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GEMP - 130

Nuclear Materials & Propulsion Operation

630A MARITIME NUCLEAR STEAM GENERATOR PROGRESS REPORT NO. 1

FLIGHT PROPULSION LABORATORY DEPARTMENT
GENERAL ELECTRIC

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630A MARITIME NUCLEAR STEAM GENERATOR
PROGRESS REPORT NO. 1

JULY 31, 1962

United States Atomic Energy Commission Contract No. AT(40-1)-2847

NUCLEAR MATERIALS and PROPULSION OPERATION
FLIGHT PROPULSION LABORATORY DEPARTMENT

GENERAL ELECTRIC

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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.

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  - 101 Cycle Analysis
  - 102 Steam Generator Assembly
  - 103 Ship Installation
  - 104 Ship Service
- 110 Reactor
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INTRODUCTION AND SUMMARY

This progress report summarizes work accomplished to June 30, 1962, since submission in April 1962, of the 630A Maritime Nuclear Steam Generator Scoping Study (GEMP-108).

The ultimate objective of the current 630A program is to establish a specific 630A configuration and to develop specifications for components and test equipment.

During this report period, work was initiated in critical experiment design and fabrication; additional fuel and materials investigations; boiler-test design and fabrication; blower studies; design of component tests; nuclear, thermodynamic, mechanical and safety analysis; and test facility and equipment studies.

Highlights of the period include:

1. Design of the critical experiment mockup and test equipment was completed and fabrication of the parts are approximately 50 percent complete. Operation of the critical experiment is scheduled October 1962.
2. A rough draft of the critical experiment hazards report was completed. A preliminary review of the report with the AEC was directed toward obtaining operational approval.
3. A fuel test in the ORR completed 876.5 hours of testing out of a planned 2200-hour test without indication of failure. The burnup was equivalent to about 6000 hours of 630A operation. Damage to the capsule during refueling of the ORR caused termination of the test. A new sample will be inserted in August.
4. The design of an MTR fuel-burnup test was completed and fabrication of the sample initiated. The sample is scheduled for test in September 1962.
5. Ni-Cr fuel sheet and cladding stock are being tested for creep and oxidation properties at temperatures up to 1750°F and have accumulated times up to 5000 hours; no failures have occurred. These tests are continuing. Additional tests will be initiated on fuel sheet with different properties.
6. Specimens of Ni-Cr were fabricated and will be tested to determine the effect of neutron irradiation. The specimens are scheduled for insertion in the ETR August 1962.
7. Cycle operating conditions with 1200°F reactor-discharge-air temperature were studied and found to be acceptable for the proposed maritime application. Increases in cycle efficiency above the 30.2 percent appear to be possible and practical.
8. Studies during this report period indicate that an acceptable power distribution can be maintained through the life of the reactor and the maximum hot spot temperature and maximum burnup location would not coincide. Specifications for the fuel loading of the critical experiment are being prepared.
9. Study of the pressure vessel resulted in selection of 304 SS rather than carbon steel which is susceptible to irradiation damage. The cost difference was estimated to be less than $15,000.
10. Containment studies indicated the practicality of designing the shield tank outer shell as part of the containment vessel.

11. A blower scoping study subcontract was completed. The study verified the feasibility of the main and afterblower concept. Alternate shaft-seal designs were proposed. The design of a performance test for the two seal types has been initiated.
12. The design of the boiler test from which control characteristics will be determined was completed and fabrication started. Tests are scheduled to start in October 1962.
13. The decision was made that the Low Power Test Facility, LPTF, at the National Reactor Testing Station will be the site used for the critical experiment.
14. A preliminary study of the power test facility requirements were completed. The study indicated that locating the facility adjacent to the LPTF would be operationally and economically feasible.

1. PROPULSION SYSTEM INVESTIGATION
(100)

1.1 CYCLE ANALYSIS (101)

The 630A Maritime Nuclear Steam Generator Scoping Study (GEMP-108) showed performance estimates for primary loop air temperatures at the reactor exit of 1400° and 1200°F, with main blower discharge pressures of 300 and 400 psia, respectively. The 1200°F, 400 psia conditions promised fewer design problems in such areas as boiler support, fuel element life, and structural and boiler tube material requirements, and were selected as the basis for the work reported herein.

During this report period, preliminary performance specifications were revised. The revised specifications define over-all thermodynamic performance for the propulsion system, including the nuclear steam generator. The specifications are shown in Figure 1. 1, a heat balance diagram calculated for 27, 300 normal shaft horsepower and normal service. The propulsion turbine throttle steam conditions are 850 psig, 950°F, with a back pressure of 1. 5 inches Hg, absolute. The blower drive turbine operates on steam extracted from the main propulsion turbine and this steam is re-inducted into the crossover of the propulsion turbine. Over-all thermal efficiency is calculated to be 30. 2 percent. The reactor total power is 67. 4 megawatts.

Several alternate steam propulsion cycles have been studied for possible application to the 630A Nuclear Steam Generator. A comparison of these alternate systems with the cycle using the re-induction turbine blower drive is shown in Table 1. 1. The three alternate cycles use topping turbine blower drives in which the total steam flow passes through the topping unit. Although the boiler dimensions are larger for the cycles using a topping turbine, thermal efficiencies can be increased by as much as 1. 8 points (to about 32 percent) relative to the 30. 2 percent efficiency of the re-induction cycle shown in Figure 1. 1.

TABLE 1. 1

COMPARISON OF ALTERNATE STEAM CYCLES FOR 630A NCULEAR STEAM GENERATOR[a]

Turbine Blower Drive	Initial Steam Conditions	Main Propulsion Throttle Steam Conditions	Thermal Efficiency, percent	Boiler Sizing		
				Heat Transfer Area	Height, in.	Diameter, in.
Re-induction	865 psia 950°F	865 psia 950°F	30. 2	Base	90. 0	90. 0
Topping	1215 psia 950°F	865 psia 870°F	31. 2	+21%	92. 0	104. 3
Topping	1215 psia 1000°F	865 psia 920°F	31. 6	+25%	92. 0	107. 5
Topping with reheat	1270 psia 950°F	865 psia 950°F	32. 0	+32%	92. 5	112. 5

[a]Performance estimates and boiler sizing are based on the assumption of constant blower power and reactor size. Fuel element plate temperature variation is negligible.

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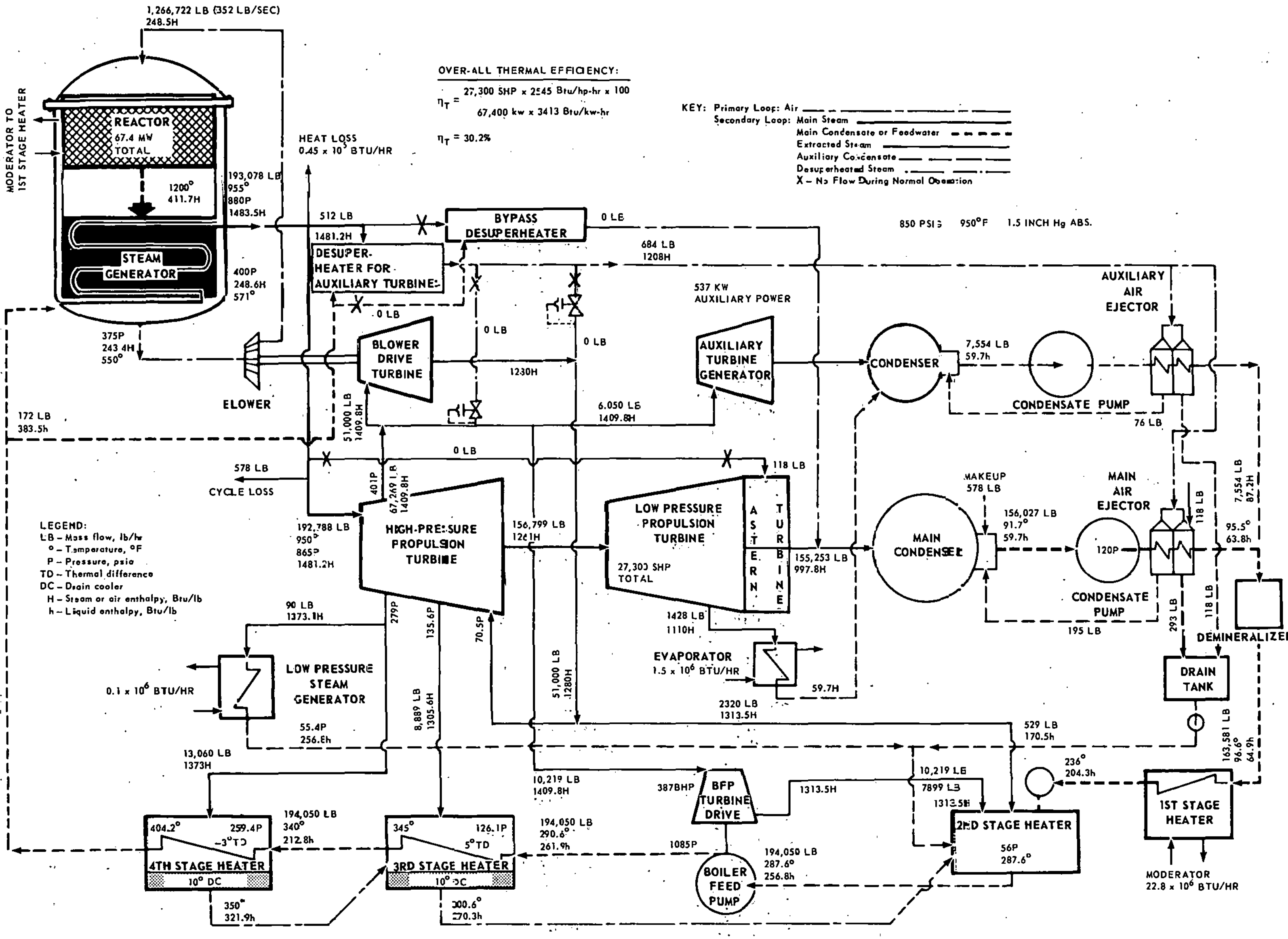

Fig. 1.1 – 27,300-SHP marine power plant heat balance at normal sea conditions

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Thermal efficiency can also be improved by using superheated steam rather than air as the primary-loop working fluid. With this approach, a primary-loop temperature range of 550° to 1200°F and blower discharge pressure of 400 psia provide an estimated over-all plant thermal efficiency of about 31.5 percent and about a 10 percent reduction in boiler height.

A combination of the superheated steam primary-loop fluid with the various topping turbine alternates shown in Table 1.1 would yield thermal efficiencies ranging up to about 33.3 percent, and increase the net heat transfer area in the boiler to about 22 percent.

Considerable effort has been expended in the attempt to optimize the trade-off between increased duct and annular flow area in the primary loop and the increase in pressure vessel diameter necessary to accommodate this flow area.

The effects upon pressure loss of varying blower inlet and exit flow areas have been evaluated and are shown in Figure 1.2. This calculation assumed each of the areas to be 650 square inches. The curves show that pressure loss decreases would be negligible if larger areas were selected.

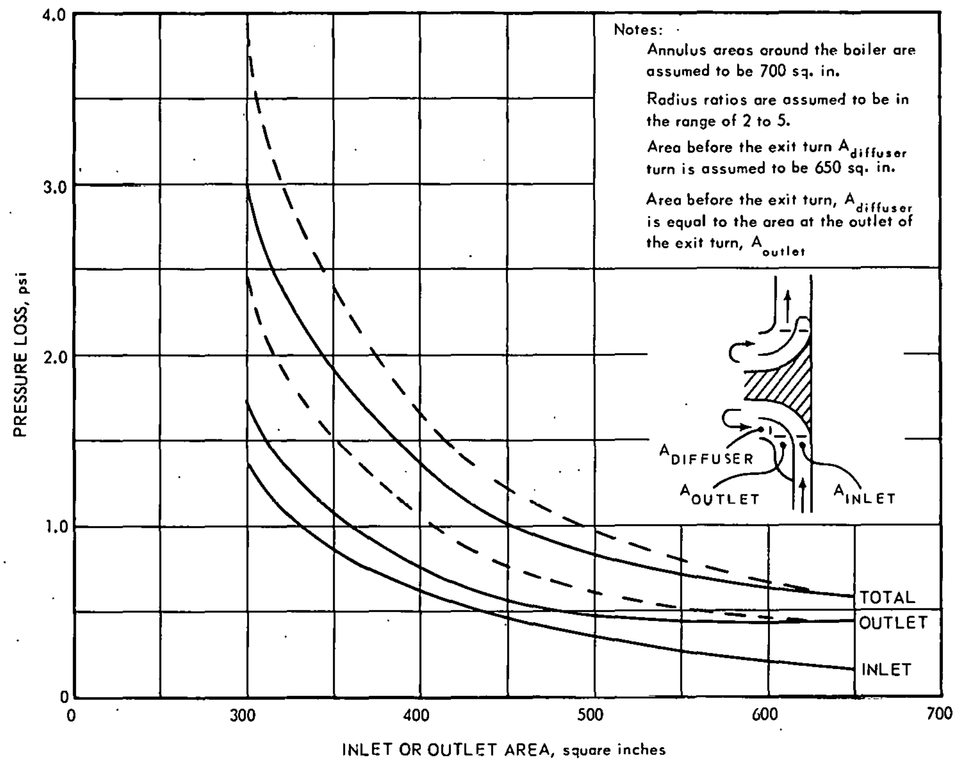


Fig. 1.2 – Effects of varying blower inlet and exit areas on pressure loss

Preliminary design studies of the blower ducting, which includes two diffusers and two 90-degree turns, indicate that pressure losses will be about 4.0 psi at the normal operating point. These losses are based upon a Mach number of 0.2 at the blower exit, diffusion to a Mach number of 0.1 before the turns, and additional diffusion to a Mach number of 0.05 before the exit to the annular duct around the boiler.

An allowance of 2.7 psi has been assigned to the pressure losses associated with the remainder of the ducting, which forms the flow path from boiler exit to blower inlet, and from the last diffuser following the blower to the fuel element inlet. The summation of the estimated pressure losses around the primary loop is 25 psi.

Figure 1. 3 shows the effect on over-all thermal efficiency of primary-loop pressure drop changes. A 1. 0 psi increase in pressure loss causes about 0. 08 percentage point decrease in efficiency, (i. e., 30. 2% to 30. 12%).

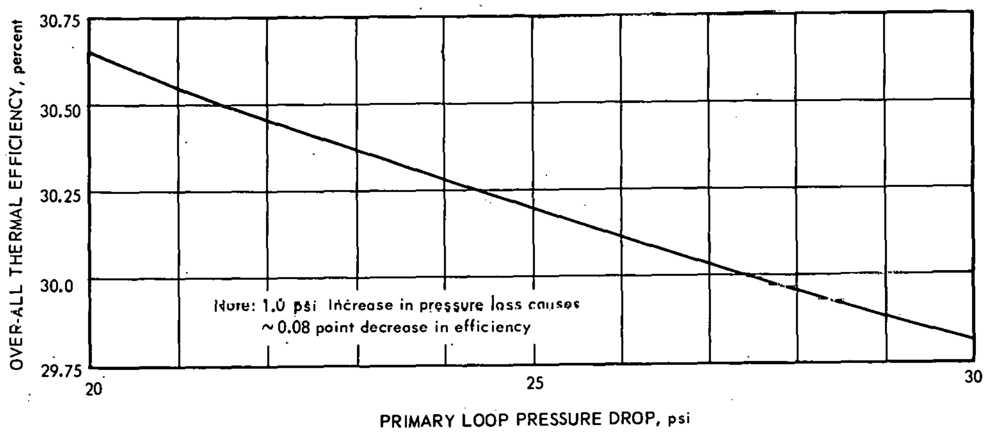


Fig. 1.3 – Effect of primary loop pressure loss on over-all thermal efficiency

1.2 STEAM GENERATOR ASSEMBLY (102)

Figure 1. 4 shows the general arrangement of the reactor, shield plug, boiler, pressure vessel, side shield, and two axial-flow blowers, arranged in a vertical assembly. The arrangement is identical to that outlined in GEMP-108, "630A Maritime Nuclear Steam Generator Scoping Study," and is repeated here for the convenience of the reader.

The boiler is located in the lower part of the pressure vessel with the tube sheet at the bottom. This tube sheet also serves as the lower head for the pressure vessel. The reactor and shield plug are permanently assembled and constitute a unit called the reactor-shield plug assembly. The reactor is suspended from the shield plug by the center moderator exit tubes and the 73 shim rod guide tubes, and is located immediately above the boiler. Part of the shield plug extends into the pressure vessel; the remainder extends above it. A flange located about half-way up the shield plug closes the pressure vessel at the top. The axial flow blowers are located within two equally-spaced nozzles that penetrate the pressure vessel near the bottom. The side shield surrounds the pressure vessel.

1. 2. 1 SCALE MODEL

A model of the 630A Nuclear Steam Generator was fabricated during the reporting period. This model will aid in determining access to components for servicing, location of components, assembly and disassembly procedures, handling requirements, and routing of piping.

The model was fabricated to a scale of 1/16 inch to the inch and depicts the reactor-shield assembly with individual containment as described in the Scoping Study.

Figures 1. 5, 1. 6, and 1. 7 are photographs of the scale model. Figure 1. 5 shows a cutaway cross section of all the components of the nuclear steam generator. Figure 1. 6 shows an over-all view of the power plant. Figure 1. 7 is a closeup of the reactor-shield plug assembly shown in cross section.

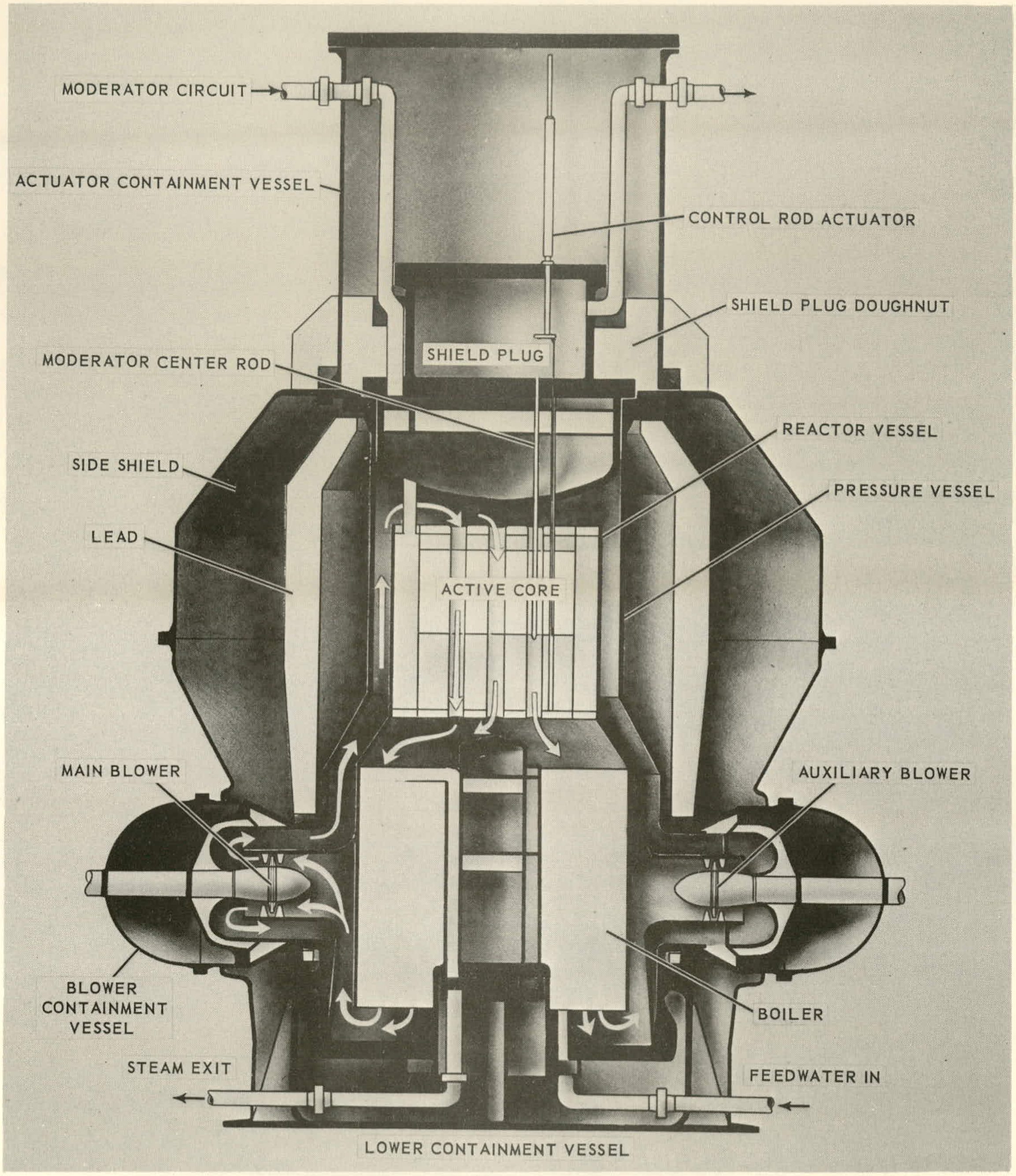


Fig. 1.4 – Artist's concept of the 630A nuclear steam generator (AS-71)

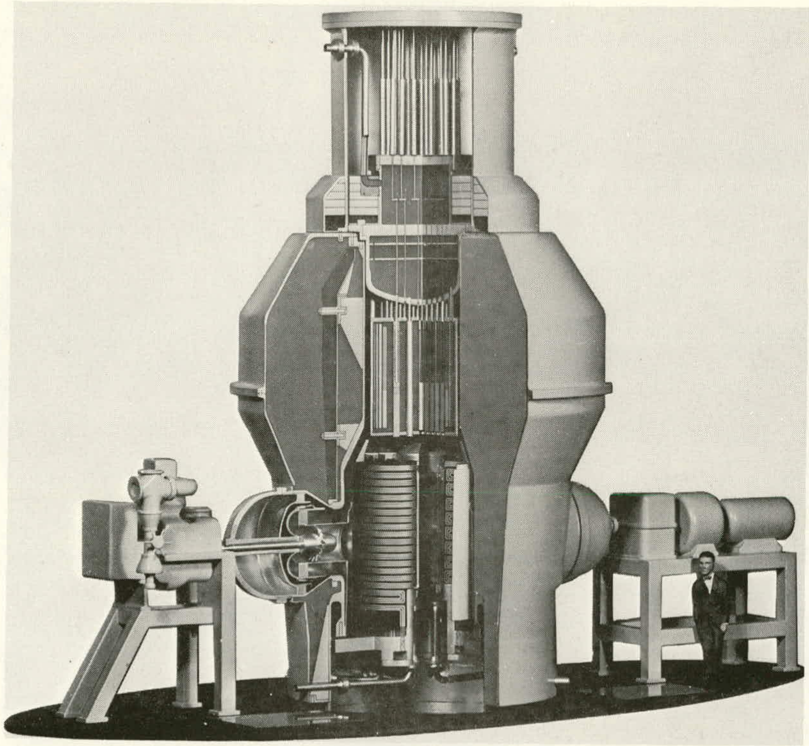


Fig. 1.5 – Cutaway cross section of the 630A scale model (Neg. P62-5-41G)

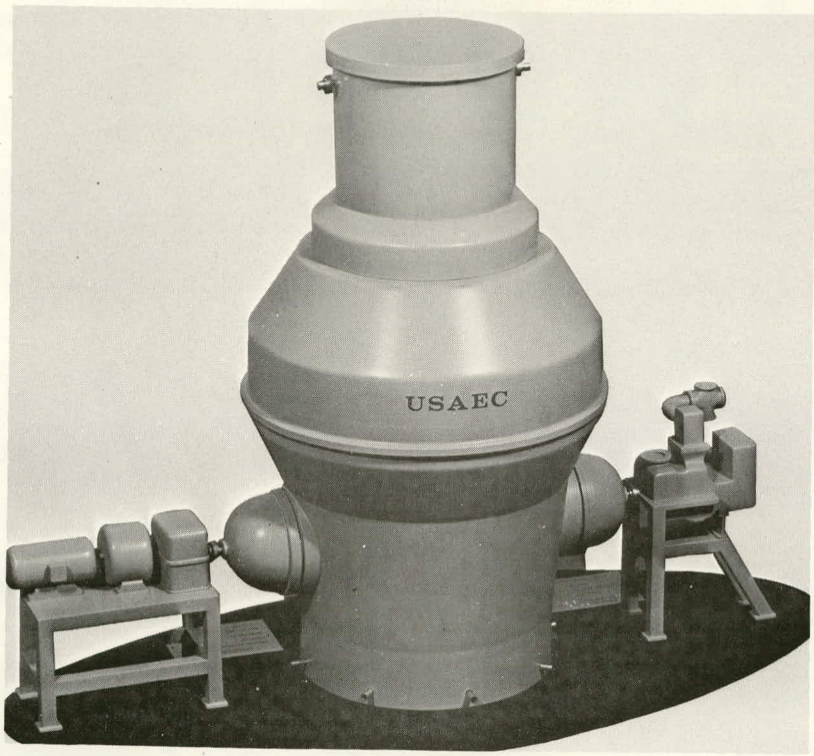


Fig. 1.6 – Over-all view of the 630A scale model (Neg. P62-5-41E)

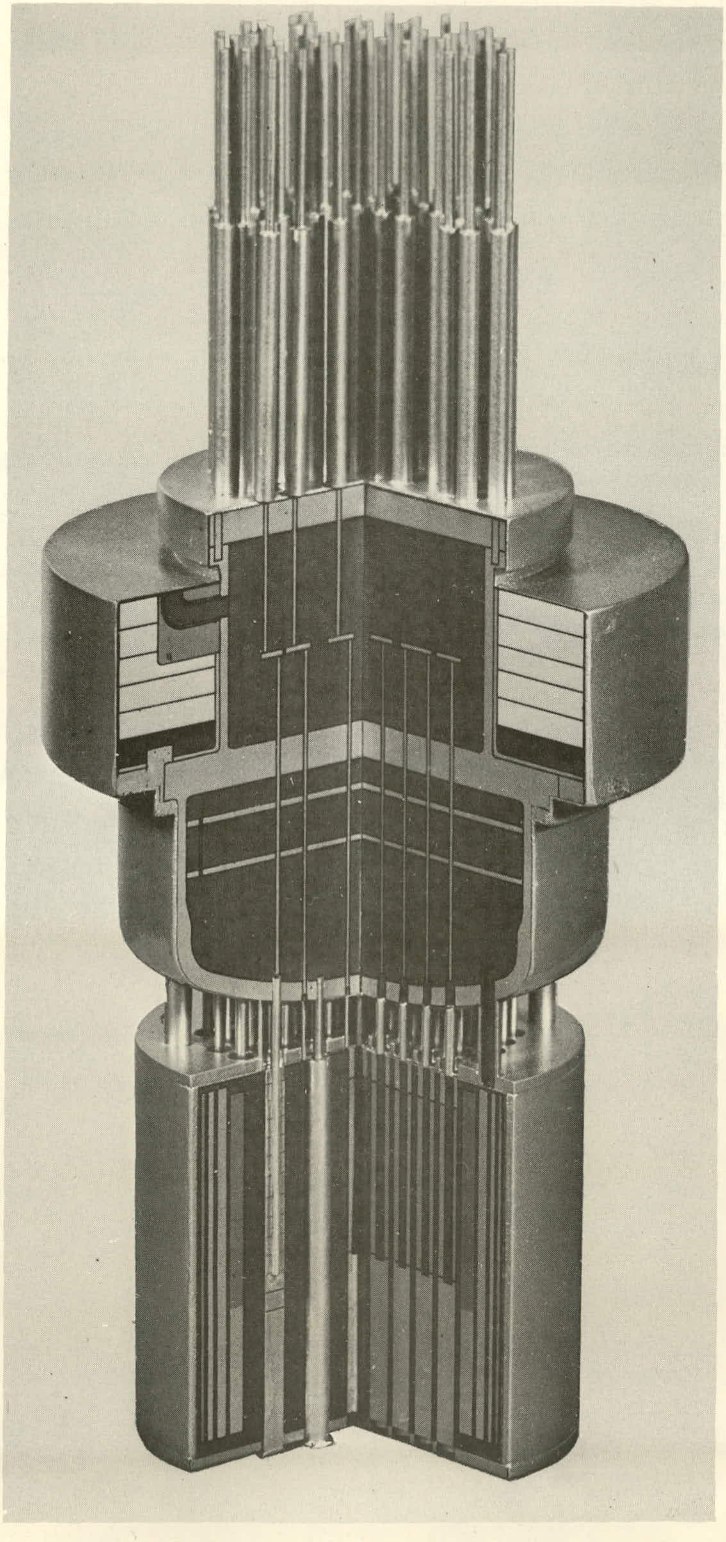


Fig. 1.7 – Scale model of 630A reactor-shield plug assembly (Neg. P62-5-41C)

1.3 WORK PLANNED FOR NEXT PERIOD

Continued evaluation of the topping turbine alternate cycles is planned, to determine the effect on boiler sizing and cost.

Continued analysis of pressure losses by analytical and scale-model testing is planned.

A 1/4-scale model flow test is being planned which will consist of a mockup of the ducting from the boiler exit plenum to the reactor inlet plenum. This mockup will include the blower flow passages as well as the annular passages around the boiler and calandria vessel. Normal design point operating conditions and possible scale model test conditions at the blower exit are shown in Table 1. 2. By use of dye or smoke injection and transparent ducting material, this model test should provide information about flow distribution to the blower and to the reactor. In addition, the relative pressure losses of various ducting sections will be measured, and extrapolation of data will provide estimates of normal operating pressure losses. During the next 2 months, detailed formulation of the test conditions and fabrication procedures are planned.

Sizing of the auxiliary blower will depend upon primary-loop pressure losses with aftercooling flow rate at atmospheric pressure. Because uncertainties exist regarding the nature of the fuel-element flow conditions and hence the pressure losses at low flow rates required in aftercooling at atmospheric pressure, a cold flow cartridge test is planned during July 1962. With a 15 pound-per-second flow rate through the fuel elements, Reynolds Number at the last stage is about 1200. Even though this is below the the transition range there is reasonable doubt as to whether laminar flow exists because there are flow path discontinuities along the fuel elements. Testing will be carried out on XMA-1 cartridge 66F59 which has geometry similar to the anticipated 630A cartridges. This cartridge was fabricated as a fueled cartridge for the ETR testing program but was never run. It is currently being returned from Oak Ridge storage.

TABLE 1. 2

SCALE MODEL AND DESIGN OPERATING CONDITIONS

Configuration	Area, in.^{2}	Mass Flow, lb/sec	Mach No.	Dynamic Head, psi	Reynolds Number	Blower Discharge Pressure, psia	Blower Power, BHP
Design	162	352	0. 2	10. 9	7.7×10^6	400	2600
1/4 model	10. 1	1. 15	0. 2	0. 4	1.5×10^5	14. 7	1. 56
1/4 model	10. 1	4. 7	0. 9	7. 4	6.2×10^5	21. 0	105

2. REACTOR
(110)

During this report period, three areas of reactor design were investigated:

1. Gross radial power shaping
2. Fine radial power shaping
3. Aerothermal characteristics

Preceding the discussion of these three areas, a general description of the reactor is presented for the convenience of the reader.

2.1 GENERAL DESCRIPTION

The reactor of the 630A Maritime Nuclear Steam Generator is air cooled, water moderated, and uses metallic fuel elements. Its design power is about 70 megawatts, thermal. The side reflector consists of beryllium and stainless steel, and serves as a thermal shield between the active core and the reactor vessel, as well as a neutron reflector.

The reactor vessel is a cylindrical tank of 304 stainless steel, closed at each end by tube sheets. The vessel is 68. 8 inches in diameter and 63. 5 inches long. Eighty-five holes, each approximately 4. 5 inches in diameter, are drilled in each tube sheet for the fuel tubes which pass through the tank and are rolled and welded to the tube sheets. This array of tubes forms the active core, which is 48 inches in diameter. A water-filled moderator tube is centered in each of the fuel tubes and extends about 38 inches into the reactor vessel from the front tube sheet. A fuel cartridge is inserted in each annular space between the center moderator tube and the fuel tube. The fueled length of the cartridge (and thus the height of the active core) is 27. 5 inches.

The fuel cartridge consists of nine identical fuel stages, each approximately 3 inches long. Each stage consists of concentrically spaced fuel rings with a nominal 0. 060-inch annulus between adjacent rings for airflow. On the average, there are 12 concentric rings in each cartridge. An inner and an outer unfueled structural ring, each spaced 0. 045 inch from the adjacent fuel ring, serve as the structural base for the elements. Thermal insulation is fastened to the outside of the outer structural ring, and the inside of the inner structural ring. The fuel inventory for the system, with a lifetime of about one million megawatt-hours, is 427 pounds of U^{235}.

Nuclear control is provided by three components; burnable poison, shim, and scram rods. The insulation on the outer structural ring of each fuel element is held in place by a fixed stainless steel foil which contains boron. The boron reduces the excess reactivity in the system at startup. As the reactor is operated and fission products accumulate, the boron in the foil is burned out nuclearly, counterbalancing the effect of the fuel burnout and fission product accumulation. The remaining excess reactivity is controlled by

72 shim rods which are located between the fuel tubes in the moderator. The shim rod guide tubes extend from the front plug through the front tube sheet and are positioned at the rear tube sheet. The shim rods are about 0.5 inch in diameter. Safety rods are located in 81 of the 85 center moderator tubes; these rods vary from 0.80 to 1.0 inch in diameter.

A complete description of the reactor is contained in the 630A scoping study.*

2.2 GROSS RADIAL POWER SHAPING (112)

Power shaping, or temperature flattening, is required in gas-cooled reactors in order to utilize the temperature capability of the metallic fuel elements. To minimize the heat transfer surface-to-coolant temperature differential, all of the surface area at a particular axial location should operate at the same temperature. In a homogeneous system or in a heterogeneous system where the basic fuel cell is the same at all radial locations, the power generated per unit of fuel is highest at the center of the reactor, and falls radially outward toward the reflectors. If the 630A system were to be operated for a short time, the fuel per unit of heat transfer area could be reduced in the central fuel elements, and increased in the elements at the outer edge of the core, thus achieving a flat gross radial power generation rate per unit of heat transfer area. However, because the 630A system is a long-lived reactor, the amount of fuel that is burned out nuclearly is quite large, and a much greater percentage of the fuel would be burned out in the lightly loaded elements at the center of the reactor than in the outer elements. This would result in a large gross radial power swing during the operating lifetime of the fuel elements. The gross radial power shaping in a system with high percentage burnup must be achieved so that the rate of fuel burnup is relatively constant in the radial direction. To accomplish this, the 630A reactor employs a varying fuel-to-moderator ratio, in the radial direction.

2.2.1 VARYING FUEL-TO-MODERATOR RATIO

During this report period, two methods of varying the fuel-to-moderator ratios were examined. The first method consists of varying the center moderator tube size in the gross radial direction, to provide less moderation at the center of the reactor and more at the outside. This was accomplished by either adding or subtracting integral fuel rings at the internal diameter of the stage. The base point was the 12-ring stage. The number of rings was varied from 11 to 15 in this study. The fuel loading per unit of heat transfer area was held constant in each of the fuel elements. The gross radial fission density for three configurations is shown in Figure 2.1. The three configurations are defined in Table 2.1.

The fuel, structure, and moderator in each region have been homogenized with the appropriate neutron flux disadvantage factors for neutron energies extending from the thermal region up to 8.315 ev. The safety rods and the shim rods are assumed to be withdrawn from the active core, and the initial amount of burnable poison is located in the stainless steel foils. The heat transfer area in the third configuration is approximately 2 percent greater than that in the first two configurations.

Note that the gross radial fission density is also the gross radial power distribution per unit of heat transfer area.

The results of this part of the study indicate that varying the center moderator size does not provide sufficient worth for complete gross radial power shaping. The fission

*"630A Maritime Nuclear Steam Generator Scoping Study," GE-NMPO, GEMP-108, April 6, 1962.

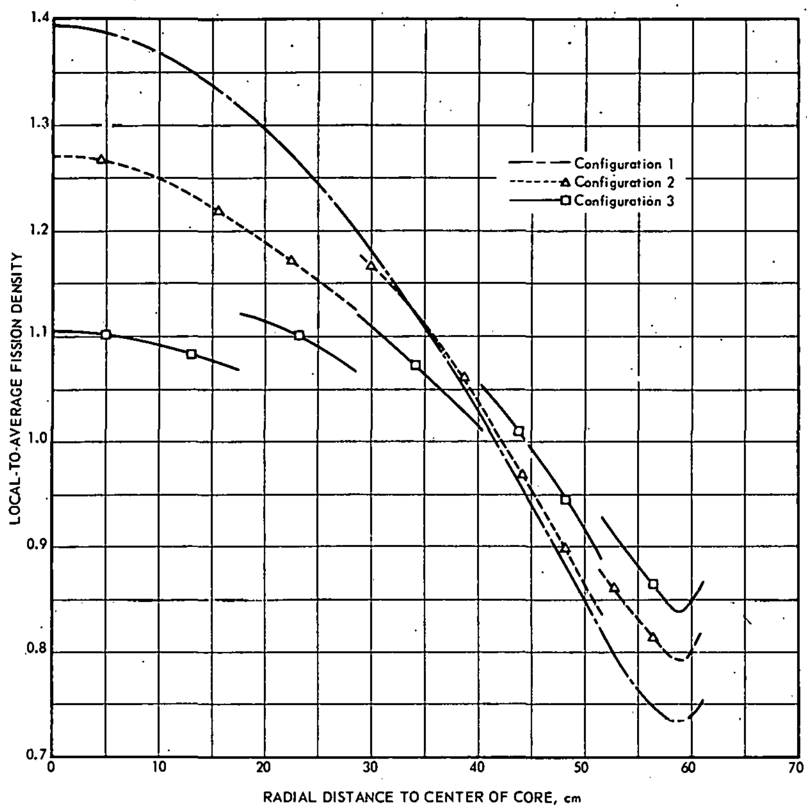


Fig. 2.1 – Gross radial fission density, center moderator variation

TABLE 2.1

CONFIGURATION VARIATIONS[a] MADE IN GROSS RADIAL POWER DISTRIBUTION STUDY

Configuration Number	1	2	3
Number of Cell Types	1	3	5
11-Ring Cells	0	24	24
Inside radius[b]	-	51.64	51.64
Outside radius[b]	-	60.96	60.96
12-Ring Cells	85	42	24
Inside radius[b]	0	28.82	40.22
Outside radius[b]	60.96	51.64	51.64
13-Ring Cells	0	19	18
Inside radius[b]	-	0	28.82
Outside radius[b]	-	28.82	40.22
14-Ring Cells	0	0	12
Inside radius[b]	-	-	17.49
Outside radius[b]	-	-	28.82
15-Ring Cells	0	0	7
Inside radius[b]	-	-	0
Outside radius[b]	-	-	17.49

[a]Variations are in center moderator size and number of rings per cell.
[b]From reactor core centerline.

distribution shown for the third configuration is not as flat as desired. The 15-ring fuel element leaves little room for safety rods in the center moderator region. Finally, the flux distribution indicated that the burnable poison at the center of the core would burn up faster than that in the outside of the core; thus, further peaking of the fission density at the center of the core would occur as the reactor is operated.

These results led to the conclusion that it would be desirable to vary the quantity of moderator on the outside of the fuel element as well as the center moderator.

The second method studied for varying the fuel-to-moderator ratio was to displace the moderator outside of the fuel tube by using aluminum shims. The shims would occupy the interstitial spaces between the tubes, and would also replace the stainless steel shim rod guide tubes as they pass through the active core. In the final configuration, the displacement of moderator outside the fuel cell may be accomplished by varying the tube spacing. The fission densities of two configurations analyzed are shown in Figure 2.2. The specifications for these configurations are given in Table 2.2. Configuration 4 uses

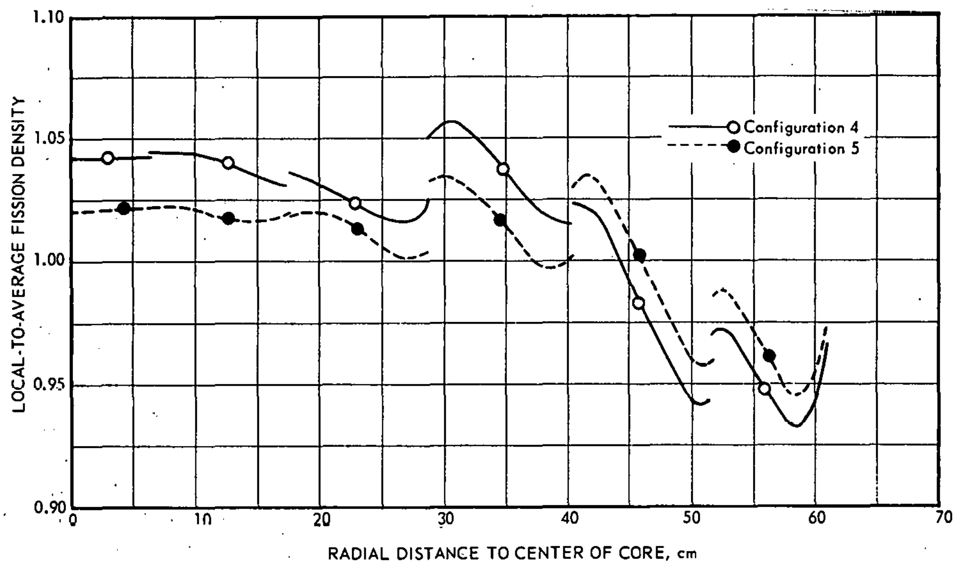


Fig. 2.2 – Gross radial fission density, center moderator variation and outer moderator displacement by aluminum shims

TABLE 2.2

CONFIGURATION VARIATIONS MADE IN GROSS RADIAL POWER DISTRIBUTION STUDY BY INCORPORATING ALUMINUM SHIMS

		Configuration 4		Configuration 5	
Fuel Cell Location	No. Of Cells In Reactor	No. Of Rings	Aluminum Volume Fraction, %	No. Of Rings	Aluminum Volume Fraction, %
Central Element	1	13	13.206	14	10.937
First Concentric Annulus	6	13	12.614	14	10.365
Second Annulus	12	13	11.507	14	8.639
Third Annulus	18	12	10.365	13	8.059
Fourth Annulus	24	12	5.725	12	5.136
Fifth Annulus	24	11	0.967	11	0.515
Average Aluminum Volume Fraction			6.75		5.382

11-, 12-, and 13-ring fuel elements; configuration 5 uses 11-, 12-, 13-, and 14-ring fuel elements. Configuration 5 has about 1.8 percent more flow area and fuel than configuration 4.

As indicated in Figure 2.2, both configurations 4 and 5 yield extremely flat fission densities at the start of the reactor's life. The burnup rate of the burnable poison is slightly higher at the outside of the reactor than at the inside, tending to flatten the fission density even more as the reactor continues operating.

On the basis of this study of displacement of both center and outer moderator, the decision was made to use this technique to achieve the required degree of gross radial power shaping.

2.2.2 WORK PLANNED FOR NEXT PERIOD

The change in gross radial fission density as a function of operating time is to be determined. This will be accomplished by first assuming uniform fuel burnup with operating time, and analyzing the reactor at the end of fuel element lifetime. If a satisfactory configuration is indicated by this study, a more detailed study of such a configuration will be made. This will consist of non-uniform burnup of both fuel and boron, at several time intervals.

In the final design, the gross radial power will probably be shaped by fuel-tube pitch variation, to reduce to a minimum the amount of moderator displaced from the active core by the aluminum shims. This work will be initiated after the best aluminum shim configuration has been defined for the critical experiment.

The aluminum displacement method was selected in preference to tube spacing because it is easier to mock up in the critical experiment.

2.3 FINE RADIAL POWER SHAPING (112)

The fission rate is highest next to the moderator and lowest toward the center of a single concentric ring fuel element. To reduce the difference in the fission rate, center moderator regions are employed in the 630A reactor to increase the low energy flux which causes most of the fissions. So that each ring of the concentric ring fuel element operates at the same temperature in a particular axial stage, the fuel inventory per unit of heat transfer area is varied roughly in inverse proportion to the fission density.

In low burnup systems, flattening the temperature of the concentric fuel rings in a particular stage is relatively easy. In the high-burnup 630A system, however, the problem is more difficult. If the fuel is distributed to produce flat temperature distribution across the stage at the start of operation, the outermost ring, in which the fission rate is highest, will have a lower-than-average heat flux at the end of life. On the other hand, fuel rings located in the middle of the cluster (e.g., the sixth ring in a 12-ring element) will experience a much lower percentage of fuel burnup and will operate at a higher-than-average temperature. The design goal in the 630A system is to reduce to a minimum the fine radial temperature swing that will occur during the operating lifetime of the fuel elements.

2.3.1 DETERMINING FINE RADIAL FISSION DISTRIBUTION

During this report period, the changes in the fine radial fission distribution for a typical cell were determined. The fuel, the fuel burnup, and the fission products were assumed to be homogeneously distributed. Figure 2.3 shows the results of this study. The fission density distributions of all the operating points considered (startup, equilibrium xenon, and

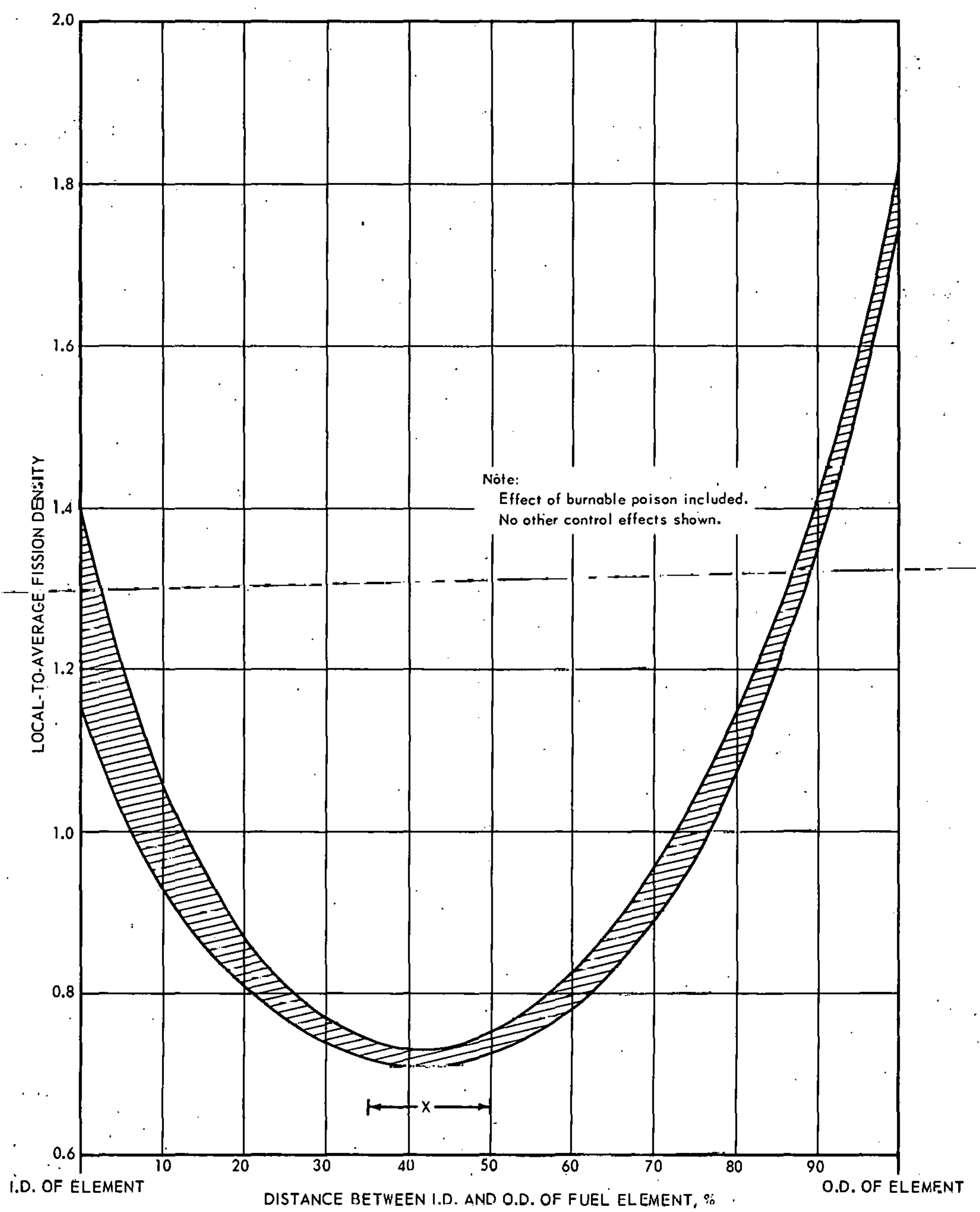


Fig. 2.3 – Fine radial fission distribution across a 12-ring fuel element

1/4, 1/2, 3/4, and end of fuel element lifetime) lay between these two curves. The analysis was made for a 12-ring element, with the burnable poison included at the start of life. Boron burnup was accounted for in each calculation.

The initial fission rate and burnup rate of the outer ring of this fuel element is approximately 170 percent of the average rate. It will fall slightly at the end of life, to about 165 percent of the average rate. Due to the larger-than-average percentage of fuel burnup in this ring, the heat generated will be only about 69.2 percent of the initial rate. Therefore, if this ring starts at the average temperature at the beginning of operation, it will run quite cold at the end (below 1200°F).

The fuel ring at the inside of the element starts operation with an average fission rate of about 124 percent of the average, and ends with an average of about 105 percent. On the assumption that the average burnup of the inside ring is about 115 percent of the average, at the end of fuel element life it will generate about 79.2 percent of the heat that it generated at startup. Although its fission density falls considerably, the average rate of fuel burnup is not very high. Therefore, the product of the amount of fuel left, compared to that left in the whole cell, times the final fission rate yields a smaller reduction in heat generation rate than in the outer ring, which had an insignificant reduction in fission rate, but a large reduction in fuel inventory.

Assuming that one ring near the center of the element (e.g., in region X of Figure 2.3) has a fairly constant fission rate of 73 percent of the average, that ring will generate approximately 112 percent of the initial heat at the end of life, because a higher proportion of its fuel will be left. Assuming that this ring operates at a maximum average temperature of 1330°F at the start of operation, its temperature at the end of life would be about 1425°F.

2.3.2 WORK PLANNED FOR NEXT PERIOD

The effects of the gross radial flux and power distributions that are achieved by varying the amount of moderation on the outside of the fuel element will be determined. This includes the effect of different rates of boron burnup at various radial positions in the core.

The effects on the fine radial power distribution of the unequal distribution of fission products and fuel within the fuel element as burnup occurs will be analyzed to determine if they are of significance in the thermal design.

The fuel distribution from ring-to-ring will be varied to determine the optimum distribution for minimizing the effect of the fine radial power swing during the fuel element operating time on the maximum temperature of the fuel elements. Consideration will be given to depressing the power in some areas and increasing it in others to minimize the temperature swing with operating life.

2.4 AEROTHERMAL CHARACTERISTICS (113)

During this report period, the General Flow Passage computer program was used to estimate the aerothermal characteristics of the reactor. The most significant of these characteristics are a calculated pressure loss of 10.2 psi and an average maximum plate temperature of 1330°F for the normal operating conditions.

The reactor and primary loop heat distributions on which the power plant heat balance shown in Figure 1.1 is based, are shown in Tables 2.3 and 2.4. The indicated accuracy is not warranted at this time but primarily serves as a basis for the heat balance and for future comparisons.

From the above data it is seen that 0.02 mw is assumed transferred to the moderator from duct air and 0.067 mw transferred to the shield from duct air. In addition, 0.038 mw is assumed transferred from the boiler to the duct air. These estimates of duct air heating and cooling are based on the current plan of having a 3/16-inch air gap insulation at both the inner and outer walls of the entire duct annulus as well as at the dished head of the shield plug.

Effects of eliminating air gaps were calculated to be as follows:

1. If there are no air gaps on either side of the annular duct opposite the boiler, an

additional 0.264 mw would be lost to shield water because of radiation from the boiler (0.5 percent heat loss).
2. If no air gap is used on the outer duct wall, an additional 2.9 mw would be lost to shield water as a result of thermal transfer from the duct air (4.5 percent heat loss).
3. If air gaps are not utilized on the calandria vessel and shield plug, an additional 1.1 mw would be transferred to the moderator (1.65 percent heat loss). When added to the steam cycle, this heat would allow about 1.1 percent additional power generation, so the net performance loss would be about 0.5 percent. In addition, the terminal difference at the moderator heater would be reduced by 22°F with a feedwater temperature of 258°F.

Estimates of the component heat distribution in the nuclear steam generator will be adjusted as more detailed evaluations of nuclear heating are obtained.

TABLE 2.3

REACTOR HEAT DISTRIBUTION[a]

Component	Component Heating mw	%
Biological Shield	0.0674	0.10[b]
Boiler	0.0539	0.08
Moderator		
Neutron plus γ	5.99	8.89
Thermal	0.6737	1.00[b]
Air	60.590	89.93
Total	67.375	100.0

[a]Outer surface of calandria is system boundary.
[b]Does not include heat received thermally from duct air.

TABLE 2.4

PRIMARY LOOP HEAT DISTRIBUTION[a]

	Component Heating mw	%
Heat Rejection		
Thermal heat from fuel to moderator	0.674	0.973
Gamma plus neutron heat to moderator[b]	5.990	8.642
Duct air heat to moderator	0.020	0.029
Gamma heat to biological shield	0.067	0.097
Duct air heat to biological shield	0.067	0.097
Gamma heat to boiler and steam	0.054	0.078
Air heat to steam	62.440	90.085
Total	69.312	100.0
Heat Addition		
Reactor	67.375	97.205
Blower	1.937	2.795
Total	69.312	100.0

[a]Pressure vessel is system boundary.
[b]This is heat to the moderator circuit, including all energy deposited in the side, front, and rear reflectors and in the front shield plug, all of which are cooled by water that is part of the moderator circuit.

3. SIDE SHIELD
(130)

During this report period, fast neutron and gamma ray dose rates were calculated for all crucial points on the surface of the shield. These calculations are discussed following a general description of the shield.

3.1 GENERAL DESCRIPTION

The principle components of the 630A shield are the side shield, shield plug, and dry shield doughnut (Figure 2. 1). The shield reduces the radiation levels outside the nuclear steam generator to a biologically safe value of approximately 2 milliroentgens per hour in work areas of the compartment.

The side shield consists of a tank of borated water and an annular cylinder of lead surrounding the pressure vessel. The shield tank water contains 0. 6 weight percent boron. The annular cylinder of lead is encased in mild steel. The lead cylinder is to be fabricated in four mating axially-split sections which are supported from the pressure vessel.

The shield plug which is attached to the reactor serves as the top cap of the pressure vessel, provides shielding for the area above the reactor, forms plenums for moderator water distribution, and provides mounting for the control rod actuators. The demineralized moderator water serves as the neutron shield; a 7-inch-thick lead slab encased in stainless steel is the primary gamma shield.

The dry-shield doughnut of polyethylene and lead is fastened above the shield tank around the shield plug. The doughnut serves as a shield cap for the clearance void required between the pressure vessel wall and the shield plug, to reduce the total radiation dose in the area to the 2 milliroentgens per hour. The doughnut shielding is fastened to a supporting cylindrical shell fabricated in three radial arc segments.

3.2 SHIELD ANALYSIS (131)

During this report period, fast neutron and gamma ray dose rates have been calculated at a number of points on the surface of the shield, including all points believed to be crucial. Secondary gamma ray dose rates from neutron interactions in the side shield have also been calculated.

The present analysis is based on the engineering drawings of the shield as presented in the Scoping Study (GEMP-108).

The analysis of this shield proceeded in two parts. In part one, the dose rate from the radiations emitted from the reactor core were computed. In part two, the gamma ray dose rate from neutron captures in the side shield were calculated.

The dose rate calculations for the radiation dose rate from the core were calculated using a computer program* based on a point kernel approach. With this method, the reactor source is borken down into a number of source regions. From each source region the distance to the receiver point and the distance through each material in the shield is calculated. These distances and the material attenuation functions are used to compute the attenuation from the source region to the receiver point. The program then integrates over the source regions in the reactor to obtain the total dose rate at the receiver point. The attenuation function for the neutron dose rate is an Albert Welton kernel. This kernel is a mathematical fit to the experimental neutron measurements in water from a reactor source. The non-water or non-hydrogenous part of the attenuation is calculated using an exponential function and a removal cross section. The removal cross sections were experimentally determined.

The attenuation function for gamma rays is an exponential function using linear absorption coefficients and buildup factors. For the gamma ray calculation, the gamma ray energy spectrum of the reactor is broken down into several energy groups and the attenuation calculations are carried out for each group. The gamma ray sources in the reactor used in this calculation included prompt-fission gammas, fission-product decay gammas, and gamma rays resulting from non-fission neutron captures in the reactor. The calculations performed for the after-shutdown dose rates were the same as the operating dose rates except that the source regions represented the fission-product decay gammas for 100 hours of reactor operation and for 3 hours after operation.

Based on the program incorporating these features, dose rates were calculated at a number of points on the surface of the shield.

The gamma ray dose rate resulting from neutron capture in the side shield is calculated by establishing the number of radiative captures per second in the shield materials. This capture rate is used as a source region in a shielding computer code. The number of captures in the shield is determined by computing the neutron flux and energy spectrum at each position in the shield, using a one-dimensional multigroup diffusion code normalized to the absolute reactor power.† The product of the multigroup results and the radiative capture cross sections of the shield materials yields the capture rate. The capture rate can be translated to Mev/sec-cm^2 released. This value is the secondary gamma ray source that can be used as input in the shielding computer code to obtain dose rates. The computer code used is the same code used to obtain dose rates from the reactor with the exception that the materials of the shield are described as the source. This procedure was used to obtain secondary gamma ray dose rates at the shield surface.

From the present calculations, a few generalized comments can be made for this shield configuration. The average dose rate level across the top of the shield plug and out through the side shield is near the levels that will be required to provide a shipboard installation meeting the radiation specifications outlined in NBS Handbook 59, "Permissible Dose from External Sources of Ionizing Radiation." The dose rate levels in the region of the plastic shield plug, in the region adjacent to the gas exit plenum between the reactor and boiler and the region directly below the boiler are in excess of the required levels. Additional shield design work will be directed toward decreasing the dose rates in these regions while maintaining weights at the prescribed levels.

*J. T. Martin, J. P. Yalch, and W. E. Edwards, "Shielding Computer Programs 14-0 and 14-1," GE-ANPD, XDC 59-2-16, January 23, 1959.

†D. J. Campbell, "Program G-2," GE-NMPO, XDC 58-4-62, April 1962.

3.3 WORK PLANNED FOR NEXT PERIOD

An analysis of the radiation leakage from the 630A reactor shield described in the Scoping Study (GEMP-108) is in progress. This analysis is to determine the fast neutron, and gamma ray dose rates at the surface of the shield, for the case of full powered operation and for the case of fission-product decay gamma rays after the reactor is shut down. The analysis will be extended to cover nuclear heating rates and material activations in the significant parts of the power plant.

4. PRESSURE VESSEL

(140)

During this report period, it was determined that the use of 304 stainless steel instead of carbon steel in the pressure vessel would eliminate the possible problems associated with radiation damage to this vessel. In addition, the results of a blower study indicated modifications in the vessel in the area of the blowers would be advantageous. A general description of the pressure vessel is presented for the convenience of the reader.

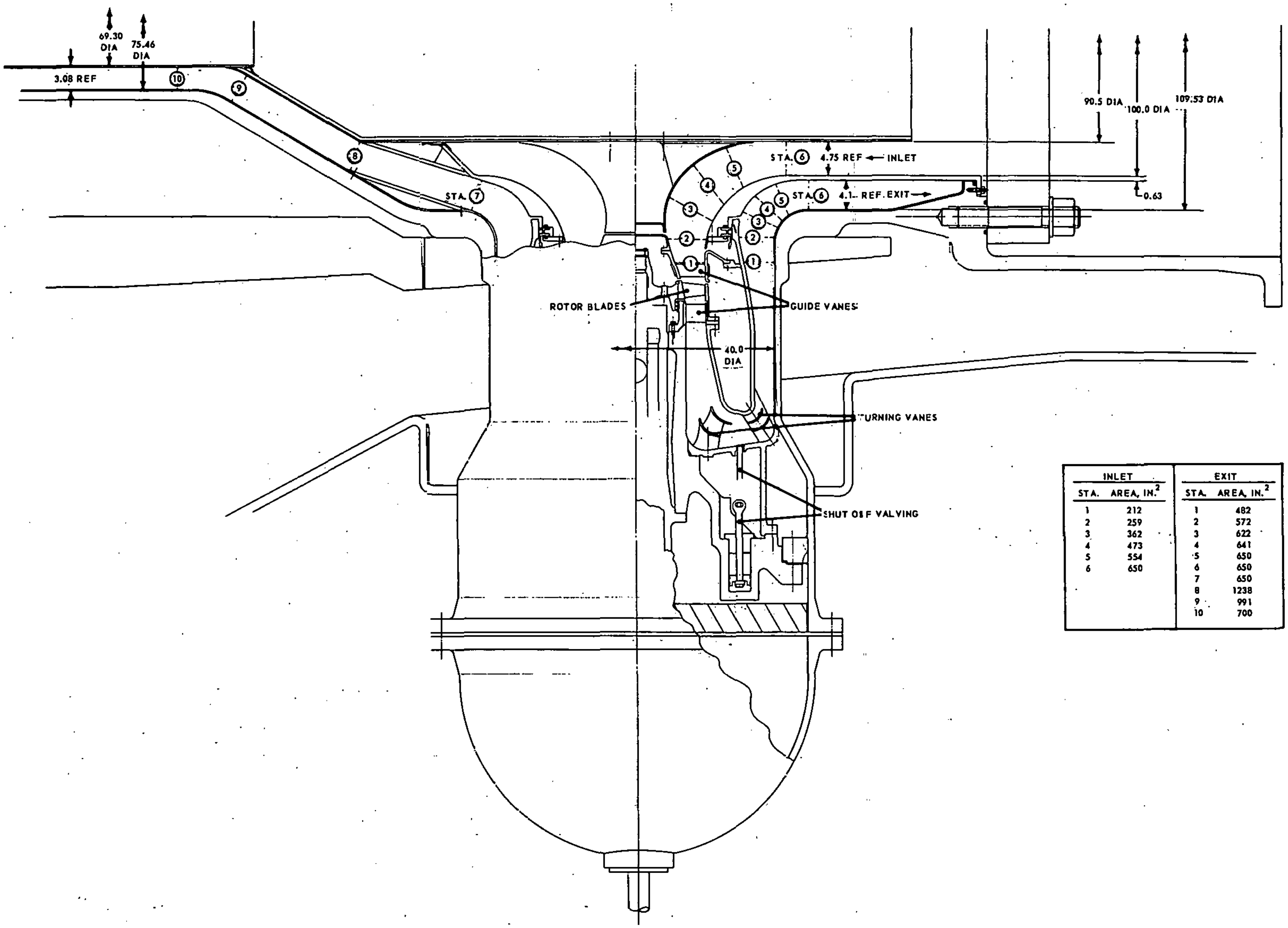
4.1 GENERAL DESCRIPTION

The vertical cylindrical pressure vessel is approximately 212 inches high and is flanged at each end (Figure 1.4). It supports and/or contains the reactor-shield plug assembly, the boiler, and the reactor primary coolant. It also supports the annular cylinder of lead, a part of the side shield, that surrounds the pressure vessel. The pressure vessel is made up of a lower section, an upper section, and a conical transition section. The boiler is located in the lower part of the pressure vessel with the tube sheet at the bottom. The tube sheet serves as the lower head for the vessel. The reactor-shield plug assembly is located within the upper section of the pressure vessel with the reactor immediately above the boiler. A flange located about half-way up the shield plug closes the pressure vessel at the top. The axial-flow blowers are located within two equally spaced nozzles that penetrate the lower section of the pressure vessel near the bottom. The side shield surrounds the pressure vessel.

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. Air exiting from the boiler passes upward between the boiler and the flow divider, enters the inlet duct to the blower that penetrates the flow divider, and is discharged from the blower into the annulus between the flow divider and the pressure vessel. The air then flows upward in the annulus formed by the outside surface of the reactor vessel and the inside surface of the pressure vessel, to the plenum formed by the lower surface of the shield plug and the upper face of the reactor vessel. The air then flows downward through the reactor and boiler to complete the loop. A lining of thin-gauge stainless steel in a waffle pattern is attached to the inside of the pressure vessel wall to produce a stagnant-air barrier to reduce the heat loss from the airstream through the pressure vessel wall to the shield water.

4.2 PRELIMINARY DESIGN STUDY

During this report period, a preliminary design study of the blower was completed. The study revealed a need to increase the flow areas at the blower inlet and exit to minimize pressure drop in these areas. Different combinations of nozzle diameter, inside diameter of the lower section of the pressure vessel, and flow-divider diameter have been studied



INLET		EXIT	
STA.	AREA, IN.^{2}	STA.	AREA, IN.^{2}
1	212	1	482
2	259	2	572
3	362	3	622
4	473	4	641
5	554	5	650
6	650	6	650
		7	650
		8	1238
		9	991
		10	700

Fig. 4.1 – Blower inlet and exit flow area study (Dwg. 219R852)

to achieve approximately 650 square inches of flow area at the blower inlet and exit. The current configuration under consideration is shown in Figure 4. 1.

The decision was made to fabricate the pressure vessel of 304 stainless steel, instead of ASTM-A302 Grade B as originally planned, to minimize the effects of radiation damage over the 20-year life of the vessel.

4.3 WORK PLANNED FOR NEXT PERIOD

Effort will be continued to achieve an optimum pressure vessel configuration in the blower inlet and exit areas. The wall thicknesses of the pressure vessel will also be checked to satisfy the lower stress allowable for the 304 stainless steel.

5. BOILER
(150)

During this report period, a preliminary specification for the 630A boiler was prepared for use in an engineering study. The purpose of this study and a summary of boiler requirements are presented after a general description of the boiler.

5.1 GENERAL DESCRIPTION

The 630A boiler is a once-through water tube boiler that supplies superheated steam (950°F and 850 psig) to the propulsion turbines. Heat is transferred in cross-counter flow from the primary-air loop to the secondary steam-water loop in the boiler by first passing over the steam superheating surfaces which comprise about 15 percent of the total heat transfer area. It then passes successively over the evaporating and economizer surfaces which contain about 54 and 31 percent of the total heat transfer area respectively.

The boiler is situated directly below the reactor core as shown in Figure 1. 4 and occupies the approximate volume of a 90-inch-high, 90-inch-diameter right circular cylinder.

The once-through boiler was chosen for two main reasons: (1) this design results in the minimum number of penetrations of the pressure vessel when compared to a recirculating design with a steam drum and (2) the high pressure nuclear heated air eliminates most of the problems associated with a once-through boiler. The absence of flue gases allows recirculation of the 400-psi air thus permitting a more compact tube arrangment without danger of fouling. Since the temperature of the primary air is at 1200°F there is little danger of "burning" out tubes with changes in water level.

The boiler is discussed more detail in the 630A Scoping Study Report (GEMP-108).

5.2 BOILER ENGINEERING STUDY

The work during this period was primarily concerned with preparing a preliminary boiler specification and negotiating an engineering subcontract with a boiler manufacturer. The Boiler Engineering Study is to provide GE-NMPO with engineering information as to the size, configuration, interfaces, weight, materials, performance, off-design characteristics, reliability, and handling and maintenance procedures for the 630A boiler so that these characteristics may be included in the GE-NMPO engineering effort. A summary of the more pertinent requirements at normal conditions for the 630A Boiler are listed in Table 5. 1.

Although the specifications defining the boiler have been available for several weeks, a delay has been encountered in establishing acceptable subcontracting procedures. The request proposal on the CPFF subcontract for the engineering study will be issued in July.

TABLE 5.1

PRELIMINARY SPECIFICATIONS FOR "ONCE-THROUGH" BOILER WITHOUT DRUM

Primary Loop Normal Operating Conditions (air)	
Flow rate, lb/sec	362
Inlet temperature, °F	1200
Inlet pressure, psia	383
Pressure at boiler exit, psia	375
Maximum desired pressure drop through boiler including entrance and exit losses at 362 lb/sec flow and inlet pressure of 383 psia	8 psi
Temperature drop through boiler, °F	650
Heat transfer rate, Btu/hr	219.6×10^6
Primary Loop Design Conditions	
Pressure, psia	455
Temperature of air, °F	1300
Secondary Loop Normal Operating Conditions (water-steam)	
Flow, lb/hr	201,680
Normal exit temperature, °F	955
Inlet pressure, psia	1015
Discharge pressure, psia	865
Maximum desired pressure drop, psi	150
Feedwater inlet temperature, °F	417
Secondary Loop Design Conditions	
Temperature, °F	1050
Internal pressure, psia	1050
External pressure, psia	455

6. CONTAINMENT (160)

During this report period, an evaluation study of the three possible methods of containment for the 630A Nuclear Steam Generator was begun. The three methods of containment available for the 630A Nuclear Steam Generator are described below.

1. Individual Containment
   In this configuration, the most critical areas such as the blowers, control actuators, the joint between the shield plug and the pressure vessel and the joint between the pressure vessel and the steam generator and its associated steam lines, are contained by separate units (Figure 2. 1).
2. Total Containment
   A pressure shell completely surrounds the reactor assembly, including the steam generator, in the total containment configuration (Figure 6. 1).
3. Modified Individual Containment
   In this system, the outer shell of the side shield is designed to contain the pressure resulting from a rupture of the pressure vessel wall. Otherwise, the arrangement is identical to the individual-containment configuration described in 1. above.

These containment methods are more fully described in GEMP-108, "630A Maritime Nuclear Steam Generator Scoping Study."

6.1 DESIGN STUDY (161)

During this report period, a study was begun to more fully evaluate the merits of each of the three methods of containment. During the study of the total containment method, recalculation of the expansion volume in the total containment vessel indicated the correct volume is approximately 6125 cubic feet instead of 2289 cubic feet previously published in GEMP-108, "630A Maritime Nuclear Steam Generator Scoping Study." The maximum pressure in the total-containment vessel resulting from the occurrence of the maximum credible accident will be recalculated, after which the wall thickness will be re-evaluated.

A weight increase of about 85, 350 pounds is required if the outer shell of the side shield is designed to contain the pressure resulting from a rupture of the pressure vessel wall as proposed in the modified individual containment method. The wall thickness of the shield wall will increase to approximately 3 inches at its maximum diameter.

This study will continue during the next reporting period.

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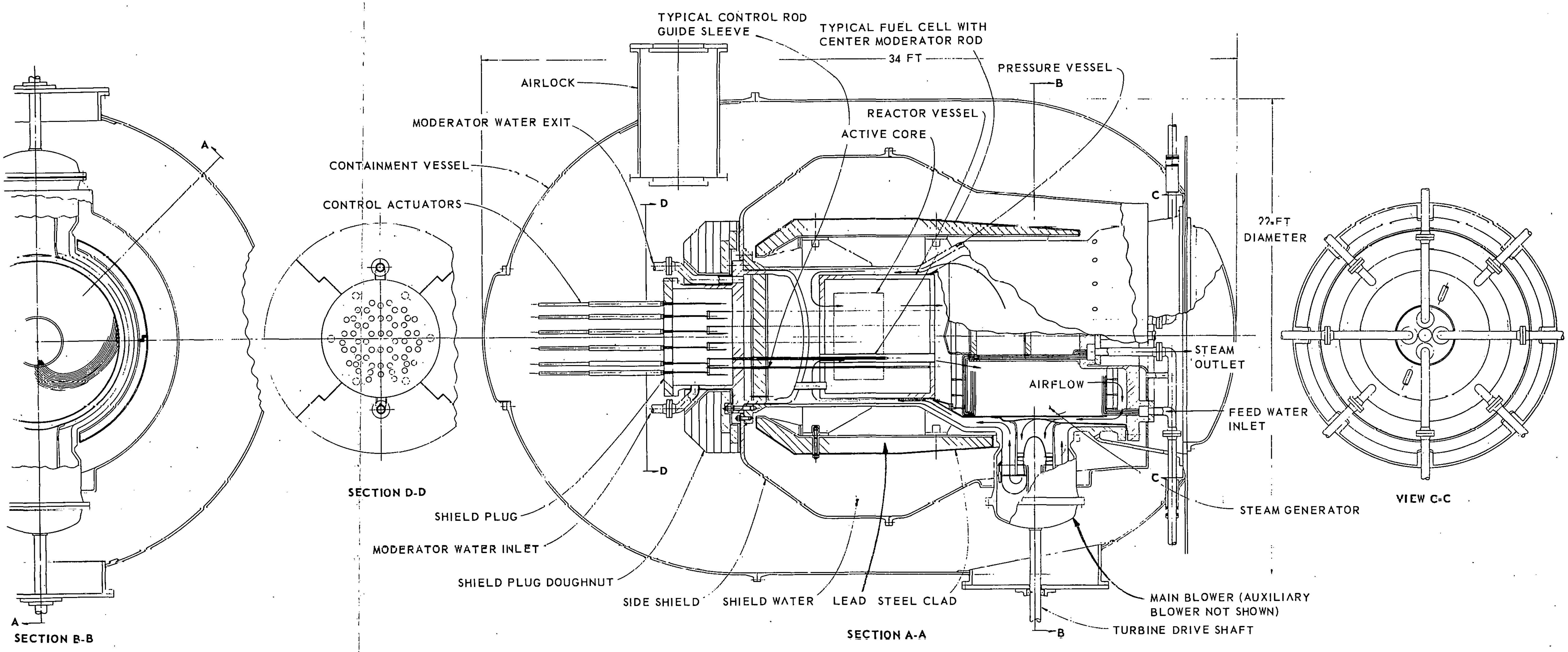

Fig. 6.1 – RSA study No. 4 with total containment vessel (Dwg. 219R810)

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7. CONTROLS
(190)

During this report period, work was initiated to define boiler test rig requirements and to place orders for the components needed to build the test rig.

7.1 TWO-TUBE BOILER TEST (198)

A test to simulate the reactor, boiler, air loop, and control system is planned to assure that the integrated system is stable under all normal and emergency operating conditions, that both steady-state and transient accuracy requirements are met, that the various control loops function together properly, and that satisfactory operating procedures are developed.

An initial 630A controls analysis was made using analog simulation of the reactor, boiler, air loop, and controls. In this preliminary systems investigation, the simulation was not complete enough to accomplish the detailed analysis needed for final design or checkout. The next step will consist of building an actual boiler with supporting accessories for analysis purposes. This approach has several advantages over analog simulation of the boiler and feedwater control system: (1) it permits operation of an actual feedwater control, which is probably the most critical loop in the system, (2) very valuable experience will be gained in working, at the earliest possible date, with actual control hardware, and (3) the cost of the boiler and associated accessory equipment is reasonable in terms of the expected yield of information.

Through the use of this test rig, such things as method of startup, control parameters, method of system integration, both steady-state and transient system accuracy, control system responses, and system stability will be determined. Although the air pressure at the inlet to the boiler will be only about 110 psia, it is expected that a significant amount of heat transfer and steam generation data will be obtained. Stability of boiling in parallel tubes will be investigated.

During this report period, the system was defined schematically as shown in Figure 7. 1. Requirements were written and orders were placed for standard type components. Demineralized water from an exisiting facility storage tank is supplied to the inlet of a high-pressure triplex piston pump where it can be boosted to pressures up to 1400 psig. A back-pressure regulating valve which bypasses flow around the pump is used to maintain the desired pump discharge pressure. The flow then passes through a valve which regulates the quantity of flow supplied to the boiler. The actual flow is sensed in a flow-measuring section just downstream of the control valve and, when the system is on automatic control, any error existing between this signal and the demanded flow causes the valve to move to correct the error. The flow next passes through a closed feedwater heater where its temperature may be increased up to 425°F by steam which is generated in the boiler. The flow is then split to pass through the parallel boiler tubes. Provisions are made in the inlet and outlet

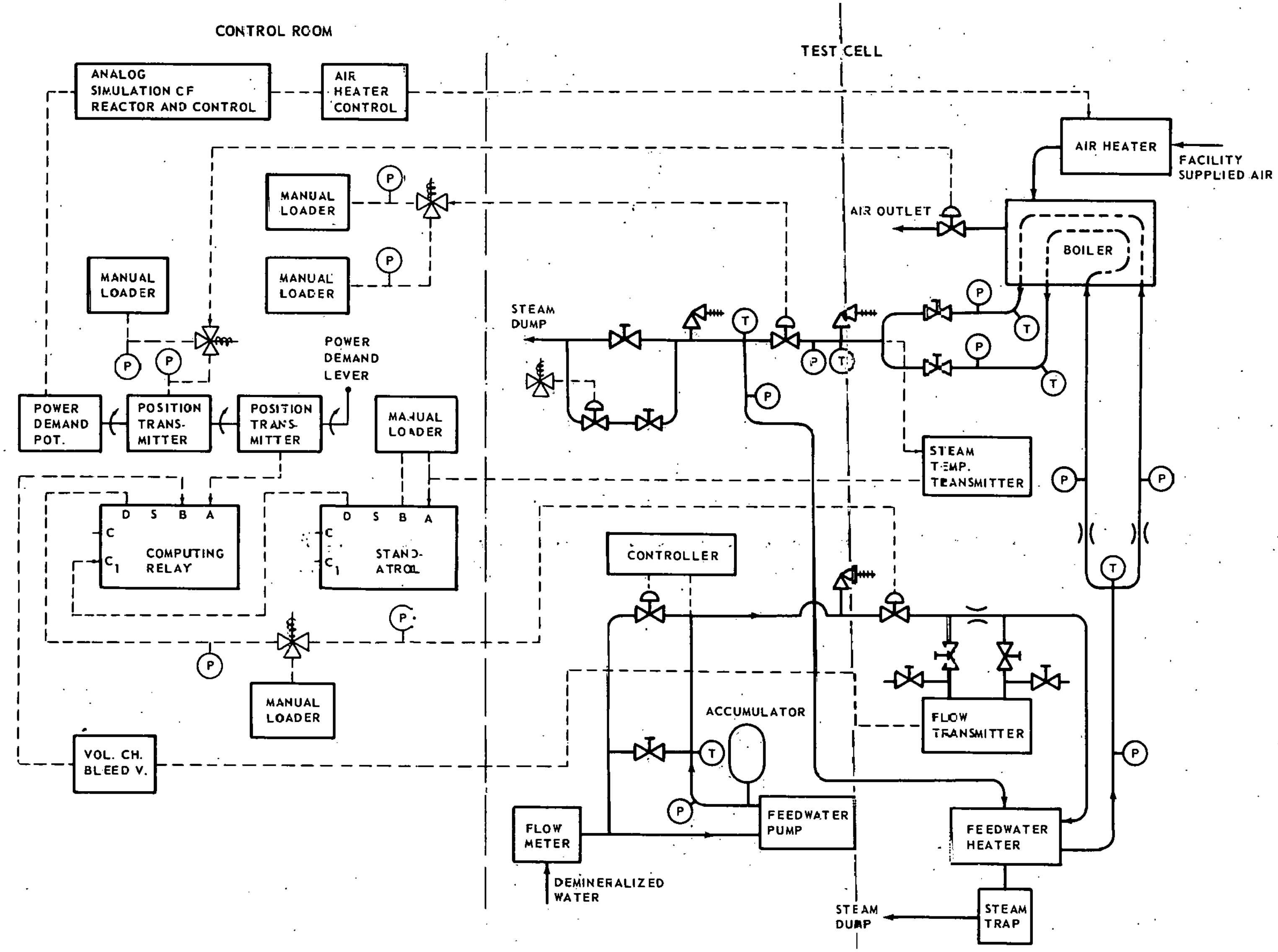

Fig. 7.1 – Two-tube boiler test system schematic (Dwg. SK125B7538)

of each tube for adding restrictions to be used in the study of stability of boiling in parallel tubes. The temperature of the mixed steam flow is sensed and the signal used to trim the feedwater flow if the temperature differs from that demanded. A throttling valve is provided to set the desired back pressure on the boiler. Steam for feedwater heating is bled off downstream from this valve. Control of the pressure in the steam side of the feedwater heater is controlled by downstream throttling valves and provisions are made for step-changing it. Control may be accomplished by manual control of each parameter or by fully integrated automatic operation. Air is provided to the boiler inlet at 110 psia and 1200°F by the existing burner test-pad facility. The heater used may be controlled manually or by an analog computer which will provide a simulation of the reactor and its control. When operating with the computer, power input to the air will be automatically varied to maintain constant pressure at boiler steam discharge.

The boiler, a sectional view of which is shown in Figure 7. 2, along with inlet and exit piping was designed and orders were placed for standard pipe and pipe fittings. Negotiations are underway for procurement of the pressure shell and boiler coils. The boiler utilizes two paralleled tubes of the same diameter, and approximately the same length and coil height as the full-scale boiler. The tube spacing and airflow gaps are sized so that normal continuous full-scale boiler steam conditions of flow, pressure, and temperature can be attained utilizing air inlet conditions of 110 psia and 1200°F.

A control stand was designed which provides the capability of integrated or individual loop operation. It also provides an instrument display for critical parameters.

Design of the assembly of the components shown schematically in Figure 7. 1, and design of the complete system into the test facility were completed.

The existing burner test-pad facility in which the boiler test rig will be installed and operated was checked out and found to be in good condition and capable of meeting the requirements specified.

In summary, the boiler test rig design was completed; orders were placed for all hardware except the boiler shell, boiler coils, boiler inner tube, boiler insulation, and miscellaneous small pipe and pipe fittings; and the facility where it will be installed was checked out. Table 7. 1 presents the status of component procurement.

7.2 WORK PLANNED FOR NEXT PERIOD

Orders for the remainder of the required boiler test rig materials will be placed early in the period. Fabrication of the assembly will be started.

TABLE 7.1

630A TWO-TUBE BOILER TEST ASSEMBLY STATUS

Drawing Or Part Title	Drawing No.	Drafting Status	Procurement Status	Shop Status
Feedwater control	Vendor part	None required	Ordered - due July	None
Flow meter	Vendor part	None required	On hand	None
Pressure regulating valve	Vendor part	None required	Ordered - due July	None
Feedwater heaters	Vendor part	None required	Ordered - due July	None
Steam trap	Vendor part	None required	Ordered - due July	None
Steam relief valves	Vendor part	None required	Ordered - due July	None
Steam flow control valve	Vendor part	None required	Ordered - due July	None
Manual throttling valves	Vendor part	None required	Ordered - due July	None
Water relief valve	Vendor part	None required	Ordered - due July	None
Valve positioner	Vendor part	None required	Ordered - due July	None
Water line orifices	Vendor part	None required	Ordered - due July	None
Feedwater pump and motor	Vendor part	None required	On hand	-
Accumulator	Vendor part	None required	On hand	-
Boiler assembly and detail drawings	219R847	Complete	Standard flanges and pipe ordered, outer shell and coils out for bid	Not started
Control stand and detail drawings	692E610	Complete	80% of parts on hand or on order	Not started
Panel arrangement and installation	848D636	Complete	All parts except for pipe and pipe fittings ordered	Not started

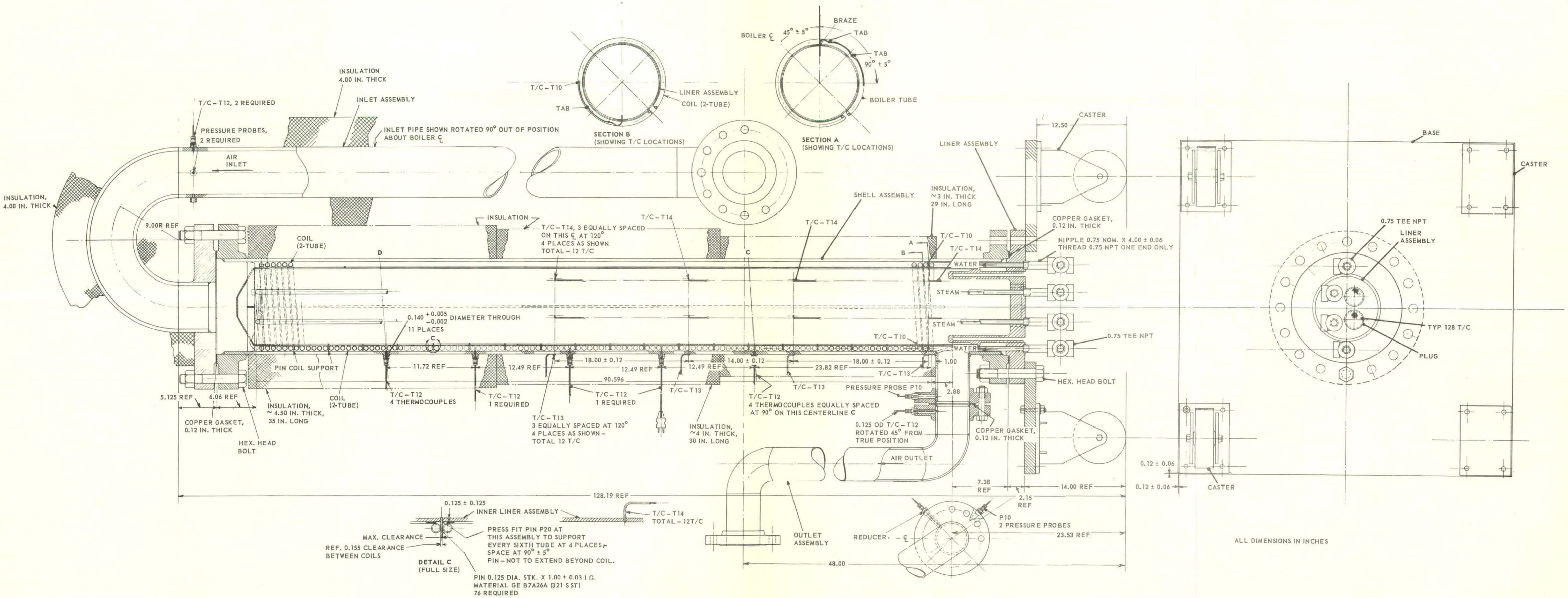

Fig. 7.2 – Boiler assembly (Dwg. 219R847)

8. PRIMARY COOLANT SYSTEM
(200)

During this report period, a feasibility study of a single blower to meet requirements of both main and auxiliary blowers was completed. The results of this study are presented, following a general description of the blower.

8.1 GENERAL DESCRIPTION OF THE BLOWER

The 630A nuclear steam generator requires two blowers, designated main and auxiliary, to circulate the high pressure air which transfers heat from the reactor to the boiler. The main blower is driven by a steam turbine and is used to circulate the air during normal power operation. The auxiliary blower is driven by an electric motor through an eddy-current coupling and speed increaser, and is used to circulate air during startup, and for both normal and emergency aftercooling. By emergency aftercooling is meant circulation of the required cooling air through the reactor during a condition of partial or complete depressurization of the primary air loop. The auxiliary blower can also be used for "take home" power in the event the main blower is incapacitated.

8.2 ENGINEERING STUDY

During this report period, an engineering study contract was placed with the Large Jet Engine Department of the General Electric Company. The purposes of this study were: (1) to determine the feasibility of designing and building a single blower to meet the performance requirements of both the main and auxiliary blowers; (2) to provide a layout of the blower which defines the envelope, interface details, general internal configuration, and air seals in sufficient detail to permit the start of a seal-evaluation test program; (3) to provide design and off-design performance predictions and practical guarantees; and (4) to provide specification details, which, when combined with the drawings and other information generated for items 1, 2, and 3 above could be used in obtaining vendor bids for final design and fabrication of the blower or blowers.

This study was completed late in the reporting period, and a report and drawings which presented the results were issued.

8. 2. 1 PERFORMANCE

A blower, the performance map for which is shown in Figure 8. 1, was designed to meet the performance requirements of the main blower. The normal continuous operating condition was selected as the design point. This design was then evaluated for the normal maximum pressure ratio requirement of 1. 10 and a 1. 3 pressure ratio for the emergency requirement. Operating at the higher pressure ratio and ambient pressure inlet conditions would require the main blower design to operate at speeds of 150 to 170 percent of normal

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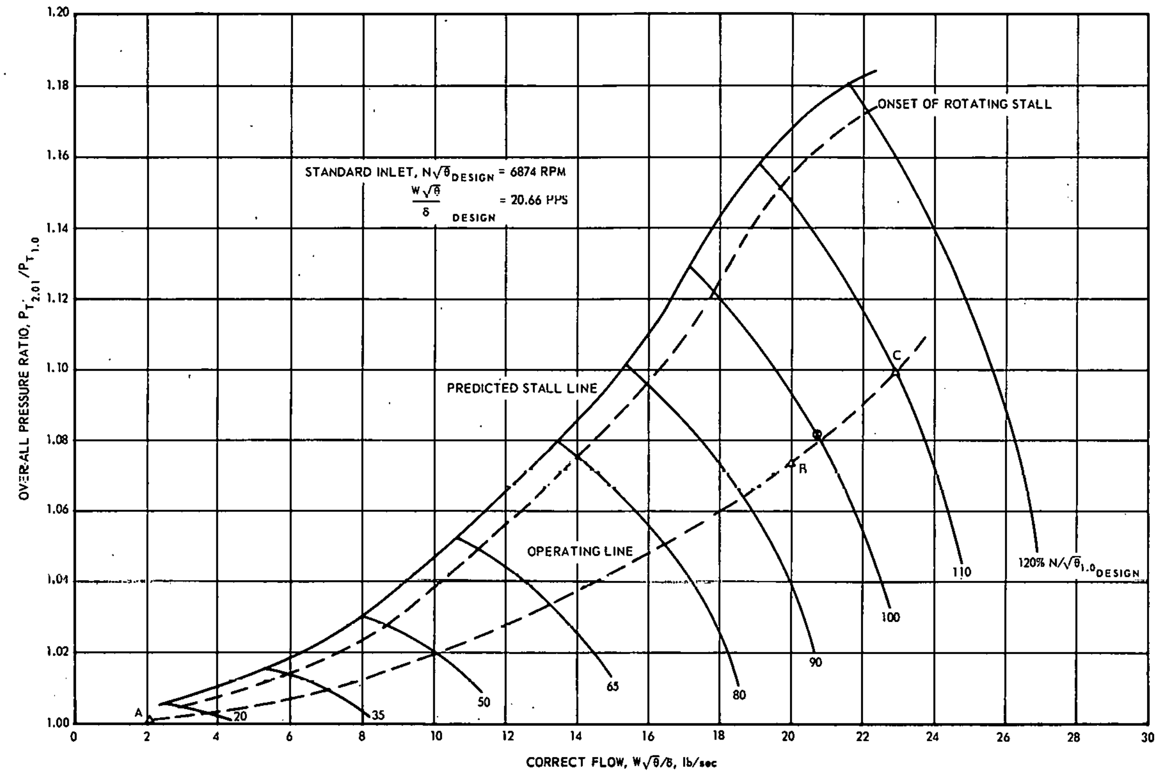

Fig. 8.1 – Continuous duty fan preliminary predicted performance – pitchline

continuous speed. In addition, it would be necessary to bypass an estimated 40 percent of the flow around the blower in order to bring the operating point below the stall line. Operating at the high physical speed requires a more complex shaft design to keep the first shaft critical above the maximum operating speed and increased tip clearances; this more complex shaft design would reduce normal continuous operating point efficiency, to allow for rotor growth. The possibility also exists that the operating point could not be brought under the actual stall line, or that the rotor or stator might become choked before the designed operating point could be reached.

An alternate blower, a performance map for which is shown in Figure 8. 2, was designed to meet all the auxiliary blower requirements. It uses the same basic arrangement as the main blower with the exception of the rotor blades, stator blades, and the outer wall of the airflow passage. Figure 8. 3 shows the main blower configuration in solid lines and the special auxiliary blower design in phantom. The rotor blading for this blower is scaled from an existing design, modified to eliminate preswirl.

8. 2. 2 BLOWER DUCTING

In order to select the airflow path with the lowest pressure drop, and one which would fit within the required envelope configuration, three different types of bends were investigated. These were: (1) a plenum bend, (2) a maximum radius, constant area bend, and (3) a bend with turning vanes. This investigation showed that a flow path having a bend with turning vanes preceded and followed by 2:1 area ratio, 12 degree diffusers gave the minimum pressure drop. This configuration is shown in Figure 8. 3.

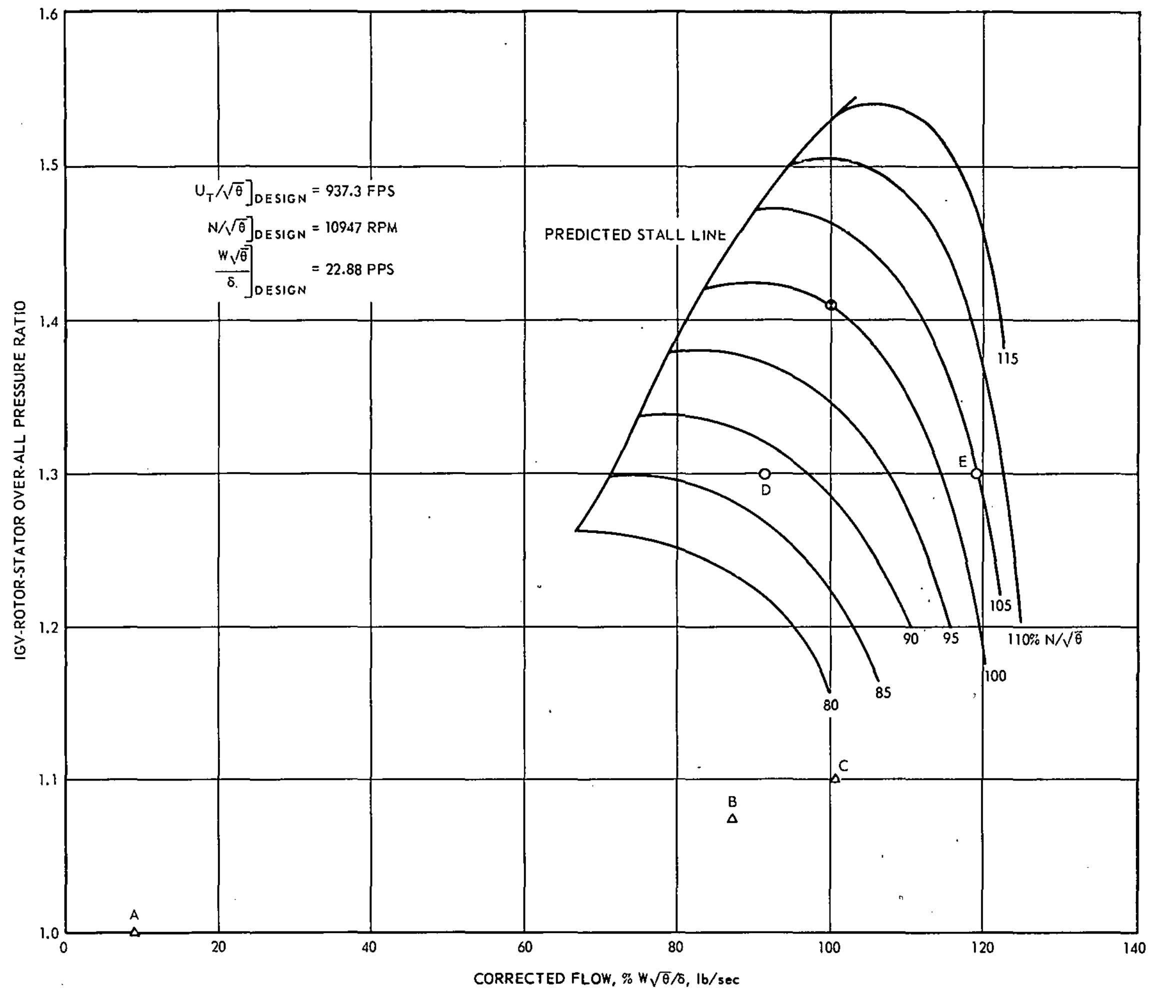

Fig. 8.2 – Preliminary predicted emergency fan map

8. 2. 3 SHAFT

A shaft configuration study was carried out to determine critical speeds for various combinations of shaft diameter, bearing size, bearing spacing, overhung moment, center hole size, and seal diameter. A simple solid shaft was deemed satisfactory if the auxiliary blower design incorporated the modifications as shown in Figure 8. 3. For the case in which the identical design was used for the main and auxiliary blower, a shaft center hole, larger in the middle portion than at the ends, would be required.

8. 2. 4 BEARING

The shaft bearing problem was studied and two journal and two thrust bearings, both of the hydrodynamic type, were recommended. The journal bearings recommended are of the type used in steam turbines. The type, quantity, and temperature of the lubricating oil required was also determined.

8. 2. 5 SEALS

Several different blower-drive shaft seal configurations designed to prevent primary-loop air from leaking through the blower to the atmosphere were considered. All were based on using fresh air supplied to seals at 1 to 3 psi above primary-loop pressure to

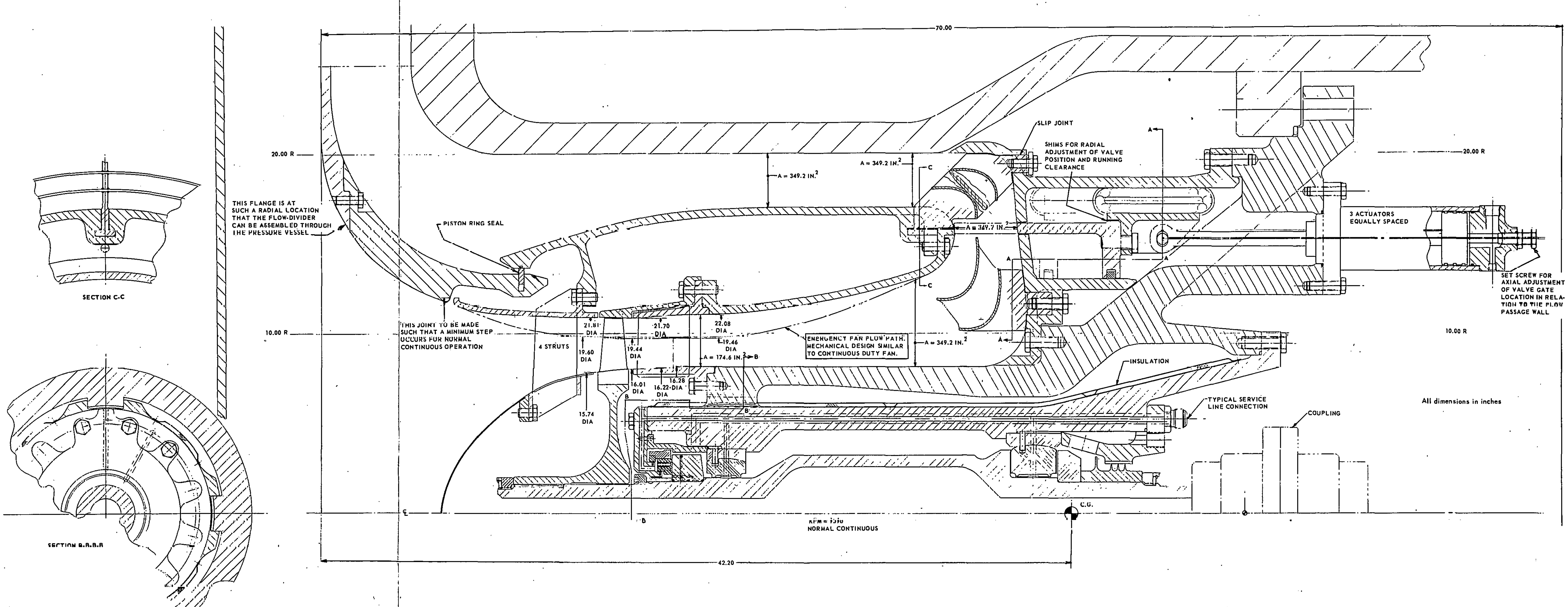

Fig. 8.3 – Main blower configuration

assure leakage into the primary loop, using a close clearance or rubbing type seal to keep the leakage into the primary loop at no more than 0.1 scfm. All seals were designed to accept fresh air leakage rates to atmosphere of up to 100 scfm. Excess air introduced into the primary loop through the seals is bled off to the waste-air handling system. Two seals were recommended for further design and development, and sufficient details were provided for starting a seal evaluation test program.

The seals are divided into two basic sections. Those adjacent to the wheel and primary-air loop are referred to as wheel seals and those on the other end of the shaft are referred to as coupling seals. The space between the two different groups of seals contains the shaft bearings and the oil sump.

One of the seal systems utilizes a wheel seal referred to as a floating-face seal. It is shown in Figure 8.4. Sealing air, which is maintained at a pressure 3 psi greater than primary-loop pressure, is introduced into the cavity behind the seal. It contains two steel piston ring secondaries which ride in carbon bores. The carbon nose piece contains three sets of orifices which vent into three pads on the sealing face. Sealing air is ported directly to the inner annulus on the nose piece, so that it will always leak into the primary loop, thus preventing primary-loop air from leaking out. Pressure, acting on the area between the two secondary diameters, creates a seating force. Air is supplied to the three pads through orifices and across a center dam. Interface pressure will build up until the system is in equilibrium. When this occurs, a very small clearance will exist, and leakage of sealing air both into the primary loop and into the ambient pressure oil sump will be very small. The sealing air which leaks into the oil sump is vented overboard. This type of seal is in the early development stages and if it proves to be successful will be a very low-leakage and long-life seal.

The other seal system proposed uses a more conventional approach and thus less risk. As shown in Figure 8.5, the wheel seal consists of two face seals within a single body. Sealing air maintained at 3 psi above primary-loop pressure is introduced between the seals and leaks into both the primary loop and oil sump. The oil sump is maintained at a pressure 50 psi lower than the sealing air by venting it overboard through a back pres-

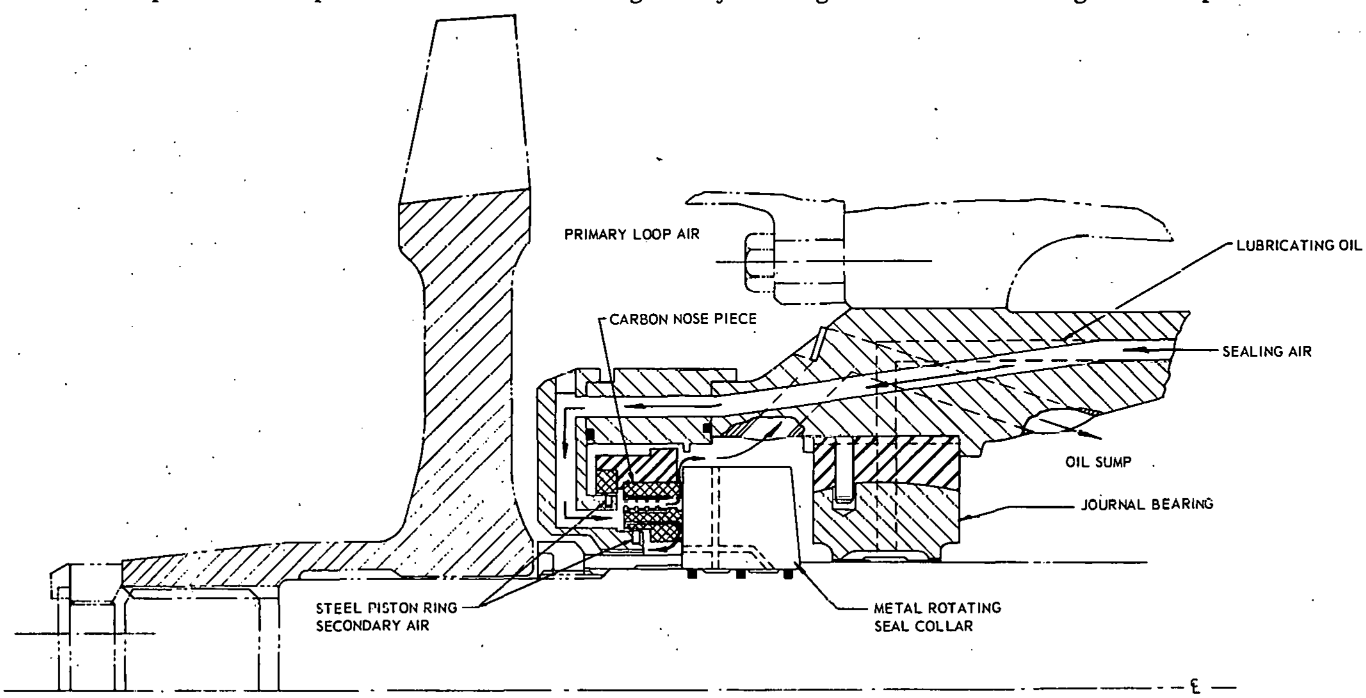

Fig. 8.4 – Blower wheel seal No. 1 (Dwg. SK 109C5532)

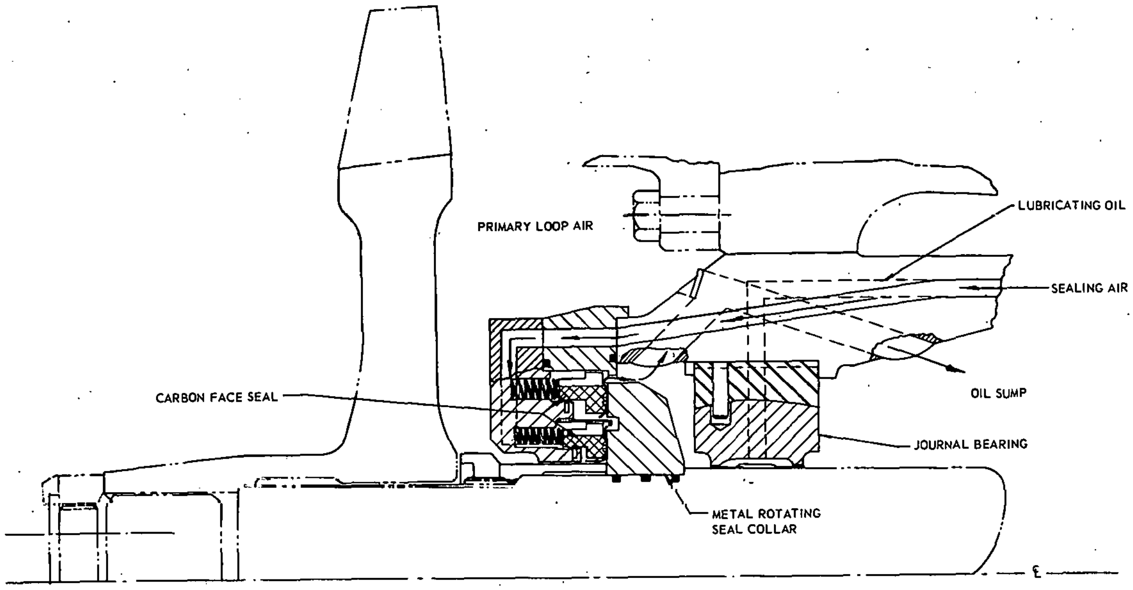

Fig. 8.5 – Blower wheel seal No. 1 (Dwg. SK 109C5533)

sure regulating valve. Maintaining this low pressure drop is necessary in order to use conventional face seals. The coupling side seal uses another face seal across which is also maintained a 50 psi drop into the sump. This is shown in Figure 8. 6. Sealing air is introduced in the cavity between this seal and a labyrinth shaft seal through which about 100 scfm will leak to atmospheric when the sealing air is at 384 psia.

In addition to the work performed as a part of the above described engineering study, an evaluation of various lubrication systems for both the pressurized and ambient pressure oil sump was begun. So far no serious disadvantages in the use of the pressurized sump have been noted. Work was begun on defining a seal development test program based on recommendations of the engineering study.

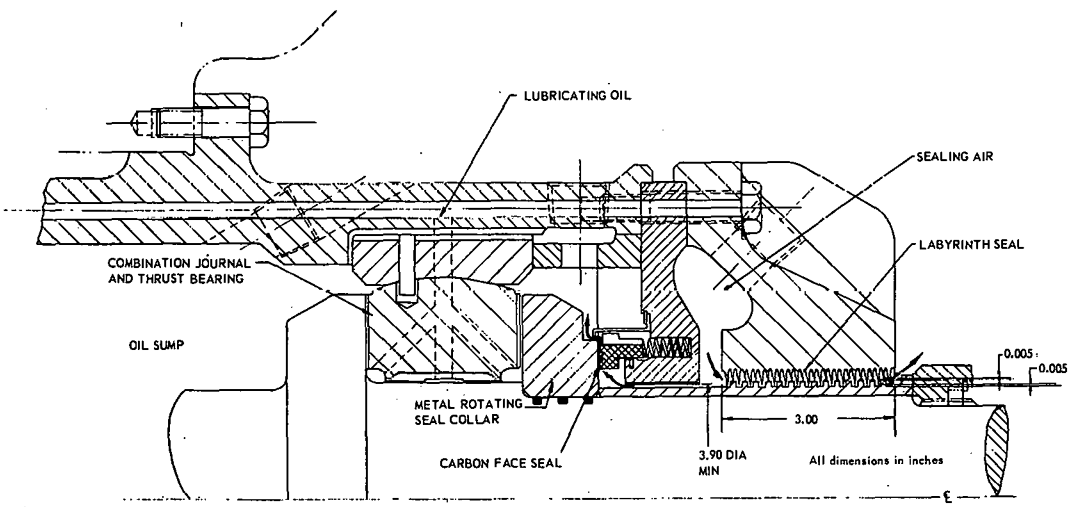

Fig. 8.6 – Coupling seal (Dwg. SK 109C5534)

8.3 WORK PLANNED FOR NEXT PERIOD

Plans for the next reporting period call for designing a seal test rig capable of testing either of the two recommended configurations; beginning negotiations with seal vendors for seal designs, prototype samples, and tests at their plants; and further evaluation of both lubrication systems and purge air system controls.

9. FUEL MATERIALS

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During this report period, studies have been initiated to determine the long-time strength and oxidation resistance of Nb-modified 80Ni - 20Cr fuel sheet cladding, and its ability to accommodate burnup levels up to 25 atomic percent U^{235} at 630A temperature levels. In addition, manufacturing equipment has been reactivated and experimental quantities of fuel sheet have been produced for fuel elements for the 630A critical experiment. This sheet will also be used to investigate improvements of materials leading to long life and reliable performance in 630A fuel elements. This work is described in detail, following a brief presentation of pertinent background information on the fuel elements. Because much of the fuel element development work was accomplished prior to initiation of the 630A program, the work described in this section of the report is not limited to the progress made during the two-month reporting period, but also covers those areas of previously performed work deemed necessary for a complete understanding of the subject matter. Subsequent reporting will be limited to work accomplished during the reporting period.

9.1 BACKGROUND INFORMATION

Concentric ring 80 Nickel - 20 Chromium fuel elements of the type planned for the 630A maritime reactor were developed during the aircraft nuclear propulsion program.* Satisfactory performance in the temperature range from 980° to 1065°C (1800° to 1950°F), for periods of 100 to 500 hours, was virtually assured. Little data was accumulated, however, on performance in the lower temperature range anticipated for the 630A fuel elements (650° to 790°C, 1200° to 1450°F). Furthermore, no data was developed on the properties of the basic fuel sheet nor the performance of fuel elements for operating times in excess of 1000 hours, while a lifetime of 10, 000 to 15, 000 hours is required for the 630A fuel elements.

9.2 IN-PILE TESTING (241)

One of the most important properties of the 80Ni - 20Cr fuel sheet to be determined is its ability to withstand high U^{235} burnup levels. In-pile tests of fuel sheet specimens in both the Oak Ridge Research Reactor (ORR) and the Materials Testing Reactor (MTR) are therefore a vital part of the over-all materials program. Preparation for the in-pile testing of fuel sheet specimens in the ORR began in November 1961 as part of the GE-NMPO High-Temperature Materials Program. A series of progress reports has been prepared detailing this portion of the in-pile test program.†

*R. C. Lever, "Metallic Fuel Element Materials," GE-NMPO, APEX-913, June 20, 1962.

†F. C. Robertshaw and R. K. Betts, "Progress Reports No. 1, 2, 3, and 4 on Long-Term Irradiation Test of 80Ni – 20Cr Fuel Specimen," GE-NMPO, TM 62-1-5, TM 62-1-10, TM 62-3-10, and TM 62-6-13, respectively.

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Plans have been made to conduct long-term, high-burnup tests in the MTR. In addition, a program was initiated to study the effects of neutron irradiation on the 80Ni - 20Cr cladding alloy.

Work performed in these areas is described in the following sections.

9. 2. 1 ORR TESTS

The F-2 facility of the ORR was chosen for the initial in-pile tests of fuel sheet specimens because of its relatively high flux level and its availability. The specimen configuration employed in these tests is illustrated in Figure 9. 1. The capsule design developed for the tests incorporates the specimen as one wall of a venturi tube. Heat generated by fission within the specimen is removed by passing air over the cladding surface exposed within the venturi. Specimen temperature is indicated by thermocouples attached to its unexposed surface, and is controlled by varying the airflow at a given flux positioning. Figure 9. 2 illustrates successive steps in the instrumentation and preliminary assembly of a specimen in the venturi tube. Details of specimen preparation and capsule construction and assembly were described in Progress Reports No. 1, 2, and 3.

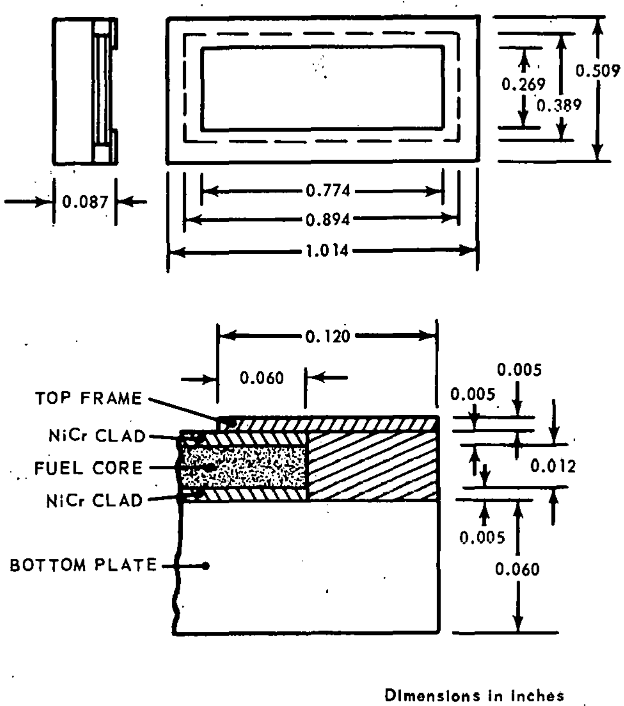
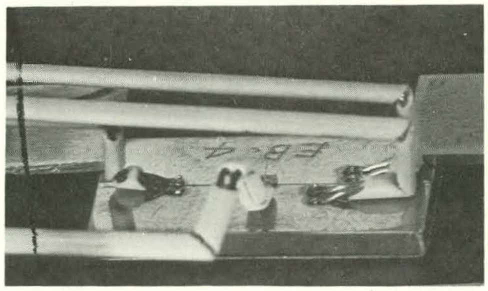


Fig. 9.1 – Plate-type 80Ni – 20Cr fuel element specimen as prepared for in-pile burnup test

Preliminary 100-Hour Test

A preliminary in-pile test was conducted at a peak specimen temperature of 870°C (1600°F) to determine the suitability of the specimen-capsule assembly for the eventual high-burnup test. This preliminary test, which was initiated on February 21, 1962, was designated ORF-1 and incorporated specimen EB-4. Table 9. 1 is a summary of the in-pile test log. In the test, thermocouples designated TC-1 through TC-4 were attached directly to the specimen back-plate. TC-5 and -6 measured proximity air temperature at the back-plate. The remaining thermocouples were attached to the capsule hardware to indicate various air and capsule surface temperatures. The performance of the assem-



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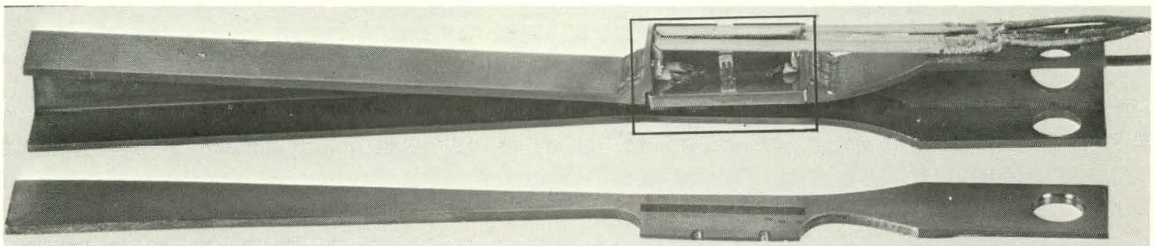

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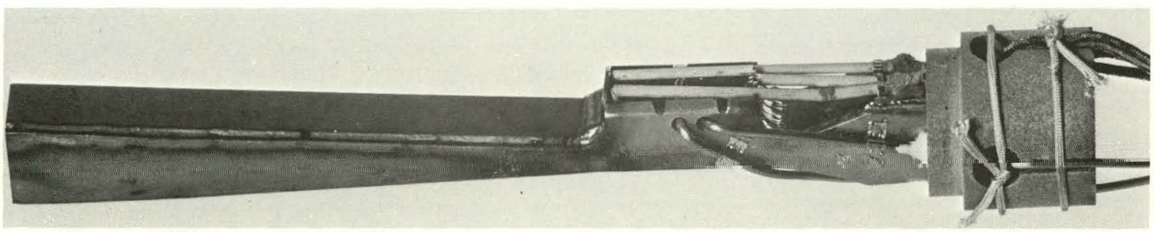

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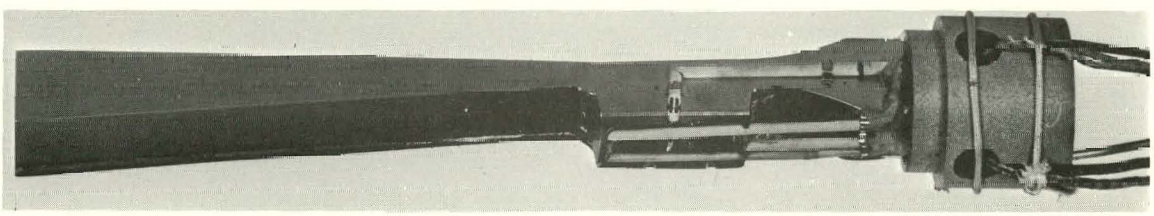

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Fig. 9.2 – Successive steps in the instrumentation and assembly of specimen EB-4 and the venturi

TABLE 9.1

SUMMARY OF IN-PILE TEST LOG OF SPECIMEN EB-4

Accumulated time, hr	0.3	1.7	6.7	17.4	42.1	66.1	100.0
Reactor power, mw	30	30	30	30	30	30	30
Temperature, $^{o}C/^{o}F$							
TC-1	738/1361	746/1375	752/1386	760/1400	768/1414	760/1400	766/1411
TC-2	843/1549	843/1549	860/1580	866/1591	870/1598	868/1594	870/1598
TC-3	790/1454	793/1459	801/1474	810/1490	818/1504	816/1501	821/1510
TC-4	484/903	488/910	496/925	496/925	500/932	504/939	504/939
TC-5	171/340	178/352	181/358	182/360	185/365	182/360	182/360
TC-6	-	-	-	-	-	-	-
TC-7	88/190	88/190	90/194	90/194	90/194	90/194	90/194
TC-8	93/199	93/199	93/199	93/199	93/199	93/199	93/199
TC-9	307/585	310/590	316/601	316/601	321/610	316/601	321/610
Inlet air pressure, psig	74	74	74	74	74	74	74
Exit air pressure, psig	22	25	30	30.5	30.5	29	23
Flow, scfm	36.75	38.50	37.00	37.00	35.00	37.00	37.00
Exit air radiation	Background throughout test						

bly during the in-pile test was excellent. All thermocouple chart readings were uniform throughout the test and no gross aberrations were observed in air pressures. There were no fission product bursts, as indicated by continuous background counts on exit-air irradiation monitors, and no reactor scrams during the test.

Figure 9.3 illustrates the appearance of the specimen after irradiation. Specimen integrity remained excellent. The microstructure of a section cut from the specimen following irradiation is shown in Figure 9.4. The structure remained sound and is quite representative of the structure prior to test.

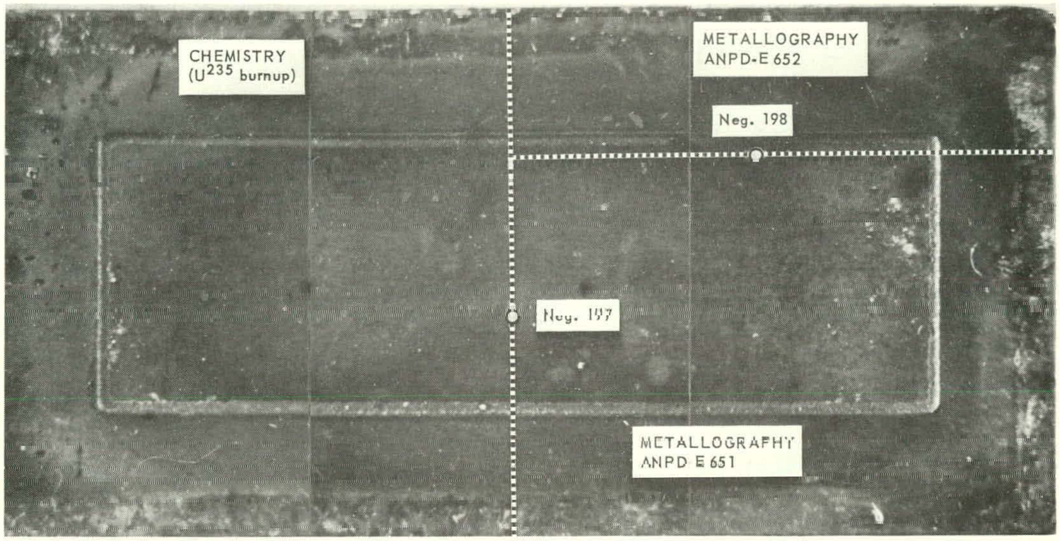
Fuel burnup analysis was performed on a sample consisting of approximately one-half of the specimen. Due to the highly radioactive condition of the piece, a chemical determination of the amount of U^{235} remaining was not performed. Rather, the burnup was determined by the difference between the pre-test fuel content and post-test fission product analysis. The initial fuel content was calculated from the specimen dimensions and pre-test data on fuel density. Allowing for tolerances in dimensioning and analysis, the U^{235} burnup was determined to be 0.98 ± 0.1 atomic percent by gamma count of cesium-137, and 1.05 ± 0.1 atomic percent by beta count of cerium-144. Thus, a burnup of approximately 1 atomic percent was experienced in 100 hours at the ORF-1 flux level.

2200-Hour Test

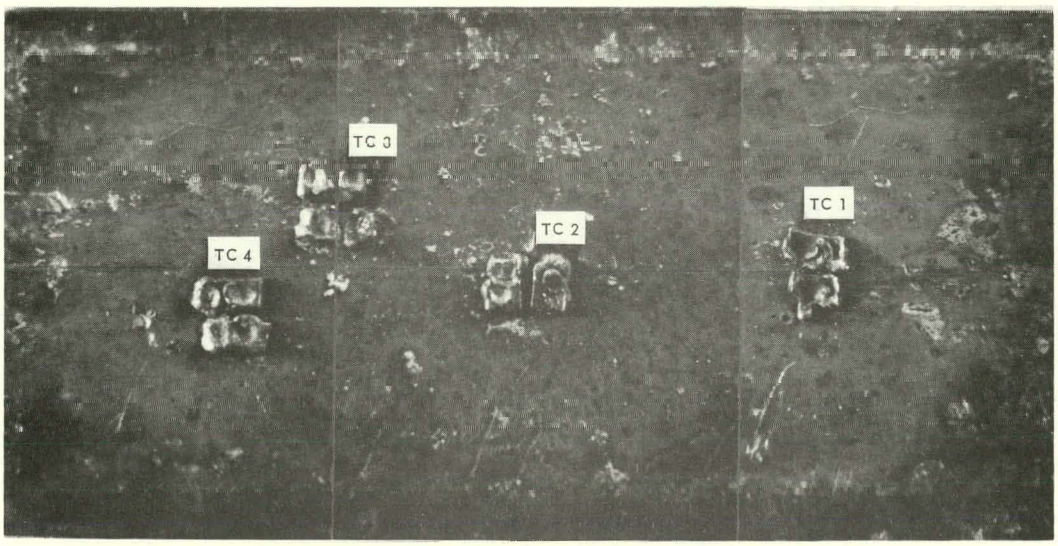
The high-burnup test (ORF-2) was initiated in the F-2 facility on April 16, 1962. The specimen utilized was designated EB-13, and the capsule assembly (referred to in Progress Report No. 4) was quite similar to that employed in the preliminary test. The thermocouple instrumentation was modified, principally by attaching a fifth thermocouple to the specimen, and by including additional hardware thermocouples.

The maximum specimen operating temperature was maintained at approximately $815^{o}C$ ($1500^{o}F$) because the estimated maximum 630A fuel element temperatures had been reduced. To date, the specimen capsule assembly has accumulated a total of 641 hours at temperature.

A summary of the in-pile test log is presented in Table 9.2. In the table, thermocouples designated TC-1 through TC-5 are attached to the specimen. The other thermocouples measured temperatures at various points on the capsule hardware. Activity level in the effluent was monitored by radiation detectors as in the preliminary test. In addition,

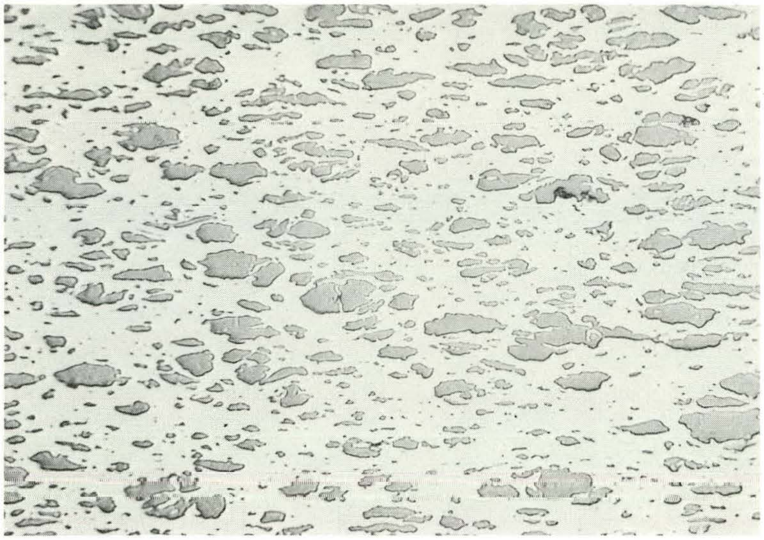


A. Cladding surface of specimen EB-4. No gross cracks, blisters, or pits are evident. Shows sectioning lines for chemical or metallographic specimens, and also the negative numbers and locations of cross-section photomicrographs shown in Figure 9.4. (RML 1865, 1866, 1867).



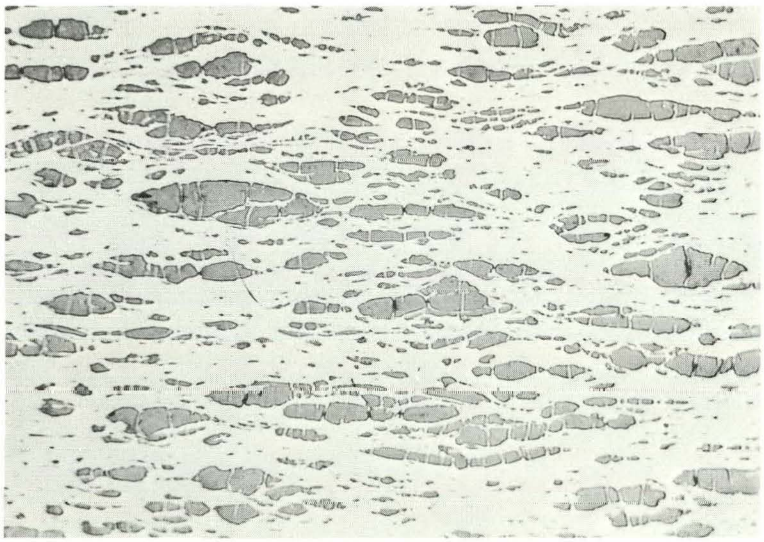
B. Back-plate surface. (RML-1862, 1863, 1864).

Fig. 9.3 – Cladding and back-plate surfaces of specimen EB-4 after 100-hour in-pile test at 870°C



Neg. 197 250 X

A. Microstructure of hottest portion of specimen as indicated by thermocouple No. 2. Section is transverse to rolling direction of fuel sheet. ANPD-E 651



Neg. 198 250 X

B. Microstructures parallel to rolling direction of fuel sheet. Area is out of hot zone and temperature was less than that indicated by thermocouple No. 1. ANPD-E 652

Fig. 9.4 – Photomicrographs of specimen EB-4 after 100-hour in-pile test at 870°C. Refer to Figure 9.3 for locations where photos were taken.

TABLE 9.2

TYPICAL OPERATING DATA TO DATE OF SPECIMEN EB-13 DURING HIGH BURNUP TEST

Time at temperature, hr	0.5	18.2	216.2	228.3	293.8	381.0	407.5	569.9	641.0
Reactor power, mw	30	30	30	30	30	30	30	30	30
Temperature, °C/°F									
TC-1	707/1305	724/1335	693/1279	693/1279	701/1294	688/1270	701/1294	704/1299	701/1294
TC-2	805/1481	827/1521	813/1495	807/1485	813/1495	801/1474	815/1499	813/1495	815/1499
TC-3	798/1468	818/1504	807/1485	801/1474	807/1485	801/1474	815/1499	810/1490	815/1499
TC-4	601/1114	629/1164	624/1155	622/1152	624/1155	610/1130	540/1004	616/1141	518/964
TC-5	715/1319	751/1384	743/1369	738/1360	740/1364	724/1335	738/1360	746/1375	746/1375
TC-6	354/669	346/655	301/574	335/635	332/630	327/621	338/640	377/711	368/694
TC-7	365/689	429/804	418/784	401/754	410/770	401/754	418/784	446/835	438/820
TC-8	240/464	296/565	268/514	285/545	285/545	277/531	293/559	342/648	332/630
TC-9	412/774	443/829	438/820	435/815	438/820	429/804	560/1040	438/820	599/1110
TC-10	149/300	163/325	160/320	154/309	154/309	149/297	152/306	154/309	157/315
TC-11	74/165	74/165	74/165	71/160	74/165	74/165	74/165	66/151	66/151
TC-12	77/171	79/175	79/175	Open	-	-	-	-	-
TC-13	46/115	46/115	46/115	54/129	57/135	57/135	57/135	49/120	49/120
TC-14	46/115	46/115	49/120	46/115	41/106	49/120	49/120	46/115	46/115
Inlet air pressure, psig	74	74	74	75	75	75	75	74	74
Exit air pressure, psig	-	56	43	55	52	-	47	55	52
Airflow, scfm	30.5	25.7	30.6	27	27	28.3	29.4	25.7	27
Exit air radiation				Background throughout test to date					
Trap No.		1	2	3					
Time on, hr		23.8	7.8	12.0					
Fission products		Nil	Nil	Nil					

charcoal traps have been inserted periodically in the system to measure fission product escape. The results of trap counting experiments are recorded in Table 9.2.

Figure 9.5 is a plot of principal thermocouple data as a function of time, and also indicates the intervals of reactor shutdown and insertion of other experiments in the F-2 facility.

Thermal Cycle Test

In laboratory bench tests, specimens identical to those used in the in-pile tests have been subjected to thermal cycling in convection air. The specimens are heated internally by their own electrical resistance. Thermal cycling is achieved by interrupting the current, which results in rapid cooling to room temperature. Several specimens were subjected to various short-time tests to verify the suitability of the EB-type configuration for in-pile thermal conditions. Then a fueled specimen, EB-6, was placed on test to determine the long-time oxidation resistance and thermal cycling characteristics for a period of at least 2200 hours at 870°C (1600°F), corresponding to the long-term in-pile test ORF-2. In rigorous inspection after 2200 hours and 21 thermal cycles, no increase in alpha count was detected and no gross external damage to the cladding was observed. There were no indications of voids, cracks, or other detectable internal effects by radiographic inspection. Thermal etching of the braze fillet was observed in the first 300 hours of testing but this condition has not worsened with subsequent testing.

The specimen was returned to test and has accumulated over 2700 hours. It is planned to continue the test to maximum endurance of this type of fuel sheet specimen.

9.2.2 PLANNED MTR TEST

The testing of an 80Ni - 20Cr fuel ring specimen in the A-19 facility of the MTR is planned for early September 1962. The test is designated 3-F-1. The specimen will be approximately 0.9 inch in inside diameter, 3 inches long, and 0.022 inch thick. It will contain 37 weight percent of fully-enriched UO_2 in the core. The purpose of the test will be to determine the performance of the fuel specimen under conditions producing burnup levels up to 25 atomic percent of the U^{235}. The power density in the MTR test will be lower than in the ORR test, thus simulating more closely the 630A operation. Maximum

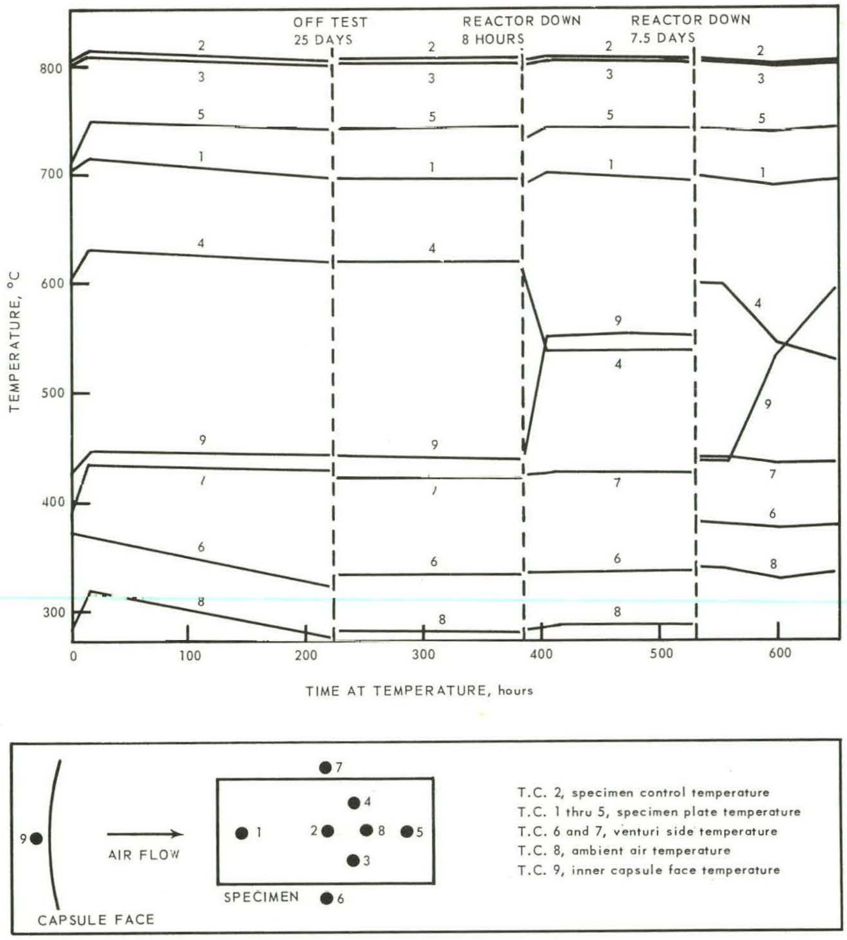


Fig. 9.5 – Temperature distribution on specimen EB-13 during high burnup experiment ORF-2

specimen temperature will be maintained at 790°C (1450°F). Since the A-19 facility is in a reflector position having a peak flux of 2.5 x 10^{13} nvt, about 5000 hours will be necessary to achieve a burnup level of 25 atomic percent.

Capsule Design

The tentative capsule design is illustrated in Figure 9.6. Fabrication of the capsule and related hardware has been initiated. No attempt has been made to design out the power scalloping inherent in the A-19 facility. This decision was based in part on the lower cost of an unmodified capsule and also on the increased positioning flexibility afforded by such a capsule. In addition, the anticipated temperature differential of approximately 150°C (300°F) is not expected to seriously compromise the performance of the specimen at a maximum temperature of 790°C (1450°F).

As in the ORR tests, air will be the cooling medium. Tentatively, only the internal surface of the specimen will be cooled. Specimen temperatures will be measured on the external surface by both attached and surface-contact thermocouples.

Continuous effluent monitoring will be conducted in or around the main effluent line. Provisions will also be made for sampling the effluent so that the presence of fission products can be confirmed in the event that high activity is detected.

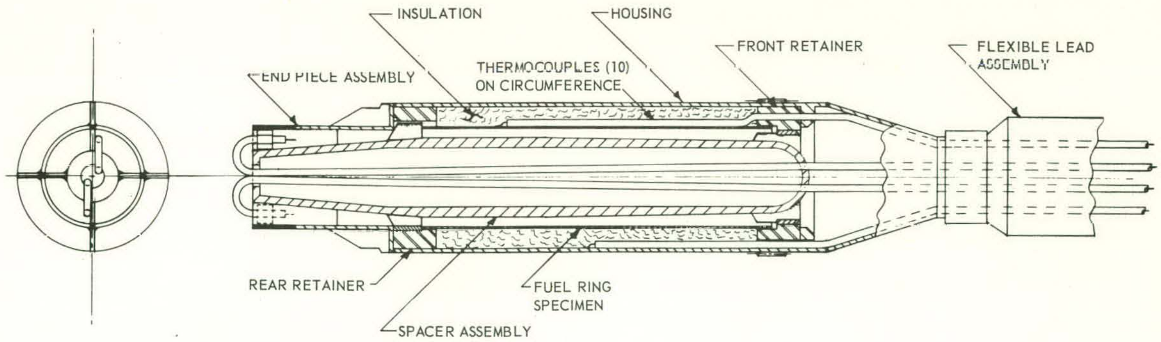


Fig. 9.6 – Final assembly, 3F1 cartridge (Dwg. 109C5675)

Fuel Ring Preparation

Specimens for the MTR test are being prepared from 80Ni - 20Cr fuel rings previously manufactured under the nuclear aircraft program. The rings were made from 0. 022-inch-thick warm-finished fuel sheet which contains 37 weight percent of fully-enriched UO_2 in the core. The rings are 1. 65 inches in diameter and 3. 090 inches long.

In preparing the MTR test rings, a section of each original ring including the joint strap is cut away, leaving a segment of proper arc length to be re-rolled into a 0. 9-inch diameter test ring. Figure 9. 7 shows one of these rings before and after sectioning. The cut edges of the segments are then sealed by leaching out the exposed UO_2, tack-welding a 0. 014-inch D-shaped 80Ni - 20Cr wire along the cut edge, and then brazing the wire in place with GE-81 braze alloy. The sealed segments are then hot-formed into rings on a 5/8-inch powered roll-former.

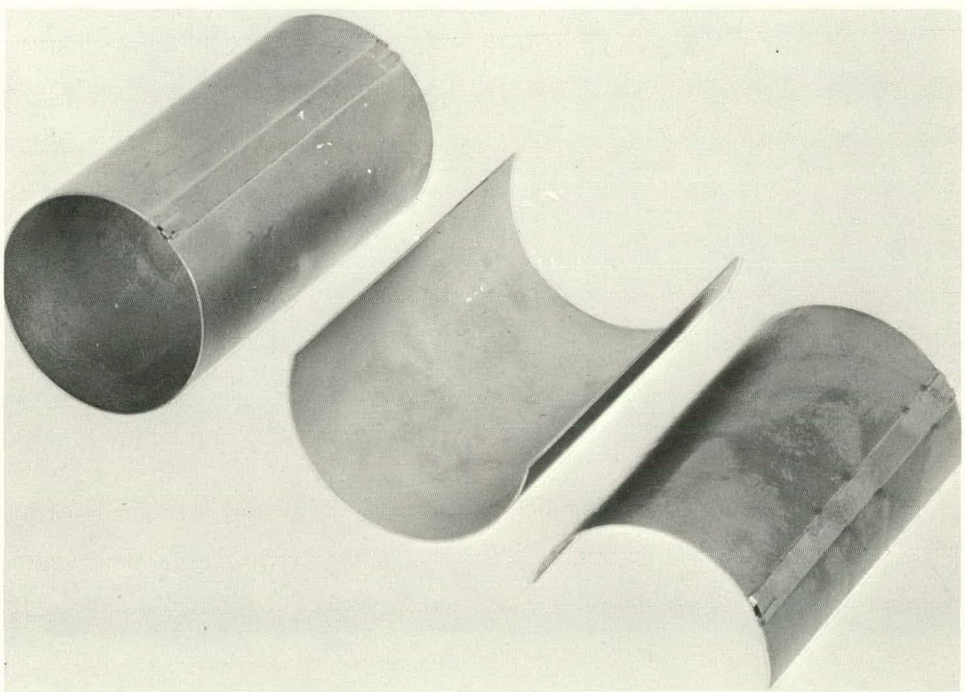


Fig. 9.7 – Typical 80Ni – 20Cr fuel ring prior to sectioning. Center segment is ready for edge sealing and roll forming (Neg. P62-6-17F)

Several ring-joining brazing techniques are being evaluated. Previous experience has indicated that a minimum of braze is desirable from the standpoint of oxidation resistance and ductility. In addition, the use of excessive tack-welding is to be avoided, particularly over the core, because tack welds are a frequent source of defects. Three ring joining methods are being evaluated for comparison with the technique devised under the previous program. The purpose is to produce a simple joint capable of maintaining the ring configuration and sealing the core beyond the useful life of the fuel sheet itself. The four techniques are illustrated in Figures 9.8 through 9.13.

The ring joining method previously used, designated the P-103 method, is illustrated in Figure 9.8. The external joint plate is tack-welded into place by means of the short tabs. After application of GE-81 braze slurry to the long groove between the ring edges, the ring is brazed in hydrogen at 1175°C (2150°F). The internal joint plate is then tack-welded into place, braze is applied to the sides of the joint plate, and the ring is given a second brazing operation. Excessive tack welds and excessive braze are the disadvantages of this method.

Method A, which is a modification of the P-103 method, is shown in Figure 9.9. The sequence of operations is similar except that both inner and outer joint straps are full length and are tack-welded at the ends only over the dead edge.

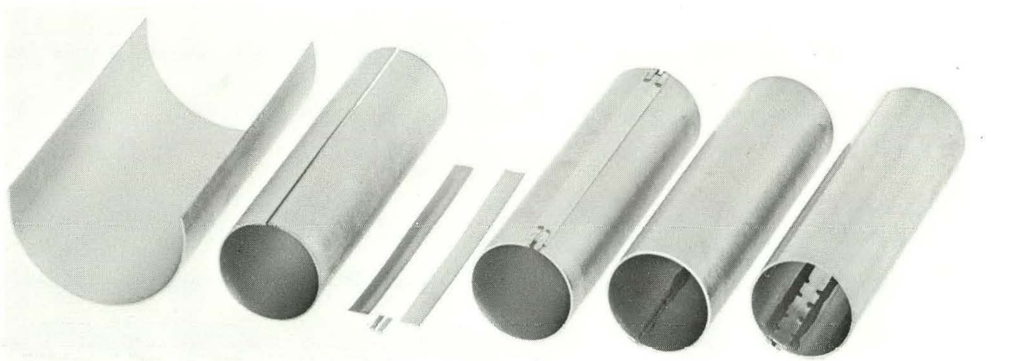


Fig. 9.8 – P-103 method of ring joining showing joint plate hardware and location of braze application (Neg. P62-6-17G)

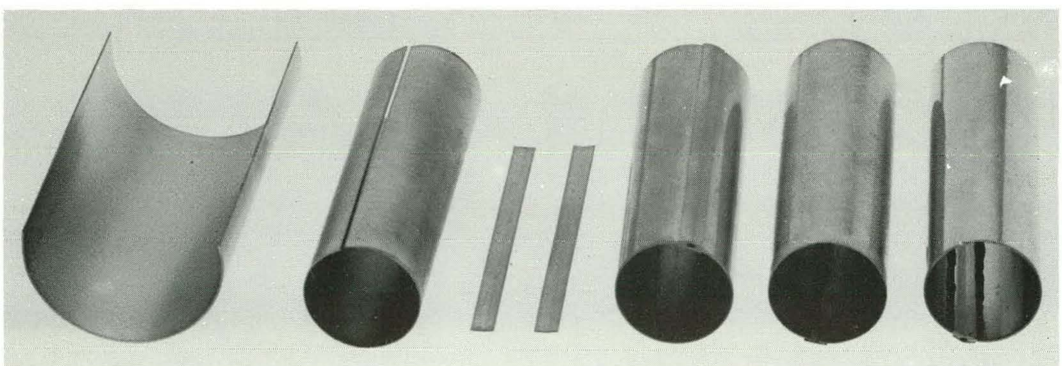


Fig. 9.9 – Ring joining by Method A, showing location of braze application (Neg. P62-6-33C)

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Method B is illustrated in Figure 9.10. A machined H-joint member is used in this method. Due to the close tolerance fit of the fuel sheet in the H-member channels, edge sealing consists of applying only braze alloy over the exposed core. No wire is used. Following edge sealing, the H-member is fitted into place, braze alloy is applied between the overhanging tabs which act as reservoirs, and the brazing operation is performed. Following brazing, the reservoirs are trimmed off.

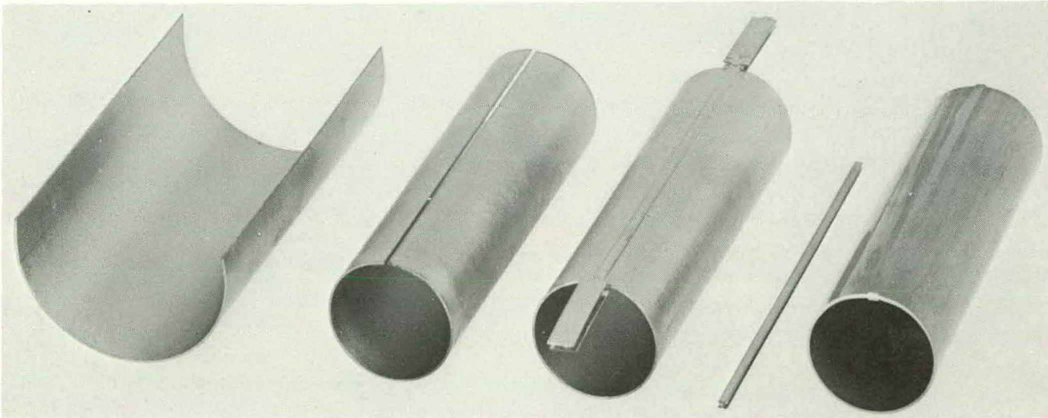

Fig. 9.10 – Ring joining by Method B. Braze slurry is placed between the overhanging tabs which are trimmed off after brazing (Neg. P62-6-17A)

Method C, currently favored for use in preparing the MTR ring specimen, is illustrated in Figure 9.11. This method consists of using joint straps that extend about 1 inch beyond the edges of the ring. The straps are press-formed to approximate the ring curvatures, and also curved lengthwise for good contact with the fueled ring. They are tack-welded only at the ends of the fuel sheet dead-edge. The extensions form a slot or reservoir into which an excess quantity of braze slurry is injected. During brazing, the liquid braze alloy is drawn from the reservoir by capillary action along the joint straps and fills the interface between the straps and the cladding surfaces. By this technique, adequate braze is introduced into the joint without the disadvantage of an excess of braze flowing over the cladding surfaces external to the joint. The braze reservoir is trimmed off after brazing.

This method is favored because: (1) no tack welding is done over the fueled core, (2) excess braze is eliminated from the vicinity of the joint, and (3) ring joining is accomplished in one brazing cycle.

Figures 9.12 and 9.13 illustrate completed ring specimens made by each of the four methods.

The 815^{o}C (1500^{o}F) tensile strength of jointed specimens prepared by the four methods are to be determined. Test specimens are being prepared from flat, 0.022-inch-thick unfueled 80Ni - 20Cr sheets, as illustrated in Figure 9.14.

<u>Planned Thermal Cycle Testing</u>

Thermal cycle testing is to be conducted on fueled ring specimens. Aerothermal conditions of the in-pile test will be simulated by inductively heating the ring as cooling air passes through the bore. The specimen will be subjected to periodic cycling to ambient temperature.

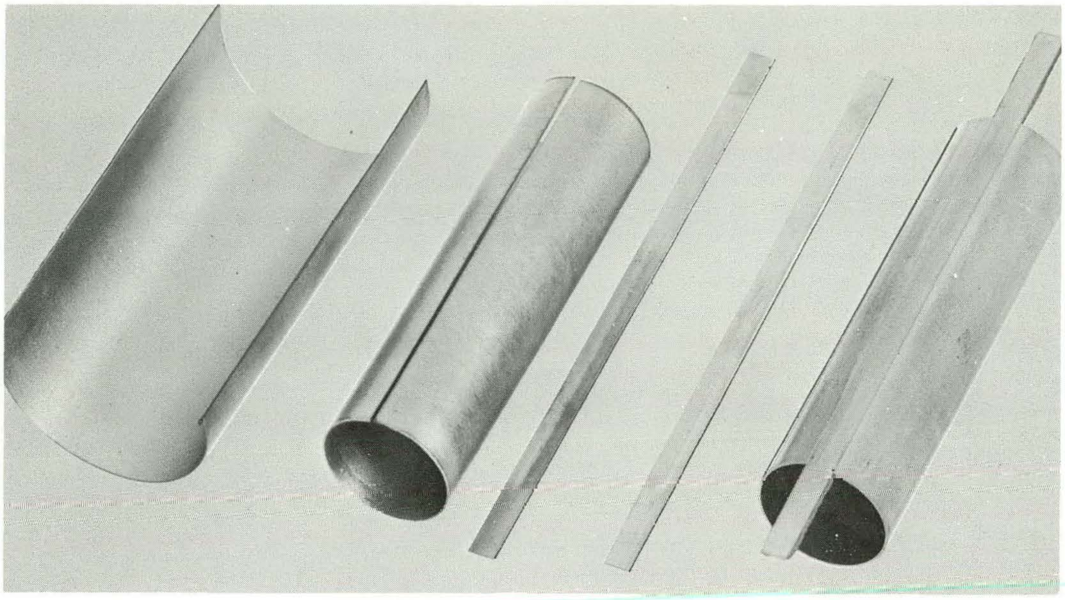

Fig. 9.11 – Ring joining by Method C. Braze slurry is placed between the extended portions of the joint straps which are trimmed off after brazing (Neg. P62-6-17C)

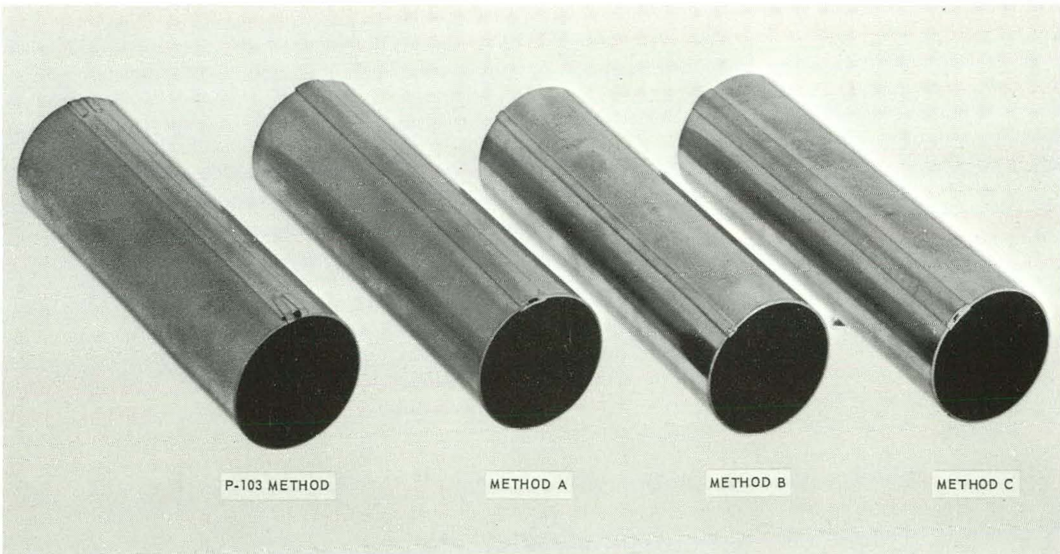

Fig. 9.12 – Samples of the four techniques of ring joining (Neg. P62-6-33a)

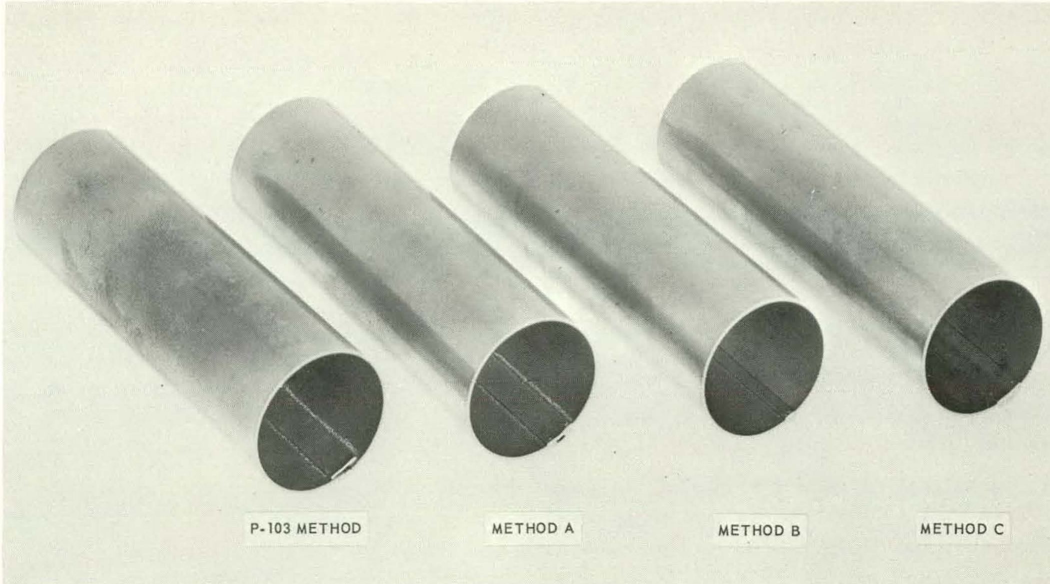

Fig. 9.13 – Samples of the four techniques of ring joining showing the excess braze associated with the P103 Method and Method A. (Neg. P62-6-33B)

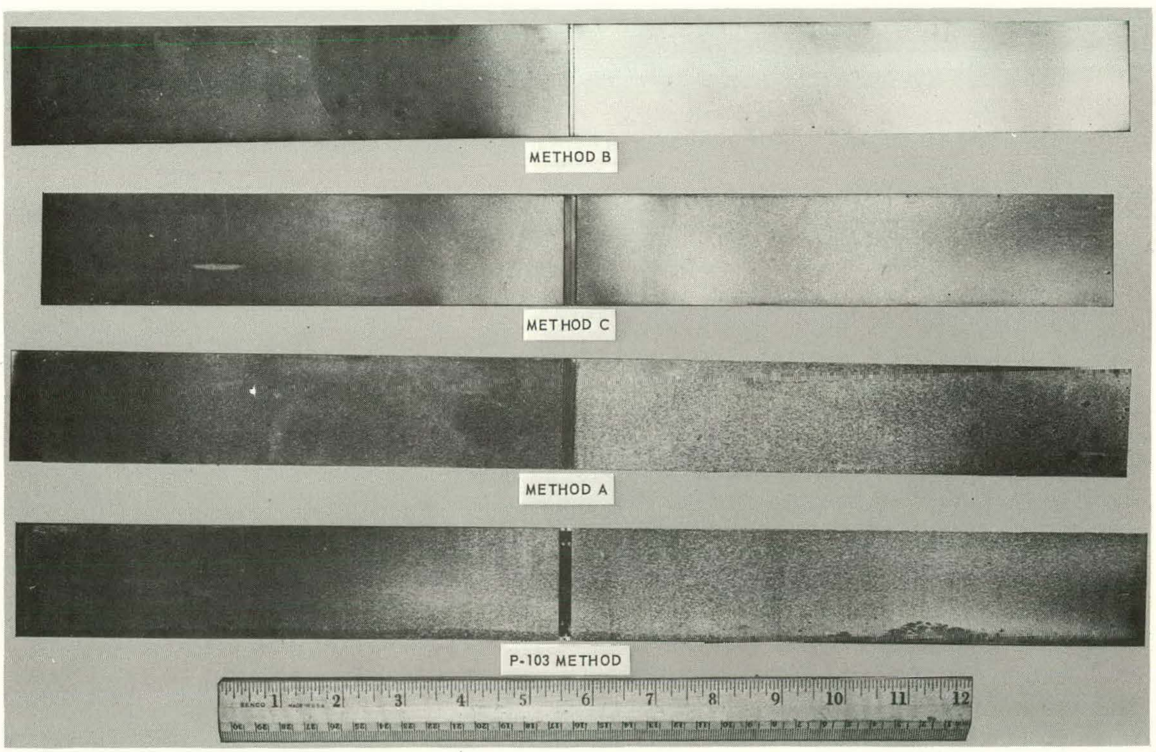

Fig. 9.14 – Tensile specimens incorporating brazed joints made by the four ring joining techniques (Neg. P62-6-33D)

An unfueled specimen of 80Ni - 20Cr cladding alloy is being instrumented with thermocouples, using various techniques for attachment. This specimen will be tested by the method described above to permit selection of instrumentation techniques and thermocouple locations to be used on the in-pile test specimen.

9.2.3 EFFECTS OF RADIATION ON 80Ni - 20Cr ALLOY

Specimens are being prepared to establish the effects of neutron irradiation on the tensile and stress rupture properties of the Nb-modified 80Ni - 20Cr alloy.

Specimen Preparation and Irradiation Plans

Specimens of the type shown in Figure 9.15 have been prepared. The specimens were made from 3/4 inch diameter bar stock of the 80Ni - 20Cr alloy of the analysis shown in Table 9.3. The bars were annealed at 1065^{o}C (1950^{o}F) for 1/2 hour, air-cooled, and then cold-swaged to a diameter of 0.450 inch. The specimens were also annealed after each 0.100-inch reduction. Specimen segments were then cut from this swaged bar stock and given a solution treatment at 1180^{o}C (2150^{o}F) for 20 minutes, followed by air cooling. A total of 40 smooth tensile specimens were prepared from this material, using low-stress grinding procedures in finishing to final dimensions. The completed specimens were examined radiographically, inspected by fluorescent penetrant techniques, and visually inspected for dimensions and surface imperfections. Twenty of the specimens have been placed in capsule MT-74, the balance being retained for unirradiated control tests at GE-NMPO.

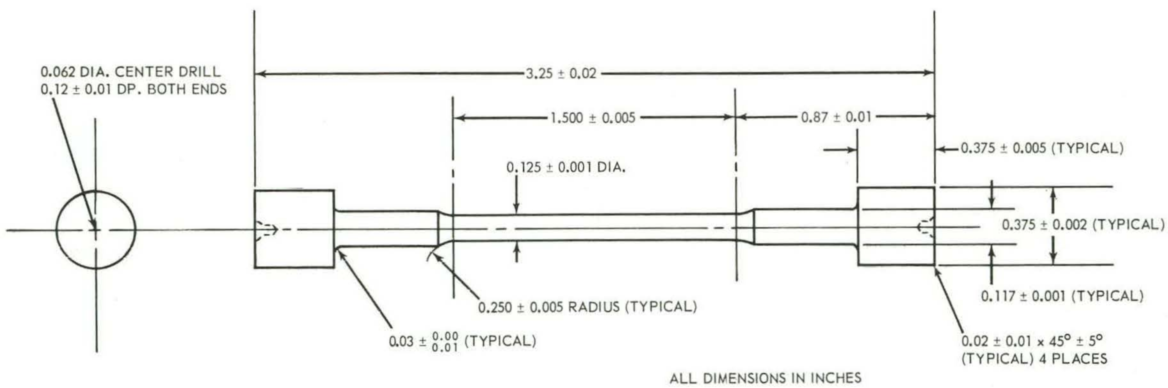

Fig. 9.15 – Configuration of specimens for tensile and stress-rupture testing (Dwg. 119B3476A)

Capsule MT-74 is awaiting irradiation at pile ambient temperature in the L-6 facility of the Engineering Test Reactor (ETR). The specimens are contained in four of the six open X-baskets comprising the capsule. Specimens for another experiment are contained in the other two locations.

The X-basket capsule design consists of two aluminum end caps held together by a 0.250-inch aluminum rod. The outside diameter of the end caps is 1.125 inches. The aluminum rod is threaded on each end to contain the specimens between the end caps. Each X-basket contains five specimens. Three of the six baskets in MT-74 will be inserted in the ETR L-6 northwest hole and the remaining three in the ETR L-6 southwest hole.

Capsule MT-74 was scheduled for irradiation during ETR cycle 47, which began on July 6, 1962.

TABLE 9.3

CHEMICAL ANALYSIS OF
Nb-MODIFIED 80Ni - 20Cr TEST SPECIMENS

Element	%	Element	%
Ni	77.81	P	0.001
Cr	19.53	Ca	0.01
Nb	1.21	Zr	0.06
Fe	0.72	Pb	<0.002
Mn	0.05	Cu	<0.04
Si	0.79	Ta	<0.01
C	0.030	W	<0.01
Al	0.08	B	<0.002
Co	0.02	Ti	<0.02
S	0.004	Hf	<0.02

Irradiation will be in water at pile ambient temperature. Fast neutron dosage is expected to be about 10^{19} n/cm^2. Dosimetry will be based upon nickel-cobalt wire flux monitors attached to the center bolt of each X-basket and on activation analysis of selected specimens following post-irradiation testing. Post-irradiation testing will be performed at the Idaho Test Station.

Irradiation of an additional 20 specimens is planned for early Fiscal Year 1963, in a reactor facility to be selected. This irradiation will be conducted at 815°C (1500°F) for at least four reactor cycles to achieve a dosage level equivalent to that anticipated during the full operational life of fuel elements in the 630A power plant.

Test Results

Post-irradiation testing of the specimens in capsule MT-74 will be conducted at temperatures ranging from 650° to 815°C (1200° to 1500°F). Control tests at 650°C (1200°F) have already been initiated. Testing is performed in air in conventional stress-rupture equipment. Special care is taken to insure accurate specimen alignment. The data obtained to date is tabulated in Table 9.4 and plotted in Figure 9.16.

9.3 PROPERTY STUDIES (242)

Testing has been initiated to determine the creep resistance of 80Ni - 20Cr fuel sheet for times up to 15,000 hours or even longer. The tests are being conducted at temperatures from 700° to 870°C (1300° to 1600°F) and at stress levels of 300 and 600 psi, under isothermal and thermal cyclic conditions. The fuel sheet used for testing is from stock retained from the previous aircraft reactor program. This work has been described in detail in a memorandum report.*

Exploratory creep tests were initiated late in 1961 on the 80Ni - 20Cr cladding alloy at temperature and stress levels which were then anticipated for 630A peak conditions. These tests are continuing.

The fuel sheet employed in creep testing was prepared late in the aircraft reactor program and was warm-finished. The sheet is available in the form of rings, 3 to 3.5 inches in diameter. The sheet is 0.022 inch thick by 3.37 inches wide, and contains 37 weight percent of partially enriched UO_2 in the core.

*R. K. Betts, "630A Materials Program – Memorandum Report No. 1 – Creep Testing of 80Ni – 20Cr Fuel Sheet," GE-NMPO, TM62-6-8, June 15, 1962.

TABLE 9.4

STRESS-RUPTURE TEST RESULTS OF Nb-MODIFIED 80Ni - 20Cr ALLOY AT 650°C (1200°F)

Specimen No.	Stress, psi	Rupture Time, hr	Elongation In 1.5 Inches, %	Reduction Of Area, %	Fracture Distance From Near Fillet, in.
21N	30,000	108.4	51.6	44.9	1/8
22N	33,000	73.2	24.2	50.7	1/8
23N	35,000	57.8	30.6	48.7	1/8
24N	30,000	149.3	52.2	42.3	3/4
25N	35,000	62.6	44.7	50.4	1/8

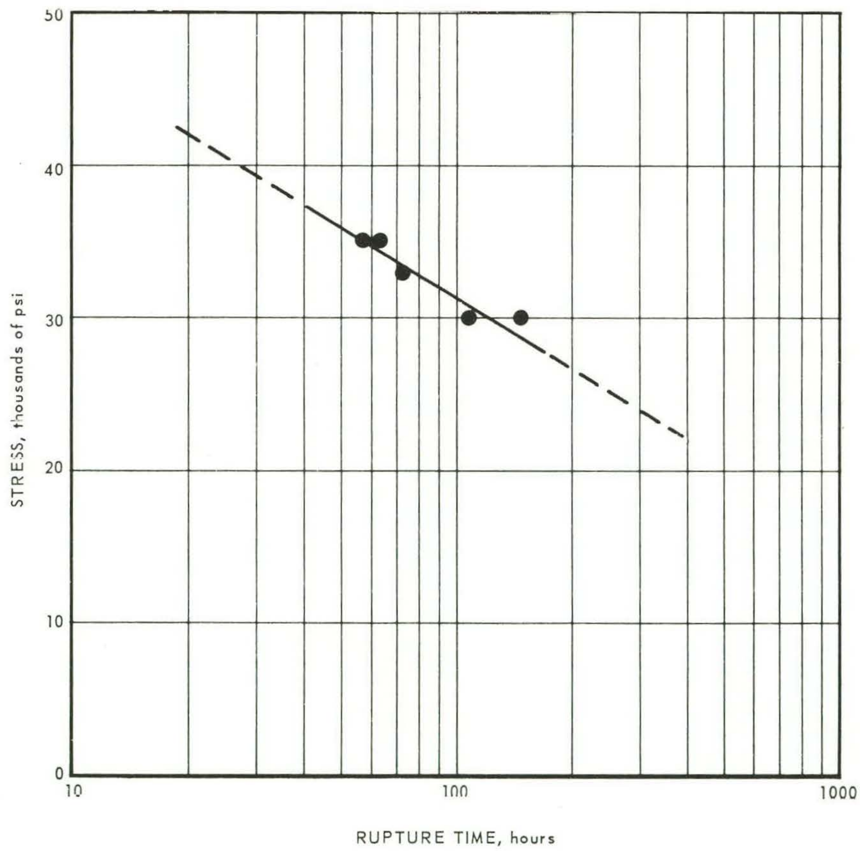

Fig. 9.16 – Stress rupture strength of Nb-modified 80Ni – 20Cr alloy at 650°C

In making specimens, the rings were cut apart through the joint plate gap, and unrolled in several passes from a furnace temperature of 980°C (1800°F). After careful descaling and inspection, specimens were sheared from the flattened fuel sheet as strips 1.25 inches wide and 8.5 inches long. None of the original dead edge was retained.

The edges were sealed by techniques similar to those used in preparing specimens for ORF-1 and ORF-2. A sketch of the configuration appears in Figure 9.17. Following brazing, the specimens were diffusion-treated at 1040°C (1900°F) for 24 hours in hydrogen to promote diffusion of silicon from the braze alloy. This homogenizes the brazed area, lessens its inherent brittleness, and improves its oxidation resistance.

Extension strips for suspending the specimens were made from 0.020-inch-thick 80Ni - 20Cr cladding sheet. The strips were resistance spot-welded in place at the ends of the fuel sheet specimen. Figure 9.18 illustrates typical specimens ready for creep testing.

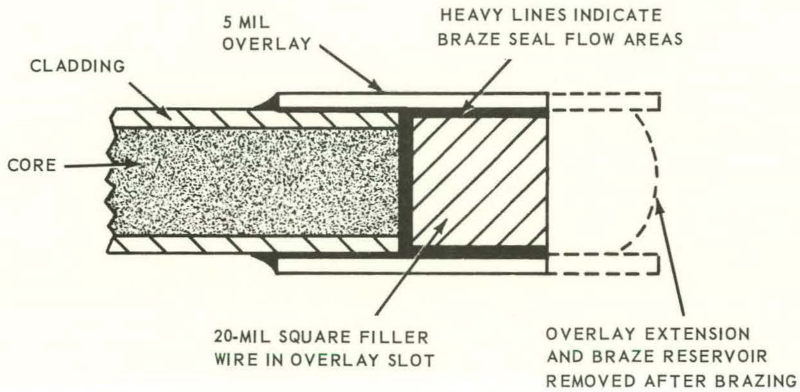

Fig. 9.17 – Method of edge-sealing fuel sheet specimens for creep tests

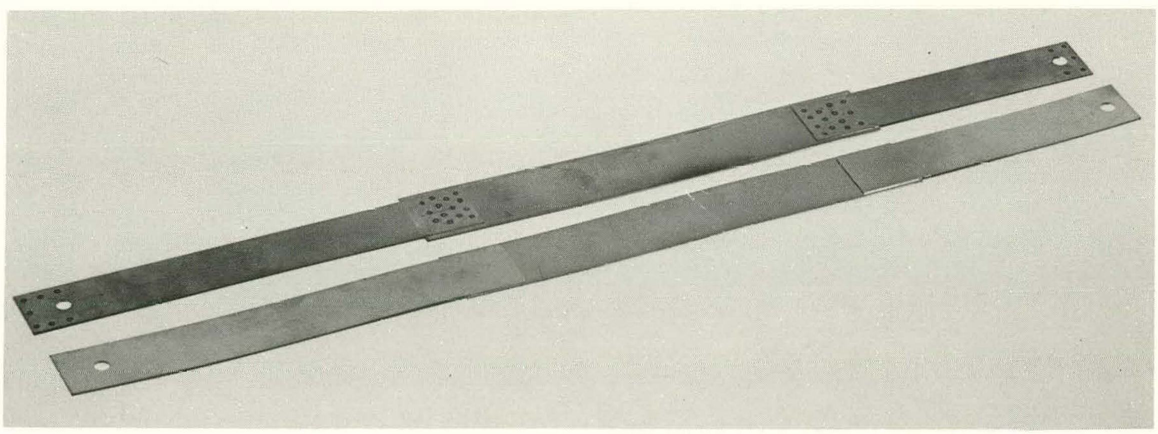

Fig. 9.18 – Two specimens prepared for creep testing. Rear specimen shows location of spot welds attaching straps to fuel strip. Ends of straps are also joined by spot welding (Neg. P62-6-171)

9.3.1 CREEP TESTING

Isothermal Tests

Testing is being conducted in facilities as shown in Figures 9.19 and 9.20. The cabinets are evacuated through "CWS" filtering systems to control spread of U^{235} in the event of specimen rupture. Specimen temperature in each furnace is indicated and controlled from a center-positioned platinum/platinum-rhodium thermocouple close to, but not contacting, the specimen. Because of the limited number of stands available, provisions have been made for parallel-mounting two specimens with separate weights in each furnace.

During the first 500 test hours, each specimen is removed at approximate 100-hour intervals, for X-ray, visual inspection, alpha counting, and elongation measurements. The specimens are then similarly checked at about 750 and 1000 hours. Thereafter, inspection will occur at 500-hour intervals for the duration of the test. Tests are to be conducted to 5 percent elongation or for at least 15,000 hours, whichever occurs first.

Table 9.5 summarizes the results obtained to date on the fuel sheet specimens. None of the specimens has developed any gross surface or internal defects, nor is there any evidence of increased surface radioactivity levels.

Thermal Cycle Tests

Several specimens are being tested under thermal cycle conditions. The specimen configuration and test stands are the same as those used in isothermal tests except that the stands are equipped for thermal cycling by means of an air blast directed through the furnace tube. In the thermal cycle tests, each specimen is exposed to air-blast cooling to about 200°C (400°F) at intervals of about 50 hours throughout the test.

TABLE 9.5

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	1713	-	nil
C-2	Brazed Channel	790/1450	300	1466	-	nil
C-4	Brazed Channel	700/1300	300	1175	-	nil
C-8	Brazed Channel	790/1450	600	1197	-	nil
B-2	Brazed Wire	(In with C-4)		70	-	nil
W-2	Welded Insert	(In with C-8)		0	-	-
Thermal Cycle Tests						
C-3	Brazed Channel	870/1600	300	1497	24	nil
C-5	Brazed Channel	700/1300	300	1200	21	nil
C-6	Brazed Channel	790/1450	600	1091	20	nil
C-7	Brazed Channel	790/1450	300	1165	21	nil
B-1	Brazed Wire	(In with C-5)		475	9	nil
W-1	Welded Insert	(In with C-6)		40	0	-

Table 9.5 also details the results obtained to date in thermal cycle tests. Again, none of the specimens has exhibited any serious surface or internal defects, or any indication of excessive surface radioactivity.

Stress Distribution

In the isothermal and thermal-cycle creep testing described above, the stresses are applied by subjecting the specimen to a load which, divided by the total specimen cross-

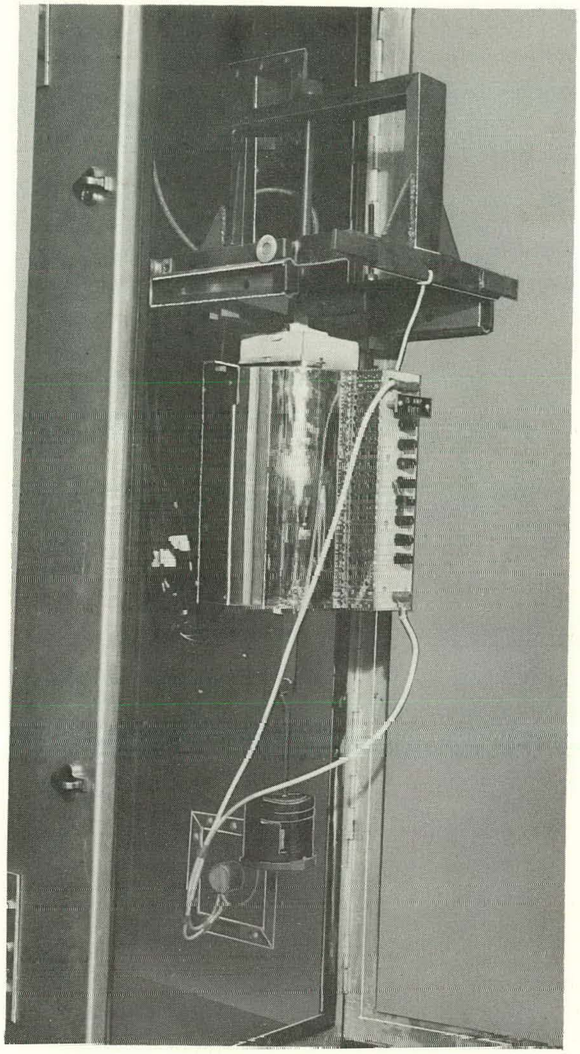

Fig. 9.19 – Enclosed CWS filtered cabinet for creep testing of enriched fueled material, showing specimen, furnace, and traversable suspension frame work (Neg. P62-5-19B)

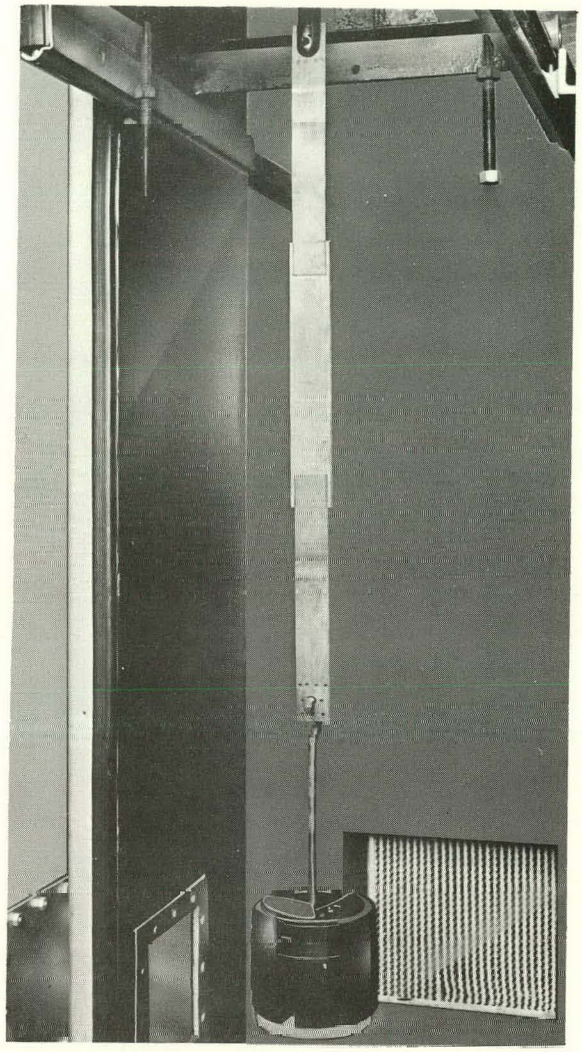

Fig. 9.20 – Interior of enclosed stand, showing enriched fuel sheet specimen and suspended weight (Neg. P62-5-19C)

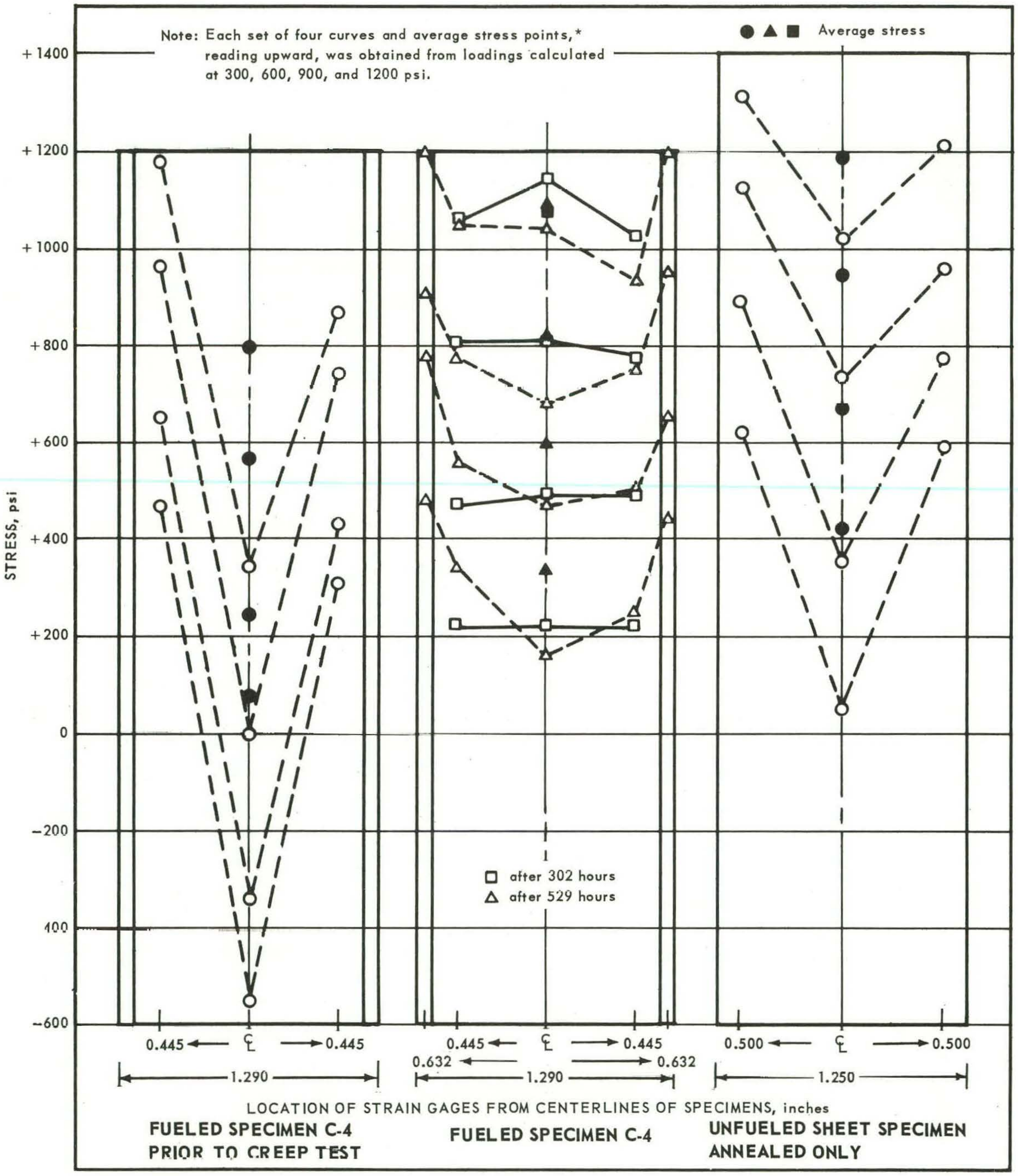

Fig. 9.21 – Stress patterns from strain gage measurements on fueled and unfueled 22-mil 80Ni – 20Cr creep-test specimens

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section area, results in the desired average stress level. In order to determine the actual stress pattern which existed in the samples, strain gages were applied at various locations on specimen C-4. Strain readings were then taken at average stresses of 300, 600, 900, and 1200 psi. Data were obtained at three intervals of test time: prior to heating, after 302 hours of testing, and after 529 hours. Figure 9.21 shows the location of gages and the stress patterns obtained.

To determine whether this non-uniform stress pattern was peculiar to the brazed channel configuration, a specimen was prepared from 80Ni - 20Cr cladding stock. The dimensions of the specimen were identical to the fueled sheet specimens. The same type of extension strap attachment was used. It did not, however, have the thickened edges peculiar to the fueled specimens with brazed channel edge seals. Results of this determination are also shown in Figure 9.21. It was suspected that the brazed edge seal caused the non-uniform stress pattern. However, it is evident from the thin-edged specimen that another cause is responsible for the irregular stress patterns. Additional testing of this type will be conducted on fuel sheet specimen C-4 at intervals during the remaining test time. Spot checks will be made on other fuel sheet specimens.

9.3.2 CREEP TESTS OF CLADDING STOCK

Test specimens were prepared from 0.040-inch-thick 80Ni - 20Cr cladding stock. Specimen RD-1 was sheared from this stock after it was cold-reduced to a thickness of 0.030 inch. Specimens RD-2 and RD-3 were hot-reduced to 0.030-inch thickness. Specimens RD-1 and RD-2 were sheared in strips 1.25 inches wide by 18 inches long. The center portion was then machined to a reduced test section 0.375 inch wide and 2.25 inches long. RD-3 was put on test as a 1/2-inch-wide by 18-inch-long strip.

The tests are being conducted isothermally in conventional creep testing facilities. The specimens are removed from test every 500 hours for elongation measurements. The results obtained to date are presented in Figure 9.22. Data will be accumulated to 15,000 hours or until 5 percent elongation occurs. The disparity in creep behavior between specimens RD-1 and RD-3 may be caused, in part, by stress variations across the specimen, as a result of the different shape of the specimens.

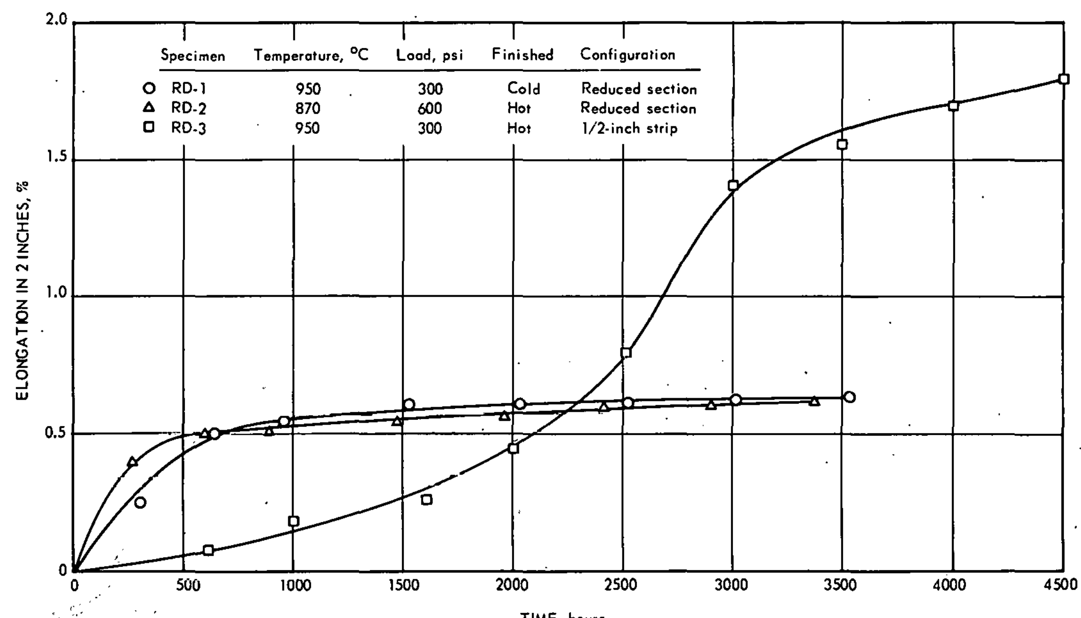

Fig. 9.22 – Results to date of creep tests of 80Ni – 20Cr cladding stock

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9.3.3 PLANNED TESTING OF TRANSVERSE STRENGTH

The transverse strength of fuel sheet, a measure of core-to-clad bonding, is indirectly an indication of the ability of the cladding to withstand internal pressure such as is generated by gaseous fission products. Specimens are being prepared to determine the short-time transverse tensile strength of recently manufactured 80Ni - 20Cr-clad fuel sheet. Specimens are simply extension bars brazed to the opposite cladding surfaces of fuel sheet segments. The assembly is then machined to a test-bar configuration. Relative strength of cold- and warm-finished fuel sheet will be determined at temperatures of 615°C and 815°C (1140°F and 1500°F).

9.3.4 PLANNED TESTING OF OXIDATION RESISTANCE OF GE-81 BRAZE ALLOY

Little information is available on the oxidation resistance of GE-81 braze alloy for extended periods of time as required in the 630A. Previous studies of 80Ni - 20Cr alloy indicated that silicon contents above 1.9 percent are detrimental to the oxidation resistance by promoting excessive grain boundary oxidation. Since the GE-81 braze alloy normally contains about 10 percent silicon, the oxidation resistance of the brazed joint areas will probably be compromised by the silicon concentration. Recent evidence of this has been observed in thermal cycle specimen EB-6 which exhibited significant thermal etching in the braze fillet area.

To study the oxidation resistance of GE-81 braze alloy, a series of compositions has been prepared containing the following weight percentages of silicon in an 80Ni - 20Cr base:

Alloy	Silicon Content, %
OX-0	0
OX-1	1
OX-3	3
OX-6	6
OX-15	15
OX-21	21

These compositions were prepared by the arc-melting of powder metallurgy compacts. They were sliced into wafers approximately 1/4- by 1/2- by 1/16-inch thick and will be subjected to air oxidation at temperatures of 650°, 815°, and 1000°C (1200°, 1500°, and 1830°F). In addition to an evaluation of surface appearances and surface area, change in weight as functions of time will be recorded.

9.4 FUEL SHEET PREPARATION (243)

Fifteen depleted fueled 80Ni - 20Cr billets were fabricated and processed into sheet following procedures established under the aircraft reactor program. Because these procedures were developed during several years of intensive effort, no problems were encountered. The fuel sheets were fabricated in order to furnish stock for evaluating the properties of warm-rolled sheet at 630A design temperatures, to activate existing laboratory equipment, and to evaluate possible improvements in the original manufacturing procedure.

9.4.1 STARTING MATERIALS

Nickel and chromium powders, cladding stock, and UO_2 were utilized from existing stocks. Each of these materials was subjected to chemical analysis to confirm its identity and to check its quality.

9.4.2 BILLET PREPARATION

Of the 15 billets which were prepared, 10 contained 37 weight percent of depleted UO_2 in the core, and 5 were prepared with no fuel in the core. These unfueled billets were used in the reactivation of several pieces of equipment. The billets were sized to produce a final sheet thickness of 0.022 inch with cladding thickness ranging from 0.004 to 0.005 inch.

An 80Ni - 20Cr powder blend was utilized in cold pressing frames and, with the addition of fuel, for the core. The as-received UO_2 was first processed through a hydrogen furnace at 1200°C (2200°F) for 1 hour to reduce excess oxygen and moisture, assuring nearly stoichiometric UO_2. It was then sintered in hydrogen at 1700°C (3100°F) for 1 hour. The purpose of this final sintering was to agglomerate the extremely fine particles. After sintering, the UO_2 was crushed manually and sifted through a minus 325-mesh screen. The particle size was 3 to 50 microns, with an average of 18 to 20 microns. The UO_2 powder was then mixed with the 80Ni - 20Cr powder blend in a proportion corresponding to 37 weight percent UO_2.

Cores were pressed in a 2.8-inch by 3.6-inch tool steel die at a pressure of 20 tons per square inch. Frames were pressed to appropriate size to enclose the cores. The core-frame assemblies were sintered for 2 hours at 1200°C (2200°F) in a dry hydrogen atmosphere. The density of the sintered compacts was approximately 70 percent of theoretical.

The compacts were then sandwiched between previously prepared 0.040-inch-thick 80Ni - 20Cr cladding sheets. To assure clean and active surfaces, the clad sheets were descaled in sodium hydroxide at 480°C (900°F) and annealed in hydrogen at 1120°C (2050°F) for 20 minutes. The sheets were spot-welded to the 80Ni - 20Cr frame. These assembled sandwiches were then hot-pressed at 1090°C (2000°F) and 1650 psi to effect bonding of the cladding stock to the sintered core and frame. After hot-pressing, the core-frame assemblies were completely encapsulated by heliarc-welding the edges of billets. Radiographic examination of the welded billets was performed to determine weld integrity, uniformity of fuel dispersion in the core, and the increase in core width caused by hot-pressing.

9.4.3 SHEET ROLLING

As welded and hot-pressed, the billets measured 0.275 inch thick. All billets were hot-rolled by the same rolling schedule. Initial reduction was performed at 1120°C (2050°F) in an 8-inch by 8-inch two-high rolling mill. The number of passes and the reductions per pass are shown in the following tabulation:

Pass	Initial Thickness, inch	Rolled Thickness, inch	Reduction, %
1	0.275	0.203	26
2	0.203	0.154	24
3	0.154	0.119	23
4	0.119	0.091	23.5
5	0.091	0.071	22
6	0.071	0.056	21
7	0.056	0.047	16
8	0.047	0.042	11

After hot-rolling, the sheets were descaled and hydrogen-annealed at 1020°C (2050°F) for 20 minutes.

Five of the fueled sheets and five unfueled sheets were warm-finish-rolled on a 4-inch by 6-inch two-high mill at temperatures between 790°C and 1000°C (1450°F and 1830°F).

The reduction from 0.042-inch to 0.022-inch thickness was obtained in seven passes. The sheet was again descaled and submitted for inspection.

The other five fueled sheets were cold-rolled on a four-high mill using 3-inch by 8-inch work rolls with 10-inch by 8-inch backup rolls. The reduction of these sheets from 0.042-inch to 0.022-inch thickness was accomplished in five passes. No descaling was necessary at this point and the sheet was submitted for inspection.

9.4.4 INSPECTION RESULTS

The inspection results obtained, on the 10 fueled sheets are listed in Table 9.6. Small sections of several of these sheets have been submitted for metallographic examination to evaluate the uniformity of fuel dispersion and to verify cladding thickness.

TABLE 9.6

INSPECTION RESULTS ON TEN EXPERIMENTAL 80Ni - 20Cr FUEL SHEETS

Billet Or Sheet No.	Average Core Width/Maximum Width Variation, in. As Hot-Pressed	As Hot-Rolled	As Warm-Rolled	As Cold-Rolled	Thickness Variation From 0.0220 inch, in.	Cladding Thickness, in. ("Dermitron" Gage)	Ultrasonic Bond Check Maximum Variation, %
DC-1	2.786/0.025	2.900/0.016	3.037/0.020	-	+0.0003, -0.0008	0.0030-0.0046	4
2	2.765/0.009	2.900/0.017	3.030/0.035	-	+0.0003, -0.0008	0.0033-0.0042	4
3	2.770/0.016	2.902/0.014	3.046/0.020	-	+0.0003, -0.0008	0.0030-0.0042	4[a]
4	2.783/0.010	2.916/0.010	3.042/0.020	-	+0.0003, -0.0008	0.0030-0.0040	4
5	2.778/0.020	2.897/0.027	3.034/0.030	-	+0.0003, -0.0008	0.0033-0.0040	4
11	2.803/0.010	2.914/0.020		2.930/0.020	±0.0004	0.0036-0.0040	5
12	2.800/0.030	2.910/0.020	-	2.928/0.030	±0.0004	0.0038-0.0042	5
13	2.787/0.020	2.910/0.020	-	2.920/0.025	±0.0004	0.0036-0.0040	5
14	2.803/0.010	Scrapped in hot-rolling	-	-	-	-	-
15	2.897/0.020	2.905/0.020	-	2.933/0.020	±0.0004	0.0038-0.0042	5

[a]One indication of poor bond due to a 0.030-inch diameter high-density spot, causing a variation of 20%.

9.5 WORK PLANNED FOR NEXT PERIOD

1. Complete preparation of MTR fuel ring specimen-capsule assembly.
2. Initiate oxidation testing of MTR fuel ring specimens in laboratory bench tests.
3. Initiate comparative creep testing of warm- and cold-finished fuel sheet.
4. Initiate fuel sheet transverse strength determinations.
5. Complete tensile tests of ring joint specimens.
6. Complete pile-ambient irradiation of 80Ni - 20Cr clad alloy specimens.
7. Produce experimental fuel ribbon with varying cladding thickness and containing spherical fuel particles.
8. Initiate the preparation of fuel sheet for the 630A critical experiment.
9. Establish procedures for preparing control rods for the 630A critical experiment.

10. CRITICAL EXPERIMENT MOCKUP

(260)

During this report period, the critical experiment mockup design was completed, and a preliminary test outline prepared. This work is described following a general description of the critical experiment objectives.

10.1 OBJECTIVES

The critical experiment is used as a primary design tool for the reactor, and to evaluate and determine operational sequences and requirements. To serve this twofold purpose, the critical experiment must be a nuclear duplicate of the 630A power reactor (see Figure 10.1) and be constructed to permit changes of the design configuration. The experiments will be scheduled to provide nuclear design data for the prototype reactor in a timely manner, compatible with the over-all program schedule for the 630A nuclear steam generator.

Basically, the geometry of the reactor mockup is identical to that of the power reactor, but with the flexibility needed to permit changes not ordinarily possible in a reactor. The fuel, for example, is in the form of uranium metal interleaved with separable layers of nichrome, instead of an oxide contained within a nichrome jacket. Using relatively thin layers of uranium metal will make it possible to determine the fission power distribution within the fuel elements with a very high resolution.

Additional flexibility is provided by varying the diameters of the central moderator rods. Thus, a combination of various moderator rod diameters, amounts of uranium, and uranium distribution, makes possible power distribution requirements experimentally tailored to provide maximum performance of the power reactor.

By suitable additions of other materials, the effects of fission product poisoning and uranium depletion will be simulated, thus allowing similar determinations of fission power distributions at progressive stages of operating life. Control rod worths, temperature rises due to secondary heating of components in the system, reflector effectiveness and source specifications for the shield are other quantities that will be determined in the critical experiment program.

The mockup will also be used to evaluate the safety of the reactor under such unlikely occurrences as the core being flooded with water, or the melting and displacement of the fuel. These safety tests, of course, will be accomplished in a finite, step-wise fashion under well-controlled conditions.

The critical experiment is a major design tool that will facilitate the confident determination of the specifications required to achieve the operating life, fuel element temperature limitations, and the assured controllability of the reactor throughout the operating life.

10.2 CRITICAL EXPERIMENT DESIGN

During this report period, the critical experiment has been designed and 85 percent of the factory drawings issued. Procurement and fabrication of the hardware was begun and was approximately 50 percent completed at the end of the report period.

A perspective drawing of the over-all critical experiment is shown in Figure 10. 2. The reactor assembly is shown in Figure 10. 3. A comparison of Figures 10. 1 and 10. 3 shows that the radial and longitudinal cross section geometries are identical in the planes of the active core region. The materials used in the reactor mockup are the same as in the power reactor except that in the mockup, mild steel is substituted for the outer stainless steel reflector ring, the borated stainless steel thermal shield and the stainless steel reactor vessel of the power reactor, and the borated stainless steel thermal shield ring of the power reactor is simulated by a 3/16-inch-thick ring of boral in the mockup. These substitutions of materials were made to reduce costs and do not significantly change the nuclear duplication. To provide the necessary ease of assembly and versatility for alteration and adjustment of components, the bottom support structure and the methods of assembly of the components are unique to the critical experiment.

The fuel cell assembly is shown in Figure 10. 4. As previously explained, the fuel elements are simulated with uranium foil interleaved with nichrome foil wrapped on the three tubes in the active core region. The outermost layer of foil is covered with a burnable poison, borated stainless steel, as it is on the prototype reactor. The three discs above the active core simulate the unfueled nichrome stage that serves as the end reflector. Likewise, the four concentric tubes below the active core simulate the two unfueled nichrome end reflector stages. It should also be noted that the three different size central moderator tubes and the center control rods are easily removable and interchangeable to provide the required flexibility.

Figures 10. 5 through 10. 12 are photographs of some of the critical experiment components at their present stage of fabrication. Figure 10. 13 is a diagram of the 630A critical experiment core.

10.3 TEST PROGRAM

During this report period, a preliminary test program was outlined which covers the test period from October 1962 through the third calendar quarter of 1963. The test program is divided into five phases:

<u>Phase I</u> covers the usual initial reactor operations of incremental core loading and system checkout.

<u>Phase II</u> includes: (1) a general power mapping of a representative sector of the core to establish the degree of power shaping with the initial distribution of fuel and moderator, and (2) the calibration of the poison rods at the interstices between fuel elements in a representative sector of the core.

If the gross radial power distribution or excess reactivity are not satisfactory, then sufficient measurements will be made to provide data necessary to calculate an improved core configuration.

<u>Phase III</u> includes detailed power distribution studies in the final, or close to final, core configuration; measurements of the moderator temperature effects on reactivity and power distributions; in-core neutron spectrum measurements; danger coefficients for

various control rod compositions; and changes in reactivity and power distributions caused by core life effects; e. g., fuel and poison burnup and fission product buildup.

Also included in this phase are reactivity measurements of flooded cells, and of the worth of central scram rods in the flooded cells, at various radial locations in the core.

Phase IV covers all secondary heating rate measurements inside the core and through the segmented reflector, thermal shield, and primary containment vessel assembly. These measurements will be made in the final core configuration, or in a configuration very similar to the final configuration.

Phase V includes all measurements required for adequate and reliable shield design, such as the slow neutron flux through the thermal shield, fast neutron spectrum as well as fast neutron and gamma dose rates outside the primary containment vessel.

The chronological sequence of experiments need not follow the sequence of this preliminary test outline, and will be designed to most economically utilize reactor time.

10.4 WORK PLANNED FOR NEXT PERIOD

During the next reporting period, the remaining factory drawings will be issued. The procurement and fabrication of the components will also be completed. Prior to shipping the components to ITS, a partial assembly of major components will be made in Evendale to insure that the fits, clearances, alignments, handling methods, and assembly procedures are satisfactory.

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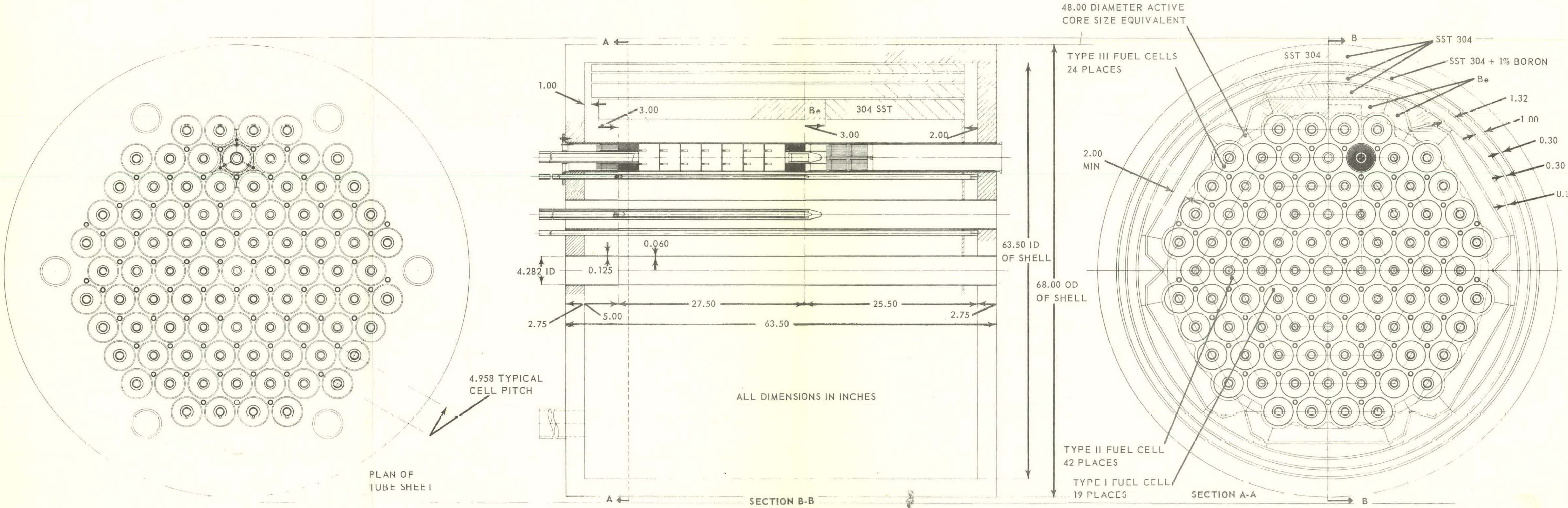

Fig. 10.1 – Second iteration of 630A reactor configuration (Dwg. 219R815)

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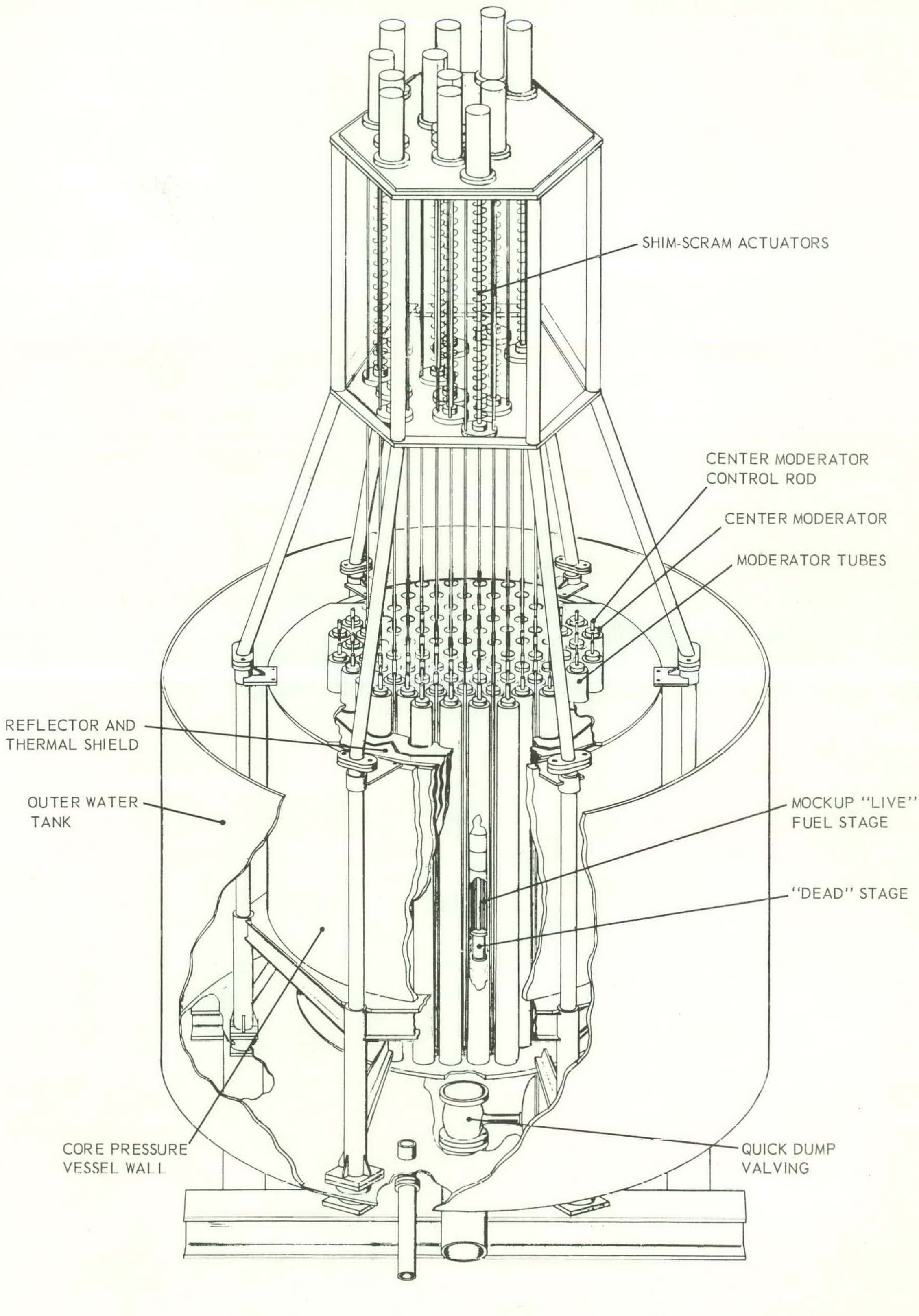

Fig. 10.2 – Perspective drawing of 630A critical experiment assembly (Dwg. 692E606)

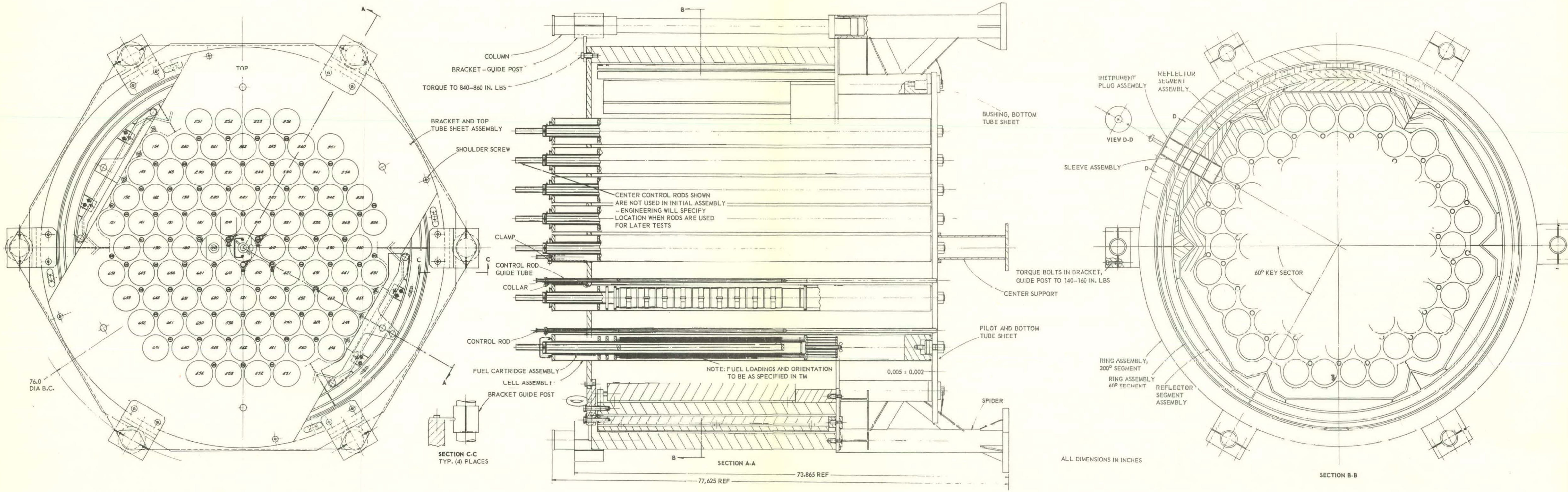

Fig. 10.3 – 630A critical experiment reactor assembly (Dwg. 219R839)

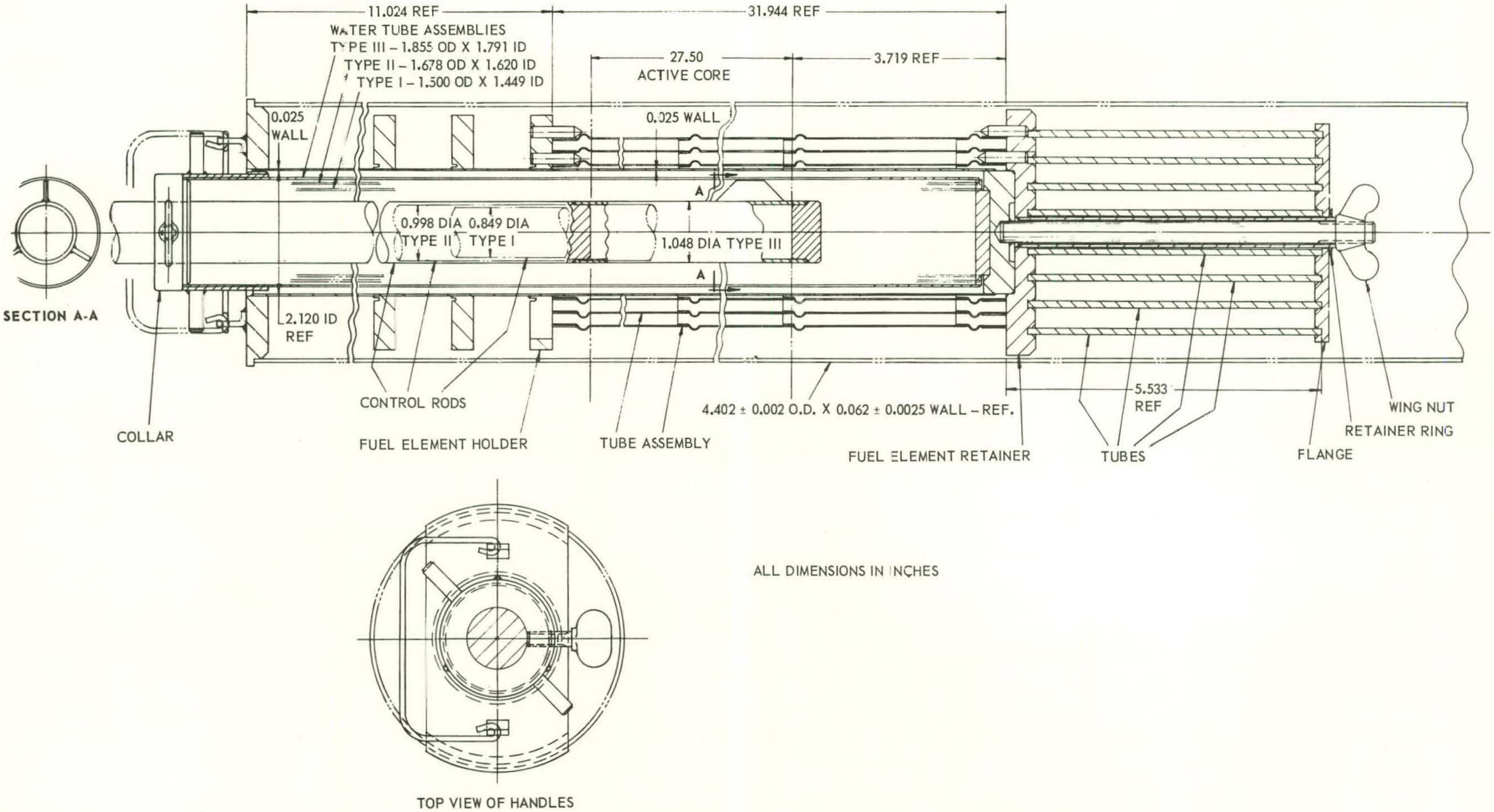

Fig. 10.4 – Fuel cell assembly, 630A critical experiment (Dwg. 848D617)

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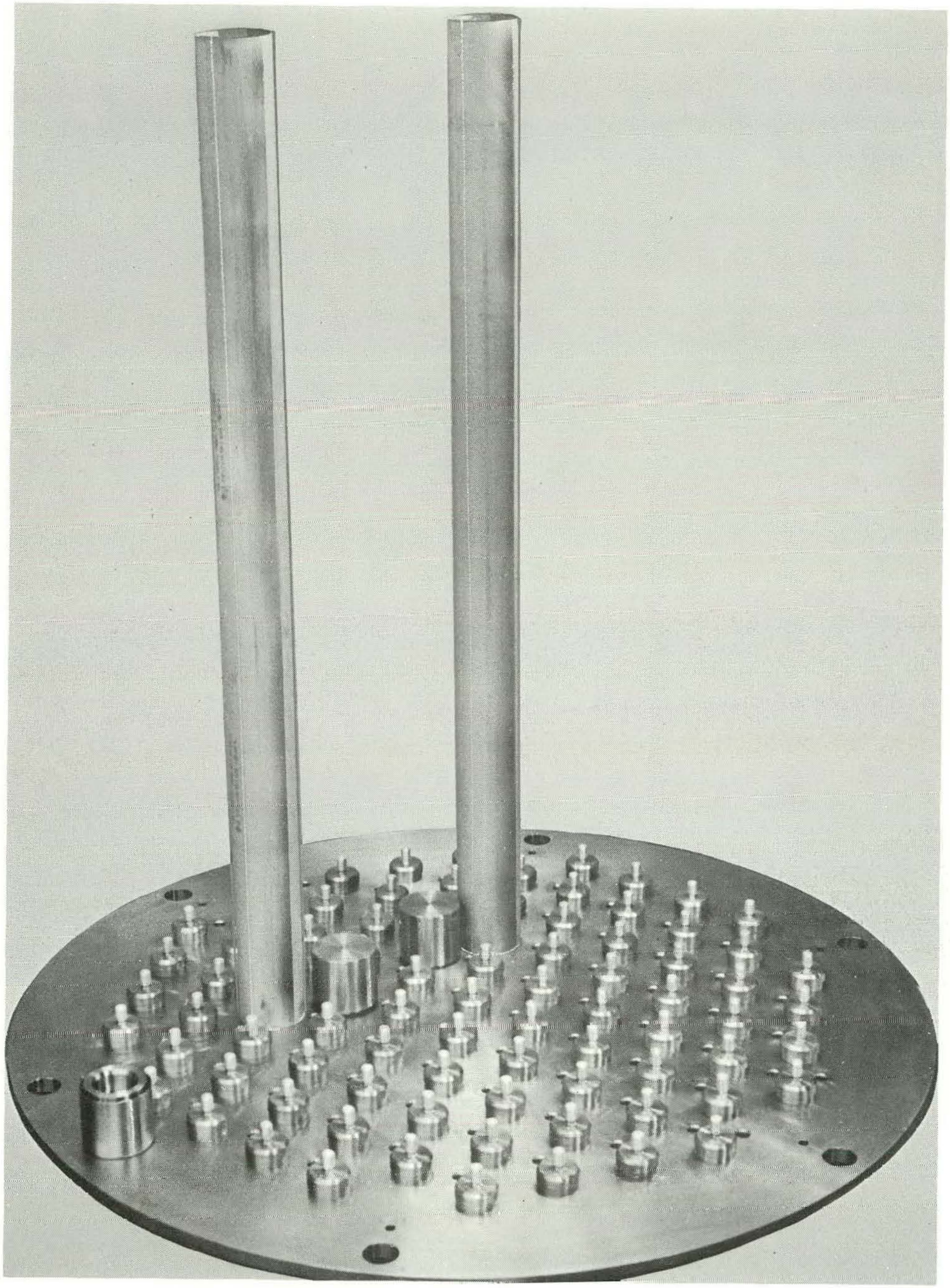

Fig. 10.5 – Bottom tube sheet with pilots and fuel tubes assembled to it
(Neg. P62-7-17A)

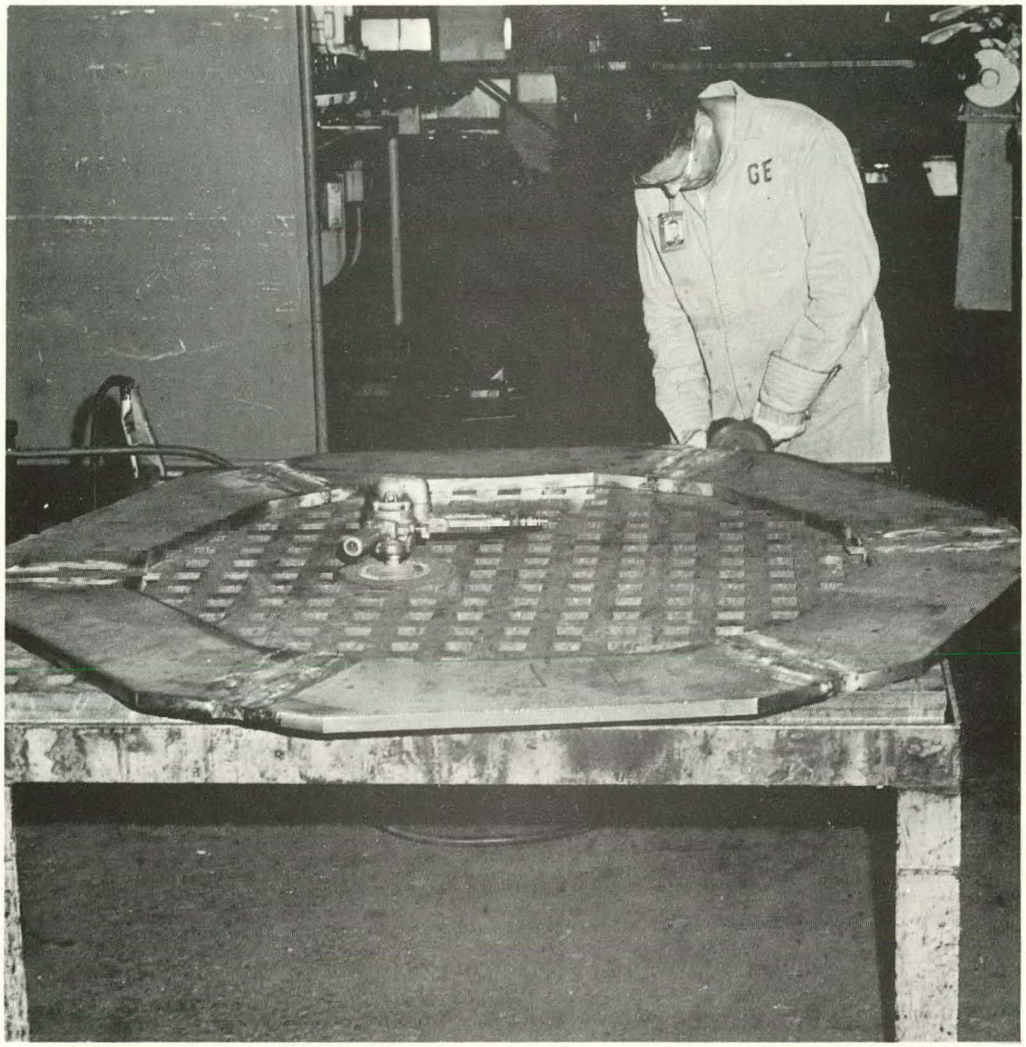

Fig. 10.6 – Top stainless steel plate of the spider assembly (Neg. P62-6-36C)

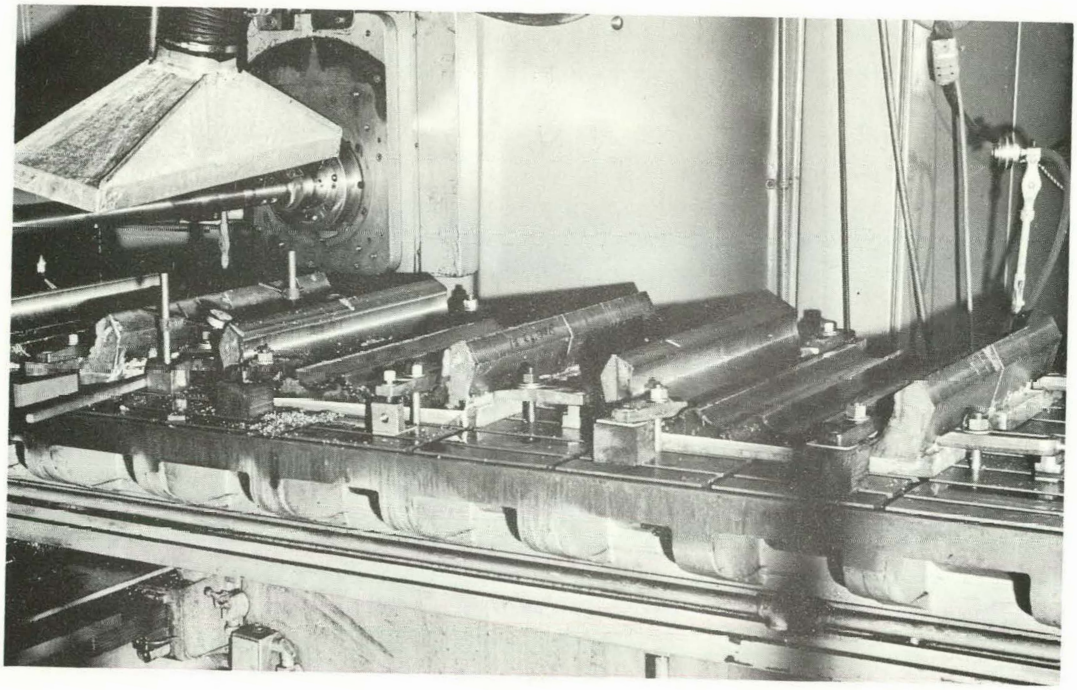

Fig. 10.7 – Machining of the beryllium reflector slabs (Neg. P62-7-16B)

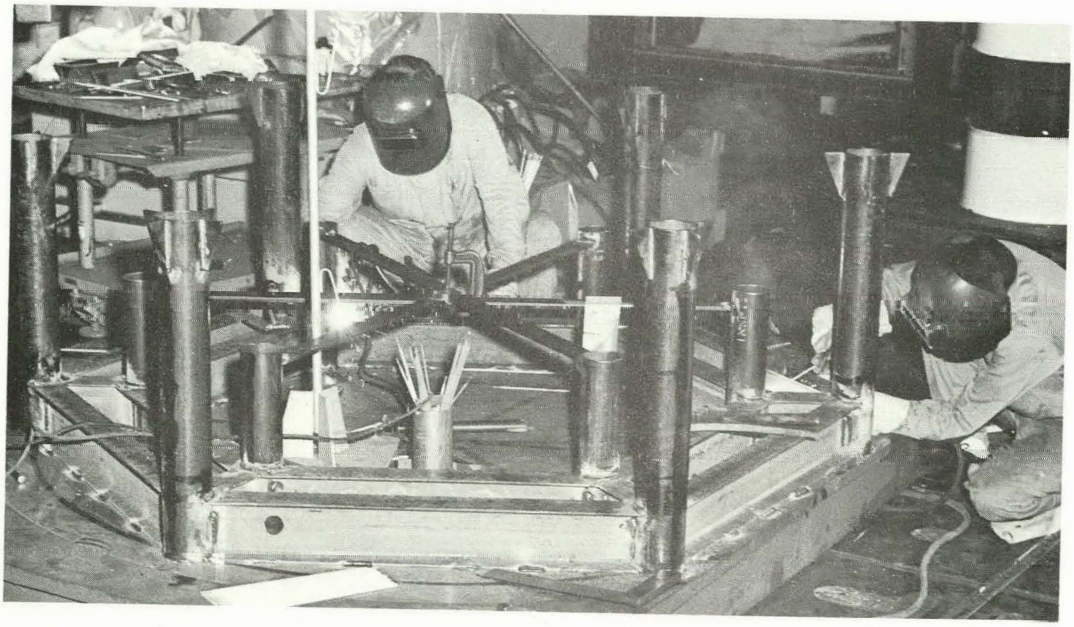

Fig. 10.8 – Fabrication of the bottom support spider (Neg. P62-6-36F)

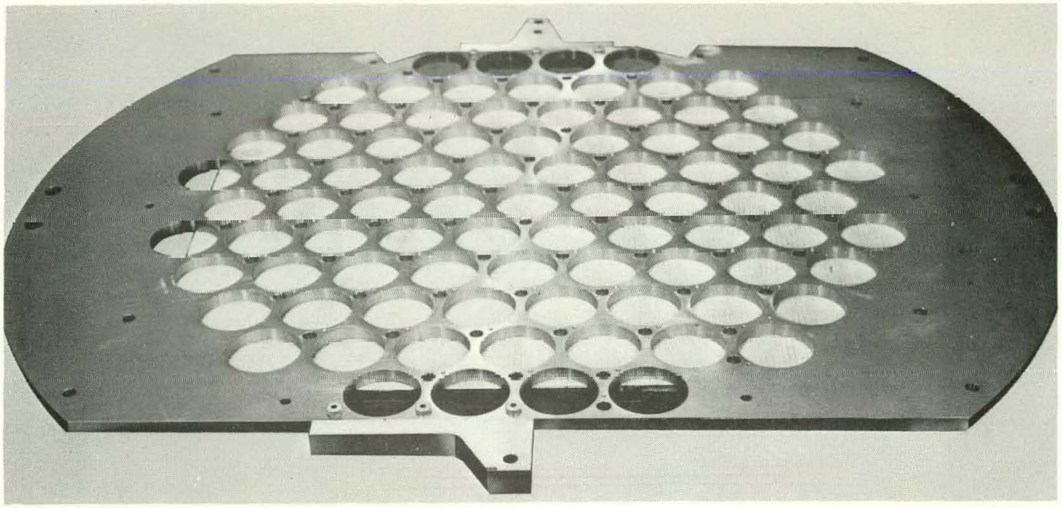

Fig. 10.9 – Completed top tube sheet (Neg. P62-7-17B)

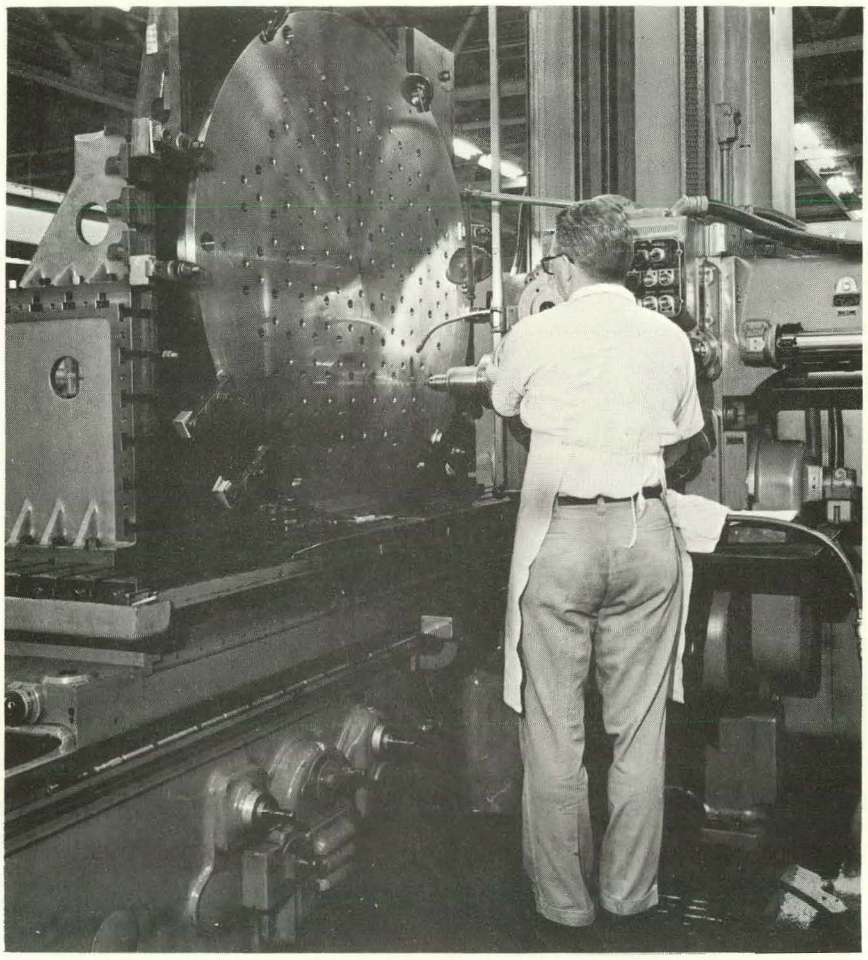

Fig. 10.10 – Machining bottom tube sheet (Neg. P62-5-20 A)

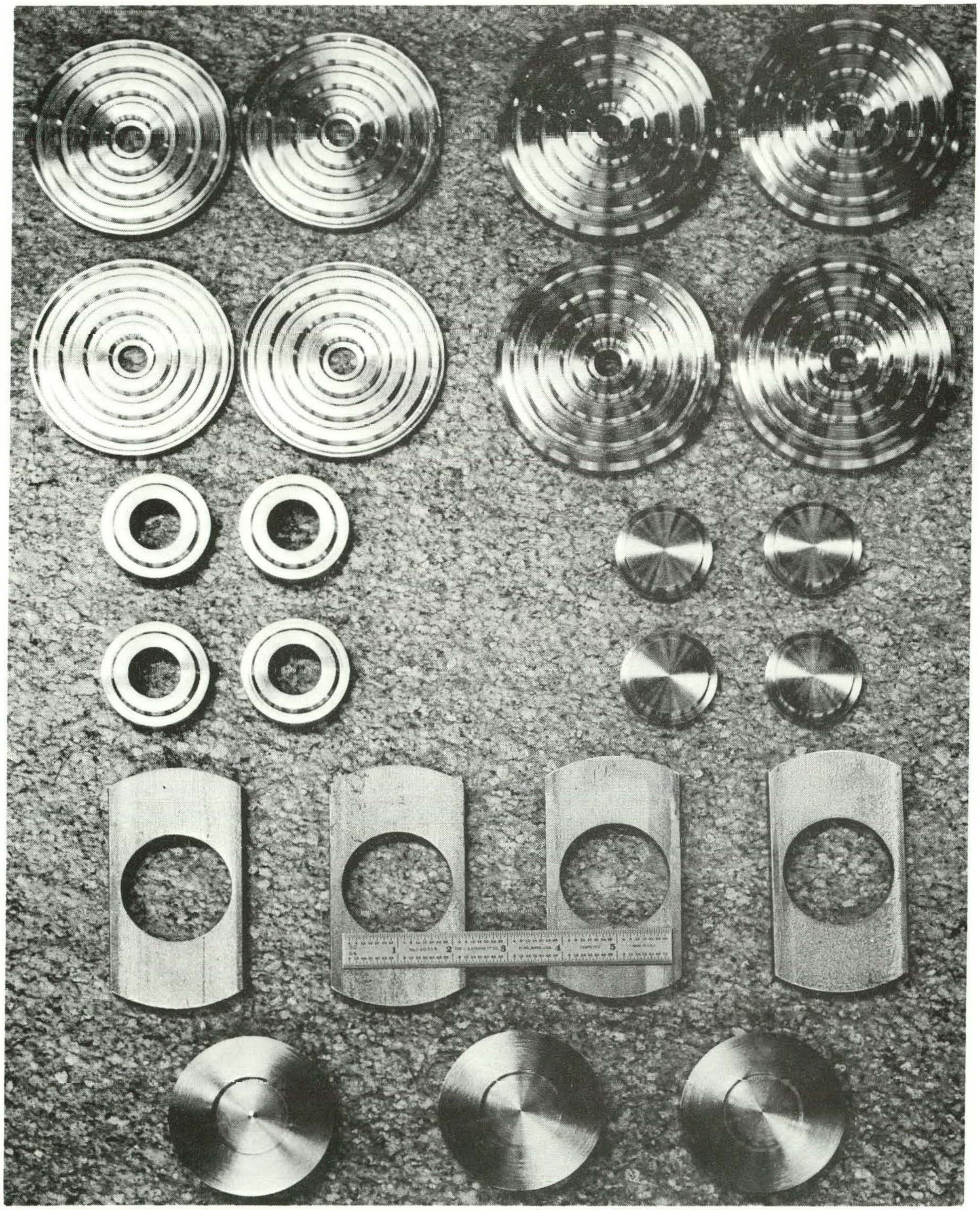

Fig. 10.11 – Components of the fuel cell assembly (Neg. P62-5-20 D)

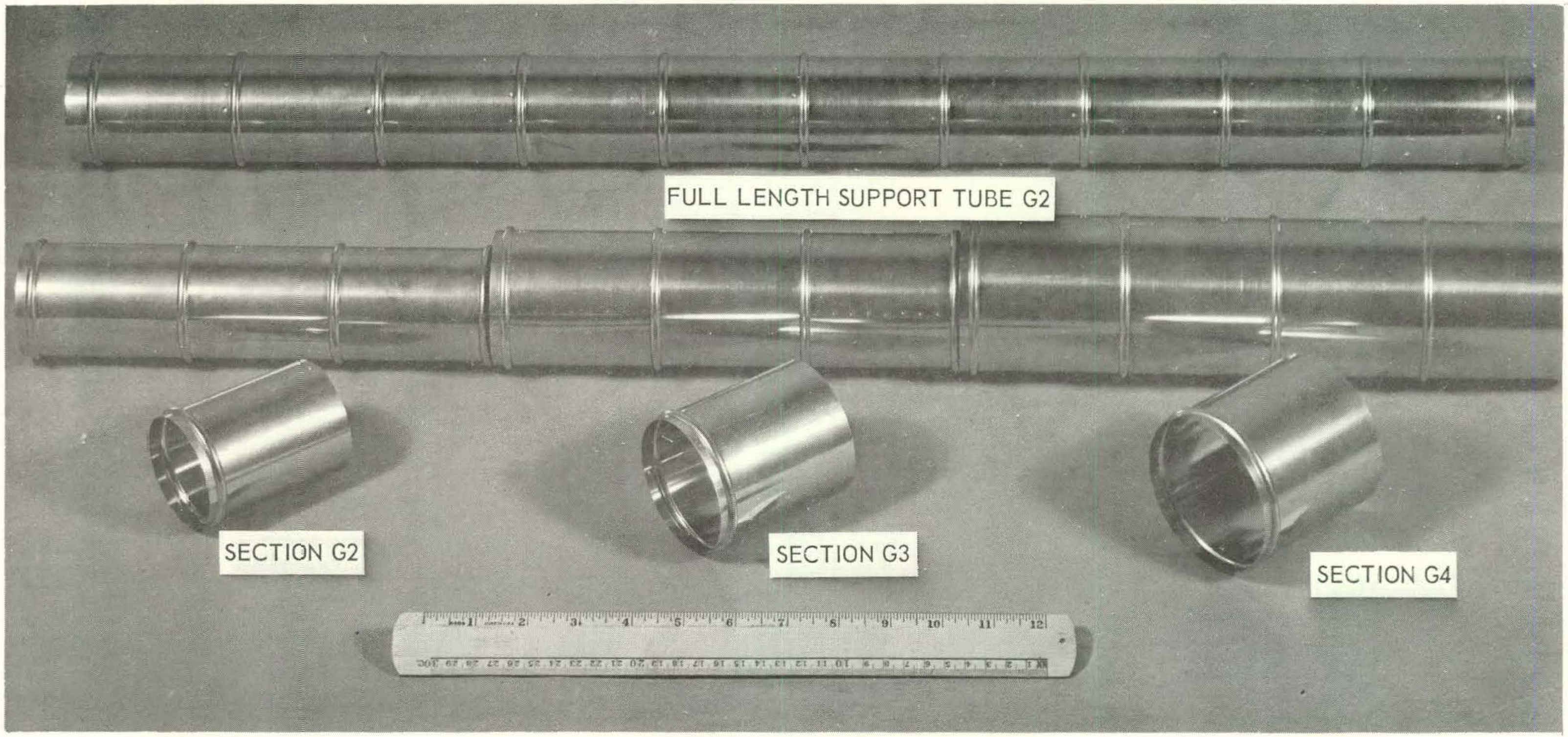

Fig. 10.12 – Fuel support tubes (Neg. P62-5-20 H)

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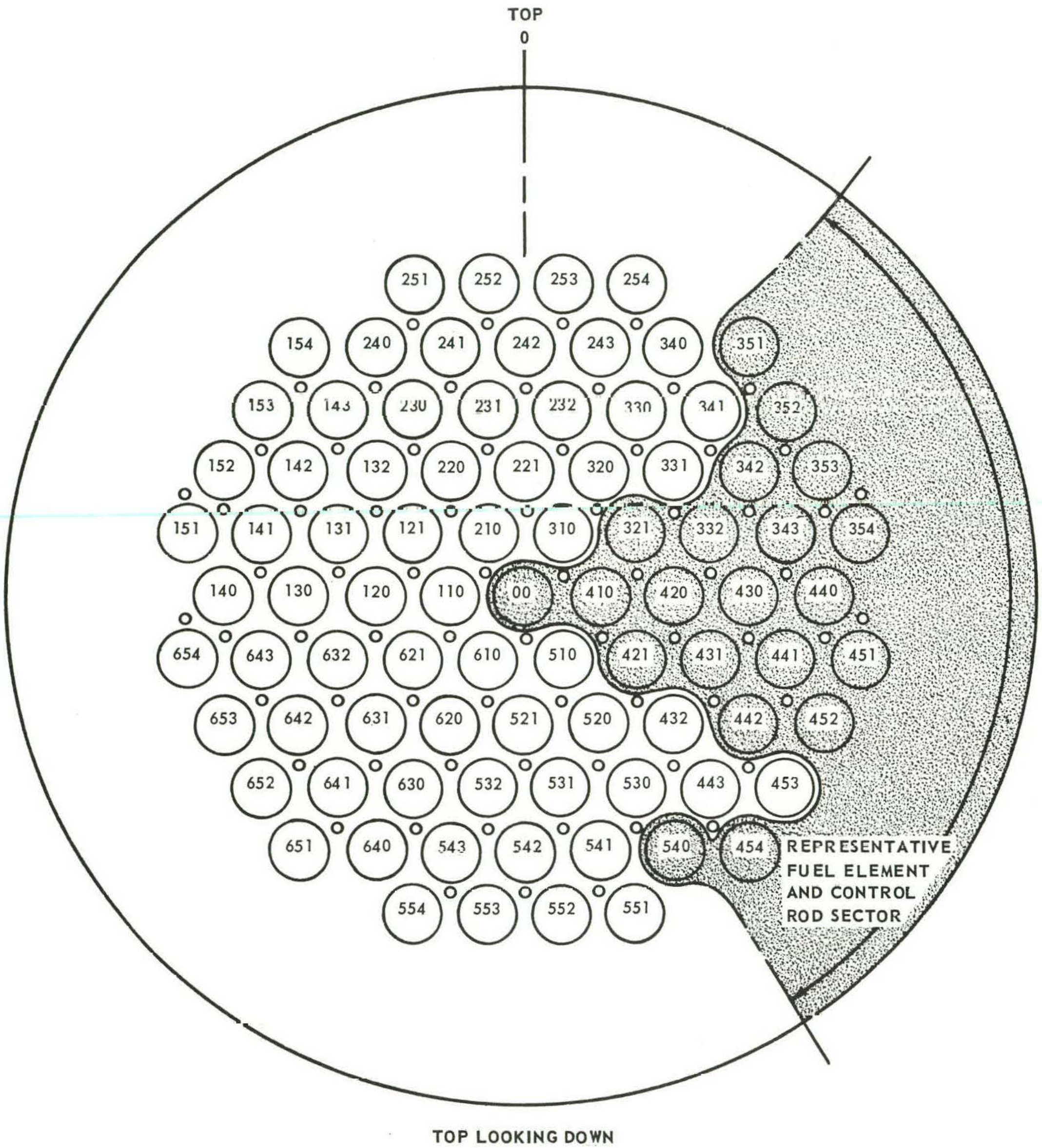

Fig. 10.13 – 630A critical experiment core diagram

11. CRITICAL EXPERIMENT TEST EQUIPMENT

(270)

During this report period, the design of most critical experiment test equipment components was completed and fabrication proceeded apace. Following a general description of the test equipment, the various components are described and the status of each at the end of this report period is presented.

11.1 GENERAL DESCRIPTION

The critical experiment consists of a tank to contain the moderator water, a reactor core mockup, actuators and their support stand, a fill and drain system, and a working platform at the level of the tank top. This assembly is being built on the 10-foot-wide trailer with detachable wheels. When completed the assembly will be moved to LPT where it will be checked out and loaded with fuel.

The use of salvageable parts from previous experiments, and the use of mild steel in certain components has kept the cost of building this assembly within very nominal limits. Mild steel is being used for parts that will be immersed in demineralized water. The mild steel will be coated with an epoxy base paint for corrosion control. Stainless steel or aluminum were used where the immersed metal surfaces were subject to scuffing from a mating surface or could not be painted.

With the exception of the reactor itself, less than $500 have been spent on materials.

The design and fabrication status of the portions of the critical experiment test equipment are shown in Table 11. 1

11.2 WORKING PLATFORM (271)

The working platform as shown in Figure 11.1 will be level with the top of the critical experiment tank to provide access to the top of the reactor. A hole, cut in the platform floor will accommodate the top rim of the tank. To prevent foreign materials from entering the reactor, removable truncated circular segments will be placed over the space between the edge of the reactor and the platform during operation. The critical experiment tank will be mounted on one end of the trailer, leaving a clear working space of 7' x 10' at the other end.

Since it is possible that the reactor will be disassembled in the cell, the working platform is designed to support the combined weight of all pieces of the critical experiment reactor.

Fabrication of the platform was completed during the report period.

TABLE 11.1

630A CRITICAL EXPERIMENT ASSEMBLY STATUS

Assembly	Drawing No.	Drawing Title	Status In Drafting	Status In Shops
630A C. E. Assembly	692E308	Test Stand - General Assembly	25% complete	
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	Component parts including tank - complete
	109C243	Float Switch Bracket	Check Print	Not started
	109C230	Support Frame Assembly	Issued	Complete
Actuator Support Stand	692E309	Actuator Support Stand	Issued	Component parts finished.
	146A6213	Actuator Stand Clamp	Issued	Complete
	848D326	Actuator Support Plate Details	25% complete	Not started
Actuators	848D325	Actuator Modification Assembly	Issued	Working on component parts (see below)
	125B7249	Actuator Mod. - Misc. Details	Issued	Parts 90% complete for 15 actuators
	125B7251	Control Rod Connector	Issued	25% complete
	125B7252	Actuator Extension Rod	Issued	Not started
	109C236	Flange, Magnet Plate	Issued	15 completed
	109C237	Flange, Bottom	Issued	15 completed
	109C240	Actuator Drive Motor Base Mod.	Issued	Component parts for 15 actuators complete.
Fill and Drain System	848D319	Piping-Fill and Drain System	Issued	25% complete - using modified system from 101 CET
	109C239	Specification Control Drawing - Heating Coil	Check Print	

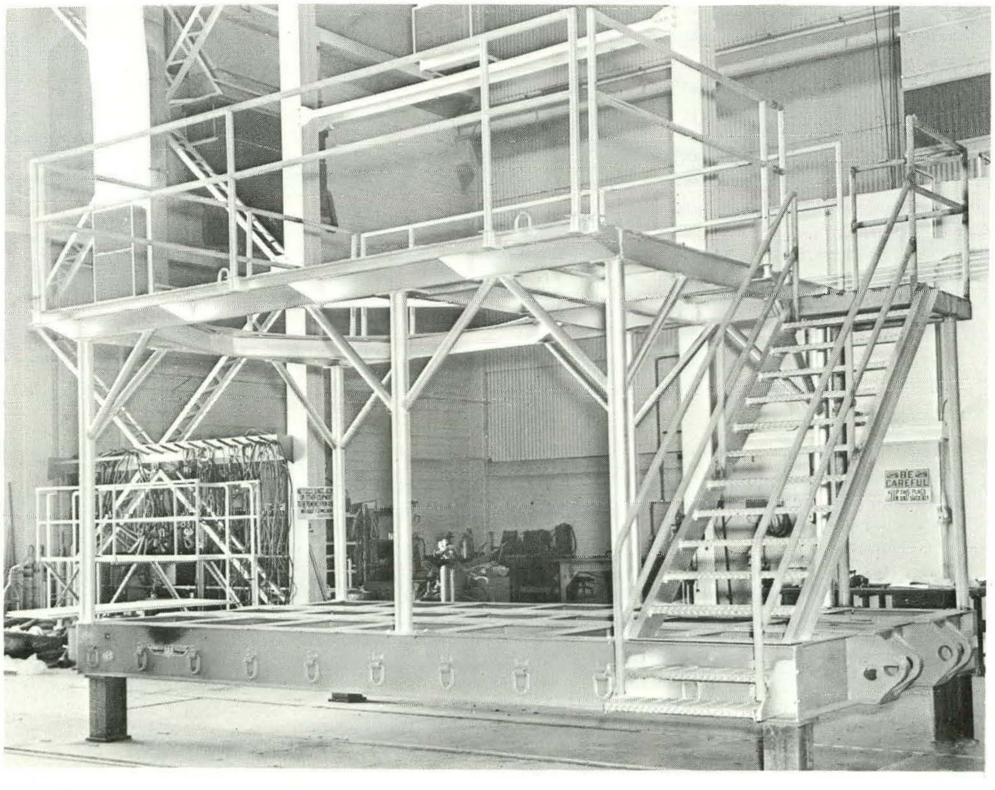

Fig. 11.1 – Working platform (Neg. 4268-1)

11.3 TANK SUBASSEMBLY (272)

The tank subassembly supports the reactor and provides containment for the moderator and shield water. The tank itself is a mild-steel tank 9 feet 2 inches in diameter and approximately 6 feet deep. There are eight instrument wells located at the horizontal midplane of the active portion of the reactor core. The support for the reactor consists of seven capped pipe pedestals that penetrate the bottom of the tank and are bolted to an I-beam support frame (Figure 11.2). The reactor support spider is bolted to these pedestals.

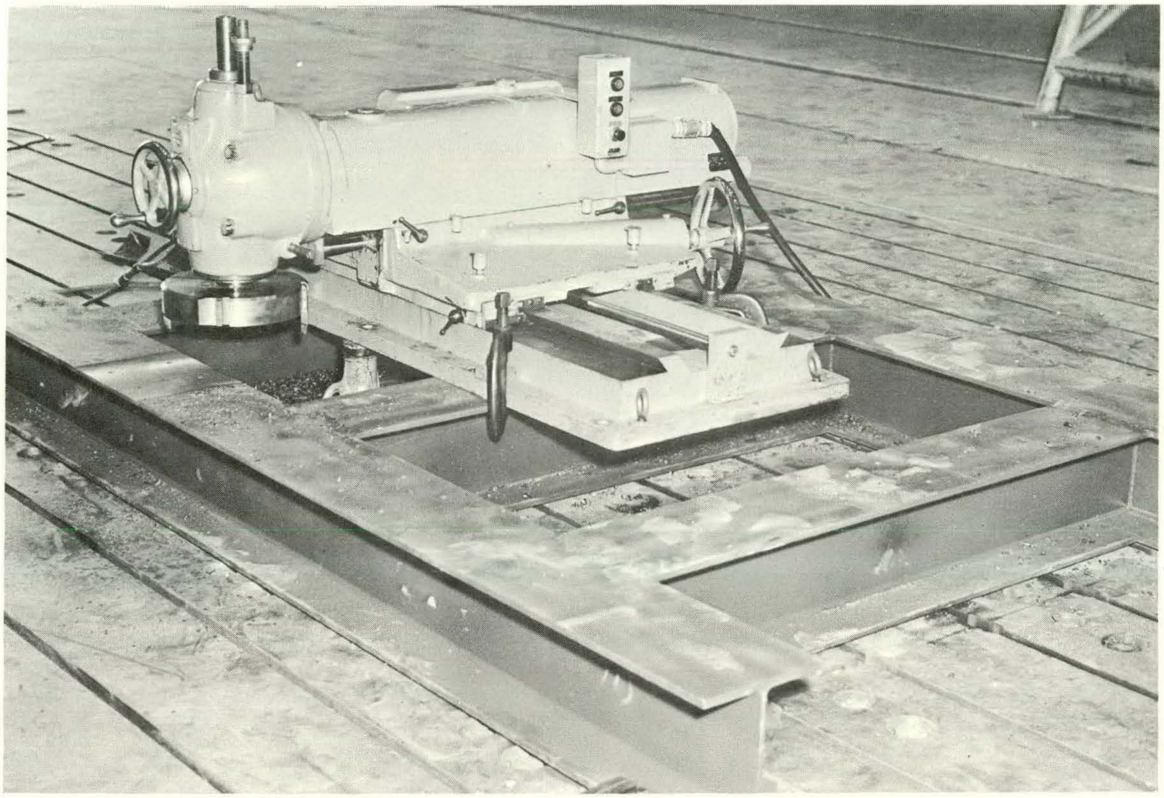

Fig. 11.2 – Machining of support frame (Neg. 4268-2)

11.4 ACTUATORS (275)

The shim-scram actuators, one of which is shown in Figure 11.3, were salvaged from a previous experiment. Modified for the present application, the actuators have a shim speed of 19.8 inches per minute, require 230 milliseconds to scram from the fully withdrawn position, and may have their maximum travel adjusted to any point between zero and thirty-four inches, with a limit switch.

The main parts of the actuators are described below:

Drive Head - Limit switches, motor and potentiometer are contained in the drive head. The potentiometer is used for position indication of the rods.

Magnet or Coupling Assembly - A movable coupling assembly is used for picking up or latching the rod connecting flange.

Lead Screw - Permits movement of coupling assembly and rods.

Coil Spring - Used to insert rods into the reactor.

Rod Connecting Flange - Rods, which may be of any size or number, are connected to this flange.

Guide Tubes - Used to position various members of the actuator.

End Flanges - Used for mounting purposes as well as a support for the actuator.

Buffer or Shock Absorber - Provides for absorption of the shock of the rods and rod connecting flange after firing.

Contact Plate - Permits pickup of rod connecting flange containing rods.

Elasticable - Stretchable wire which travels with the coupling assembly to provide current flow through the magnet coil.

Extension Rod Connector - Used to make the connection between the rod connecting flanges and the extension rods.

Extension Rods - Provide space between the actuator mounting plates and the top of the reactor so that fuel elements may be removed from the reactor without removing the actuator (see Figure 11.4).

Control Rod Disconnect - Provides a quick means of disconnecting the poison tip from the extension rod to facilitate removal of the fuel elements (see Figure 11.4).

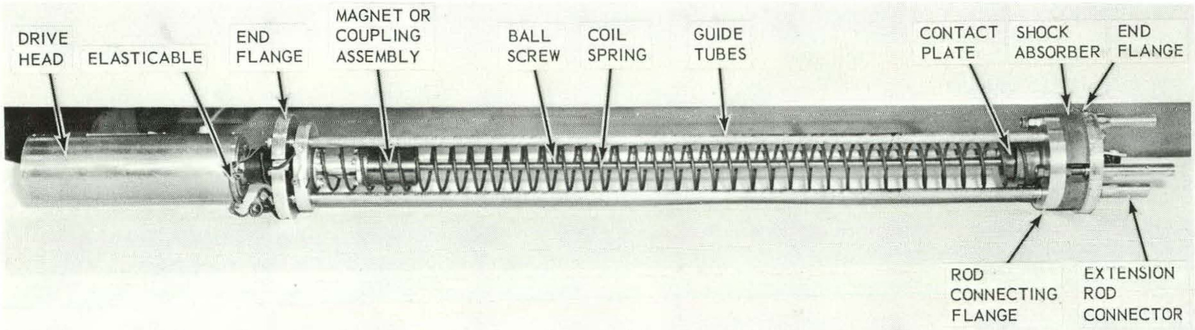

Fig. 11.3 – Critical experiment actuator (Neg. 4268-3)

In order to test the integrity of the actuator, a scram endurance test was conducted on two actuators mounted in a test stand as shown in Figure 11.4. A total of 822 full scrams were executed with the actuators. A summary of the problems encountered during the test is given in Table 11.2.

The actuator scram test results and the configuration of the 630A CE control rods require that the following modifications be made to the actuators:

1. End flanges - increase the diameter to accommodate the control rod spacing for the 630A CE reactor
2. Rod connecting flange - increase the diameter and add a second flange 3-1/4 inches from, and rigidly fastened to, the existing rod connecting flange to minimize cocking.
3. Drive head - increase the size of the potentiometer mounting studs to overcome a metal fatigue problem with the original studs. Also, countersink a small set screw that keeps the ring gear in the limit switch drive from rotating. This ring gear slipping has apparently been the reason for the limit switch settings changing during past operation.

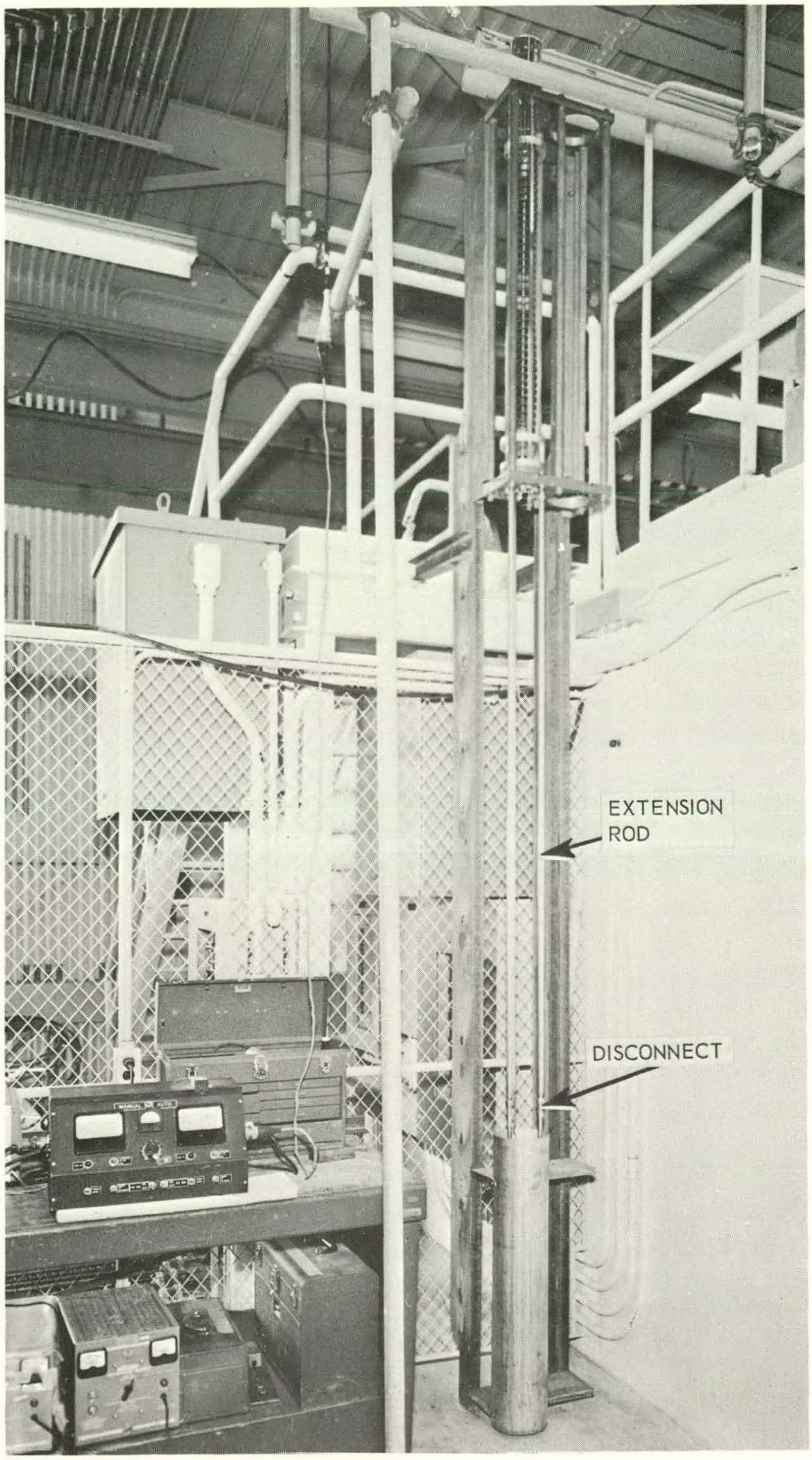

Fig. 11-4 – Actuator test stand (Neg. 4268-4)

TABLE 11.2

SUMMARY OF FAILURES DURING ACTUATOR SCRAM TESTS

Number Of Scrams After Repair To Next Failure	Type Of Failure	Remedy
64	Lower limit switch out of adjustment	Readjusted limit switch
82	Lower limit switch out of adjustment	Readjusted limit switch
73	Spurious Scramming	Straightened lead screw and honed magnet
140	Loose extension rods	Tightened extension rods
15	Motor mounting studs broken	Replaced studs and readjusted cams
19	Lower limit switch out of adjustment	Adjusted limit switch
13	Lower limit switch out of adjustment	Adjusted limit switch
10	Lower limit switch out of adjustment	Adjusted limit switch
23	Lower limit switch out of adjustment	Adjusted limit switch
16	Bottom magnet plate unscrewed from magnet	Tightened and replaced roll pin that had vibrated out
35	Potentiometer mounting studs broken	Changed drive units
58	Potentiometer mounting studs broken	Replaced studs. Drilled counter drive pinion gear housing for set screw.
274	Potentiometer mounting studs broken	Redesigned studs

4. Latching circuits - add a 60-volt power supply which will be used to latch the magnets. After three seconds a time delay relay will automatically reduce the voltage to a hold-in value of 24 volts, to prevent overheating of the coils. This is to overcome a drop-out problem that resulted from the magnet not always aligning correctly with the contact plate, by providing an additional force to pull the magnet and contact plate into alignment before the voltage is reduced. The effect of the magnitudes of the pull-in and holding voltages on the scram time for the actuator was checked and found to be negligible because the release time of the magnet is short compared to the travel time for control rods and connecting flange.

11.5 ACTUATOR SUPPORT STAND (276)

The actuator support stand sits on extensions to the six outer legs of the reactor support spider. The legs of the support stand are long enough to allow a fuel tube to be removed from the reactor without removing the support stand.

The actuators are supported top and bottom within the stand by horizontal plates. Since the bottom horizontal plate is counterbored to accept the bottom flange of each actuator, the bottom plate absorbs all of the vertical loads resulting from a scram. The top horizontal plate has a clearance hole for the top flange of each actuator so that it restrains only the horizontal movement of the top of each actuator. An actuator may be removed from the support stand by disconnecting the poison tips from its extension rods, loosening a single hold down bolt, and lifting out the actuator vertically with the bridge crane.

For major changes in the reactor, the actuator support stand may be removed from over the reactor after the poison tips have been disconnected from their extension rods and the legs of the support stand unbolted from the reactor support structure.

11.6 FILL AND DRAIN SYSTEM (277)

The fill and drain system shown in Figure 11.5 consists of a 3000-gallon storage tank for demineralized water, a steam and an electrical heating system, and the necessary valves, piping and pumps to circulate the water between the critical experiment tank and the storage tank or through the electrical heater. Water is pumped to the critical experiment tank by a 150-gpm pump. This pump is bypassed by valve V8 and a second pump returns the water to the storage tank under normal conditions. This pump also is used to circulate the water through the electrical heater if required. The quick operating valve is a 6-inch valve with straight through ports. In the case of an emergency this valve will rapidly reduce the water level below the active core region.

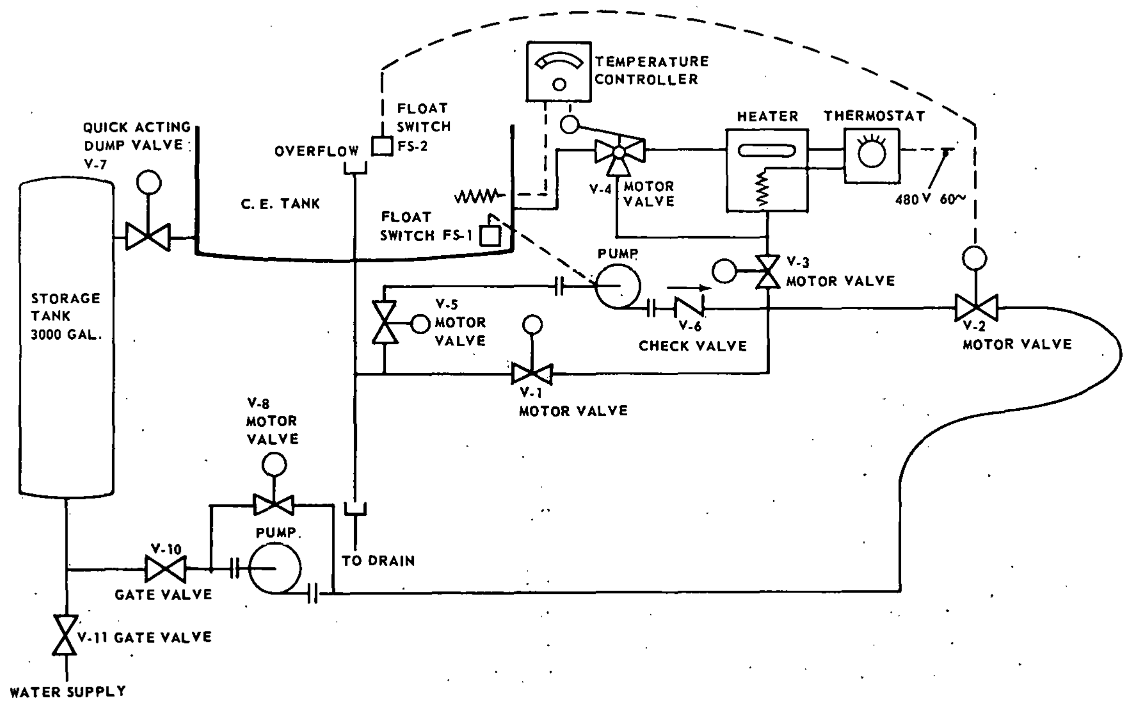

Fig. 11.5 – Critical Experiment test fill and drain system

The steam heating system is used to warm the water for measurements of temperature coefficient. The electrical heater is used to maintain the water temperature after it has been heated by the steam heater.

11.7 TERMINAL BOX (278)

The terminal box serves as a junction box for all of the actuator control cables so that they may be branched in an orderly fashion. This box also makes it possible to point by point check the dolly wiring without disconnecting the cables. The terminal box was used on a previous experiment and it is anticipated that it can be used for the 630A critical experiment with very little rewiring.

11.8 WORK PLANNED FOR NEXT PERIOD

The work outlined in Table 11. 1 will be completed by the next reporting period.

12. SAFETY ANALYSIS
(280)

Nuclear safety analysis efforts during the reporting period have been confined, almost exclusively, to estimation of the hazards associated with the 630A critical experiment to be operated at the Low Power Test Facility at NRTS. The initial draft of the hazards summary report was completed and is currently being circulated for internal review. The general content of the report was discussed with AEC-ORO personnel on June 26, 1962, at Oak Ridge, with the result that minor changes were recommended. Six revised draft copies are to be sent to ORO for review prior to final publication and distribution. Copies of the draft will also be circulated for internal review and for preliminary dissemination of information to AEC-IDO.

Criticality safety rules and procedures have been prepared for the storage of 243.5 kilograms of uranium foils in the LPT Facility storage vault and for the handling of these foils during the assembly and disassembly of the mockup fuel cartridges. Procedures are also being prepared for stripping and replacing the Teflon-like coating on the foils. These rules and procedures have not yet been approved by either AEC-IDO or ORO.

12.1 CRITICAL EXPERIMENT HAZARDS (281)

12.1.1 POSTULATED ACCIDENTS

The Postulated Maximum Credible Accident

The maximum credible accident, as postulated for the 630A critical experiment, is assumed to occur as a result of the following sequence of events. During the course of the experimental program, it is determined that a change in core configuration is necessary. An analytical evaluation of the desired change indicates that reactivity will be increased by about one dollar. To compensate for this added reactivity, an equivalent amount of poison is to be added to keep the reactivity within the limits of the control system. During the changeover, a mistake in the sign of the change is made and a positive contributor is added. The resulting net increase in system reactivity in now approximately two dollars.

At the next scheduled startup, the moderator water is pumped into the core at the normal rate of 150 gallons per minute. The rate of reactivity insertion decreases with time in accordance with calculated curves shown in Figure 12.1. With complete failure of the scram system and no interference by the operator, such as draining water or shutting off water supply, the reactor will go critical when the water level is 22.9 centimeters below the top tube sheet. Although the resulting excursion would be terminated by core melting before the normal operating water level is reached, it is assumed that filling is continued to the normal level.

Under these conditions the excursion would peak at 7000 megawatts with 0.73 percent $\Delta k/k$ excess reactivity in the system; i.e., the reactor would not go prompt critical. At

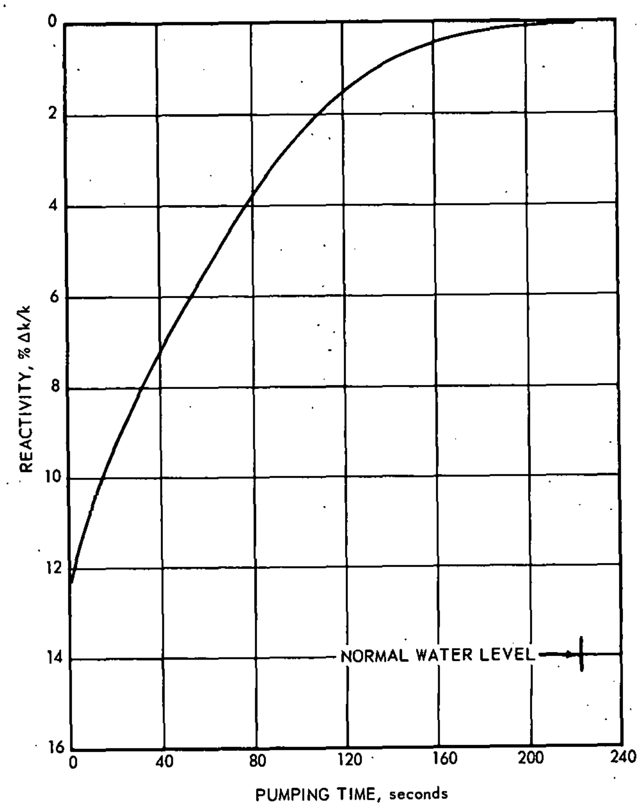

Fig. 12.1 – Reactivity worth of interstitial moderator as a function of pumping time

22.95 seconds after criticality, the reactor is again subcritical by -0.008 percent $\Delta k/k$. Shutdown would be caused by the melting of about one-third of the reactor core. About 423 megacalories of fission energy would be released with 90 percent of the release occurring within a span of 3 seconds. A total of 1.1 x 10^{20} fission fragments would be generated and about 7.8 x 10^{18} of these would be released from the core.

A startup excursion of the type postulated above could possibly be initiated by other events, but this type of accident is not considered credible for normal day-to-day operations which are not preceded by changes in core configuration. The above power excursion will be considered the "controlling accident" after initial loading, after all subsequent reloadings, and on all startup operations following significant changes in excess reactivity, whether or not compensating changes have been made. "Controlling accident," as used herein, means the postulated accident upon which protective measures and pre-planned emergency measures will be based.

Other Postulated Accidents

An accident of low credibility can be postulated by assuming that one control rod assembly (three poison tips) is withdrawn at the normal shim velocity of 0.597 centimeters per second. Again the scram system is assumed to fail and the moderator water is not drained. The reactor is assumed to be operating at a nominal power of 1 watt, and the subject rods are assumed to be fully inserted at the initiation of withdrawal action. The worth of three rods is about 0.3 percent $\Delta k/k$ and a uniform axial incremental worth is assumed.

Complete withdrawal requires 119 seconds. During this period the reactor power rises on an increasingly shorter period until it peaks at 50 megawatts at 203 seconds after start of rod motion. The minimum period is 8.78 seconds. As in the postulated MCA, the excursion is terminated by meltdown of the central one-third of the core. This accident would result in the release of 248 megacalories of fission energy and the generation of 6.4×10^{19} fission fragments. A total of about 4.5×10^{18} fission fragments are released from the core.

Another possible initiating event is the flooding of a number of the fuel assemblies by failure of the 0.060-inch-thick stainless tubes which separate the fuel assemblies from the water moderator. The concurrent flooding of more than one tube, coupled with failure of the scram system, is not considered credible because the tubes are not subject to significant loads, and their tops are above the top of the moderator tank.

The flooding of one tube would result in the addition of 0.14 percent $\Delta k/k$. Again assuming failure of the scram system and lack of corrective action (such as draining the moderator tank) by the operator, a destructive power excursion would follow the development of such a leak. Assuming steady state operation at 1 watt and a large enough leakage rate, an accident can be postulated having roughly the same profile and yield as the rod withdrawal accident described above.

The credibility of both the rod withdrawal and leakage accidents suffers because of the long time period during which the operator can take corrective action. The tank drain system would provide a means of reducing system reactivity in time to terminate the excursion before any damage was sustained.

12.1.2 RADIOLOGICAL HAZARDS

The radiological hazards discussed in the following sections have been estimated for the postulated maximum credible accident (the startup accident). The doses for the rod withdrawal or fuel cell flooding accidents would be about a factor of 2 lower.

Prompt Doses

The doses resulting from direct and scattered radiation from the reactor are estimated from data generated during tests of the Hot Critical Experiment reactor at the LPT. Appropriate corrections have been made for the fact that the 630A critical experiment will be operated in the I. C. cell rather than the C. E. cell and for differences in test cell and reactor shielding. As shown in Figure 12.2, the doses in all normally accessible areas in and surrounding the test facility will be below significantly damaging values except for the fenced area immediately outside of the cells, and within the cells themselves. Access to these areas will be controlled in accordance with a pre-arranged safety plan.

Effluent Doses

The following assumptions are made in order to estimate the doses resulting from fission products expelled from the core.

1. About five percent of the noble gases, one percent of the halogens, 0.8 percent of the volatile solids, and 0.06 percent of all other solids formed in the excursion are released from the reactor.
2. While plateout within the reactor was assumed in determining the above release fractions, no additional allowances were made for removal within the test cell or cloud depletion during downwind transport.
3. Release from the test cell is via the exhaust outlet in the roof, with a rate governed by the 2400 cfm fan capacity. Uniform mixing is assumed within the cell so that the release rate decreases exponentially with time.

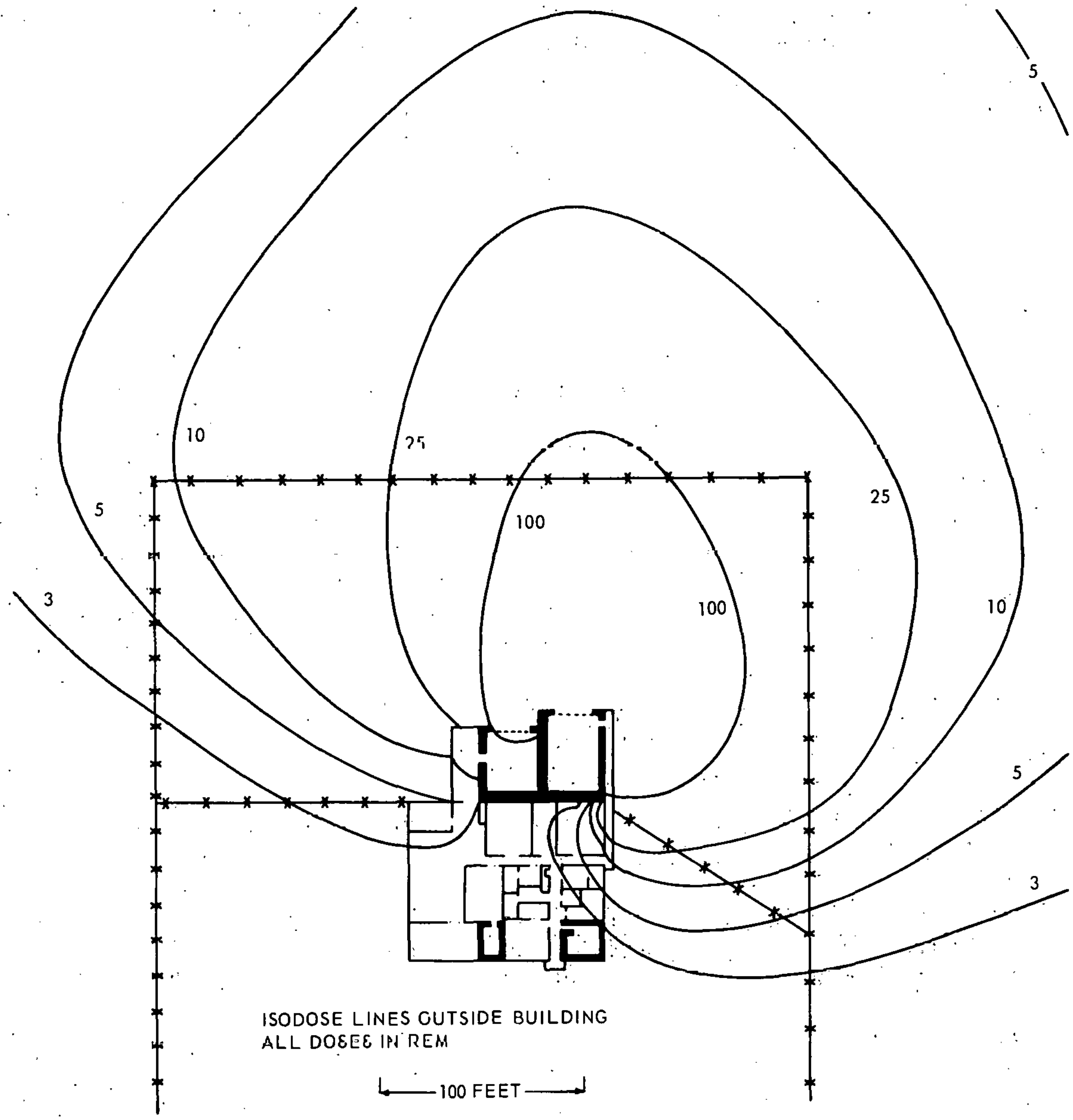

Fig. 12.2—Prompt dose from the maximum credible accident—630A critical experiment

Effluent doses are calculated by means of equations used successfully in meteorological control procedures used during the aircraft nuclear propulsion testing program at nearby facilities. Basically, these are modifications of Sutton's atmospheric diffusion expression.* The biological terms are those recommended by the International Commission on Radiological Protection.

Effluent doses to personnel remaining within the facility could occur during periods when an eddy forming in the lee of the test building brings the effluent down to a point outside of the ventilation system intake louvers. Contrary to established emergency procedures, the ventilation system is assumed to continue in operation so that the contaminated air is drawn into the building. A very pessimistic estimate of cloud diffusion is used

*Sutton, O. G., "Micrometeorology," New York, McGraw-Hill Book Company, 1953.

in place of the Sutton term of the dose equations, which cannot be directly applied in this case. The resulting doses are not considered to be unacceptably high (see Table 12.1) and they can be reduced drastically by shutting down the ventilation system.

TABLE 12.1

INFINITE CRITICAL ORGAN DOSES FROM EFFLUENT EXPOSURE WITHIN THE TEST FACILITY

Critical Organ	Dose, rads
Thyroid	31.7
Whole Body	3.0
Lung	3.6
Bone	0.01

Outside of the facility, the doses at inhabited locations will depend upon wind direction and speed as well as the stability of the atmosphere. Prevailing winds are southwesterly (roughly toward the nearest offsite inhabited areas), but northerly winds are also quite frequent. The typical diurnal cycle results in southwest winds at night. Lapse conditions, usually persist in the daytime while inversion (poor diffusion) conditions occur almost nightly.

Effluent dose calculations have been performed for four different wind speed groups and four stability classifications. For simplicity, however, only the ends of the spectrum will be defined herein by presenting a typical strong lapse with a moderatly strong wind and a strong inversion with a light wind. Effluent doses to the thyroid, lung and whole body as a function of distance downwind are shown in Figures 12.3 and 12.4 for these conditions. Calculations were also made for the dose to the bone, but due to the low reactor inventory this critical organ is of no concern. Since the effluent release from the facility occurs over a fairly long time, the cloud centerline dose predicted by the Sutton equation would result in over-estimation of the dose to a stationary receptor. This is true because directional meander of the cloud would shift the centerline back and forth around the mean. Compensation for this effect has been accomplished by assuming uniform variation within a 22-1/2^{o} sector. The effluent doses to unprotected persons as close as the STPF are well within acceptable bounds and can, of course, be further reduced if the receptor evacuates or seeks shelter. Inhalation and cloud immersion doses off-site are low enough to be of little concern.

Of special interest in the case of off-site effects is the possibility of additional thyroid dose due to ingestion of milk from cattle grazing on contaminated pasture land. Infants are the receptors of interest in this case because the mass of the thyroid gland is less, which results in a greater concentration of radio-iodine per unit mass, while the daily intake in milk may equal that of an adult. Although there are no Grade A dairy cattle in the areas immediately surrounding the NRTS, a number of farm families in the Mud Lake Basin keep milk cattle for their own use, selling the surplus to dairies for the manufacture of cheese and other process products. Because of the semi-arid climate, pasture lands in the area will not fully support milk cattle; that is, pasture grazing is supplemented by grain feeding. During the colder months grazing accounts for little or none of a milk cow's food intake. Because of this variability in grazing intake, it is difficult to estimate the contamination of milk which would result from a fission product release. If it is pessimistically assumed that 100 percent of a cow's intake is from grazing, calculations indicate that the ingestion dose to a child's thyroid can be as great as 100 to 300 times the inhalation dose calculated for the same conditions. Thus, for the worst meteorological conditions, the infinite ingestion dose from the MCA could be 10 to 30 rads at the distance of the nearest farm having milk cattle (1.1×10^4 meters). If desired, the infinite dose can easily be reduced by temporarily restricting milk consumption. Terminating consumption after one day would reduce the dose by a factor of 11.

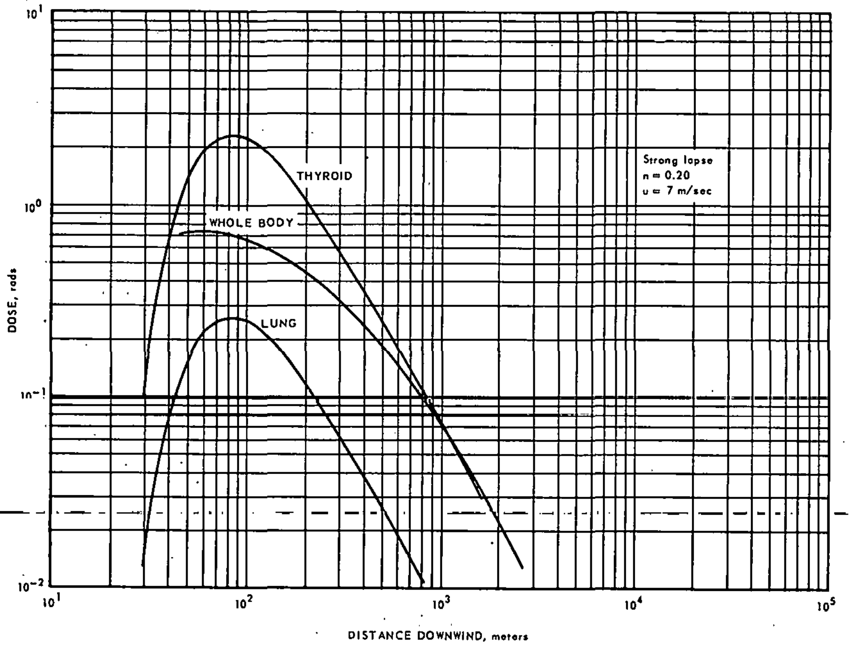

Fig. 12.3 – Effluent dose from the maximum credible accident – 630A critical experiment – typical lapse

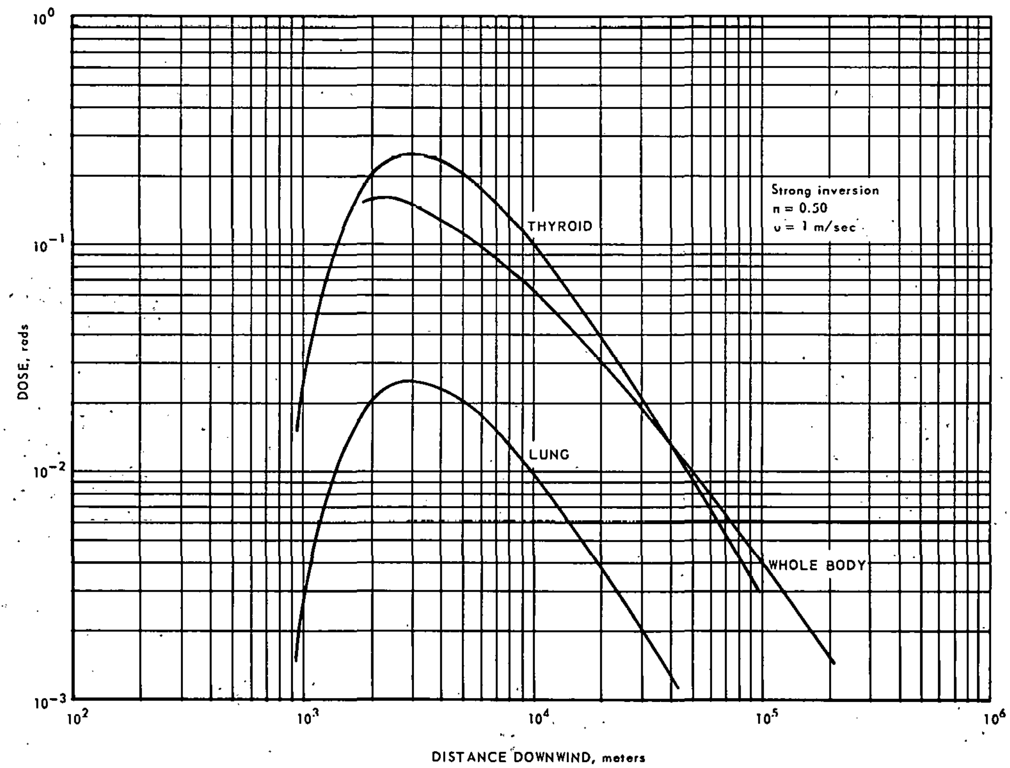

Fig. 12.4 – Effluent dose from the maximum credible accident – 630A critical experiment – typical inversion

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13. FACILITIES AND EQUIPMENT

(330)

Little, if any, modification to the Low Power Test Facility or existing nuclear instrumentation will be required for the 630A critical experiment. Some modifications to the control and safety circuitry, and to the counting room, are planned. These modifications are discussed following a general description of the Low Power Test Facility, including the Initial Criticality (I. C.) and Critical Experiment (C. E.) cells. Preliminary studies on the Power Test Facility, and the possible location and arrangement of the test cell are discussed.

13.1 LOW POWER TEST FACILITY (331)

The 630A critical experiment will be conducted at the Low Power Test Facility (LPTF), located in the northeastern part of the National Reactor Testing Station (NRTS), Idaho Falls, Idaho. Figure 13. 1 is an aerial photograph showing the general configuration of the LPTF and its location relative to other NRTS facilities.

The LPTF provides a complete and essentially self-contained facility for conducting all operations associated with critical experimentation and low power reactor testing. All normal services and support equipment are provided and the facility maintains a high degree of versatility with respect to the type or size of reactor to be tested and the types

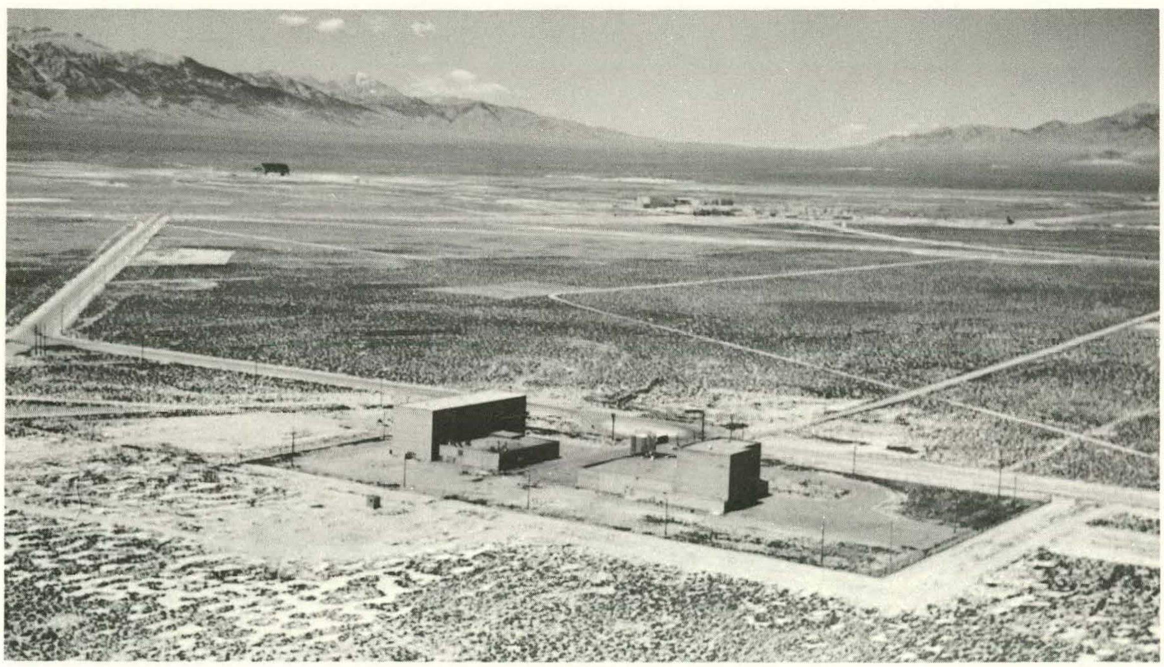

Fig. 13.1 – Aerial view of LPTF (Neg. U-3343-4)

of tests to be performed. Consequently, little, if any, facility modifications will be required for the 630A critical experiment testing.

Figure 13. 2 is a floor plan of the facility showing the general layout of office, shop, test, and service areas. The test cells, counting room and the SS storage vault are of poured concrete construction. The remainder of the facility has a structural steel frame with exterior walls of cinderblock construction, and interior walls and partitions of wood frame and wall board construction.

Shielding between the test cells and the occupied portions of the building is provided by a 5-foot-thick concrete wall which extends to a height of 30 feet and then steps to a reduced thickness of 18 inches for the remainder of the cell's height. All other cell walls are 2-feet thick with the exception of the wall separating the two cells which is 4-feet thick; and a portion of the C. E. cell wall which is 3-feet thick. The ceilings of both cells are of reinforced concrete, 1-foot thick with built-up roofing. The ceiling height is 43. 5 feet and the I. C. and C. E. cells are equipped with overhead bridge cranes having capacities of 5 and 10 tons respectively. Both cells have entrances 22 feet wide and 30 feet high, with industrial-type roll-up doors. The entrances open onto an asphalt ramp which connects with the normal facility vehicle roadway. Heavy equipment is delivered to, and positioned in, the cells using truck and trailer units. Each cell is provided with 120-volt and 480-volt electrical outlets. A total of 1000 kva of electrical power is supplied through separate breaker panels in the cells to provide for electrically heated experiments. Industrial water and 125-psi air is supplied to each cell and, in addition, de-ionized water is supplied to the I. C. cell. Floor drains are provided for contaminated and process liquid waste, with the contaminated liquid being drained to a 3000-gallon underground storage tank outside of the facility. Lines for the control circuitry and nuclear instrumentation are brought into the cells through conduits to a terminal box in each cell. Connections between the test device and the facility are made at this location.

The control rooms are directly behind the cells, shielded by the 5-foot-thick wall. Each control room includes a primary reactor control console and a secondary rack containing supplementary equipment such as: recorders, power supplies, and amplifiers. Closed-circuit television is provided between each cell and its associated control room. Electrical conductors from the control room pass into cable trenches in the floor and then to a master terminal board situated below floor level adjacent to the control rooms. Interconnections are made here and the conductors then proceed through conduits into the cell. A surplus of conduits has been installed to provide for a variety of test devices.

Since a major portion of the experimental program involves the counting of activated material, a counting room equipped with a high-capacity counting system is provided. The counting room has 16-inch poured concrete walls to maintain the background to less than 0. 1 milliroentgen per hour. The counting equipment is further described in section 13.1.3.

The assembly room where fuel elements are assembled has a linoleum floor for easy decontamination, and a stainless steel sink for decontamination of small components. The floor drain and the sink drain, as well as the wash bowls in the adjacent change rooms, drain into the underground contaminated liquid storage tank outside the building. Adjacent to the assembly room is an SS storage vault.

Other operating areas in the facility include office space for approximately 10 people and a shop area for performing instrument maintenance, repair, and storage.

The north portion of the building is used to house service equipment: two boilers, an auxiliary diesel generator, heating and ventilating equipment, and electrical switch gear.

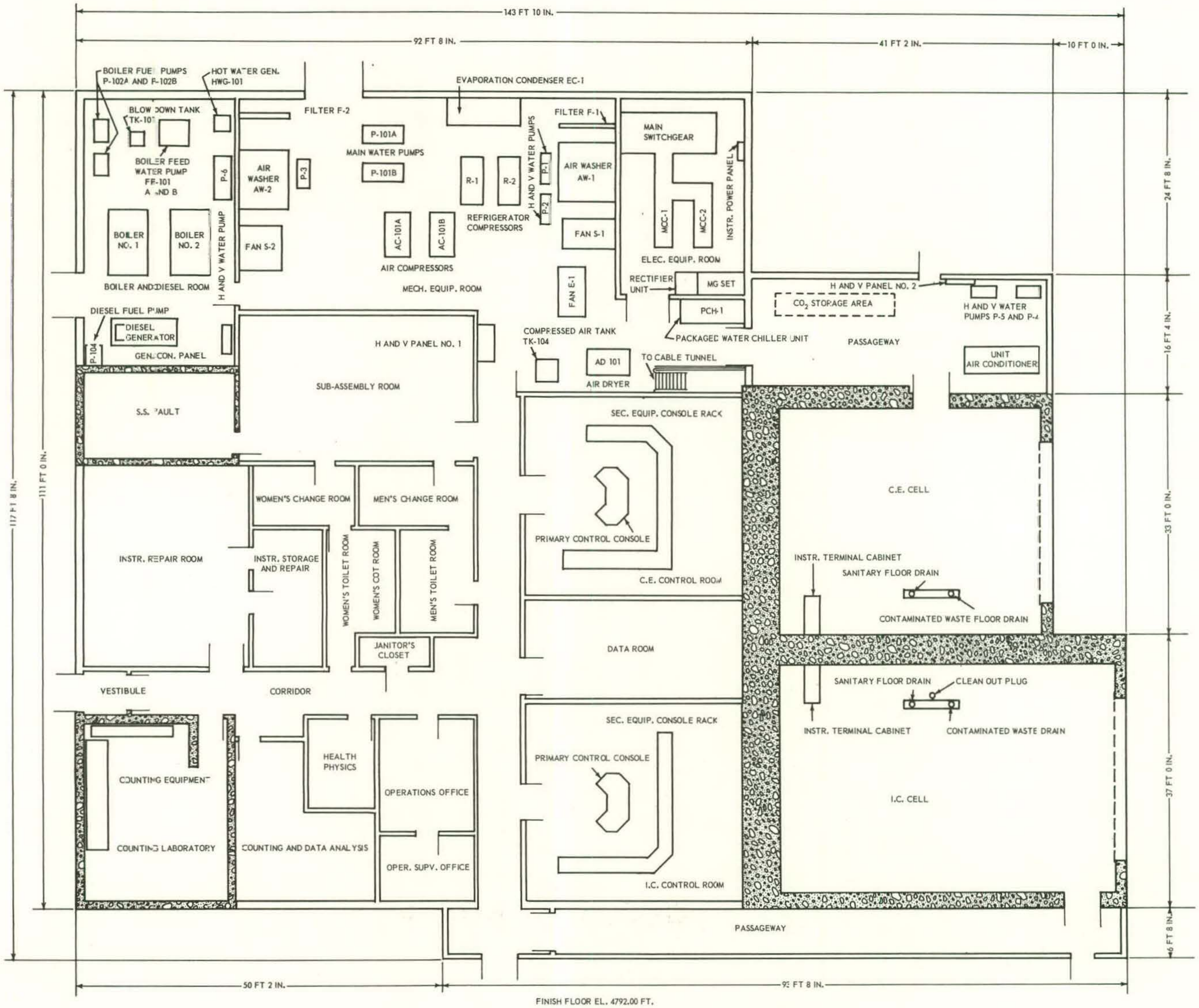

Fig. 13.2 – Floor plan of Low Power Test Facility

The electrical system consists of a 13. 8 kv primary circuit and a 480-volt secondary circuit obtained through outdoor transformers. Four transformers are furnished: one for the well house, one for instrument power, one for the heating panels in the cell, and one for general lighting and power. Appropriate 440-volt switch gear is used for control and protection of the electrical circuits. In addition to the normal electrical circuitry, a diesel generator is provided which automatically picks up a portion of the electrical load in the event of a power failure.

The water system consists of a well, located approximately 700 feet east of the main facility. The well house contains a chlorination system and a 400-gpm electrically driven pump with a 40-horsepower gasoline engine coupled through a disconnecting clutch for emergency use. Water is stored in a 75, 000 gallon tank located adjacent to the main building and water pressure is maintained by the use of two centrifugal pumps, one of which runs continuously.

The ventilation system consists of refrigerated air or steam heating for the control rooms, the data room, and the counting room, and evaporative cooling or steam heating for the remainder of the equipment and control portion of the building. The I. C. cell has a steam heating system and a ceiling exhaust, and the C. E. cell has a separate refrigerated air and steam heating system.

For security purposes, the facility is surrounded by a fence with one personnel entrance and one vehicle entrance, both of which are guarded by security personnel. A signal system is incorporated in the facility to notify the main guard station in case of fire, tampering with the vault or unauthorized opening of doors. For radiation protection, fences are installed adjacent to the rear of the test cells and extend to the security fence in order to prevent access to the area outside the test cells during operation. The fences are equipped with electrical interlocks which signal the operating crew in the event that any of the gates is opened. Surrounding the security fence is a radiation fence that prohibits an approach of less than 1000 feet from the unshielded side of the test cells.

13. 1. 1 NUCLEAR INSTRUMENTATION

Figure 13. 3 shows the nuclear instrumentation to be used during the 630A critical experiment. These instruments are the same as have been used for past reactor testing at the facility and little, if any, instrument modification will be required. Figure 13. 4 shows the general arrangement of the control room.

The nuclear instrumentation consists of nine channels of neutron-level and period-sensing devices. Included are: two linear flux channels, two log flux channels, two log count rate channels, and three count rate channels. These instruments provide a visual indication of reactor power and period, three automatic level scrams, two automatic period scrams, and interlock circuitry.

The linear flux channels are operated by compensated ion chambers with chamber voltage being supplied by either batteries or an electronic power supply. Chamber output current is measured by a zero-stabilized micromicroammeter having a range from 10^{-12} to 10^{-3} amperes. Chamber output current is indicated by meters situated on the instrument chassis, and also by strip chart recorders situated in the primary and secondary control consoles. Each linear flux channel is included in the scram bus and provides automatic high-level scram protection.

The log flux channels employ compensated ion chambers which are identical to, and interchangeable with, the chambers used on the linear flux channels. Chamber output current is amplified by a logarithmic amplifier and indicated by a 4-decade log flux meter situated on the amplifier chassis. A -30 to +3 second period meter is also situated on the

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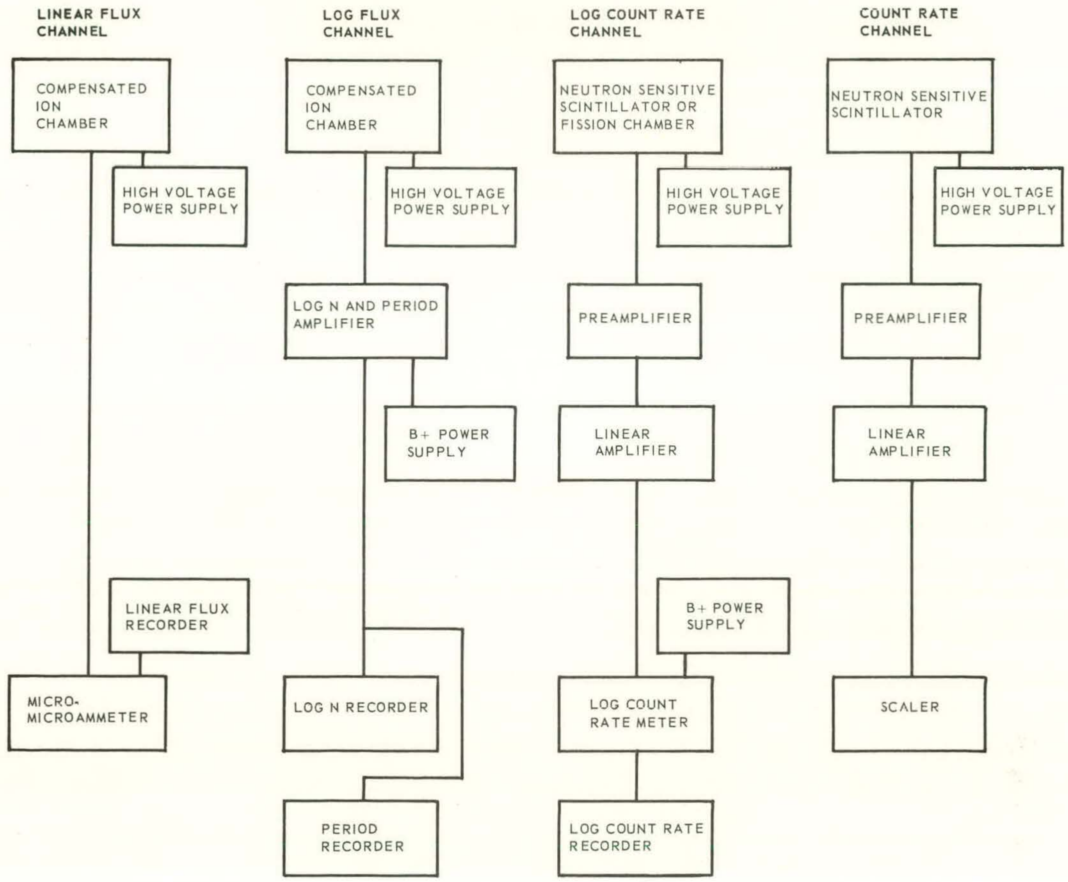

Fig. 13.3 – Block diagram, LPTF nuclear instrumentation

Fig. 13.4 – Control room (Neg. U-3963-10)

amplifier chassis. Log flux and period information is retransmitted to strip chart recorders and to additional panel meters situated on the primary console. The amplifier is equipped with trip devices which serve to prohibit rod withdrawal when the period becomes shorter than 15 seconds, and which initiate a scram when the period becomes shorter than 7 seconds. An additional interlock device is provided which operates in conjunction with the "source-in" limit switch and prevents rod withdrawal at low power levels unless the source is fully inserted.

Log count rate channels 1 and 2 are operated either by fission chambers or neutron sensitivity photomultipliers. Chamber output is preamplified at the chamber location and is amplified by a linear amplifier in the control room. The linear output is converted to a logarithmic output by a log count rate circuit and is indicated by 5-decade strip chart records and panel meters. Channel 1 is equipped with an automatic level trip device which is included in the scram bus and which serves to initiate a scram at a preset count rate.

Count rate channels 1, 2, and 3 are identical and are operated by neutron sensitive photomultipliers or BF_3 proportional chambers. The amplified signals from these chambers operate electronic scalers which are used in conjunction with an automatic timer to determine low-level count rate.

13. 1. 2 CONTROL AND SAFETY CIRCUITRY

The control and safety circuitry to be used on the 630A critical experiment is a modification of the circuitry used for previous reactor operations at LPTF. Essentially, all circuits and control equipment are in existence in the facility; however, modifications and additions will be made which include:

1. Provision for changing from two-phase to single-phase actuator motor drives.
2. Provision for 20 shim-scram actuators.
3. Transfer of the fill and drain system controls to the primary control console.
4. Alteration of the mode of operation of the actuator position indicator lights.

Prior to the start of these modifications, a considerable effort will be required to update and re-establish a satisfactory print file for the control room circuitry. This work is currently in progress and is about 50 percent complete.

An inspection of the wiring revealed several areas where interconnections had deteriorated (the taper pin connectors had sagged allowing the pins to drop out and some pins were very easily loosened). It was also noted that several of the connector blocks were quite inaccessible. These items are scheduled to be corrected in conjunction with the modification rewiring. The checkout of the existing wiring is 50 percent complete.

The control system includes actuator drive circuitry, position-indication circuitry, and actuator-selection circuitry. The safety system includes: the scram bus, the auxiliary scram system, withdraw interlocks, and bypass circuits.

Figure 13. 5 illustrates actuator drive, selection, and position-indication circuitry and Figure 13. 6 illustrates the main scram bus circuit.

13.2 COUNTING ROOM

The equipment in the counting room is being reactivated with the major attention being directed to the semi-automatic foil and wire counter.

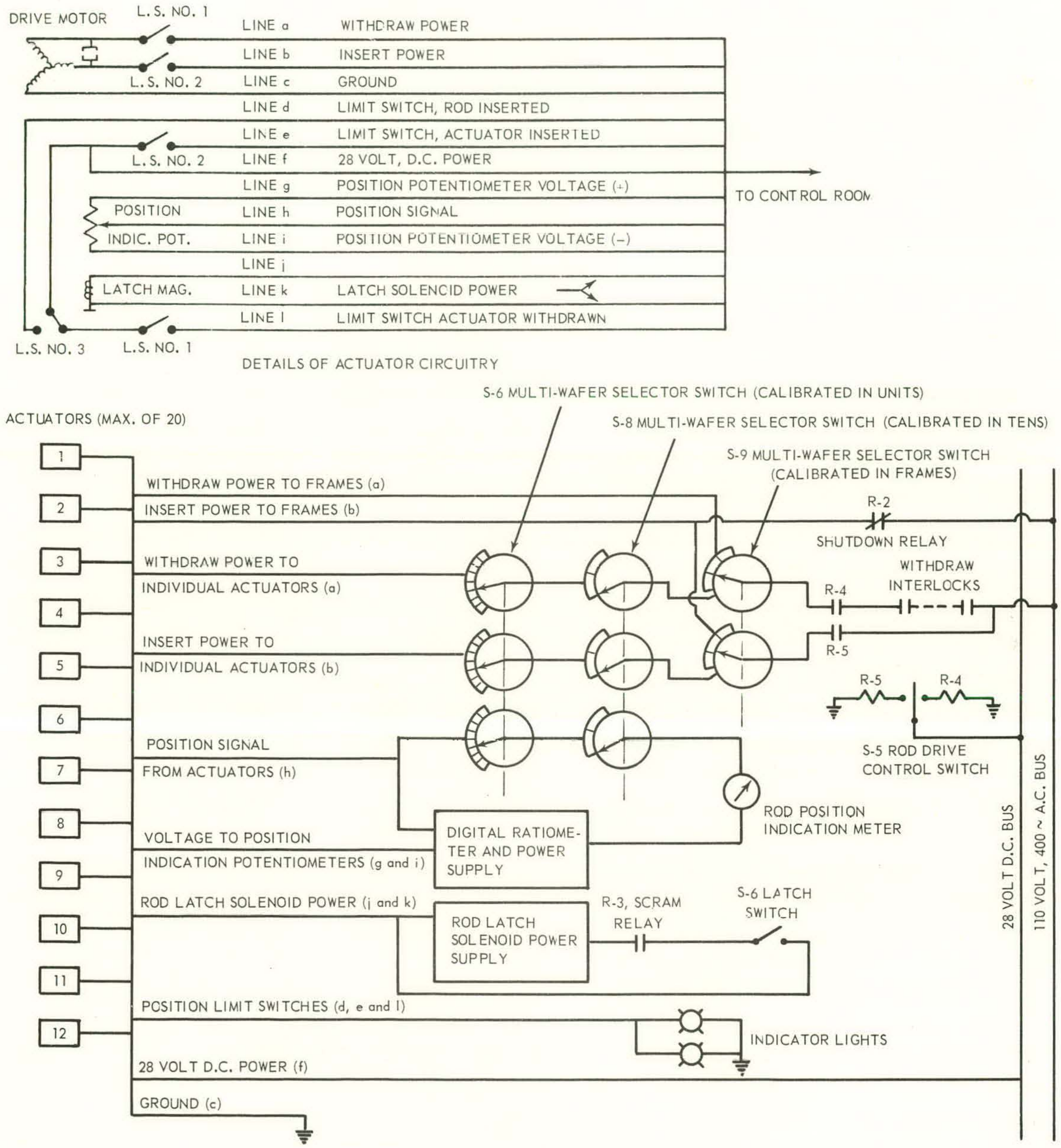

Fig. 13.5 – Control system

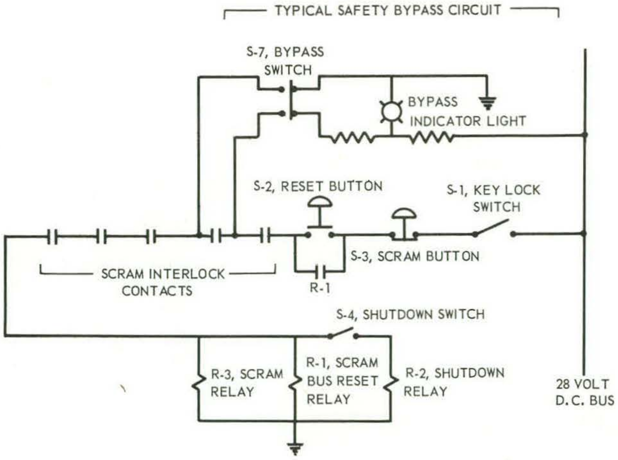

Fig. 13.6 – Scram bus

A complete rewire and layout of the basic control chassis has been undertaken to permit easier maintenance and give better reliability. Figure 13. 7 is a view of the front panel showing the controls. Figure 13. 8 is a top view which shows the internal wiring, connections, and component layout as modified.

A complete interconnection rewire job is being performed, with wires fully identified at every terminal point and laid out in open ducting, to allow quicker and easier maintenance on the system. The entire system is being cleaned up, and faulty and doubtful components replaced. This work is 90 percent complete.

During previous operation, higher foil activations were desired and a design project was begun to reduce the dead time of the counting system. A prototype of a high-speed (10+ megacycle pulse-pair resolution) nuclear system was built. This prototype consists of a modified beta scintillation head to replace the gas proportional detector, a nonlinear, transistorized preamplifier to replace the tube preamplifier- and amplifier-discriminator, and modification of the readout scalers for high speed operation. The old tube-type components that will be replaced with this new system are shown in Figure 13. 9. Figure 13.10 shows the prototypes. A contract was placed with an outside vendor, Technical Measurements Corporation, for construction of the heads and preamplifiers. Delivery of these units is expected momentarily. Their installation and checkout will follow the completion of the control system functional checkout.

The LPTF foil system provides for parallel operation of six counting channels and the sequential recording of the integrated count rate from each channel. A seventh counting channel provides a means of controlling the counting time interval so that count rates in the six counting-recording channels are normalized and decay-corrected relative to a "standard."

13.3 POWER TEST FACILITY (332)

During this report period, preliminary planning of the power test facility was begun. A detailed scoping document will be started during the next reporting period.

The preliminary study indicates that the power test facility should be a new facility with the steam generator permanently installed for the following reasons:

1. The continuity and long duration of the 630A nuclear steam generator testing makes it impractical to share the test facility with another reactor.

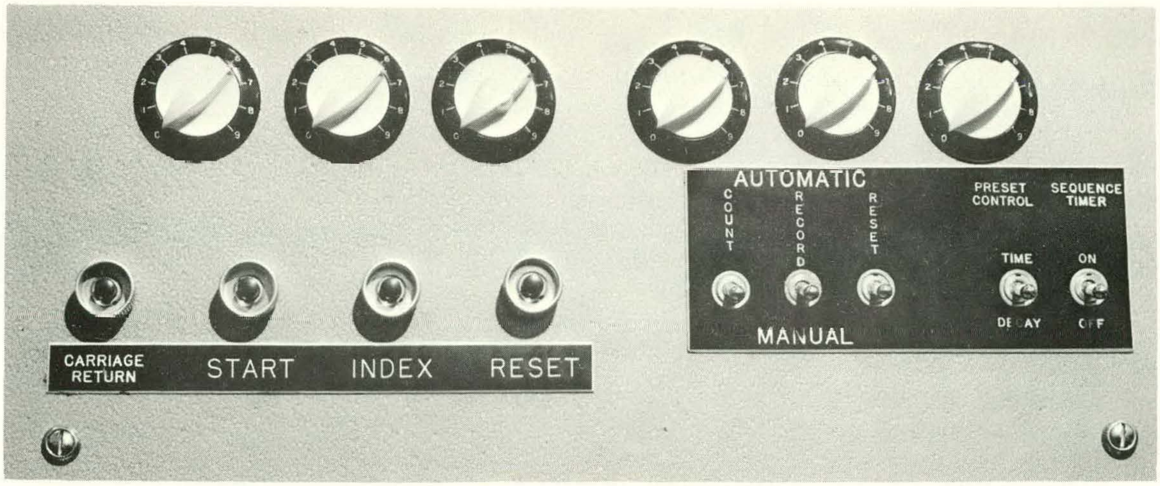

Fig. 13.7 – Control chassis front panel (Neg. U-4268-5)

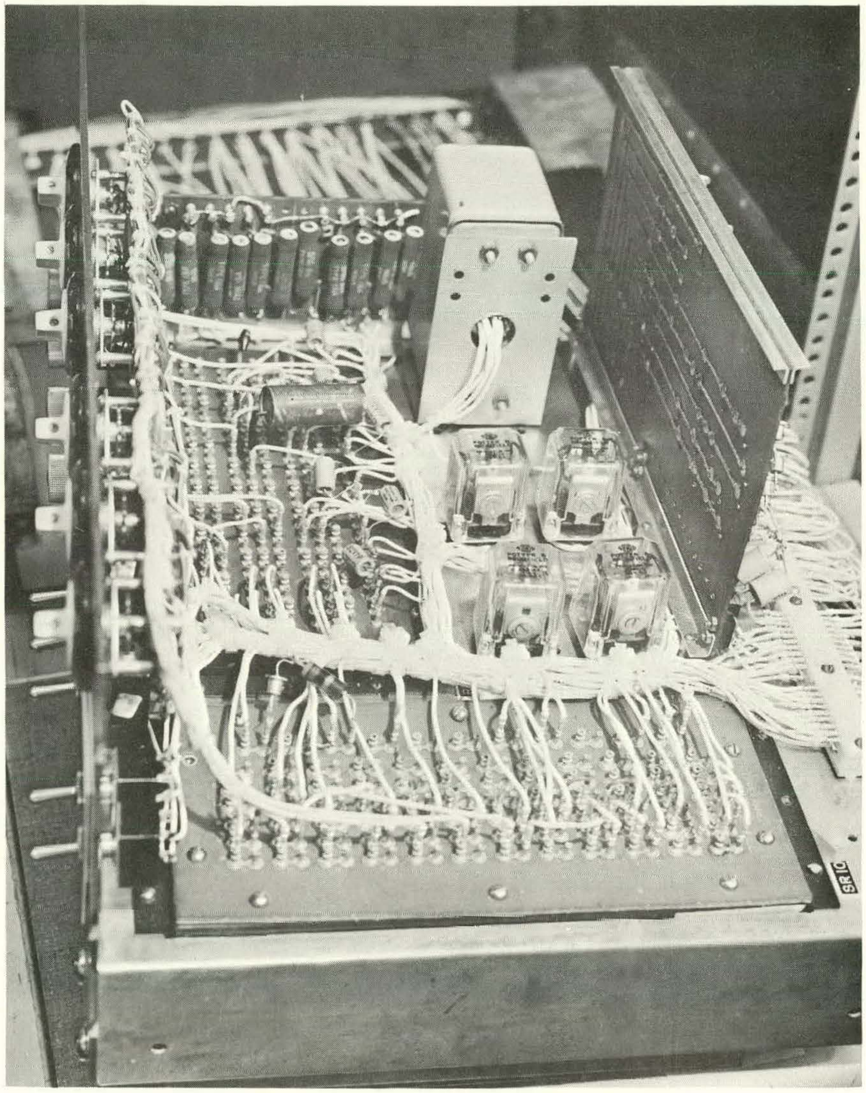

Fig. 13.8 – Control chassis, top view (Neg. U-4268-10)

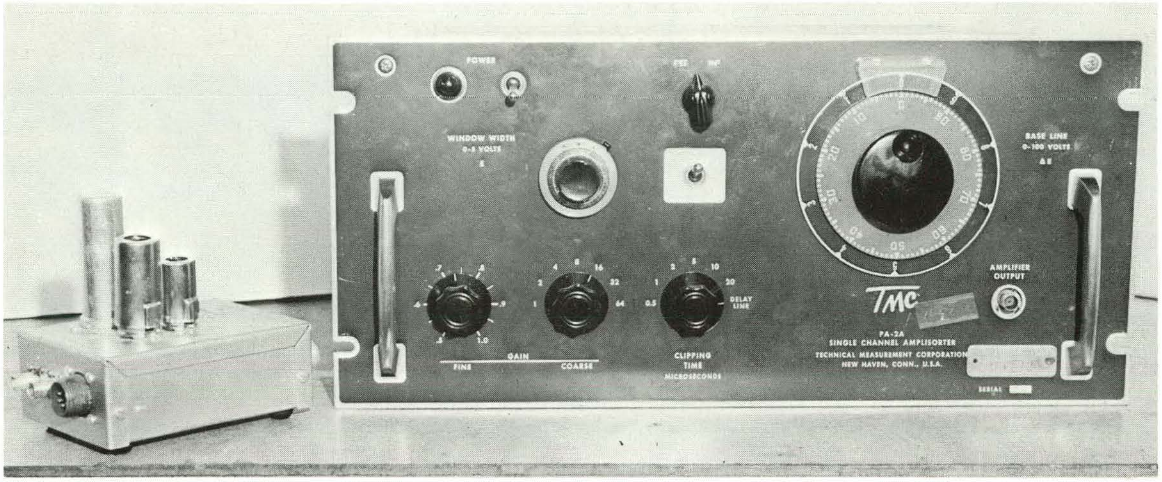

Fig. 13.9 – Counting system tube component (Neg. U-4268-9)

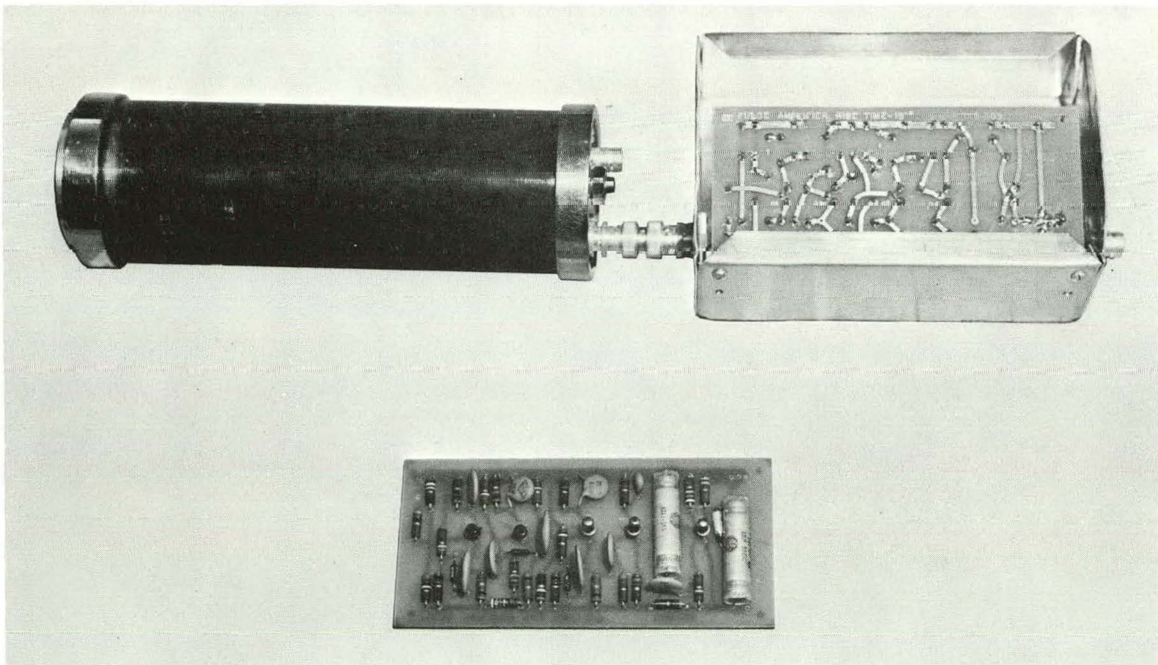

Fig. 13.10 – Counting system – transistorized prototypes (Neg. U-4268-8)

2. All of the existing "north end" test facilities have reactor tests scheduled.
3. A new 630A power test facility would be a relatively simple and inexpensive facility to build.

Although the specific location of the MPTF (Maritime Power Test Facility) has not been chosen as yet, it is proposed that it be a part of the LPTF-STPF complex so as to share some of the existing facility services, which should result in greater economy of operation. The two most likely locations are shown on the plot plan in Figure 13. 11. The choice will be made after a detailed appraisal of such factors as radiation hazards, ease of access to the test cell with a large trailer, ease of common connection to existing facility services, etc.

The conceptual layout of the power test facility is shown in Figure 13. 12 and is the basis for the following discussion of the test cell.

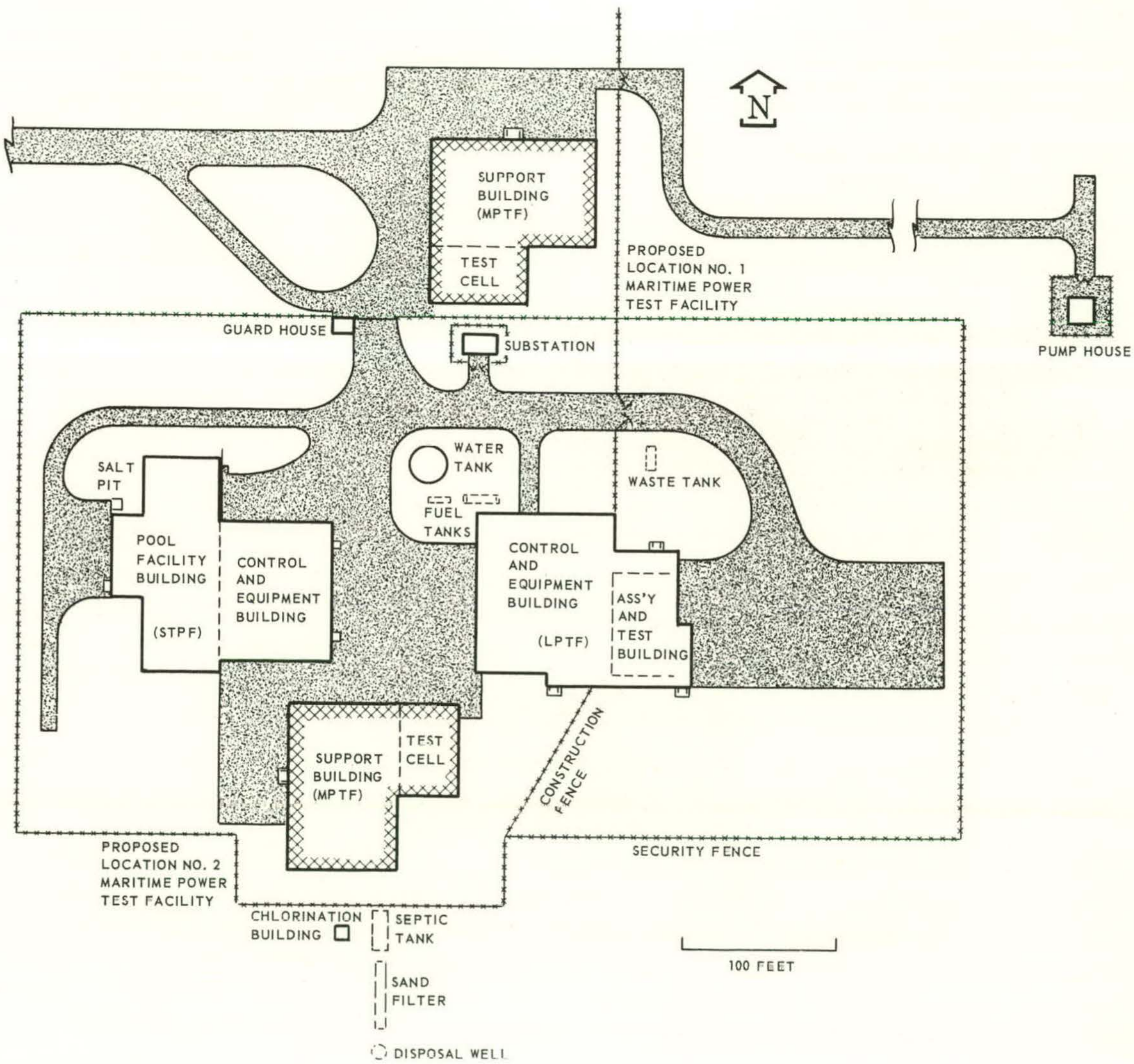

Fig. 13.11 – Plot plan LPTF-STPF complex

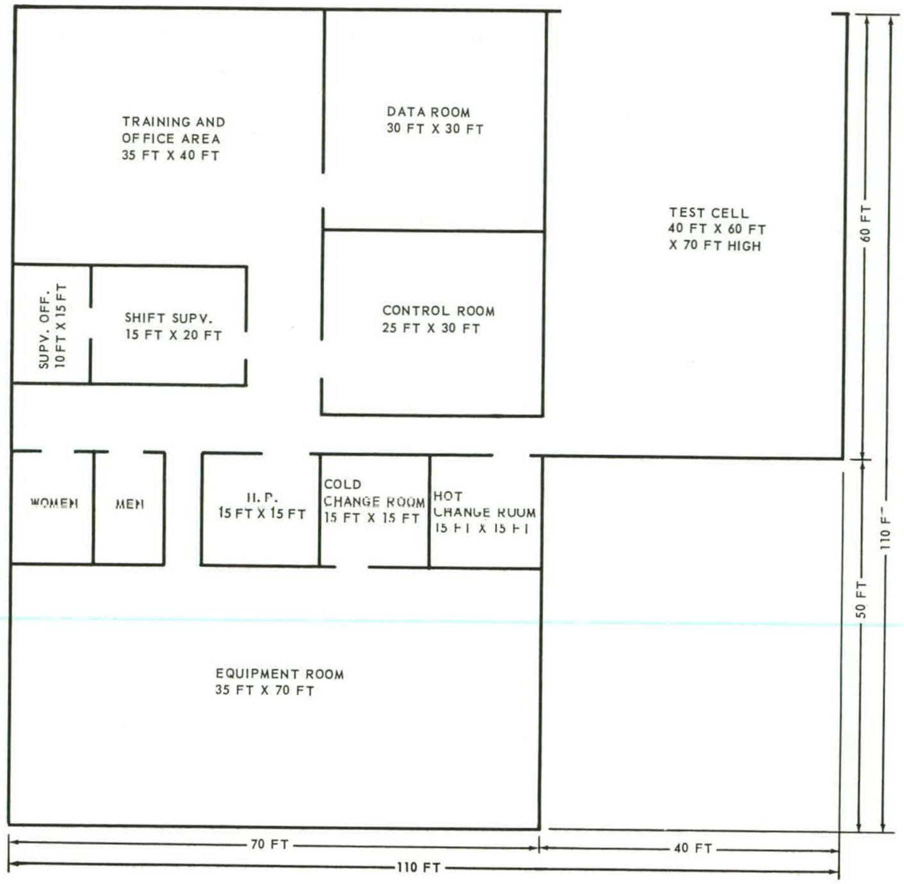

Fig. 13.12 – Conceptual layout of maritime power test facility

13. 3. 1 TEST CELL

The test cell is essentially a standard high bay with a support structure for the steam generator built into the floor.

The framing will be of steel and the siding a single layer of corrugated Transite* or similar material. The cell will be heated to prevent freezing of the water systems during the winter months. The 40 foot by 60 foot floor area will allow for the placement of the steam generator in one end of the cell and a clear working area at the other end of the cell. This clear area will be utilized during assembly and disassembly operations.

Installing or removing the reactor core and shield plug requires that the cell be about 70 feet high and be equipped with a 100-ton bridge crane.

13. 3. 2 SUPPORT BUILDING

The support building is comprised of three general areas – an equipment area, a control and instrumentation area, and an office and training area. The equipment area is about 35 feet by 70 feet and will contain such systems as the demineralizer, air supply, waste handling, auxiliary diesel generator, auxiliary cooling system, and facility heating and ventillation equipment. A pipe gallery will connect this room with the test cell.

The control and instrumentation area is comprised of a control room and a data room. The control room will contain all of the controls and instrumentation necessary to operate

*Registered trademark Johns-Manville.

the steam generator heat dump and auxiliary equipment. This room will be made to resemble a shipboard control room as much as possible. The data room will contain the instrumentation and data readout devices that will be required to check the performances of the prototype power plant.

Steps have been taken to secure the data system from the Initial Engine Test for use with the 630A. This data system, although old, will be repaired, and used to record temperatures and pressures in digital form for reduction on a computer.

The third general area will supply the office space required for the permanent personnel as well as the office and training space for the crews that are sent to Idaho for operator training on the 630A steam generator.

FLIGHT PROPULSION LABORATORY DEPARTMENT

GENERAL ELECTRIC