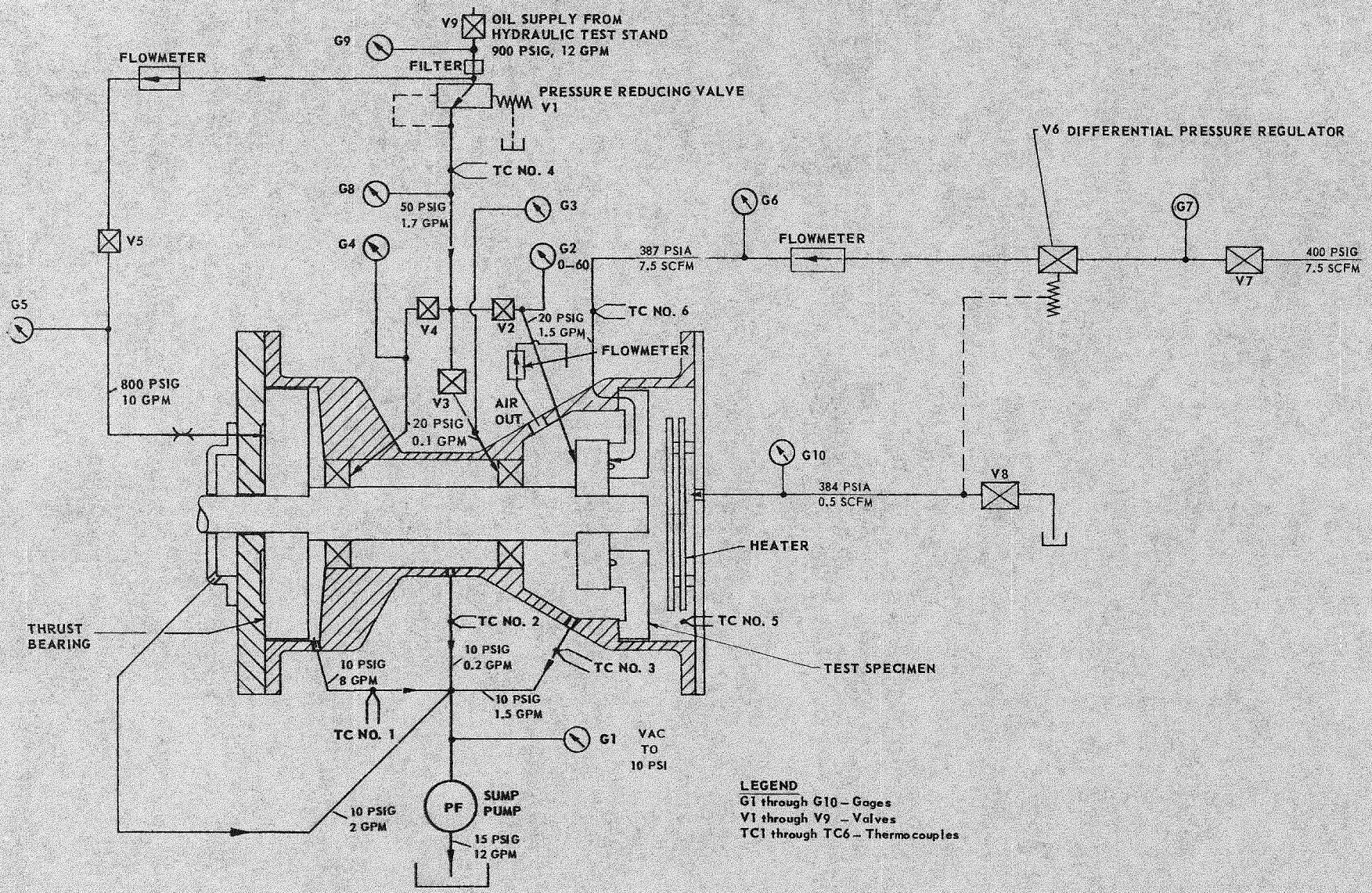

Fig. 8.3 – Schematic No. 1 seal test rig hydraulic – pneumatic (Dwg. 848D652)

9. FUEL MATERIALS

(240)

During this reporting period, the long-term in-pile test of 80Ni - 20Cr fuel sheet, ORF-2, was terminated because of mechanical damage to the test capsule. Prior to termination the specimen had been on test at 815°C for 876.5 hours, and had reached an estimated burnup level corresponding to approximately 8 percent of the U^{235} atoms. Preliminary examination of the specimen, EB-13, indicated no evidence of damage or defects. Post-test investigation is continuing. A new test, designated ORF-3, is being prepared.

Thermal cycle specimen EB-6 has accumulated to date over 3200 hours at 870°C (1600°F), with 28 rapid cooling cycles to room temperature, and appears to be in excellent condition.

The design of the capsule for the MTR fueled ring test, 3-F-1, has been completed and fabrication is underway. The fueled ring specimen for this test has been prepared.

Pile ambient irradiation of smooth stress-rupture specimens of the 80Ni - 20Cr cladding alloy has been completed. Additional specimens are being prepared for irradiation at 815°C (1500°F).

Creep specimens of fueled sheet have been on test for times up to 3400 hours at temperatures of 700° to 870°C (1300° to 1600°F) and stresses of 300 and 600 psi. Maximum elongation measured to date has been 0.30 percent in a 2-inch gauge length. Creep specimens of cladding stock show elongations of <0.8 percent after approximately 5000 hours on two specimens, and 2.3 percent in approximately 6500 hours on one specimen.

Oxidation tests have been initiated on GE-81 brazing alloy containing varying amounts of silicon.

Preparations are being made to produce approximately 750 fueled rings for the 630A critical experiment and to begin the processing of experimental ribbon to evaluate the effects of varying cladding thickness and spherical fuel on processing and properties.

9.1 IN-PILE TESTING (241)

9.1.1 ORR TESTS

2200-Hour Test

The results of this test are described in detail in High-Temperature Materials Program Progress Report No. 15, Part A, GEMP-15A, dated September 30, 1962. A brief description of the results is presented in the following paragraphs.

The long-term in-pile test (ORF-2) was initiated in the F-2 facility of the ORR on April 16, 1962. By June 25, 1962, the test specimen had achieved a total exposure time

of 876.5 hours at approximately 815°C (1500°F) when the reactor was shut down for refueling. This time is equivalent to approximately 6000 hours of 630A operation at average temperature higher than that expected in the 630A. At this time, the capsule was removed from the facility for insertion of another experiment and stored in-pool. Later on during the shutdown period, the capsule was observed to have been damaged and an extensive indentation was visible as shown in Figure 9.1, a telephotograph of the damaged capsule as it lay in-pool on the reactor grating. Two additional indentations were observed on the opposite side of the capsule shell. The cause of the damage and the time at which it occurred are not known. As a result of the damage, the ORF-2 test was discontinued, and the capsule was severed from its in-pile hardware and returned to GE-NMPO. At the time of discontinuance, an estimated burnup of approximately 8 percent of the U^{235} atoms had occurred in the specimen fuel sheet.

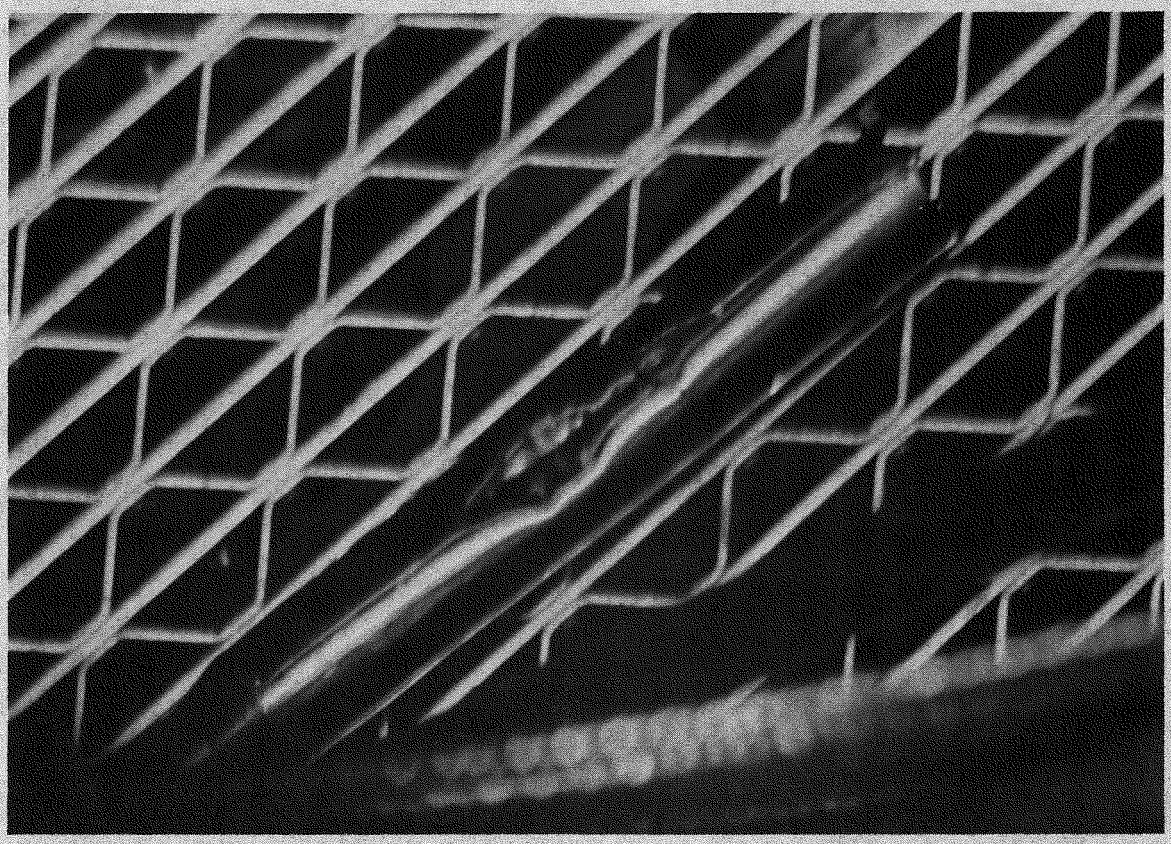

Fig. 9.1 – Telephoto view showing damaged capsule resting in pool on the reactor grating (57972)

During capsule disassembly at GE-NMPO, it was observed that the damage was apparently limited to the outer shell. The two inner cans and the venturi were not damaged. The condition of the specimen was found to be excellent. The cladding surface was oxidized but no blisters, pin-holes, cracks, or ruptures could be detected after thorough inspection. Figure 9.2 illustrates the appearance of the cladding and back-plate surface of the fueled specimen. The post-test investigation is continuing.

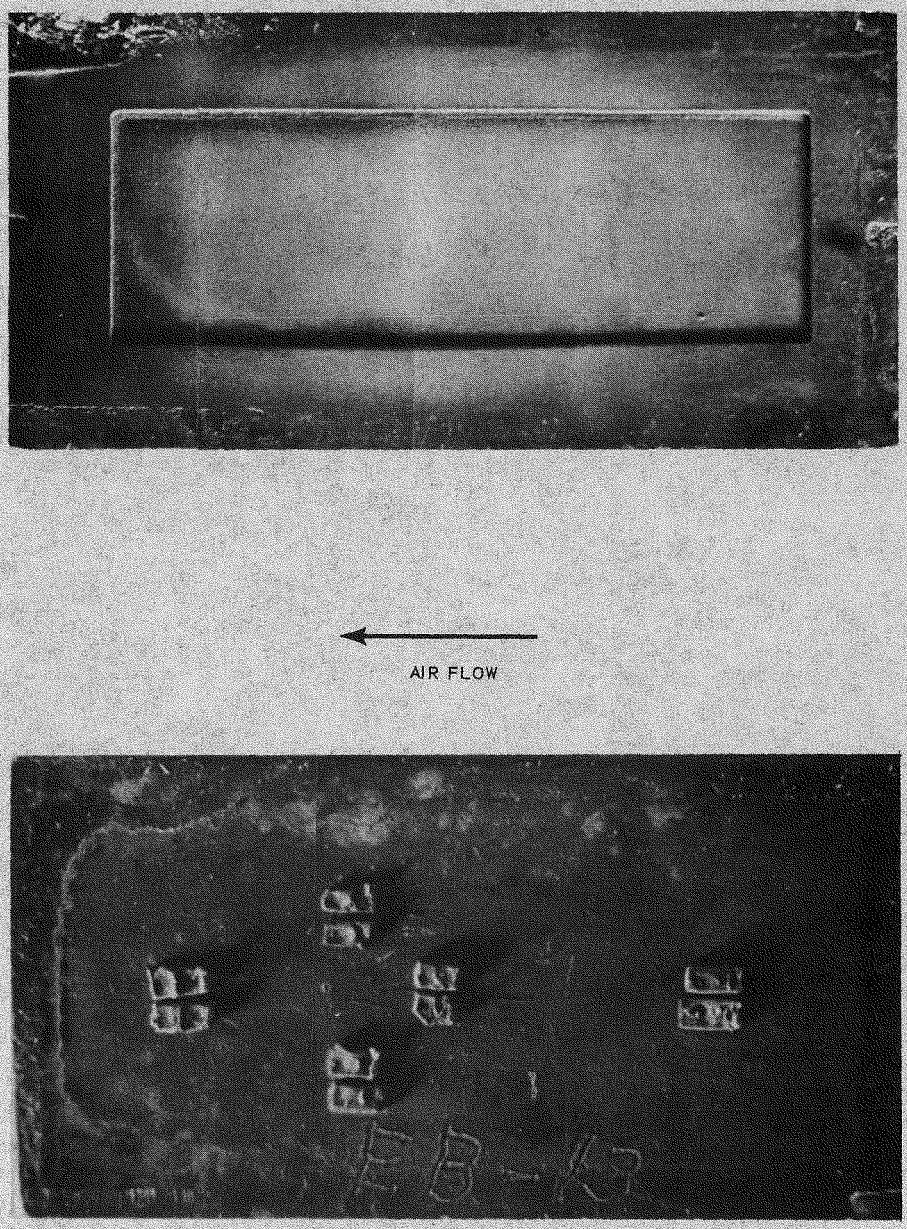

Fig. 9.2 – Cladding and back plate surfaces of 80Ni – 20Cr fuel sheet specimen EB-13 after 876.5 hours at 815°C and 5×10^{13} nvt flux in the ORR

During the 876.5-hour test period, five charcoal traps were introduced at intervals into the system to determine effluent activity. The first four traps showed no significant levels of fission products. The fifth trap which, unlike the other four, was not refrigerated and was maintained on the line for 115 hours, showed some evidence of fission products even though line monitors during this period indicated no significant levels of activity above background. The isotopes found to be present in the fifth trap were principally Xe^{133} and Xe^{135}. Only a trace of I^{131} was detected. Based on the appearance of the specimen and the low iodine activity, and assuming that the fission products came from the specimen, it is tentatively concluded that the fission products detected resulted from thermally enhanced diffusion and not from escape through any defect in the cladding. It is also regarded as possible that back diffusion products from the main effluent line might have been responsible for the trap activity. This has been observed in other tests on unfueled specimens.

At the close of the reporting period, a new specimen-capsule assembly, designated ORF-3, was being prepared for irradiation in the ORR in another attempt to study the effect of high burn-up levels on 80Ni - 20Cr fuel sheet.

Thermal Cycle Test

Fuel sheet specimen EB-6, identical to the one used in the long-term in-pile test, has been on test in the laboratory for approximately 3200 hours at 870°C (1600°F) in air. During this period, it has been rapidly cycled 28 times to room temperature. The specimen was given a detailed inspection after approximately 3150 hours. No blisters or other defects of any consequence were observed. Figure 9.3 illustrates the generally excellent condition of the cladding surface. EB-6 will remain on test indefinitely.

Fig. 9.3 – Cladding surface of thermal cycle specimen EB-6 after ~3150 hours at 870°C (1600°F) and 28 rapid cooling cycles to room temperature (P62-8-3C and P62-8-3D)

9.1.2 MTR TEST

Preparations are continuing for irradiation test 3-F-1 in the A-19 facility of the MTR. This is to be a high burnup test of a 0.9-inch diameter fueled ring specimen at 790°C (1450°F). Capsule design has been completed and fabrication of components has begun.

Fuel Ring Preparation

A fuel ring specimen containing 37 weight percent enriched UO_2 in the core has been prepared and inspected. Ring joining was accomplished by Method C as described in the previous progress report.* During its preparation, the ring specimen was subjected to repeated visual and radiographic examination to determine the integrity of the fuel sheet and to detect any adverse effects caused by processing. After the ring joining step in the process, the specimen was given a diffusion heat treatment in hydrogen at 1040°C (1900°F) for 25 hours to homogenize the composition in and adjacent to the braze joints. Following this treatment, the ring was proof tested at 425°C (800°F) for 24 hours in static air to detect any gross core exposure. Final inspection consisted of stereomicroscopic and radiographic examination for surface, joint and core defects. The specimen was also alpha counted to detect if any abnormal surface contamination existed. Having passed inspection, the specimen has been stored to await encapsulation in the 3-F-1 test assembly.

Detailed chemical analysis and metallographic examination of a segment from the same fuel sheet used for the 3-F-1 test specimen are being conducted.

Instrumentation

Ten chromel-alumel thermocouples are to be used to measure the temperature at various positions on the other surface of the ring specimen. Eight of the ten specimen thermocouples will be of the "contact" type in which the thermocouple leads are attached to a small 80Ni - 20Cr tab which rests against the specimen. The other two thermocouples are to be spot welded to the joint plate and dead edge of the specimen. To avoid any adverse effects on specimen performance, it is not planned to spot weld any thermocouples directly to the fuel sheet cladding.

The accuracy of the contact thermocouples is being checked out using an instrumented dummy specimen in an induction coil. Preliminary data show that they indicate temperatures within 30°F of the actual surface temperature as measured by spot welded thermocouples. Efforts are being made to improve the accuracy of these contact thermocouples.

Thermal Cycle Test

Oxidation and thermal cycle bench testing of a fueled 3-F-1 type ring specimen was initiated during this report period. A fueled ring was instrumented and inductively heated to 790°C (1450°F) while 30 scfm of filtered shop air was passed through the bore. The outside diameter of the specimen was wrapped in a blanket of thermoflex insulation as planned for the 3-F-1 specimen. Examination of the specimen after 207 hours of operation under these conditions revealed no defects. The temperature was then increased to 1040°C (1900°F) in order to develop in a relatively short time any time-temperature effects which might be anticipated in 5000 hours at 790°C (1450°F). After 18.5 hours at 1040°C (1900°F) the test was terminated due to localized melting and burn-through of the specimen. Subsequent investigation revealed that failure was caused by irregular spacing of the induction heating coil which caused local overheating. Corrections have been made in the test equipment and a dummy ring specimen has since been heated to 1040°C (1900°F) for 135 hours with no adverse effects.

These tests are continuing in order to improve instrumentation techniques and, ultimately, to indicate the endurance of these ring type specimens at 790°C (1450°F) under oxidizing and thermal cyclic conditions.

Tensile Testing of Brazed Joint Configurations

The tensile strength of the several types of ring joint described in the previous report are to be determined at 650°C (1200°F) and 815°C (1500°F) using unfueled 80Ni - 20Cr

*Ibid., p. 63.

sheet. Preparation of the specimens is 90 percent complete and testing is expected to be completed during the next reporting period.

9.1.3 EFFECTS OF RADIATION ON 80Ni - 20Cr ALLOY

The irradiation of twenty niobium modified 80Ni - 20Cr smooth stress rupture specimens was completed. The specimens were irradiated in capsule MT-74 in the L-6 facility of the ETR and received 565 hours of exposure in water at pile ambient temperature. Pending activation analysis of the nickel-cobalt dosimeters attached to the capsule, total neutron dosage is estimated to be approximately 2×10^{20} nvt ($E_n \geq 1$ Mev). The capsule has been removed from the reactor and is currently awaiting disassembly at GE-ITS.

Material has been prepared and an order placed for fifty additional niobium modified 80Ni - 20Cr smooth stress rupture specimens. Twenty-two of the specimens have been received and are of good quality; the balance were rejected and returned to the vendor for rework. Approximately one-half of these specimens will be irradiated at 815°C (1500°F) for at least four reactor cycles in the L-6 facility of the ETR in order to achieve a dosage level equivalent to that anticipated in the cladding during the full operational life of the fuel elements in the 630A power plant. Design of the test capsule is underway.

9.2 PROPERTY STUDIES (242)

9.2.1 CREEP AND OXIDATION TESTING

Fuel Sheet Creep Tests

Data are being accumulated on the creep resistance of 80Ni - 20Cr clad fuel sheet to establish its applicability to the 630A maritime nuclear steam generator. Specimen dimensions are approximately 1-1/2 inches by 9 inches by 0.020 inch. The fueled core contains approximately 42 weight percent UO_2. Detailed descriptions of the specimen configuration and preparation were presented in the previous progress report.*

The test conditions and current data are recorded in Table 9.1. "B" series and "W" series specimens are being tested in tandem with "C" series specimens; i.e., two specimens are mounted in parallel in the same furnace but are separately loaded. In the thermal cycle tests, the specimens are rapidly cooled to approximately 200°C (400°F) every 50 hours. All specimens are removed from test for inspection and measurement approximately every 500 hours. Inspection consists of surface alpha count, X-ray and stereomicroscopic examination of the cladding and edge seal. No gross increase in alpha activity has been detected nor have any significant cladding defects been observed.

Since the tests are indirectly a measure of edge seal integrity, the methods of sealing will be briefly reviewed. The eight "C" series specimens are edge sealed using brazed 80Ni - 20Cr foil and wire to form a channel around the specimen edges. The two "B" series specimens are sealed simply by a brazed wire, providing a specimen with a more or less constant cross section. The "W" series specimens are sealed by electron beam welding the edges which were previously slotted and filled with an 80Ni - 20Cr wire insert.

Cladding Stock Creep

Over 5000 hours of test time have been accumulated by the three cladding stock creep specimens. One of the specimens has been on test for 6500 hours with less than 2.3 percent elongation at the last measurement. The other two specimens show less than 0.8 percent elongation and a linear creep rate of 0.004 percent per 1000 hours over the previous 4000 hours. The test conditions and data to date are presented in Figure 9.4.

*Ibid., pp. 74–76.

TABLE 9.1

CREEP TEST DATA ON 80Ni - 20Cr FUEL SHEET

Specimen	Edge Seal Type	Temperature, °C/°F	Average Stress, psi	Time, hr	Cycles	Elongation, %
Isothermal Tests						
C-1	Brazed Channel	870/1600	300	3413	-	0.10
C-2	Brazed Channel	790/1450	300	3271	-	0.15
C-4	Brazed Channel	700/1300	300	2833	-	0.20
C-8	Brazed Channel	790/1450	600	2940	-	0.10
B-2	Brazed Wire	On test with C-4	300	1740	-	nil
W-2	Welded Insert	On test with C-8	600	864	-	nil
Thermal Cycle Tests						
C-3	Brazed Channel	870/1600	300	3115	54	0.25
C-5	Brazed Channel	700/1300	300	2920	51	nil
C-6	Brazed Channel	790/1450	600	2597	46	0.25
C-7	Brazed Channel	790/1450	300	2910	50	0.30
B-1	Brazed Wire	On test with C-5	300	2206	38	nil
W-1	Welded Insert	On test with C-6	600	917	16	nil

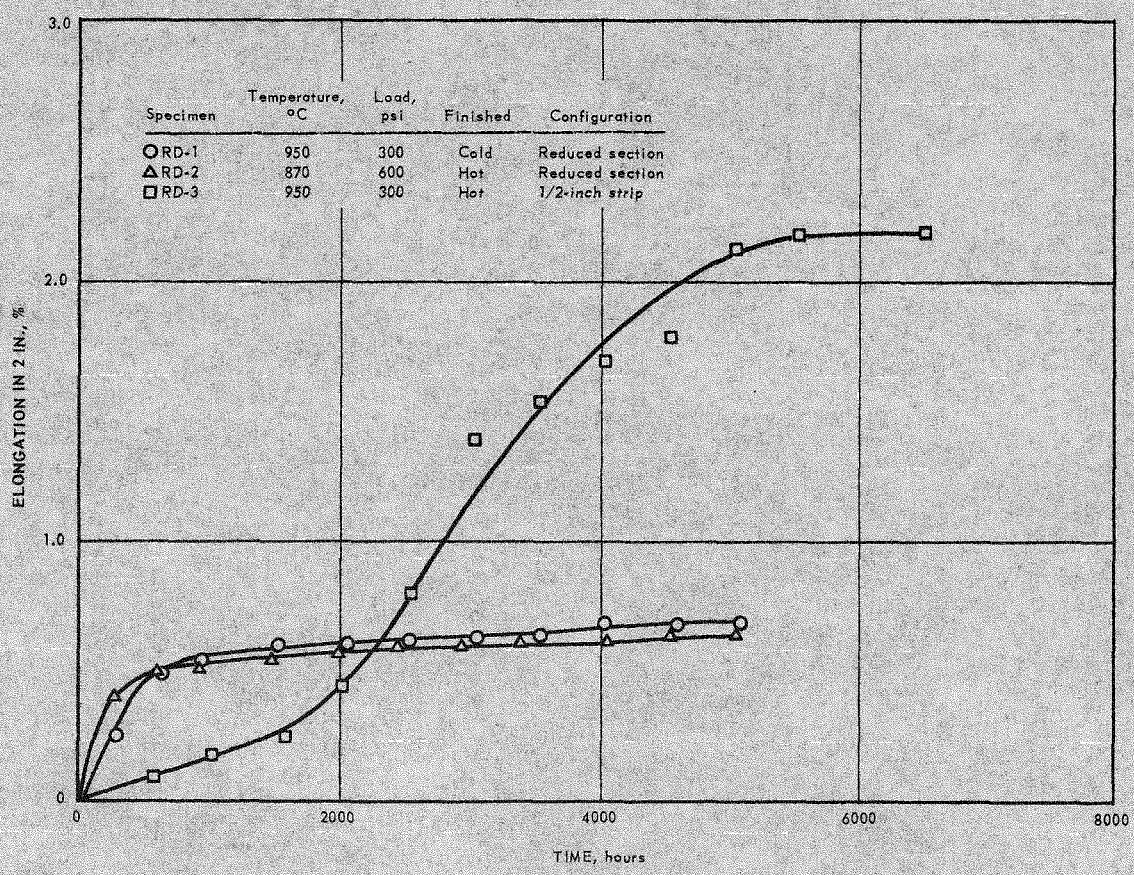

Fig. 9.4 – Results to date of creep tests of 80Ni – 20Cr cladding stock

Stress Distribution

In the previous report, it was stated that a non-uniformity in stress exists across the width of specimen C-4. This was based on strain gage measurements of the specimen made at room temperature prior to and after several periods of testing. Since then, an unfueled 80Ni - 20Cr fuel strip of constant thickness and the same over-all dimensions as C-4 showed a similar stress distribution, indicating the effect is not due to the edge frame. The most significant aspect of the stress distribution across these specimens is the relatively high edge stresses which exist.

It has been suggested that the disparity in creep behavior in the cladding stock specimens (Figure 9.4) is the result of relatively high stresses near the edge of uniform width specimen RD-3 as compared to those which exist in specimens RD-1 and RD-2, each of which has a reduced section. These relatively high edge stresses occur near the specimen gage marks, which are located in the frame and are used to measure elongation. Evidence to support this suggestion has been found in strain gage measurements which have been made across the width of a flat sheet specimen in which a reduced section was machined. The test specimen was approximately 1.25 inches by 12 inches by 0.020 inch, with a 2-inch-long section at the center reduced in width to 0.5 inch. Results of these measurements are shown in Table 9.2 as compared to similar measurements on a specimen of uniform width. It is obvious that the reduced section configuration markedly equalizes the stress distribution across the narrow width and that a significant differential exists in the specimen of uniform width.

TABLE 9.2

COMPARATIVE STRESS PATTERN MEASUREMENTS FOR SHEET CREEP SPECIMENS WITH AND WITHOUT A REDUCED SECTION

	Specimen With Reduced Section				Uniform Width Specimen			
		Average Stress, Front and Back, psi				Average Stress, Front and Back, psi		
Load, lb	Nominal Stress, psi	Edge	Center	Edge	Nominal Stress, psi	Edge	Center	Edge
3	300	372	341	310	120	403	-186	443
6	600	651	620	620	240	558	-124	558
9	900	961	961	930	420	775	0	744

In the case of fuel sheet specimen C-4, the results of additional stress measurements at 1988 test hours indicate that the previously reported non-uniform stress distribution has not been significantly altered with time and temperature. The stress pattern was identical to that reported previously for 529 hours.*

9.2.2 COMPARATIVE CREEP AND OXIDATION TESTING

Fuel sheet has recently been prepared utilizing both cold and warm finishing after initial hot reduction. Studies made during the aircraft program indicated that warm finishing produced improvement in the creep properties of fuel sheet, at least at the higher temperatures and relatively short times associated with that program. Whether warm finished fuel sheet possesses a creep strength advantage over the cold finished sheet for the relatively long times and low temperatures associated with the 630A application will be evaluated in a series of creep tests to be conducted on the new warm and cold finished sheet. The specimens will incorporate reduced sections where possible, and tests will be conducted similarly to the current studies.

*Ibid., p. 72.

9.2.3 TRANSVERSE STRENGTH OF FUEL SHEET

Specimens have been prepared which will permit determination of the short-time transverse tensile strength of cold and warm finished 80Ni - 20Cr fuel sheet. The fuel sheet is of recent manufacture as previously reported.* A sketch of the brazed specimen assembly appears in Figure 9.5. Figure 9.6 is a photograph of two specimens machined into tensile bars ready for testing. Each bar contains two fuel sheet segments, either both cold finished, both warm finished, or one of each finish. Specimens containing each combination will be tested at room temperature, 650°C (1200°F) and 815°C (1500°F) to determine comparative strength properties.

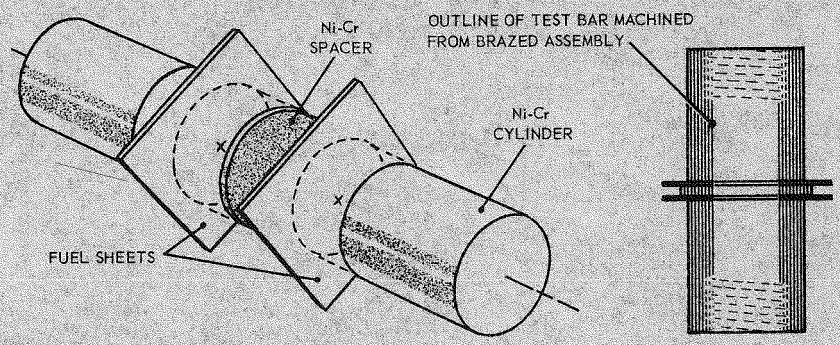

Fig. 9.5 – Components and assembly of brazed transverse tensile test specimens

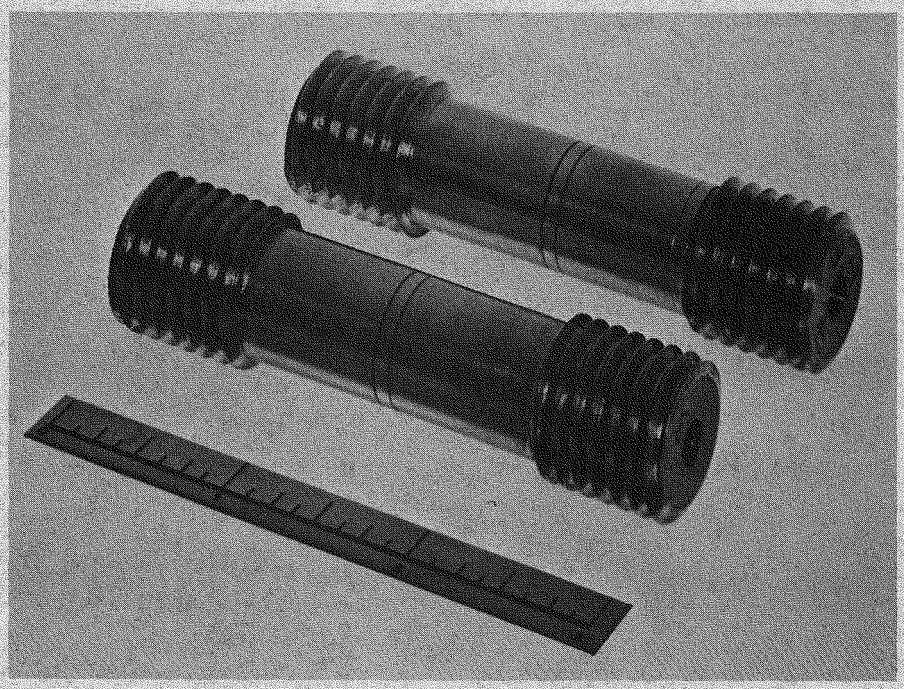

Fig. 9.6 – Tensile specimens for use in determining transverse strength of fuel sheet (P62-8-9)

9.2.4 OXIDATION RESISTANCE OF GE-81 BRAZE ALLOY

The oxidation resistance of braze joint areas is believed to be compromised somewhat by the high concentration of silicon contained in the GE-81 braze alloy. Normally, silicon content

*Ibid., p-74.

of the braze alloy approximates 10 weight percent. Thermal cycle specimen EB-6 and the long-term in-pile test specimen EB-13 both show significant thermal etching of braze areas. Accordingly, the long-term oxidation resistance of GE-81 alloy and other 80Ni - 20Cr base braze alloys with gradations of silicon are being studied. Wafers cut from a series of arc-melted button heats containing 0, 1, 3, 6, 10, 15 and 21 weight percent silicon are being subjected to air oxidation at 815°, 980° and 1090°C (1500°, 1800°, 2000°F). Significant differences in appearance and weight change have not been observed in the first 200 hours of testing.

9.3 FUEL SHEET PREPARATION (243)

9.3.1 EXPERIMENTAL FUEL SHEET

Preparations have begun to produce warm finished fuel sheet with different cladding thicknesses. All of the fuel sheet currently on test possesses a cladding thickness of approximately 0.004 inch. The experimental sheet to be prepared will possess cladding thicknesses of 0.006 inch and 0.008 inch. Comparative studies of creep strength and oxidation resistance will be conducted. Fuel preparation for this study has been started.

9.3.2 FUEL SHEET FOR THE 630A CRITICAL EXPERIMENT

It will be necessary to produce approximately 750 fueled rings of good quality, and of various diameters and thicknesses, for use in the 630A critical experiment's cell flooding tests. To date the necessary equipment for the production of fuel sheet and rings has been checked out, a work area has been selected and decontaminated, fuel specifications have been prepared, and a fuel allocation request has been submitted for approval. Thus far, three vendors have submitted price and delivery quotations on both depleted and enriched UO_2. An order will be placed with each vendor for a small quantity of depleted UO_2 to establish comparative quality for future reference. A larger order for the enriched fuel will be placed as soon as allocation approval is received so that billet fabrication can begin. All other raw materials, including metal powders, cladding stock and brazing alloy, are on hand.

The use of spherical fuel particles is being considered as a possible means of improving the uniformity of fuel dispersion in warm rolled sheet. It is planned to produce an experimental quantity of this sheet to establish rolling characteristics and the uniformity of dispersion which can be attained. A quantity of fuel of the proper particle size has been prepared and spheroidization will be attempted during the next reporting period. Eventually it is planned to produce experimental fuel sheet containing Ni- or Cr-coated spherical fuel particles.

9.4 WORK PLANNED FOR NEXT PERIOD

1. Complete post-test investigation of ORF-2.
2. Complete preparation of ORF-3 specimen-capsule assembly.
3. Initiate in-pile testing of ORF-3 in the ORR.
4. Complete preparation of 3-F-1 specimen capsule assembly.
5. Initiate in-pile testing of 3-F-1 in the MTR.
6. Initiate determination of transverse strength of fuel sheet.
7. Complete tensile tests of ring joint specimens.
8. Initiate preparation of experimental fuel sheet.

10. NON-FUELED MATERIALS

(250)

During this report period, procedures were established for the processing of poison tips for the 630A critical experiment control rods.

10.1 CONTROL ROD FOR THE 630A CRITICAL EXPERIMENT

Procedures have been formulated for the preparation and loading of the poison tips for the 630A critical experiment control rods. The procedures covered tube cleaning operations, welding sequence, loading of the filler material, and inspection of loaded tubes.

The filler material is 60-mesh technical-grade boron carbide powder containing approximately 70 weight percent boron. The powder is added to previously prepared AISI 304 stainless steel tubes 0.4-inch ID by 0.05-inch wall by 27.5 inches long. Vibratory packing is employed to produce a boron density of 1.00 ± 0.030 g/cc over the tube length.

11. CRITICAL EXPERIMENT MOCKUP

(260)

During this report period, factory drawings for the critical experiment mockup were completed, including those for the aluminum shims. The shims are being used for gross radial power shaping. Procurement and fabrication of components is about 85 percent complete.

11.1 COMPONENT DESIGN

During the report period, aluminum shims were added to the initial assembly of the critical experiment to displace the moderator outside of the fuel tube. The fuel-to-moderator ratio is varied with these shims for gross radial power shaping. Volume fractions and predicted gross radial power distributions with the initial aluminum configuration are shown in section 2.1.1. Figure 11.1 is a sketch of a radial cross section of the core and a longitudinal view of a cell tube showing the initial arrangement of the aluminum shims. As shown in Figure 11.1, the aluminum shims consist of an inner tube, that fits over the cell assembly tube, to which an additional tube and strips are fastened as required to achieve the desired moderator displacement. The additional tube and strips are located radially and longitudinally to the innermost aluminum tube, which rests on the rear tube sheet, by tabs that fit into clips at the bottom end and dimples that fit into a slot at the top end. The strips are added to reduce the circumferential scalloping due to moderator thickness variation around the cell. Longitudinally, these additional pieces extend from 3 inches above the active core to 4-7/8 inches below the active core. By the addition or removal of these pieces, the moderator displacement may be easily varied to achieve the desired power shaping in the critical experiment.

The factory drawings of the aluminum shims as well as the basic component drawings not previously issued have been issued during this report period.

11.2 PROCUREMENT AND FABRICATION

This phase of the work was curtailed by temporary funding limitations. Presently approximately 85 percent complete, the work has been fully reinstated. The major components have been shipped to the Idaho Test Station. Prior to shipping, a partial assembly of the components was made to insure that the fits and clearances were satisfactory. The status of the procurement and fabrication of the critical experiment components is shown in Table 11.1 which lists the required quantity of principal components and individual subassemblies or parts.

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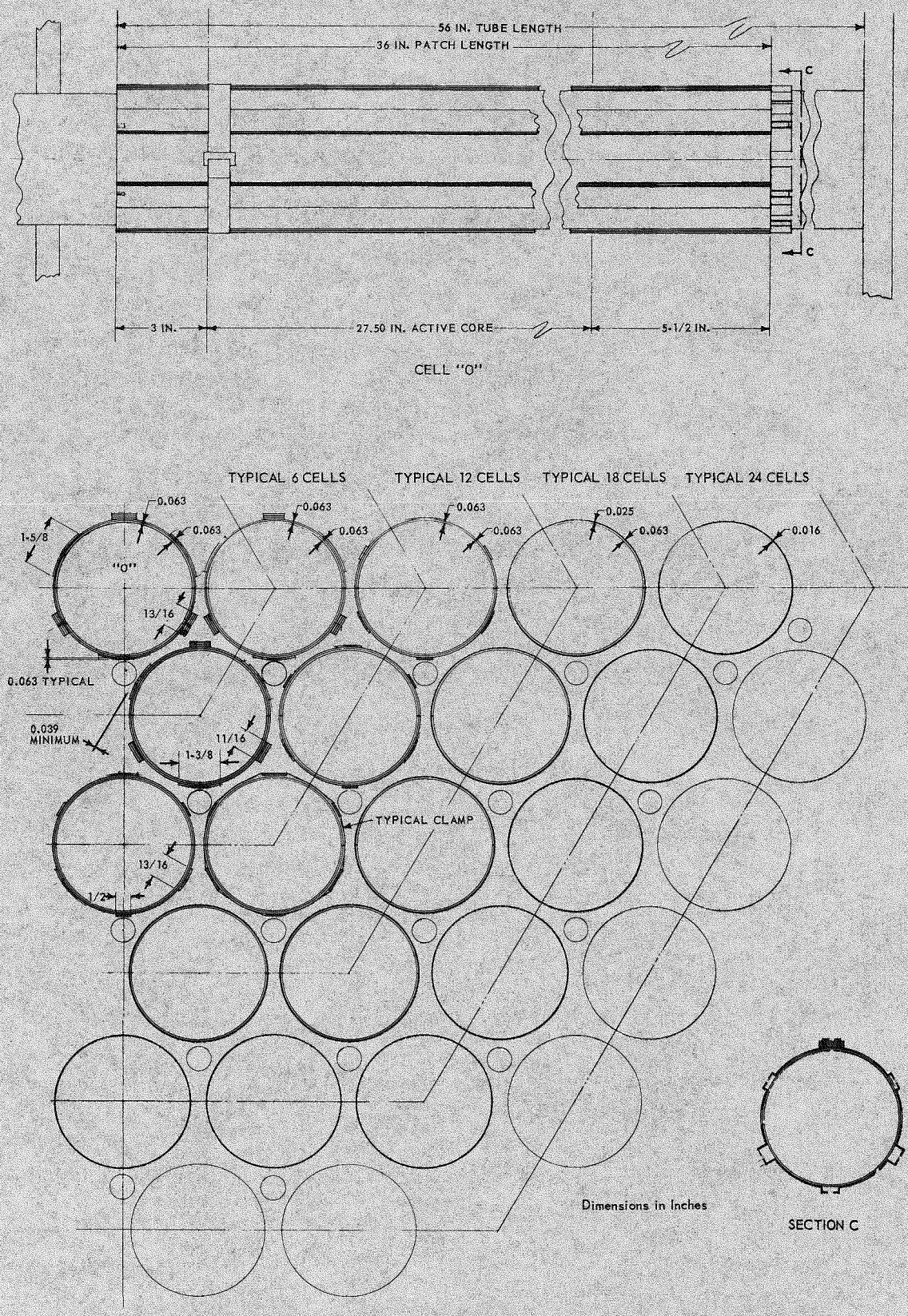

Fig. 11.1 – Arrangement of the aluminum shims employed to displace the moderator outside of the fuel tube to vary the fuel-to-moderator ratio for gross radial power shaping (219R868)

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TABLE 11.1

PERCENT COMPLETION OF THE PROCUREMENT AND FABRICATION OF THE CRITICAL EXPERIMENT COMPONENTS AS OF 9-1-62

Principal Components And Individual Subassemblies Or Parts	Required Quantity	Completion Of Procurement And Fabrication, %
Support, Center	1	100
Spider	1	100
Pin, locating	16	100
Bottom Tube Sheet and Pilot Assembly	1	100
Bottom tube sheet	1	100
Pilot	85	100
Hex-nut	85	100
Rollpin	85	100
Bushing, Bottom Tube Sheet	6	100
Screw	6	100
Ring Assembly (300-degree segment)	1	100
Ring, reflector No. 1	1	100
Ring, reflector No. 2 (300-degree segment)	1	100
Ring, reflector No. 3 (300-degree segment)	1	100
Slotted round head cap screw	18	100
Washer	13	100
Washer	13	100
Hex-nut	13	100
Internal hex socket head cap screw	13	100
Rollpin	1	100
Sheet-boral	1	100
Eye bolt	4	100
Ring Assembly (60-degree segment)	1	100
Locating pad	1	100
Boral sheet	1	100
Spacer	6	100
Reflector ring No. 2 (60-degree segment)	1	100
Reflector ring No. 3 (60-degree segment)	1	100
Dowel pin	2	100
Socket head cap screw	2	100
Slotted round head cap screw	8	100
Nut, regular jam	6	100
Socket head cap screw	6	100
Eye bolt	3	100
Pin	1	100
Reflector segment assembly, 60 degree	6	100
Beryllium reflector slab	1	100
Right-hand reflector corner	1	100
Upper reflector slab	1	100
Cap	1	100
Lower reflector slab	1	100
Left-hand reflector corner	1	100
Locating pin	6	100
Shot pin	2	100
Socket head cap screw	2	100
Be spacer	4	100
Socket head cap screw	4	100
Socket head cap screw	4	100
Dowel pin	2	100
Bracket, eye bolt L. H.	1	100
Bracket, eye bolt R. H.	1	100
Socket head cap screw	4	100
Cell Assembly	85	95
Base	1	100
Tube	1	100
Bracket and Top Tube Sheet Assembly	1	100
Top tube sheet	1	100
Bracket	1	100
Bracket	1	100
Socket head shoulder screws	6	100
Bushing Top Tube Sheet	4	100
Bracket, Guide Post	6	100
Socket Head Shoulder Screws	16	100
Column, Actuator Support	6	100
Shot Pins	7	100
Tube, Guide	72	80
Tube Assembly (source rod)	1	80

TABLE 11.1 (Cont'd)

PERCENT COMPLETION OF THE PROCUREMENT AND FABRICATION OF THE CRITICAL EXPERIMENT COMPONENTS AS OF 9-1-62

Principal Components And Individual Subassemblies Or Parts	Required Quantity	Completion Of Procurement And Fabrication, %
Clamp	60	85
Collar	13	100
Thumb Screw	13	100
Strap	13	100
Socket Head Cap Screws	73	100
Control Rod, Interstitial	72	50
Tip	1	85
Tube	1	100
Rod	1	85
Flange	1	100
Filler (boron carbide)	81.70 g	100
Control Rod Center, Type I	2	50
Plug	1	100
Tube	1	100
Rod	1	85
Lug	3	50
Filler (boron carbide)	247.60 g	100
Control Rod Center, Type II	8	50
Plug	1	100
Tube	1	100
Rod	1	85
Lug	3	50
Filler (boron carbide)	323.60 g	100
Control Rod Center, Type III	2	50
Plug	1	100
Tube	1	100
Rod	1	85
Lug	3	50
Filler (boron carbide)	409.85 g	100
Collar, Type I	2	100
Collar, Type II	8	100
Collar, Type III	2	100
Thumb Screw	12	100
Fuel Cell Assembly	66	75
Holder, fuel element	1	50
Retainer, fuel element	1	100
Water tube assembly, type I		50
Water tube assembly, type II	1	50
Water tube assembly, type III		50
Flange	1	100
Tube	1	100
Tube	1	100
Tube	1	100
Tube	1	100
Retaining ring	1	100
Nut, wing	1	100
Fuel Cell Assembly (measurement sector)	19	75
Holder - fuel element	1	50
Retainer - fuel element	1	100
Water tube assembly, type I		50
Water tube assembly, type II	1	50
Water tube assembly, type III		50
Flange	1	100
Tube	1	100
Tube	1	100
Tube	1	100
Tube	1	100
Retaining ring	1	100
Nut, wing	1	100
Key	1	100
Screw, internal hex head	2	100
Sleeve Assembly	1	90
Flange	1	85
Tube	1	100
Flat Head Cap Screws	2	100

TABLE 11.1 (Cont'd)

PERCENT COMPLETION OF THE PROCUREMENT AND FABRICATION OF THE CRITICAL EXPERIMENT COMPONENTS AS OF 9-1-62

Principal Components And Individual Subassemblies Or Parts	Required Quantity	Completion Of Procurement And Fabrication, %
Instrumentation Plug	1	50
Plug, reflector	1	0
Spacer	8	0
Plug	1	100
Plug	1	100
Plug	1	100
Plug	1	100
Spacer	2	0
Rod	2	100
Hex-nut	2	0
Bar	1	100
Disk	1	0
Rollpin	2	0
Rollpin	2	0
Plain flat washer	6	0
Plate, boral	1	50
Aluminum Shim Assembly No. 1	1	0
Cylinder and clip assembly	1	0
Ring split	1	0
Strap, flat	9	0
Strap, curved	3	0
Bands	2	0
Aluminum Shim Assembly No. 2	6	0
Cylinder and clip assembly	1	0
Ring split	1	0
Strap, flat	9	0
Strap, curved	3	0
Bands	2	0
Aluminum Shim Assembly No. 3	12	0
Cylinder and clip assembly	1	0
Ring split	1	0
Strap, flat	3	0
Strap, curved	3	0
Bands	2	0
Aluminum Shim Assembly No. 4	18	0
Cylinder and clip assembly	1	0
Ring split	1	0
Bands	2	0
Aluminum Shim Assembly No. 5	24	0
Sleeve	1	0

Figures 11.2 through 11.10 are photographs of some of the critical experiment components at various stages of completion.

Figure 11.2 shows the fuel element holder assembly. This assembly is the counterpart of the prototype reactor fuel cartridge and center moderator rod in the critical experiment mockup. The fuel elements are simulated with uranium foil interleaved with 80Ni - 20Cr foil wrapped on the three concentric ridged tubes (only the outermost tube is visible) to form the active core region. The outermost layer of foil is covered with a borated stainless steel foil, a burnable poison, as it is on the prototype fuel cartridge. The three disks above the active core region simulate the unfueled nichrome stage that serves as end reflector. Likewise, the four concentric tubes below the active core region (only the outermost tube is visible) simulate the two unfueled 80Ni - 20Cr end reflector stages. The flange at the top end of the assembly centers the assembly in the cell assembly tube and locates the fuel element holder assembly longitudinally in the critical experiment assembly.

Three different size water tubes, to be filled with water, fit within the fuel holder assembly to duplicate the center moderator rods. Center control rods of three comparable sizes fit within the water tubes to duplicate the safety poison rods in the prototype reactor.

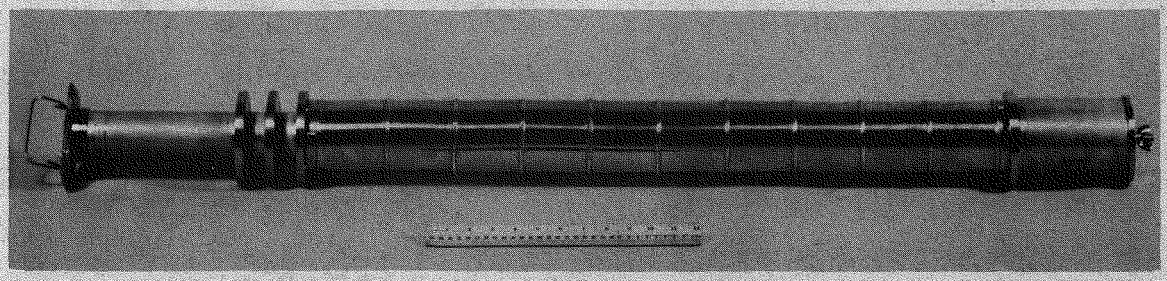

Fig. 11.2—Fuel element holder assembly (P62-8-12A)

Figure 11.3 shows the support spider as the 300-degree reflector ring assembly is being lowered into position. This assembly supports and positions the major components of the critical experiment mockup. The support spider is located in the lower portion of the outer water tank and is bolted to short columns that are mounted on the I-beam structure and extend through the bottom of the outer tank.* The lower tube sheet and pilot assembly is bolted to the bottom face of the inner posts of the spider. The 300- and 60-degree reflector ring assemblies rest on the machined top surface of the spider and are radially positioned by dowel pins in the top surface. Six columns that support the actuator support frame fit into the top of the outer posts of the spider.

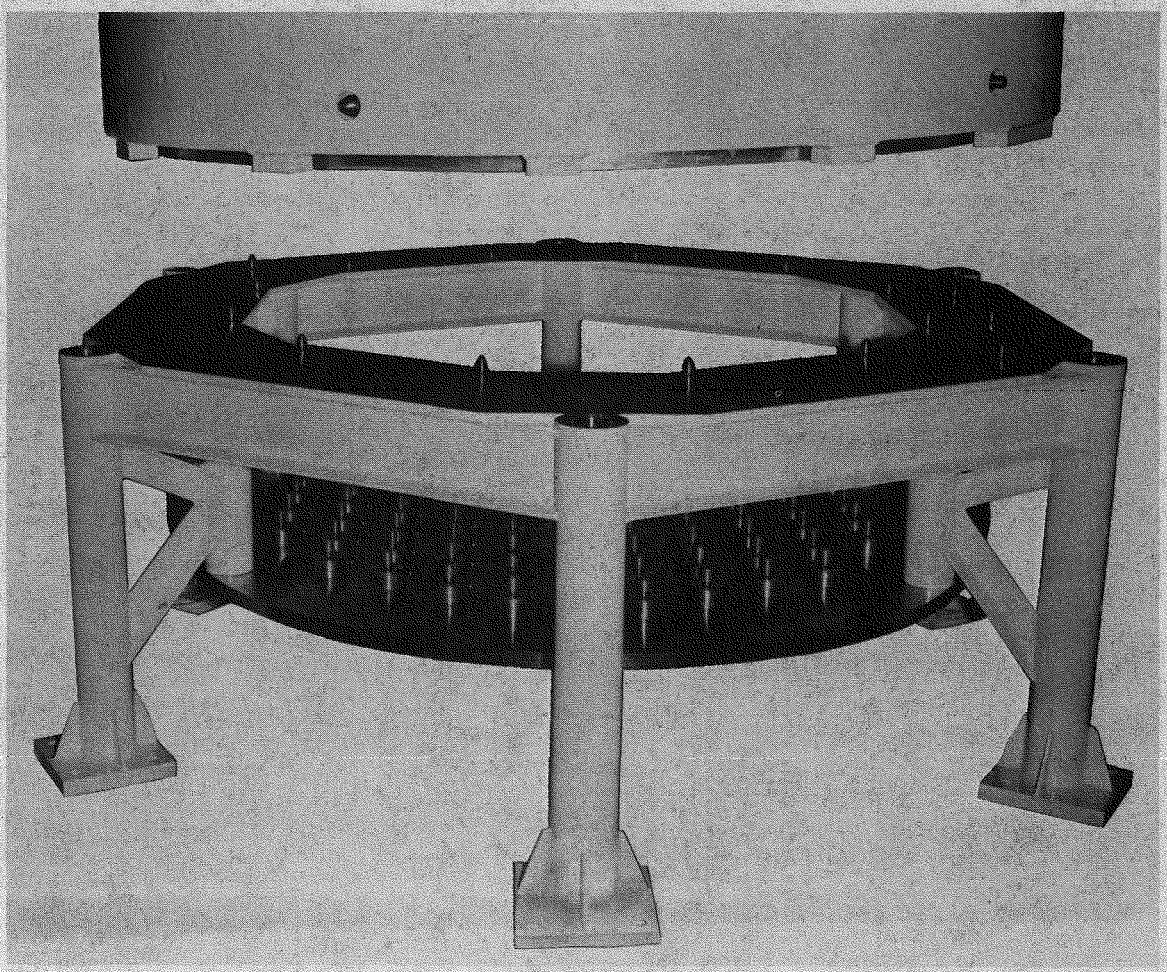

Fig. 11.3—Support spider (P62-9-2K)

*Ibid, p. 81.

Figure 11.4 shows a completed beryllium and stainless steel reflector assembly. Six of these assemblies are required in the critical experiment assembly to completely surround the 85-cell assembly tubes in the active core region. The inside surface of the beryllium is contoured to conform to the shape presented by the outside row of cell assembly tubes in the hexagonal array. The beryllium is 2 inches thick at the thinnest section and is 33.5 inches long, extending 3 inches above and below the active core. The beryllium is bolted to the rear stainless steel slab. Spacers maintain a 0.3-inch water gap between these components. The odd-shaped stainless steel corner pieces are held in position by dowel pins through the top stainless steel piece and the bottom piece. The 0.3-inch water gap is maintained between the corner pieces and the beryllium. The assembly rests on the top surface of the spider and is radially positioned by two dowel pins in the spider, and one dowel pin that extends through the top tube sheet to locate the inner surface of the beryllium approximately 0.4 inch from the outside diameter of the cell assembly tubes. Figure 11.5 shows six beryllium reflector slabs assembled on the spider as the trial assembly was made.

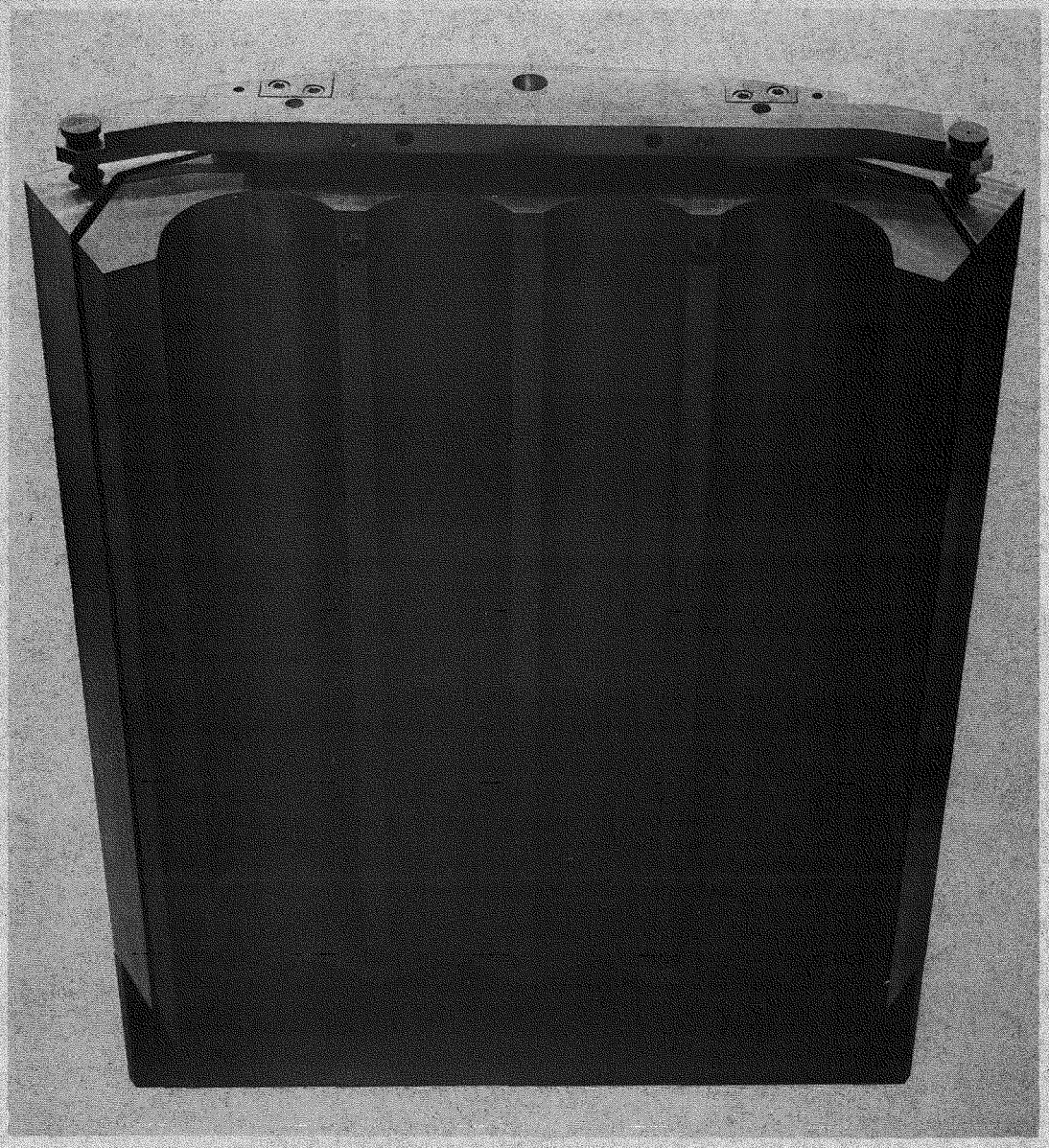

Fig. 11.4 – Beryllium and stainless steel reflector assembly (P62-8-16C)

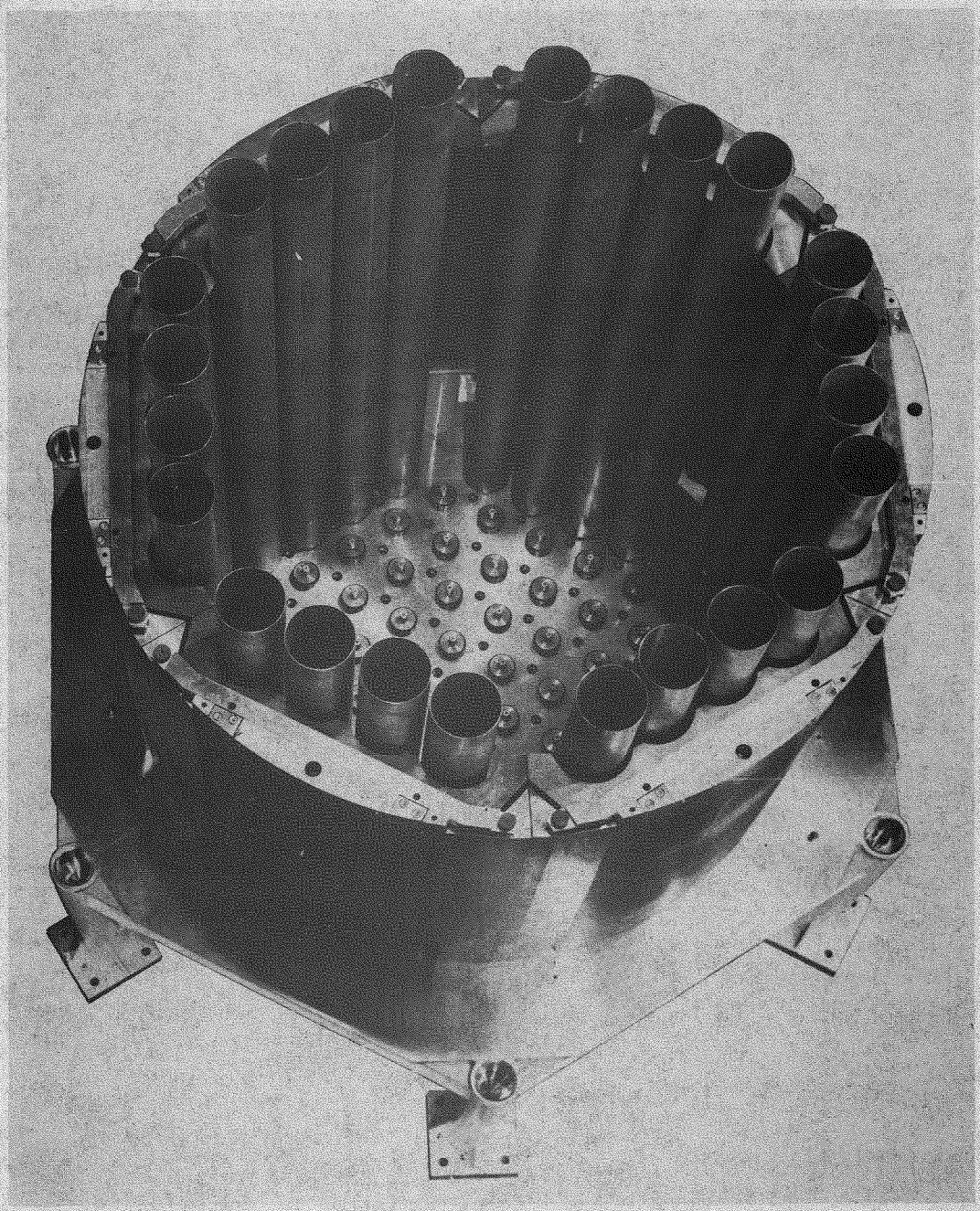

Fig. 11.5 – Be reflector assembled on spider (P62-9-2H)

Figure 11.6 shows the mild steel ring simulating the cylindrical shell of the prototype stainless steel reactor vessel. This ring rests on the top surface of the support spider and is radially located by two dowel pins in the support spider that extend into the bottom surface of this ring. The top tube sheet assembly and the brackets to locate the actuator support frame columns are secured to the top surface of the outer ring with shoulder screws. Three hundred degrees of the two inner mild steel reflector rings are bolted to the inside diameter of this ring. Water gaps of 0.3 inch are provided for cooling by spacers between each of these rings. The remaining 60 degrees of the inner reflector rings form an independent assembly that rests on the support spider and is radially positioned by dowel pins. The radial position of this 60-degree reflector ring assembly may be varied so the effect of varying the water gap can be determined. All mild steel components employed in the critical experiment assembly are painted with an epoxy resin paint to prevent rusting.

Figure 11.7 shows a 300-degree segment of one of the two inner mild steel reflector rings. These reflector rings simulate the inner stainless steel reflector cylinders in the prototype reactor. As previously explained, these segments of the reflector rings bolt to the outer reflector rings.

Figure 11.8 shows the completed assembly of the outer mild steel ring and the two 300-degree segments of the inner rings previously described.

Figures 11.9 and 11.10 show the partial assembly made to check the fits and clearances of the components prior to shipping the component to ITS.

11.3 WORK PLANNED FOR NEXT PERIOD

The procurement and fabrication of components will be completed during the next reporting period and all components will have been delivered to the Idaho Test Station.

Fig. 11.6 – Mild steel outer reflector ring (P62-8-4D)

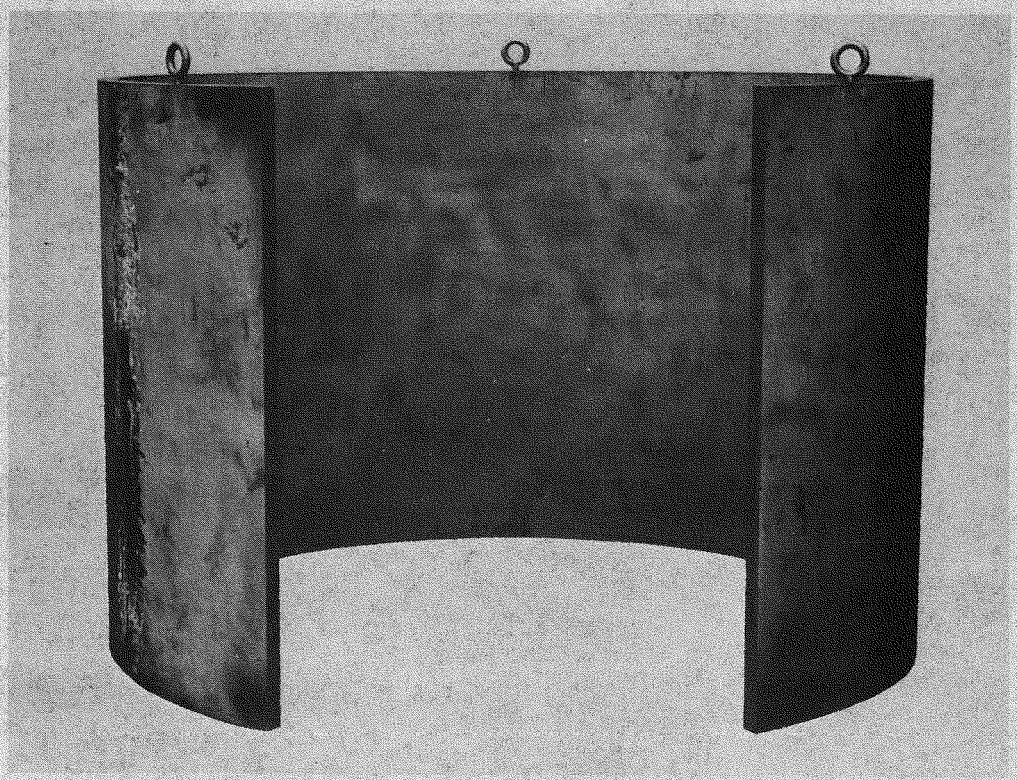

Fig. 11.7 – 300-degree segment – inner mild steel reflector ring (P62-8-4E)

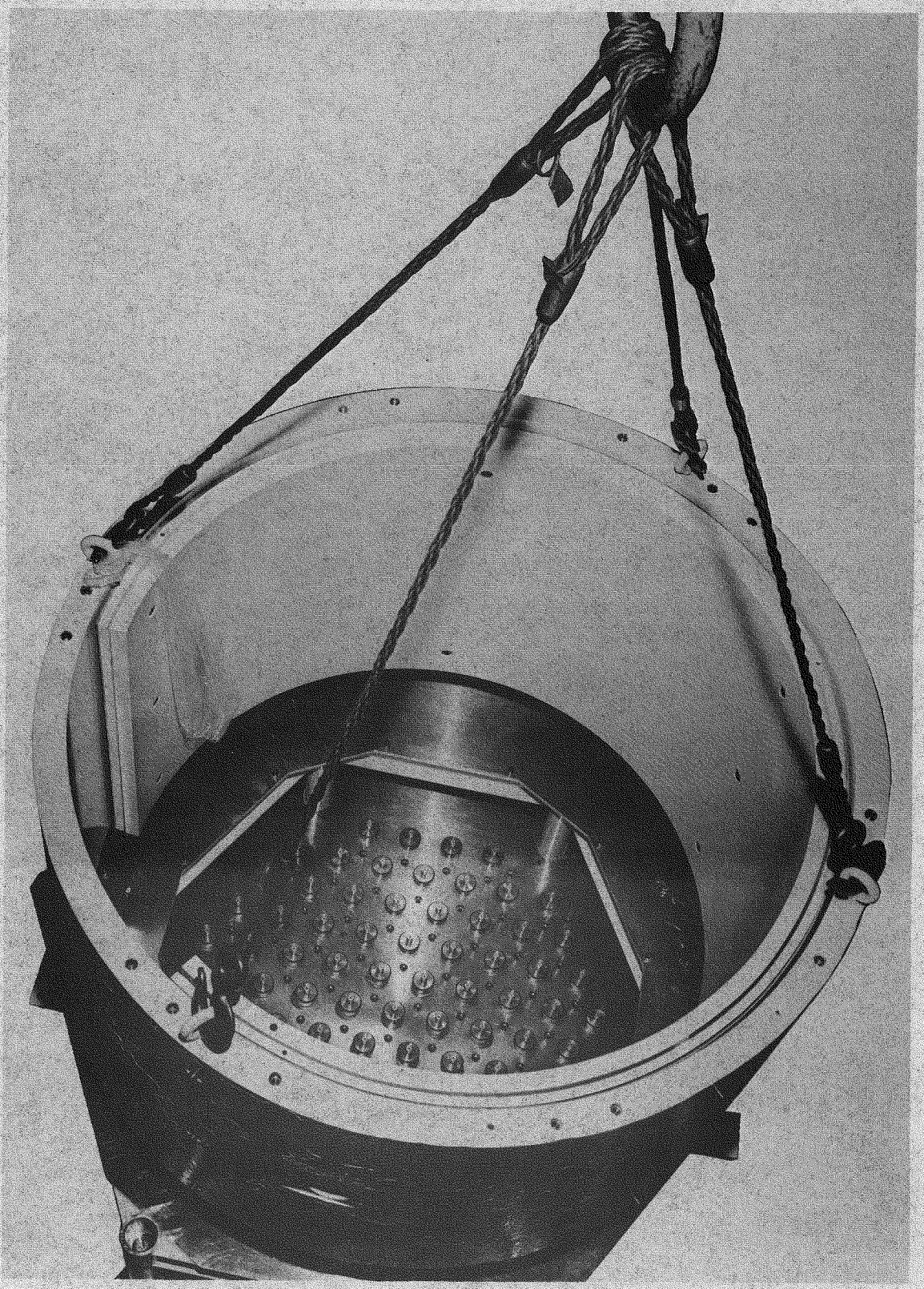

Fig. 11.8 – Completed assembly of outer ring and two 300 degree segments of inner rings (P62-9-2J)

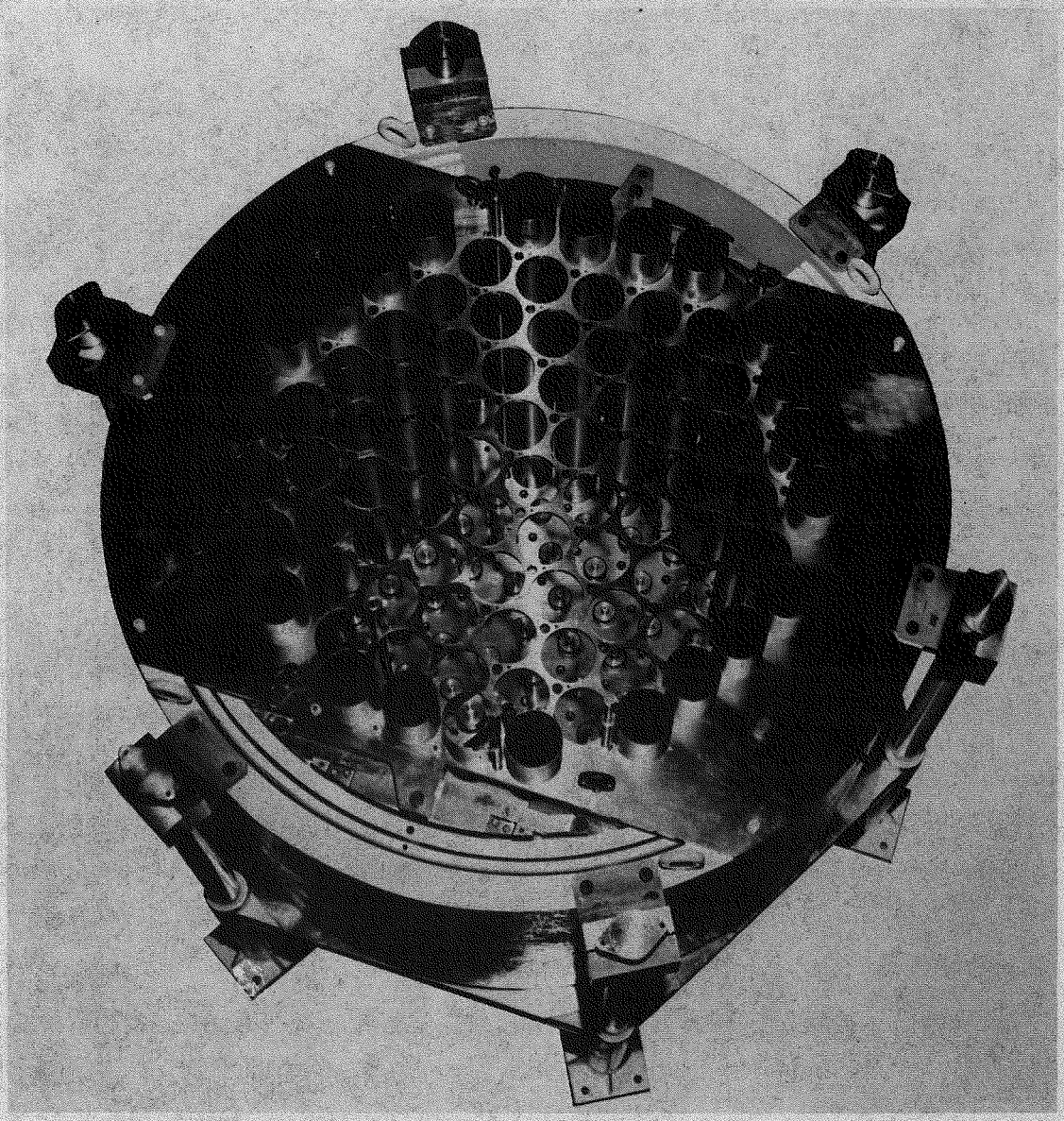

Fig. 11.9 – A plan view of the critical experiment mockup partially assembled (P62-9-2M)

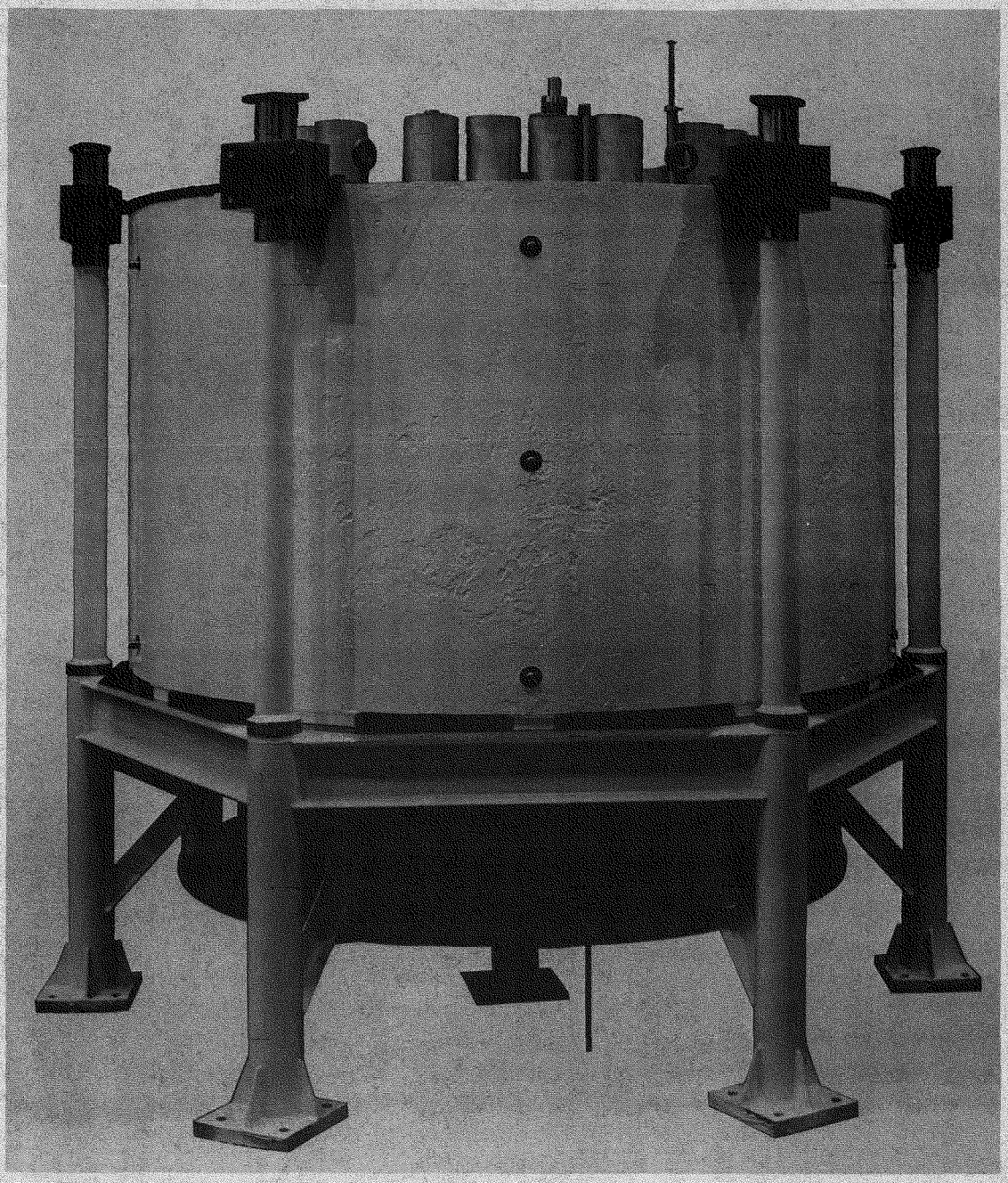

Fig. 11.10 – A partial assembly of the critical experiment mockup (P62-9-2P)

12. CRITICAL EXPERIMENT TEST EQUIPMENT

(270)

The design of the test equipment and the drawings for this equipment were virtually completed during the report period. Although fabrication in the shops was slowed down due to the temporary funding limitation, the test equipment for the critical experiment mockup will be completed when the mockup is delivered.

A summary of the status of the test equipment is given in Table 12.1 and amplified in succeeding sections.

12.1 FABRICATION OF CE ASSEMBLY

12.1.1 TANK SUBASSEMBLY

The tank subassembly was installed on the dolly as shown in Figure 12.1. Painting and adding a temperature and a water level probe will complete work on this subassembly.

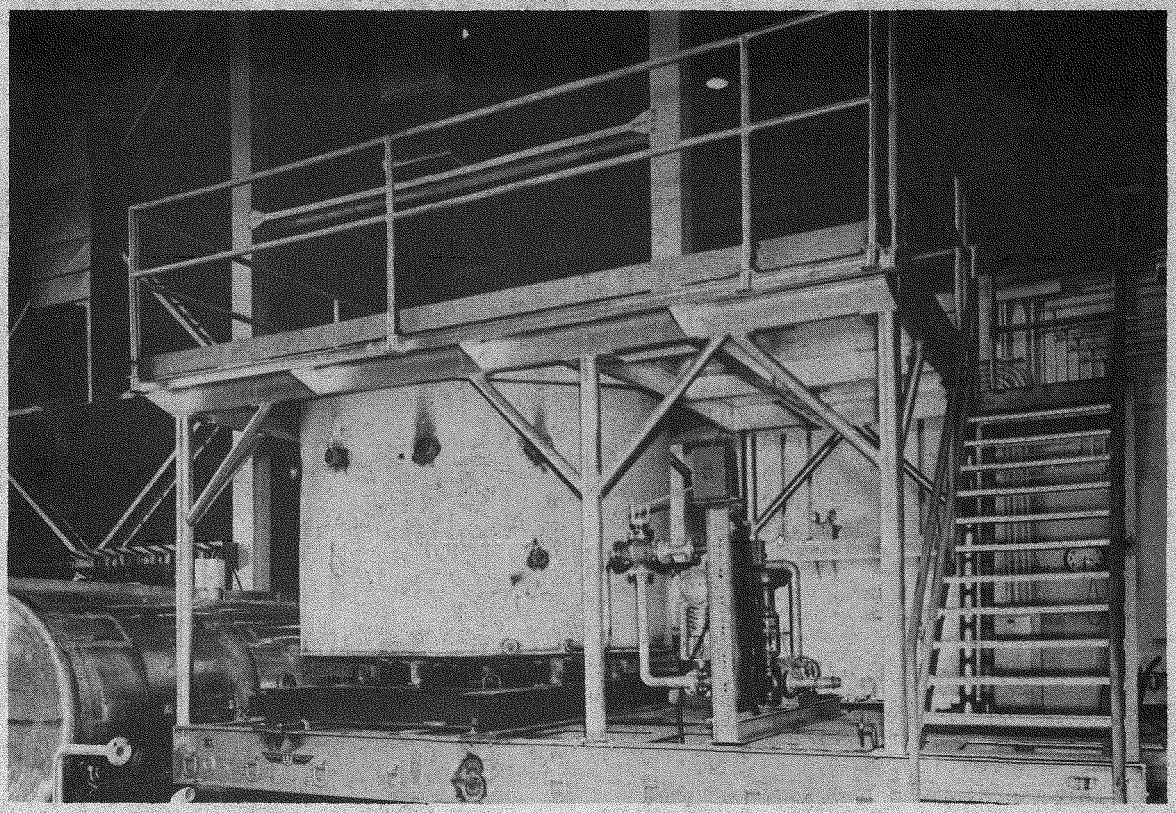

Fig. 12.1 – Critical experiment test assembly (U-4287-1)

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TABLE 12.1

630A CRITICAL EXPERIMENT ASSEMBLY STATUS

Assembly	Drawing No.	Drawing Title	Status In Drafting	Status In Shops
630A CE Assembly	692E308	Test Stand - General Assembly	Adding wire trays	Materials being accumulated
	692E317	Cable Trays - Subassembly and Details	Issued	
Working Platform	692E310	Platform Modification	Issued	Complete
	692E311	Handrail and Stairway Subassembly	Issued	Complete
	692E312	Working Platform Subassembly	Issued	Complete
	125B7240	Deck Plating - Removable	Issued	Complete
Tank Subassembly	692E313	Tank Subassembly	Issued	85% Complete
	125B7253	Platecoil Hanger Detail	Issued	Not Started
	109C243	Float Switch Bracket	Issued	Complete
	109C230	Support Frame Assembly	Issued	Complete
	125B7261	Flange, Overflow	Issued	Complete
	125B7262	Support Dump Valve	Issued	Not started
Actuator Support Stand	692E309	Actuator Support Stand	Issued	Complete
	146A6213	Actuator Stand Clamp	Issued	Complete
	848D326	Actuator Support Plate Details	Issued	Complete
	146A6219	Actuator Hold-Down Lugs	Issued	Complete
Actuators	848D325	Actuator Modification Assembly	Issued	Parts for modifying 21 drive heads complete. 15 drive heads completely modified. Installation of spacer button on flange not started.
	125B7249	Actuator Modification - Miscellaneous Details	Issued	Complete
	125B7251	Control Rod Connector	Issued	Complete
	125B7252	Actuator Extension Rod	Issued	Complete
	109C236	Flange, Magnet Plate	Issued	Complete
	109C237	Flange, Bottom	Issued	Complete
	109C240	Actuator Drive Motor Base Modification	Issued	Component parts for 20 actuators complete, 25 required
	848D336	Quick Disconnect - Control Rod	Issued	Prototype completed, 40 required
	146A6218	Spring, Quick Disconnect	Issued	Not started - 40 required
	125B7254	Bail-Lifting, Actuator	Issued	Complete
	109C256	Source Holder Subassembly and Details	Issued	Not started
Fill and Drain System	848D319	Piping - Fill and Drain System	Issued	90% Complete
	109C244	Piping Support	Issued	Complete
	109C254	Support, Motor Control Panel	Issued	Complete
	109C239	Specification Control Drawing - Heating Coil	Issued	Complete
	848D337	Support - Tank Demineralized Water Storage	Issued	Complete
	848D338	Piping, Fill and Drain - Demineralized	Issued	30% Complete
	692E319	Schematic Diagram Fill and Drain System	Check Print	Not started
	109C262	Control Air Piping - CE Tank	Issued	Not started
	692E318	Storage Tank, Miscellaneous Details	Issued	30% Complete
		Remote Level Indicator	Not started	Parts on hand
Facility Modification			Prints to be marked	Not needed until after Test Program starts.

12.1.2 ACTUATORS

The test actuator with a modified drive-head was scrammed from the fully withdrawn position 505 times during an endurance test. The actuator performed very well with no maintenance required during the last 286 scrams. This indicates that there should be very little maintenance to perform on the actuators during the performance of the critical experiment.

A prototype control rod quick disconnect shown in Figure 12.2 was manufactured and evaluated. Some redesign was performed to facilitate manufacturing.

12.1.3 ACTUATOR SUPPORT STAND

Fabrication of this item, shown in Figure 12.3, was completed during this report period. The completed top actuator support plate is shown in Figure 12.4.

12.1.4 FILL AND DRAIN SYSTEM

The fill and drain system was modified slightly to permit the pumping of water from a tank truck to the storage tank for refilling the system.

A revised schematic diagram, the dolly-mounted portion of the fill and drain system, and the storage tank are shown in Figures 12.5, 12.6 and 12.7, respectively. The storage tank, which was used in the vertical position for a previous experiment, was modified to lie on its side so the critical experiment tank could be drained by gravity with the quick dump valve. The piping for pump No. 2, which pumps from the storage tank, is being fabricated.

The control panel for the fill and drain system is being completely rewired to give safer and easier operation. Existing panels and components will be used.

12.2 PREPARATION OF FACILITY TO RECEIVE DOLLY

No major modifications to the Low Power Test Facility have been necessary. Some modifications to the control room wiring were necessary in order to accommodate the control rod actuators and the fill-drain and heating system on the CE tank and dolly. The steam supply to the test cell for the CE tank water heater has not been provided. The LPT counting system has been improved by increasing the count rate capacity of the counters. A semi-automatic plotter has been added to the counting system which will plot the counts as they are printed out from the foil counters.

The Low Power Test Facility will be shared with another contractor who will operate a reactor in the CE cell. This contractor will occupy the CE cell, CE control room, instrument repair room, instrument storage room, operations office and operations supervisor office. General Electric will occupy and be responsible for the remaining portion of the building.

The 630A critical experiment reactor will be placed in the southwest corner of the IC cell.

The fuel cartridges will be assembled in the subassembly room. Two hoods have been added to the room in which the loading and fuel handling will be accomplished. The air is drawn through the hoods with a single 2200-cfm blower. The air is filtered before it is exhausted to the atmosphere.

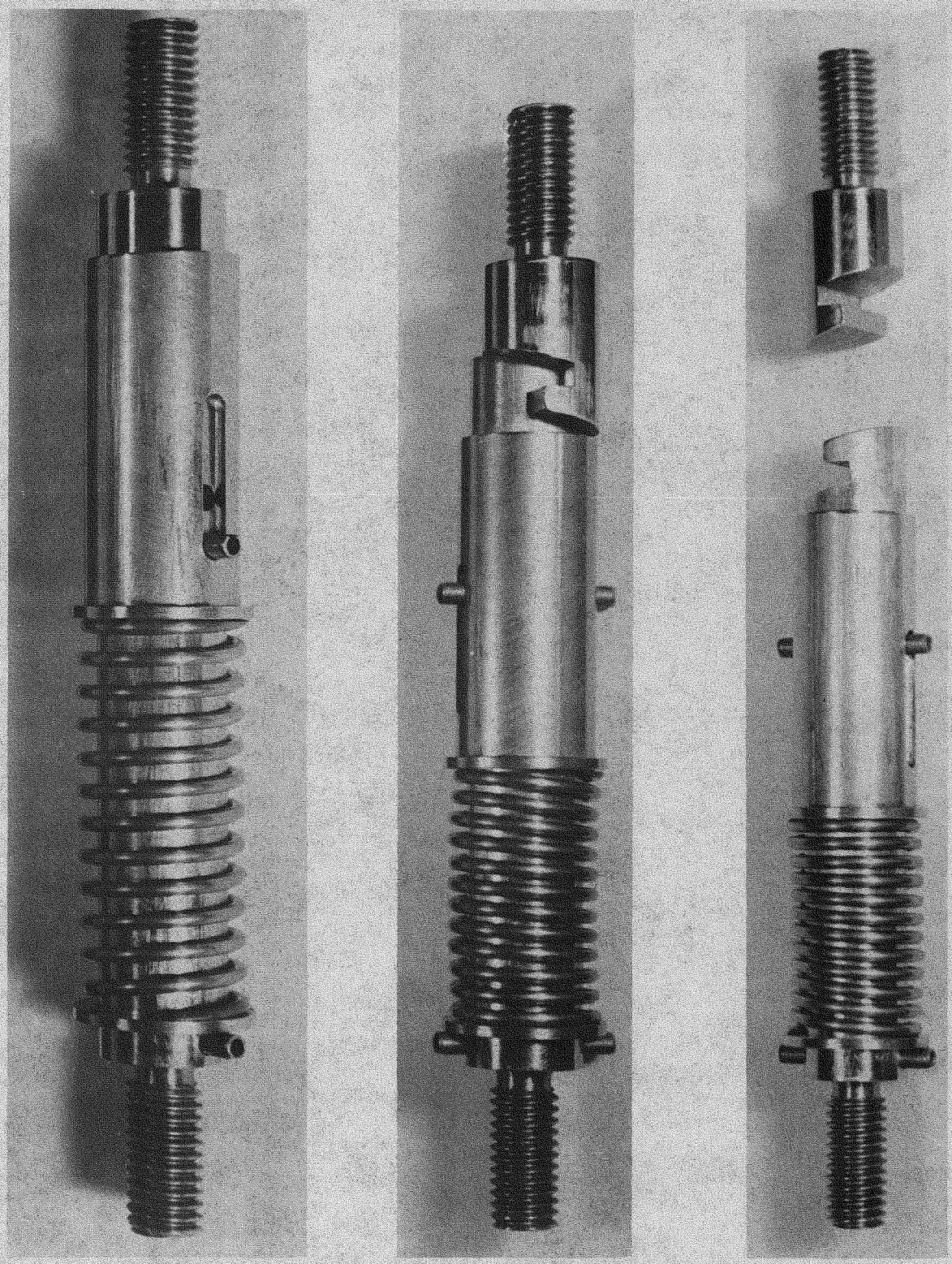

Fig. 12.2 – Control rod quick-disconnect (U-4287-6)

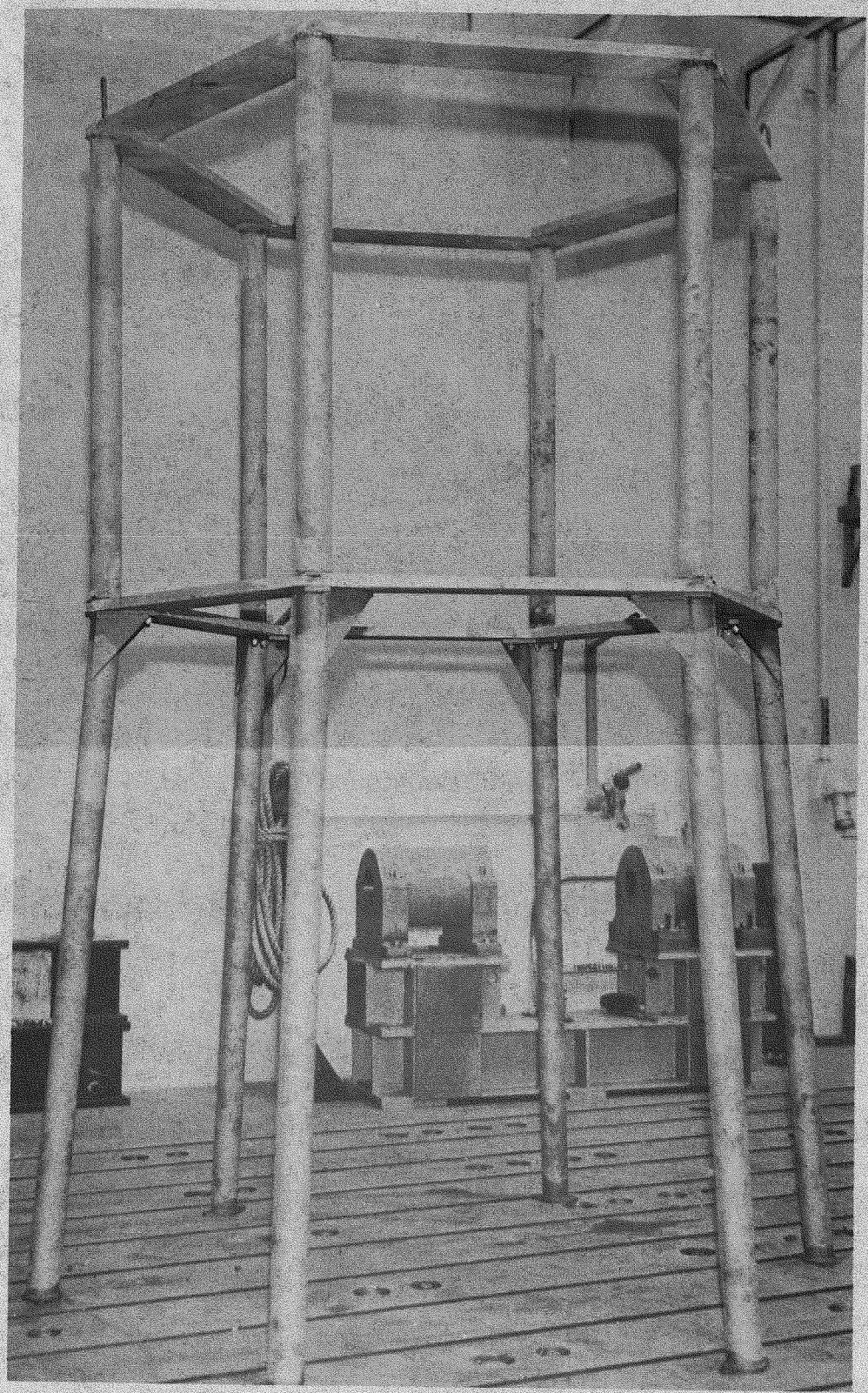

Fig. 12.3 – Actuator support stand (U-4287-5)

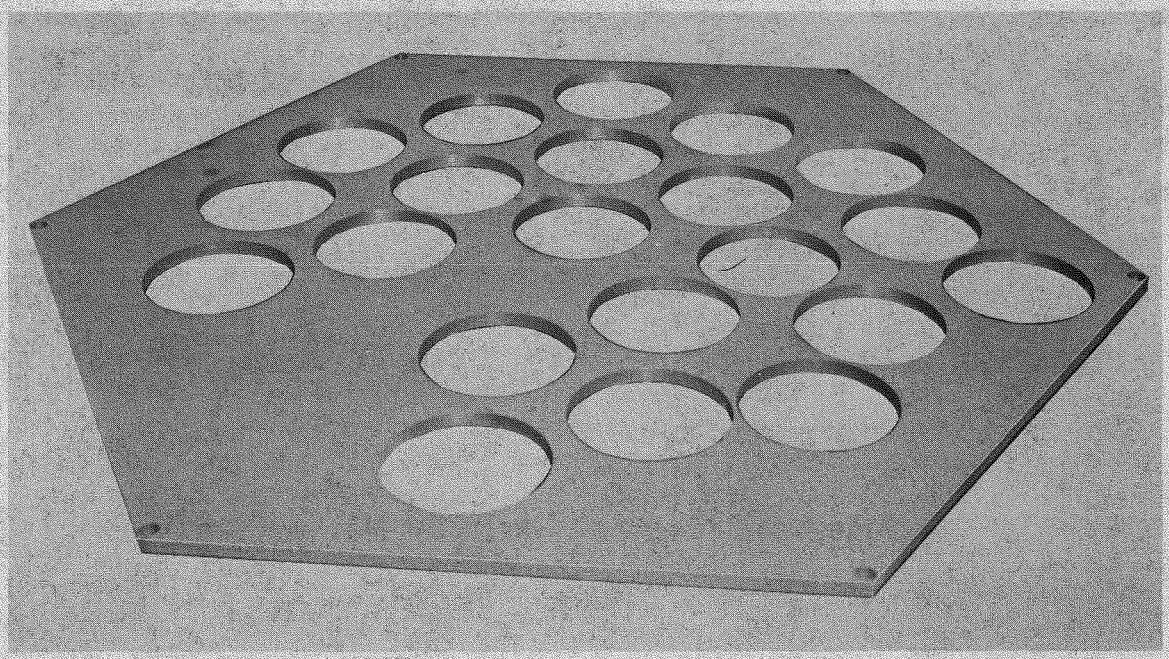

Fig. 12.4 – Top actuator support plate (U-4287-4)

12.2.1 NUCLEAR INSTRUMENTATION

No modifications will be made to the CE nuclear instrumentation described in the previous progress report.*

12.2.2 CONTROL AND SAFETY CIRCUITRY

Modifications and additions to the control and safety circuitry have been completed with the exceptions of installation of the control rod latching switches and addition of a moderator water heater switch to the control console. Primary effort was directed toward alteration of the mode of operation of the actuator position indicator lights, repair or replacement of wire connectors, and modification of actuator motor drive and latching circuitry. Control rods were previously latched with 28 volts direct current power and held with 14 volts direct current. The present system will latch with 60 volts direct current power and hold with 30 volts direct current.

All corrections and additions to the prints of the control room circuitry have been completed. Final prints are currently being drafted.

*Ibid., p. 108.

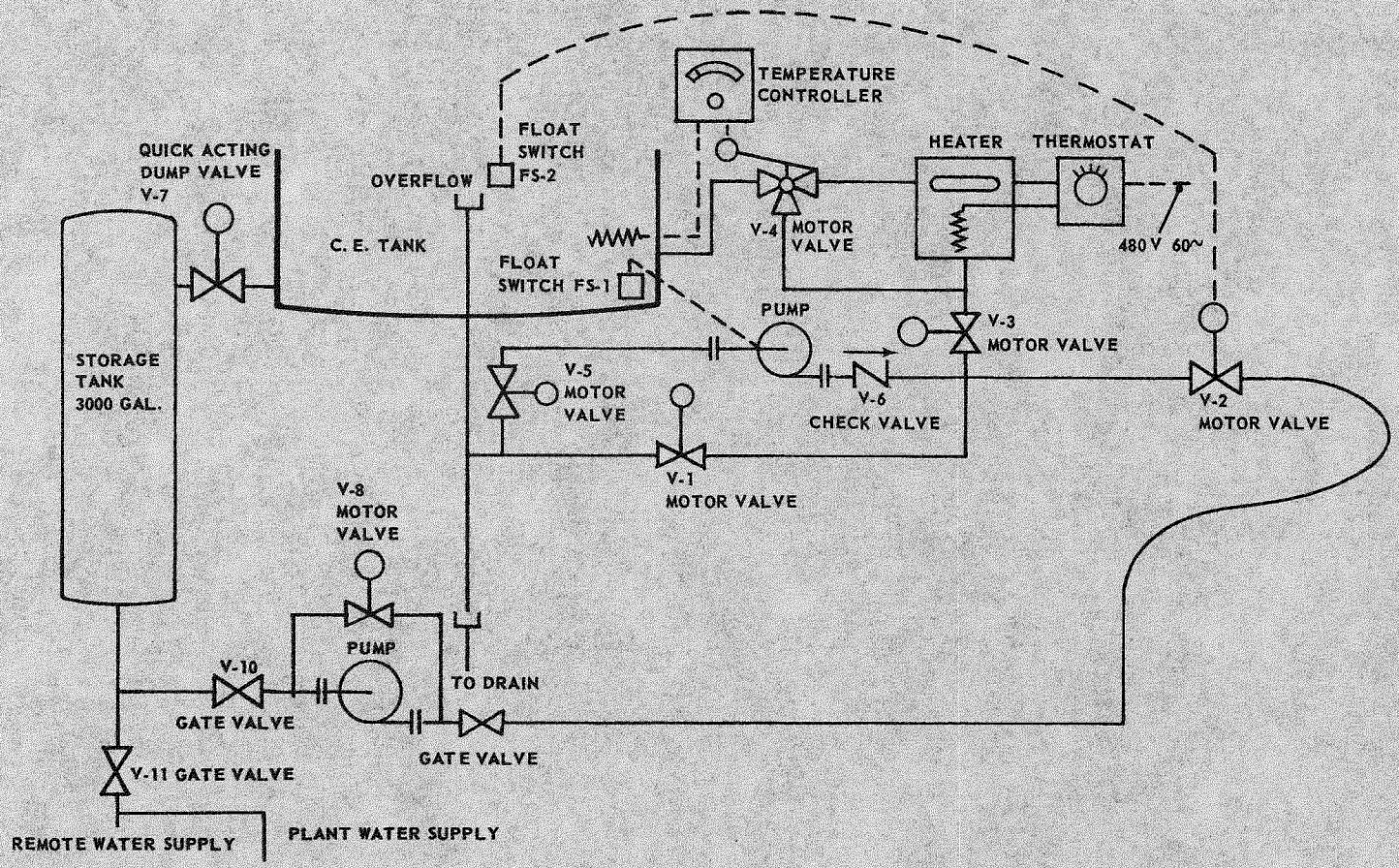

Fig. 12.5 – Schematic of 630A CE hydraulic system

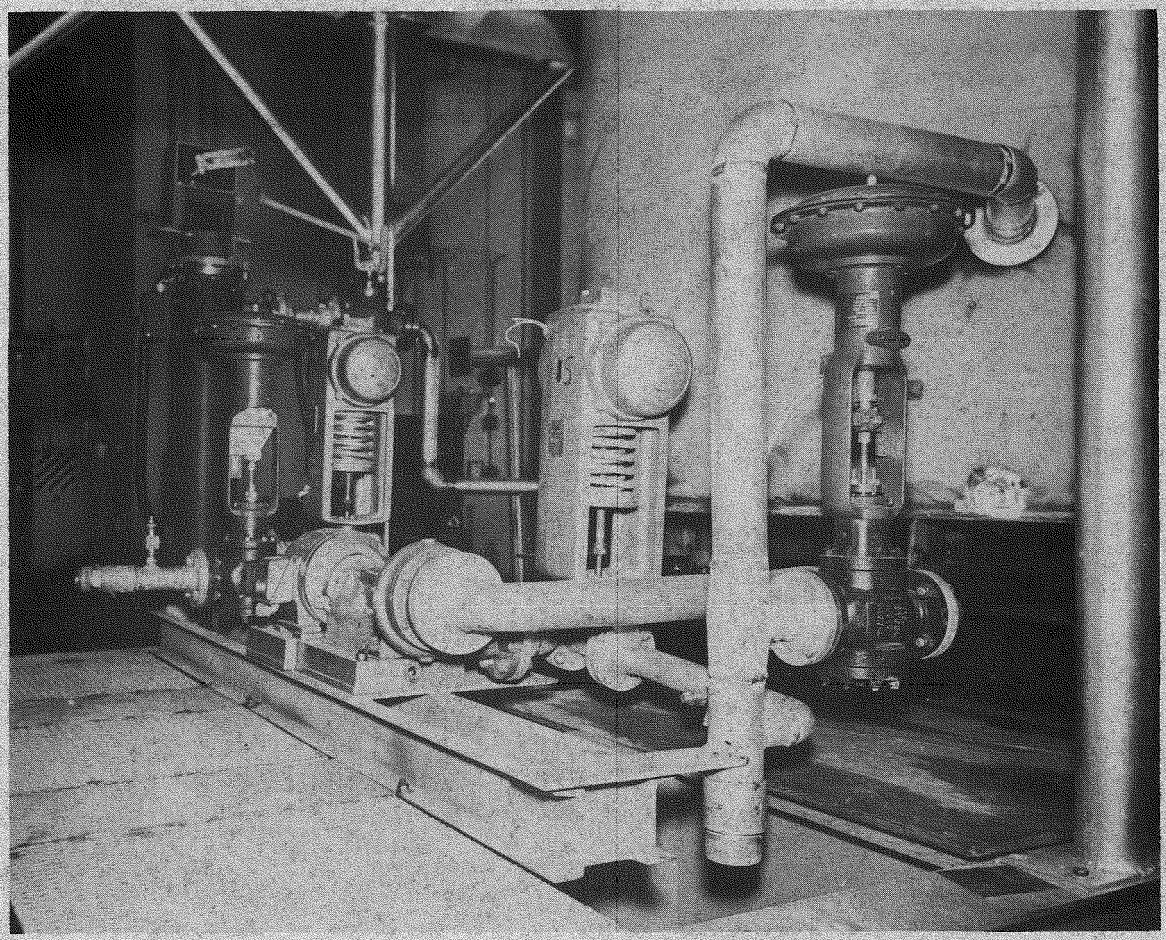

Fig. 12.6 – 630A CE fill and drain system (U-4287-2)

Fifteen actuators have been assembled and tested. Parts are available for the modifications to ten additional actuators. Four of the ten actuators require decontamination before they can be reassembled. It is expected that these ten actuators will be assembled and tested during the early part of the next report period. The total of twenty-five shim-scram actuators will provide five spares.

12.2.3 COUNTING ROOM

The rewiring of the counting system control chassis and component interconnections has been completed. The modifications have been made to the electronic counters to permit the counting of foil activities up to 10^6 cpm with less than 0.1 percent deadtime correction. The scalers were also modified to accept positive polarity pulses. The old system counted negative pulses.

The beta scintillation heads and the transistorized preamplifiers have been received from the vendor. These components have been checked out with the complete counting system with very satisfactory results.

A test was made to determine the operating plateau (count rate versus photomultiplier high voltage) of each channel. It was found that an individual voltage adjustment for each detector would reduce the differences in counting efficiencies of the channels. This additional feature is approximately 80 percent completed.

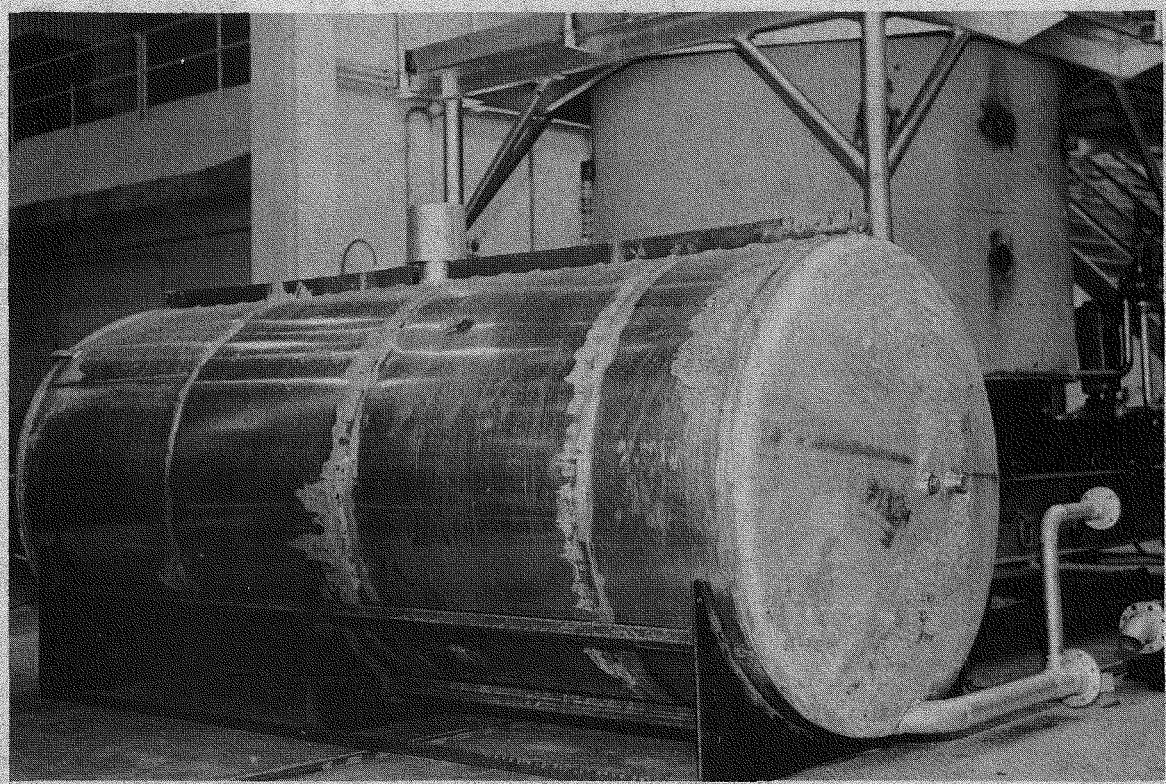

Fig. 12.7 – CE water moderator storage tank (U-4287-3)

It was also found that the focusing potentiometer in the detector head did not provide a wide enough range of adjustment to compensate for differences in counter geometry. The detector sleeves are being threaded so that a collar nut can be used to vary the vertical position of the detector. This work is approximately 90 percent complete.

A semi-automatic plotting system has been incorporated into the recording system. The plotting system includes an Electronic Associates Vari-plotter plotting table, a modified Electro Instruments decimal-to-binary converter, and the necessary resistor and relay logic to provide the plotting coordinates. This added feature will generate plots of relative circumferential power distributions in parallel with the existing recording devices (electric typewriter and paper-tape punch). The plotting system has been checked out with very satisfactory results. A final checkout will be made as soon as the new counting heads are installed.

12.3 FINAL PREPARATIONS FOR OPERATION

The initial plan was to deliver the CE tank and dolly assembly to LPT for the checkout of the actuators, water fill and drain system and water heating system before the reactor was assembled and placed in the CE tank. This schedule has now been changed. The CE tank, dolly and reactor will be delivered to LPT at the same time late in September. However, the actuator stand will be taken to LPT early in September for a complete checkout with the new facility wiring.

12.4 CRITICAL EXPERIMENT FUEL ELEMENTS

Permission to handle the fuel for the mockup 630A fuel cartridges was received August 6, 1962. Since that date, 14 birdcages of fuel have been sorted. This represents approximately 98.5 kilograms of U^{235}. Of this amount approximately 10 percent of the fuel has been discarded as scrap and about 25 percent has been or will be cleaned and recoated. The remainder was considered usable in its present form. The sorting, cleaning and coating of the fuel is being done in the actuator building.

Numerous problems were encountered in finding a satisfactory procedure for cleaning the fuel without a significant loss in fuel weight. The process which was finally accepted is dipping the fuel sheets in boiling water for a few seconds, drying them with a soft cloth and then washing them in acetone. The sheets are allowed to dry and then coated with a thin layer of fluorocarbon. The coating material is the same as has previously been used to coat bare sheet fuel.

There is on an average a 4 percent difference between the accountable weight of the fuel which has been cleaned and the measured weight after cleaning, the cleaned fuel being lighter. Part of this loss is due to the cleaning process.

The coating on some of the fuel was in very poor condition and would flake off with little or no handling. Figure 12.8 shows some of the fuel as it was removed from the plastic bags in which it was wrapped.

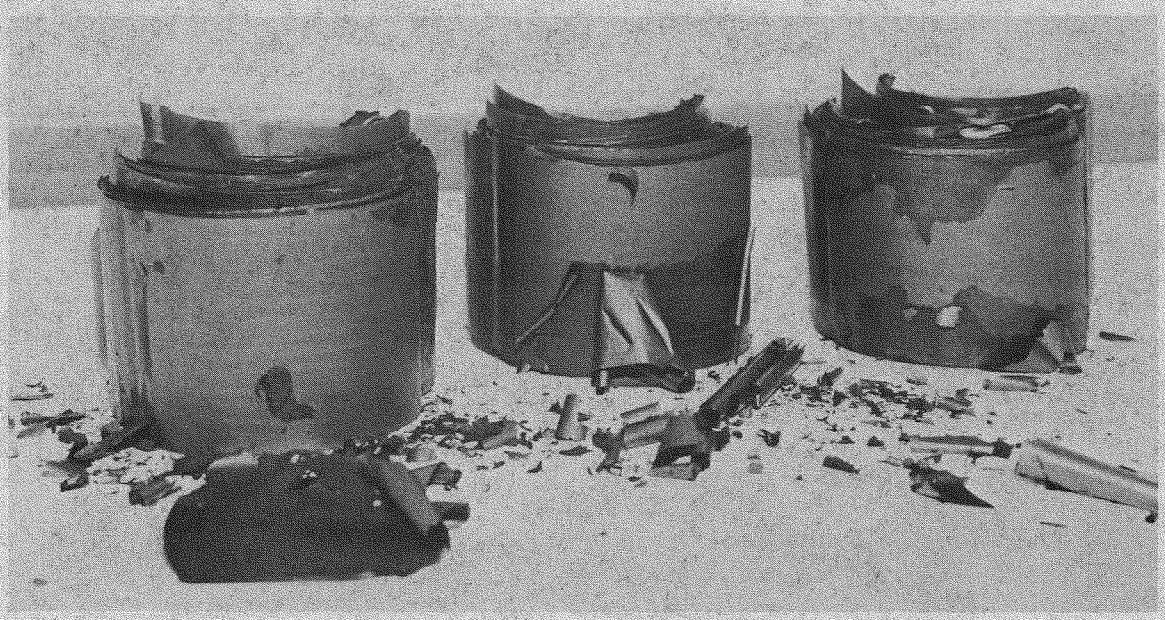

Fig. 12.8 – Illustration of deterioration of fuel coating (U-4280-4)

Loading of the mockup fuel cartridges was started August 29, 1962, in the LPT subassembly room. One difficulty which has been encountered in loading the fuel cartridges is in being able to determine the exact weight of U^{235} loaded on each fuel cartridge. This is a very important factor in obtaining the correct reactor fuel loading. The weight of the fuel which is cleaned and coated in Idaho is known exactly. However, the weight of the fuel which is not cleaned is questionable. The fuel is carefully weighed in small batches but the exact weight of the coating is not known. An attempt is being made to determine an average coating weight.

12.5 OPERATOR TRAINING

Operator training for the 630A critical experiment has been started. Operators have been familiarized with the test equipment drawings, facility wiring and the test program and procedures.

12.6 WORK PLANNED FOR NEXT PERIOD

All of the items in Table 12.1 will be completed. Preparation of the facility to receive the dolly and reactor will be completed. The actuator support stand containing the actuator assemblies will be checked out at the facility. The CE tank, dolly and reactor assembly will be delivered to LPT for checkout. The mockup fuel cartridges will be loaded. Operator training will be completed and the critical experiment started.

13. SAFETY ANALYSIS

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The rough draft of the "Hazards Summary Report for Operation of the 630A Maritime Reactor Critical Experiment at the Low Power Test Facility" was completed and copies were forwarded to AEC-ORO on July 9, 1962. Following review by various AEC components and compilation of the comments, the report was discussed at a meeting of ORO and GE-NMPO representatives on August 23, 1962. The report has been revised to incorporate necessary changes and to update the sections covering the test program, system description and administrative controls. Issuance of the final draft was scheduled during the month of September.

The criticality safety rules and procedures for storing, cleaning and recoating the uranium foils and for assembly of the mockup fuel cartridges were completed and approved by AEC-IDO during the report period. The work covered by these procedures is now under way.

13.1 CRITICAL EXPERIMENT HAZARDS (281)

13.1.1 POSTULATED ACCIDENTS

The consequences of an additional postulated accident-initiating sequence have been investigated and are included in the final issue of the hazards report. This accident is assumed to occur as a result of the unrestrained withdrawal of a single actuator frame (four actuators or twelve poison tips). Assuming that at delayed-critical the frame is fully inserted and that withdrawal proceeds at the normal shim velocity of 0.597 centimeters per second, the peak power of 2.56×10^9 watts is reached in about 46 seconds. It is assumed, of course, that the automatic scram system fails and that the operator fails to take appropriate safety action. As in the postulated maximum credible accident (MCA), the excursion is terminated by meltdown of the central one-third of the core. Such an accident would result in the release of 1.37×10^9 watt-seconds fission energy and the formation of 8.49×10^{19} fission fragment. Assuming the same fractional release as in the MCA, the hazards of such an excursion would be about 77 percent of those estimated for the MCA.

13.1.2 RADIOLOGICAL HAZARDS

The effluent doses outside of the facility resulting from the MCA have been re-evaluated because the assumption of uniform emission into a 22-1/2-degree sector, for which the prior evaluations were made, is not consistent with the remainder of the assumed release conditions. The calculated cloud gamma doses have also been further corrected for radioactive decay. Corrected curves for the effluent doses to the thyroid, lung and whole body, as a function of distance downwind, are shown in Figures 13.1 and 13.2 for the same typical lapse and inversion conditions presented in the previous progress report.*

*Ibid., p. 104.

147 75

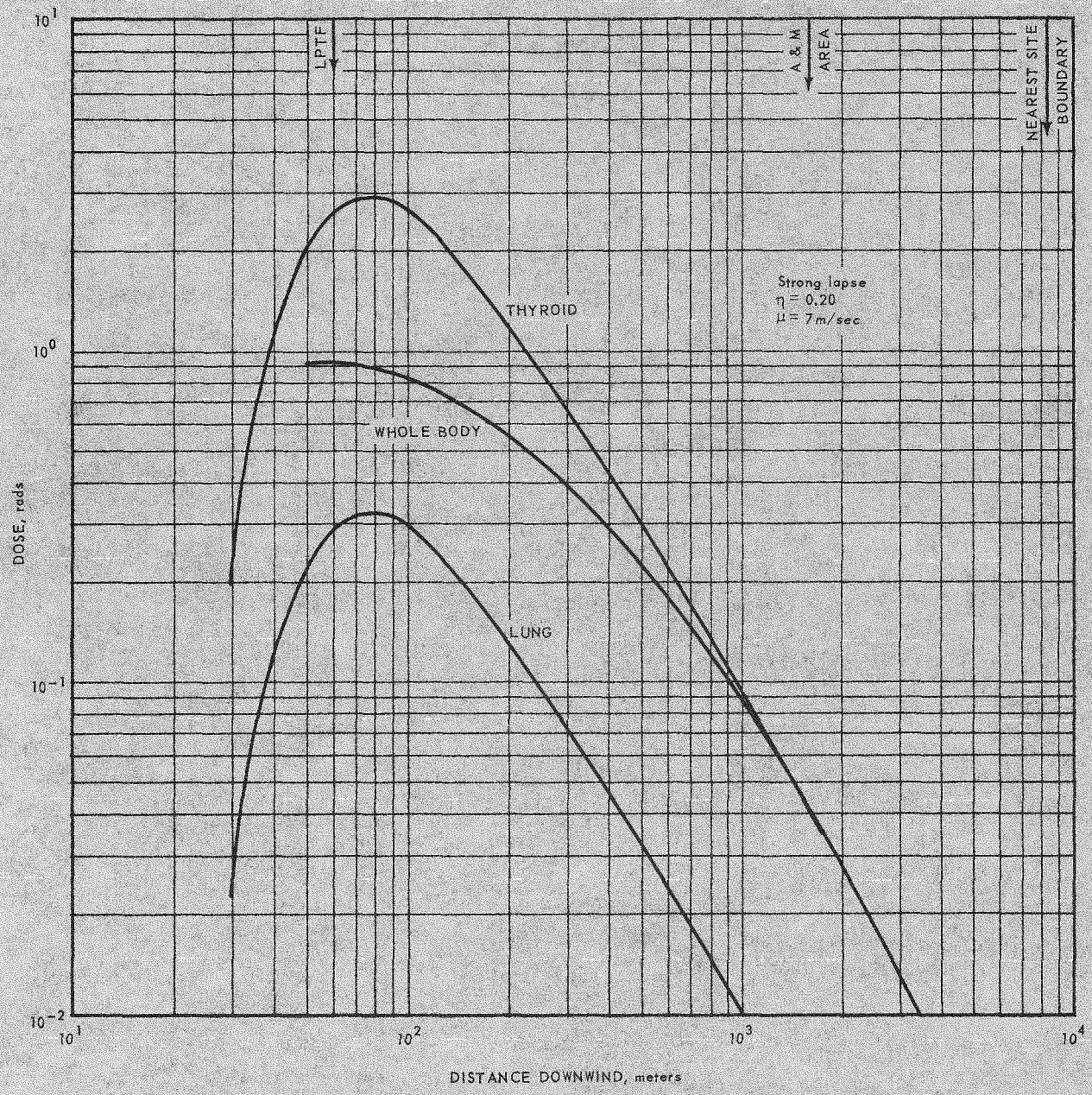

Fig. 13.1 – Effluent dose from the maximum credible accident – 630A critical experiment – typical lapse (daytime)

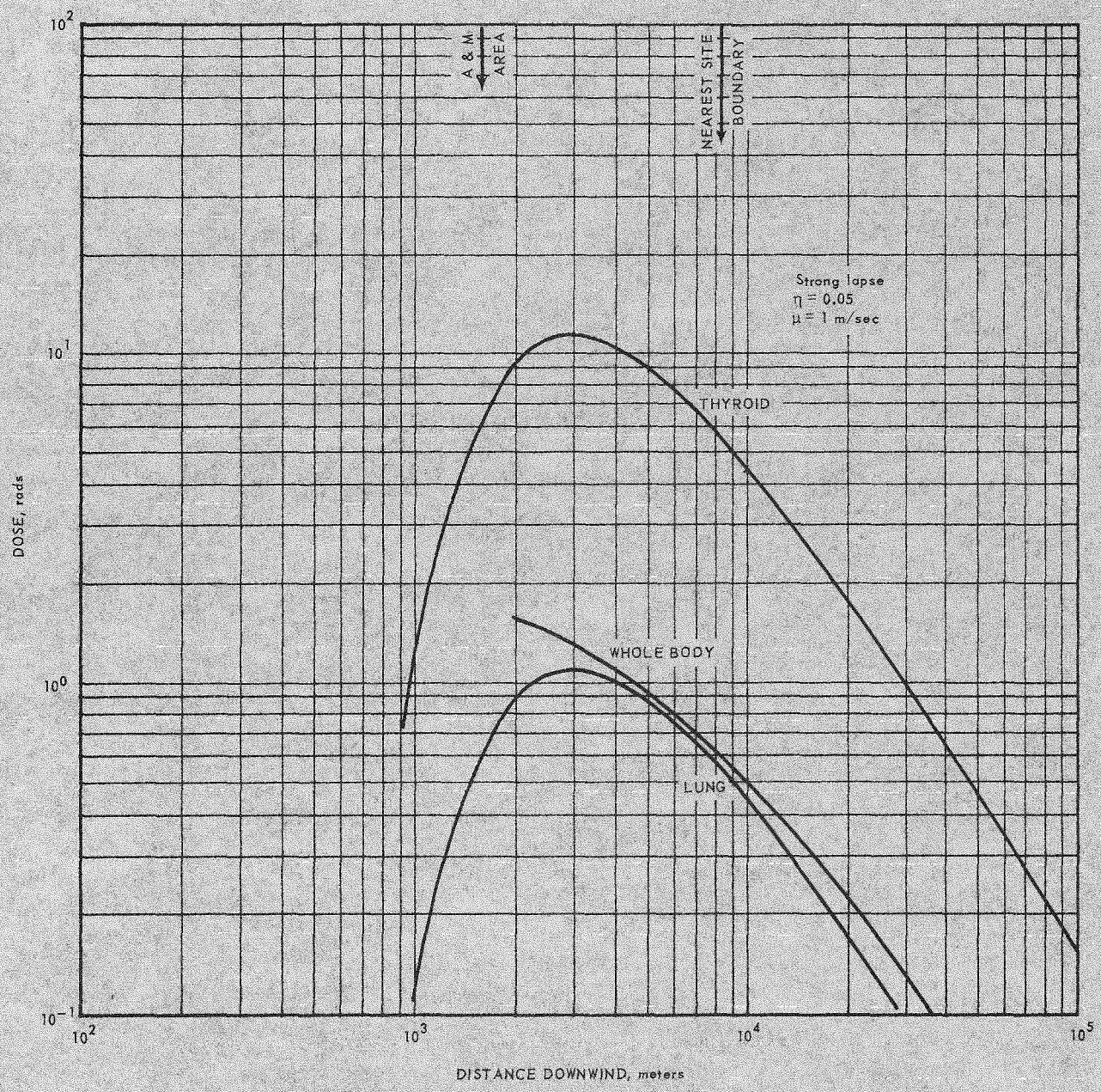

Fig. 13.2 – Effluent dose from the maximum credible accident – 630A critical experiment – typical inversion (nighttime)

14. CRITICAL EXPERIMENT TEST OPERATIONS

(290)

During this report period, a rough draft of the proposed detailed test program and procedures for the critical experiment were written. The test program outlined differs slightly from that given in the previous report.* However, the tests to be conducted remain essentially the same with some additions. The additions include the following measurements.

1. Change in K_{ex} as a function of water moderator level.
2. Radiation dose levels within the Low Power Test Facility resulting from the 630A mockup reactor, for radiological hazards evaluation.
3. Reactivity worth of typical fuel cartridges.
4. Power distribution and K_{ex} changes caused by changes in the amount of center and outer moderator associated with typical fuel cartridges.
5. Power distribution and K_{ex} changes due to the addition of a 2-inch-thick, 60-degree sector of top tube sheet.
6. Power distribution and K_{ex} changes caused by varying the separation distance between the reflector and the reactor.

A rough draft of the Standard Operating and Maintenance Procedures for the 630A Critical Experiment Reactor was written during this report period. It will soon be issued for review.

The writing of the Low Power Test Facility Operating Manual was started during this report period. It is approximately 30 percent complete.

14.1 WORK PLANNED FOR NEXT PERIOD

The test program, operating procedures and facility operating manual documents will be completed and issued with appropriate approvals.

*Ibid., p. 78.

15. FACILITIES AND EQUIPMENT

(330)

During this report period work was confined to selection of the site and preparation of the design criteria for the power test facility. A summary of this work follows.

15.1 POWER TEST FACILITY (332)

The discussion of the power test facility, presented in the previous progress report,* assumed that the facility would be located in the LPTF-STPF complex. Subsequently, the IDO office of the AEC reviewed the proposed location and indicated that the 630A power test facility should be located a minimum of 4,000 feet southeast of the LPTF-STPF complex because of a high calculated Experimental Beryllium Oxide Reactor (EBOR) accident dose at STPF.

Since the hazards associated with the operation of EBOR at STPF curtail the operation of LPT to the same degree as they do the 630A, it might be better to operate EBOR at a different site and place the 630A in the STPF. The 630A Nuclear Steam Generator assembly will have a built-in biological shield and triple containment so its operation will not constitute a hazard to the nearby LPTF.

A preliminary study was conducted to determine the feasibility of performing the power testing of the 630A at the STPF. The pools themselves would not require modification. An additional structure would be required to house the heat dump equipment, emergency diesel generator and other minor equipment. It was concluded that the STPF would be a satisfactory site for the 630A. The proposed 630A installation would not affect the flexibility of the existing facility. Considerable savings could be achieved by utilizing the STPF rather than building a new facility located 4,000 feet southeast of the LPT-STPF complex.

The proposed installation of the 630A nuclear steam generator is in the north pool (Pool No. 1) leaving the south pool (reactor pool) available for other programs. Figures 15.1 and 15.2 show the 630A nuclear steam generator installed in the north pool. The total-containment version was used in this study because of its slightly larger size over the individual-containment version. The outlines shown for the blower drive also include the area required for assembly and removal of the blowers. The elevation of the power plant relative to the floor of the building is flexible. A minimum of forty feet crane-hook clearance is required above the mounting deck for removal of the core and plug assembly. Fourteen feet of vertical clearance below the mounting deck is required for assembly and removal of the boiler. A steel support structure, not shown on the drawing, will distribute the power plant load over the bottom of the pool.

The assembly area adjacent to the north pool would be used for major component assembly and receiving or shipping equipment via motor truck. The existing 50/5 ton bridge crane

*Ibid., p. 105.

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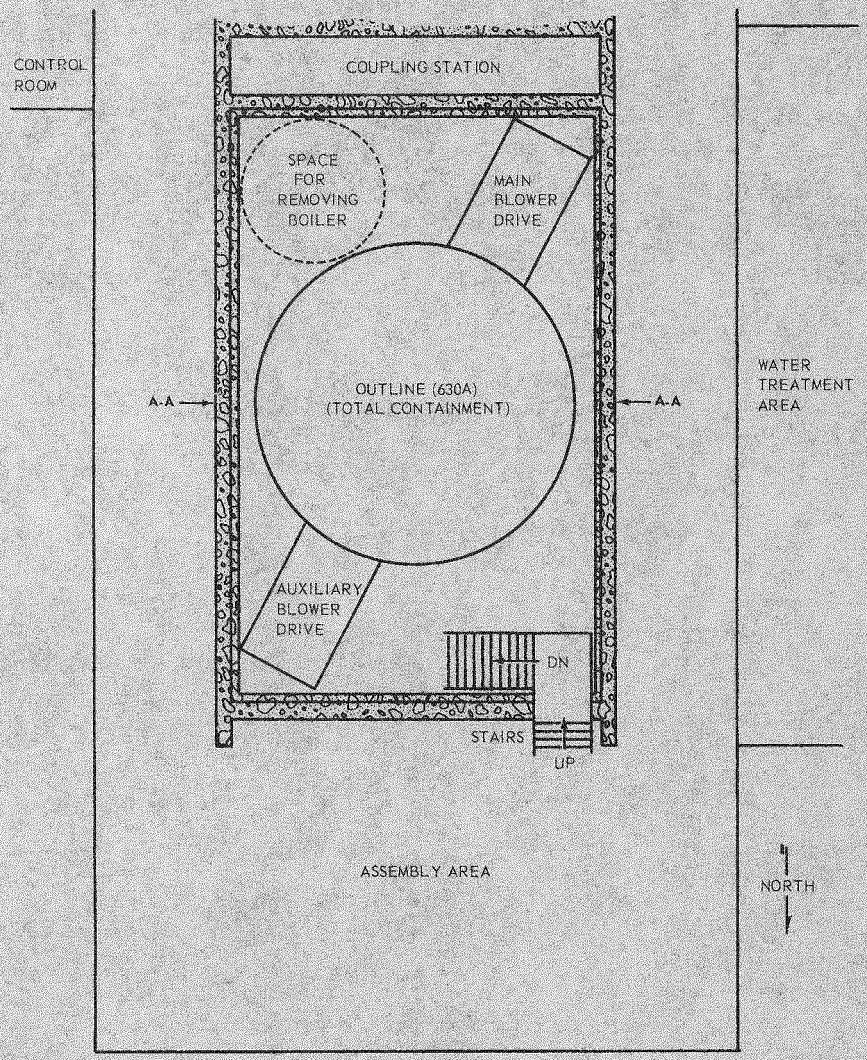

Fig. 15.1 – 630A reactor in pool facility

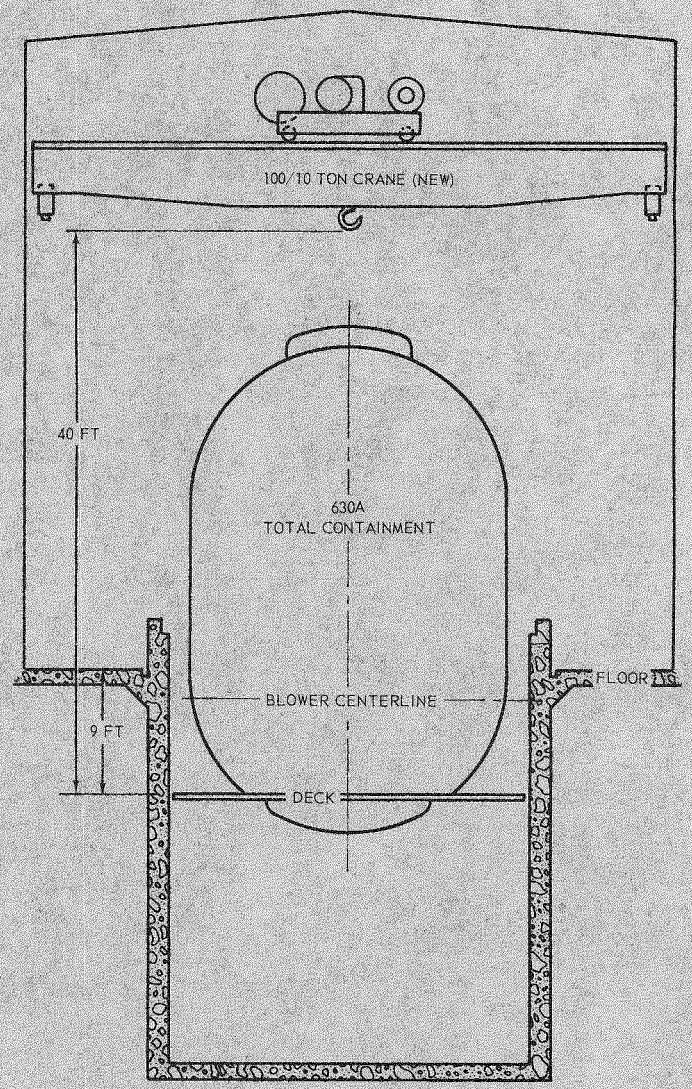

Fig. 15.2 – Section A-A 630A reactor in pool facility

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would be replaced with a 100/10 ton unit. Preliminary calculations indicate that the existing crane rail system could be used. The existing control and equipment building would be used with only minor additions (control console and racks) required. The existing facility services would be utilized and would be supplemented with a second well, and greater electrical capacity. To safely dissipate the heat generated by the 630A power plant, a heat exchanger would be installed in a new structure located adjacent to the northwest corner of the pool building shown in Figure 15.3. Also contained within this building would be the large emergency diesel generator and other supporting equipment required for the 630A test.

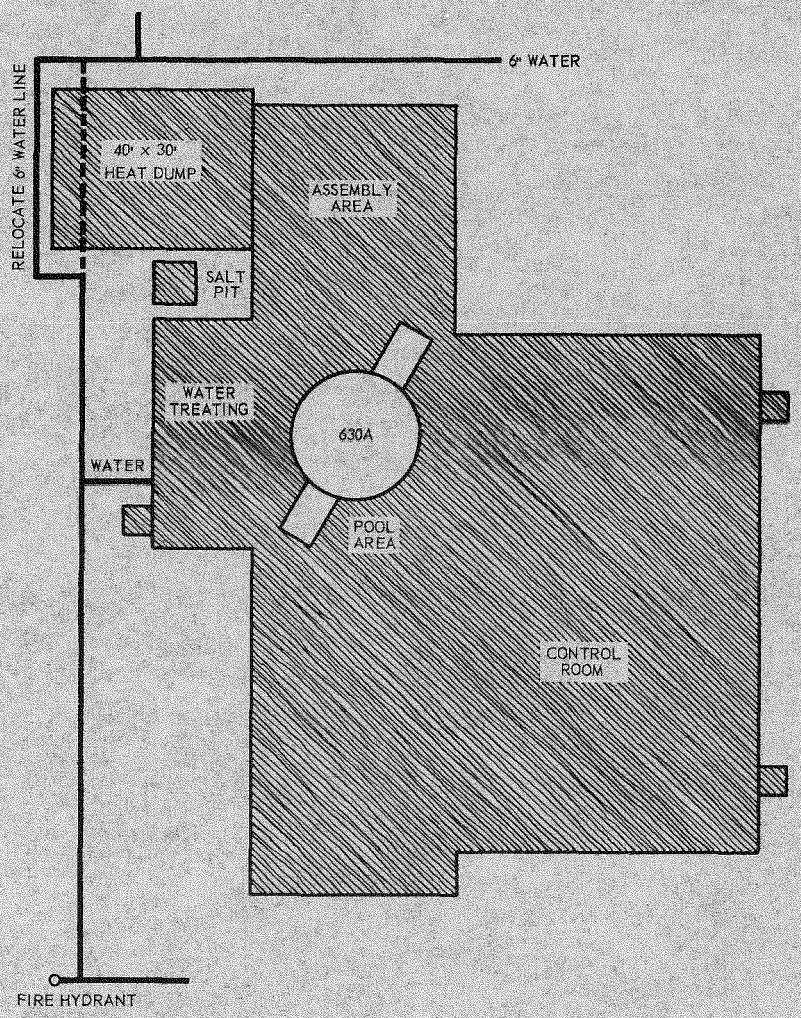

Fig. 15.3 – Heat dump and pool facility plot plan

An alternate installation would locate the 630A nuclear steam generator above the north pool. While this installation would provide easier access around the power plant and more room to remove the boiler, the cost of modifications to the pool building (raising the roof approximately 10 feet and providing new crane trackage) might be prohibitive.

15.2 WORK PLANNED FOR NEXT PERIOD

Investigation of the problems involved in conducting the 630A power tests at the STPF will be continued. The design criteria for the power test facility will be prepared and issued.