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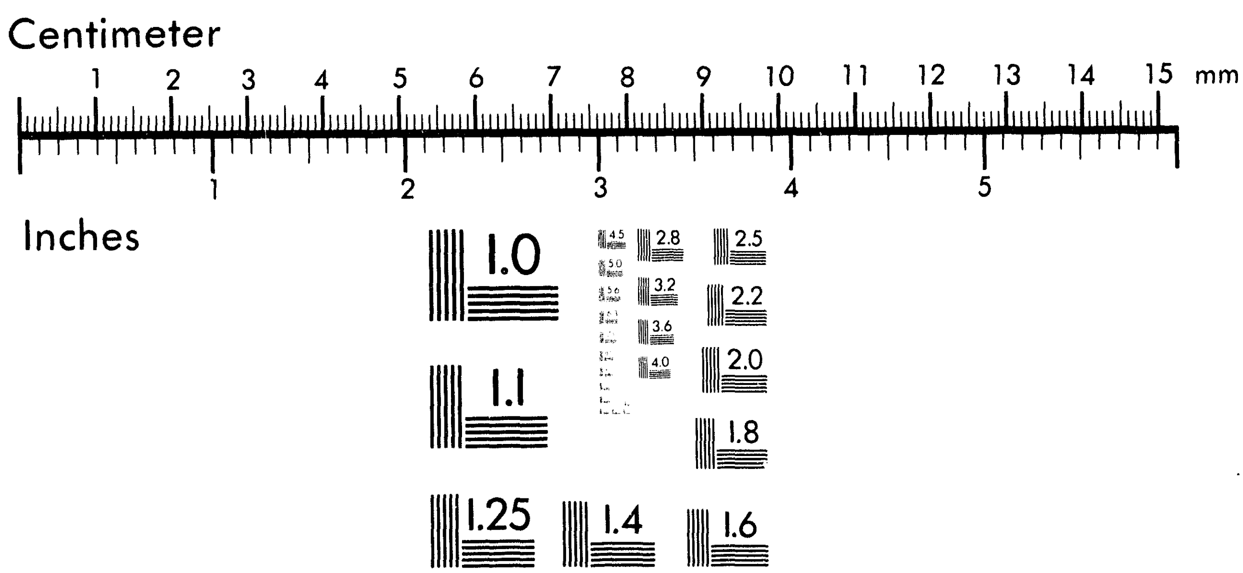

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HANFORD ATOMIC PRODUCTS OPERATION - RICHLAND, WASHINGTON

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PRODUCTION TEST IP-376-D

IRRADIATION OF MGCR-HDR-3 TEST ELEMENT

AUTHOR

EC Bennett

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PRODUCTION TEST IP-376-D

IRRADIATION OF MGCR-HDR-3 TEST ELEMENT

November 29, 1960

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

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PRODUCTION TEST IP-376-D
IRRADIATION OF MGCR-HDR-3 TEST ELEMENT

OBJECTIVE

The objective of this test is to irradiate a test fuel assembly for the MGCR (Maritime Gas Cooled Reactor). The irradiation of this assembly will be carried out in the DR-1 Loop under controlled conditions to determine the feasibility of the heterogeneous 19-rod bundle fuel element concept. Specific areas of interest are:

1. Diffusion of fission products through metal cladding.
2. Fission gas retention of fuel bodies.
3. Dimensional stability of fuel bodies.
4. Satisfactory performance of the creep-shrink process for maintaining pellet position in the fuel pin.

BASIS

The design, construction, and installation of the DR-1 Loop was jointly sponsored by the Maritime Reactors Branch and the Army Reactors Branch of the Division of Reactor Development, Atomic Energy Commission. The facility is now supported solely by Maritime Reactors and will be devoted to tests in support of the Maritime Gas Cooled Reactor Program. This test will be performed at the request of, and under the technical direction of, General Atomic, Division of General Dynamics Corporation. General Atomic is responsible to Maritime Reactors for the development of the MGCR and related gas cooled reactor concepts. The Hanford Irradiation Request No. GA-C-184(1), requesting this irradiation, has been approved.

SCHEDULE

The test assembly will be installed during the first outage following the arrival of the test assembly at HAPO. General Atomic plans to ship the assembly early in December 1960. Requested test duration is approximately four months.

COST

Cost Code - 5R51 - XXX.76
Elevator time - none required.
Shutdown time - none required.

DESCRIPTIVE DETAILS OF THE TEST

The test assembly consists of the test fuel element; an enclosure to direct flow across the element and to prevent by-pass flow; a lead tube to carry instrument leads out of reactor to position the element during irradiation and to retrieve the element upon discharge; and a gas-seal component at the out-of-reactor end to form the gas seal between the assembly, in-reactor tube, and instrument leads. (See Figures 1, 2, 3, and 4 for illustrations of the assembly and test fuel element.)

The test element consists of nineteen rods arranged in a circular cluster and surrounded by a circular shroud. The rods are held in position by support grids so that the rear fuel rod support grid is welded to the fuel rod end caps, thus providing a fixed position for the rods. The front support grid is constructed to provide a slip fit around the fuel-rod end caps, thereby allowing for differential axial expansion.

Both support grids are welded to the circular shroud to prevent rotation of one grid relative to the other. The support grids are welded to support sections which rest on the in-reactor tube. A gas seal is provided by piston rings at each end of the element. The element is connected to a thermocouple lead tube which protects the thermocouple leads, positions the element in the in-reactor tube, and provides a means of withdrawing the element after the test is completed.

The fuel rods consist of Hastelloy-X tubes 0.375 inch OD by 0.010 inch wall thickness, 15.5 inches long, containing UO_2-BeO ceramic fuel pellets. The pellets are sealed into the tubes with quarter-inch thick end caps welded in place, with a Hastelloy-X insert in the tube to eliminate any spacing between the fuel pellets. Voids drilled in the insert provide an expansion volume for fission gas released from the fuel pellets. Eighteen of the rods will be fueled with UO_2-BeO pellets, 0.361 inch OD x 0.5 inch long, 28 pellets per rod. The fuel mixture will be 30% UO_2, (3.65% enriched) and 70% BeO by volume. The center rod will contain fifteen inches of BeO sleeves 0.315 OD x 0.187 ID, but will not contain fuel. The center rod will also contain the element thermocouples to give an indication of the cladding surface temperature. Fuel rod spacing is maintained by a spiral fin spot-welded around each rod, providing a clearance of 0.044 inch between cladding surfaces.

The circular shroud around the rod bundle is made of Hastelloy-X, 0.010-inch wall thickness and 2.106-inch maximum diameter.

The end support pieces are made of Hastelloy-X. The piston ring gas seals of 430 stainless steel are positioned in grooves in the end support pieces. The element is instrumented with five chromel-alumel thermocouples; two to measure inlet gas temperature, two to measure outlet gas temperature, and one in the center rod near the hot end of the element to give an indication of cladding temperature. The actual cladding temperature during operation will be calculated from correlations of the element heat generation, coolant flow, and inlet coolant temperature. The fuel element contains a total of 1230 grams of U-238 (3.65% enriched). Based on extrapolation of Hanford Test Pile measurements of a nuclear mockup the generated power is estimated to be 40 kw. Maximum deviation from this nominal power level is estimated to be $\pm$ 9.2 kw.

The desired reference conditions for this test will be a cladding temperature at the hot end of the element of 1500 F $\pm$ 25 F, and a coolant gas temperature of 1250 F or less at the outlet end of the fuel element. Coolant flow and preheater temperatures will be adjusted to maintain these conditions, provided that loop temperature limits are not exceeded. These loop limitations are:

1. Outer tube wall temperature: 1275 F maximum at 215 psig.
2. Maximum preheater outlet temperature: 1000 F.
3. Maximum reactor outlet temperature: 1200 F.

HAZARDS

The principal hazards to reactor safety and continuity of operation will be loss of coolant flow which may result in a melt-down of the element, and the release of fission products through cladding failure. A detailed analysis of the heat transfer characteristics and the fission product release for the element has been reported (2). (In summary, the element cladding temperature will not exceed the Hastelloy-X melting point of 2350 F unless the element suffers a complete loss of coolant flow, with no emergency cooling, for a period of 3.8 minutes and with the loss of coolant flow occurring at equilibrium reactor power). Response time tests of the loop reactor scram system show that under these conditions the reactor would remain at power a maximum of four seconds after loss of flow. If such a loss of cooling occurred, the inner ring of fuel rods would reach a peak temperature of 2415 F and would probably suffer structural failure, but the damage would not extend to the outer ring of fuel rods due to the higher heat capacity of the outer ring and radiation losses to the tube assembly. Peak temperature of the outer ring is calculated to be 2290 F. The possibility of physical damage to the reactor is extremely remote since the test element is isolated from the reactor proper by the inner tube, gas annulus, and outer tube of the in-reactor tube assembly. None of these components would reach temperatures exceeding their specified limits. (See Table II). Melting points for the test materials[4] are as follows:

Hastelloy-X	2350 F
Stainless Steel	2540 F
BeO	4610 F
UO_2	4530 - 4980 F reported

It is possible that distortion of the test element would be sufficient to make removal by usual methods impossible. In this case, the entire in-reactor assembly would be removed. Procedures and equipment have been developed for this purpose.

In the event of a cladding failure of any type, fission products generated in the fuel bodies would be released to the coolant stream. Such a release would create a potential hazard to personnel and probably would interfere with reactor operation to the extent that a minimum outage would be required for the discharge of the element. The hazard to personnel would consist of increased gamma radiation intensities on the X-0, X-1, and X-2 levels, and possible ingestion hazards due to leakage from the loop to the building atmosphere. The possibility of contamination spread to the surrounding environs exists, although the rate of coolant release to the building exhaust stack can be controlled to maintain the rate of fission product stack emission below limits. The test analysis[2] indicates the fission product release to be about five curies, including 0.3 curies of I_{131}, based on total release of the fission products from two fuel rods. The maximum fission product release, based on failure of all the fuel rods, is estimated at 42 curies, including 2.5 curies of I_{131}.

The loop radiation level is monitored by two ionization chambers located near the loop piping in the process area. For control purposes, a sudden increase of the loop radiation level, as indicated by the monitors, to 100 mr/hr (from nominal values of less than 20 mr/hr) will constitute evidence of a cladding failure, and will make a shutdown of the reactor mandatory.

[redacted]

Since the test element contains BeO, the test was reviewed with Industrial Hygiene Operation, RO, for potential beryllium poisoning hazards to personnel. This review indicated that no significant hazards exist. All the beryllium is present in ceramic form, with a melting point of 4610 F, well above any anticipated element temperatures. The BeO sleeve in the central tube is not subject to gas flow, and any BeO dust would have to traverse a circuitous route to reach the coolant stream. Furthermore, normal clothing and respiratory protection required for handling the test element will provide sufficient protection against the potential beryllium hazards.

OPERATING PROCEDURE

Immediately after inserting the test element into the DR-1 Loop, the thermocouples will be connected and temperatures monitored to obtain base readings for the unirradiated element. The loop will then be operated at full coolant flow with various inlet gas temperatures for additional base readings. Loop radiation levels will be recorded at each temperature.

During the initial reactor start-up, full coolant flow (450 lbs/hr) will be maintained until the reactor has reached equilibrium conditions. The element power will then be calculated from the coolant flow rate and temperature increase as it passes the test element. Operating conditions, necessary to maintain reference conditions, will be calculated from operating charts based on Figure 5. These reference test conditions are 1500 F cladding temperature and 1250 F or less outlet gas temperature.

When the reactor has reached equilibrium, the inlet temperature will be gradually increased to the operating level selected, and the coolant flow adjusted as necessary to reach reference conditions.

Coolant samples will be taken for chemical and fission product analyses prior to reactor start-up to provide base readings. Chemical analyses of this coolant will be provided at the loop site by the Beckman GC-2 chromatograph to monitor O_2, N_2, CO, and CO_2 in the coolant. Sample frequency during operations will be dictated by experience as the test progresses. Fission product samples will be analyzed by Hanford Laboratories Operation for xenon and iodine until suitable equipment can be installed at the loop. During the initial start-up, it is proposed to monitor the isotopic content of the coolant with a 128-channel analyzer provided by Radiological Engineering, IPD. If this instrument is not available, samples will be taken at several temperature increments during start-up, and analyzed by Hanford Laboratories Operation.

In the event of failure of the element thermocouples from which the test conditions are controlled, the next most representative thermocouples will be used, controlling them at temperatures corresponding to the desired reference conditions as corrected by prior experience. If all thermocouples fail during the test, preheat will be reduced to zero and the gas flow increased to maximum. Disposition of the test will then be as agreed upon between DR Processing Manager and the Supervisor, Irradiation Testing.

Sudden increase of gamma activity in the process area to a level of 100 mr/hr or more shall constitute evidence of cladding failure. In this case a reactor shutdown is mandatory. As soon as the reactor is down, the loop technician will: 1) bypass all scram trips, 2) turn off the compressors, and 3) initiate a purge of the loop with the purge valve positioned for maximum flow. The element will be discharged before the reactor resumes operation.

A slow, steady increase in loop activity can be due to two causes: 1) diffusion of fission products through the element cladding or 2) an impurity in the make-up gas, such as argon. In this case, the reason for the activity increase will be determined by analysis of the loop coolant. The loop engineer will then direct appropriate action after conferring with the Supervisor, Irradiation Testing and the Manager, DR Processing, to insure the safety of the building personnel and the reactor. The Radiation Protection Standards established by Radiological Engineering will be observed.

SCRAM SETTINGS AND PHILOSOPHY

The scram settings, operational settings, and scram philosophy remain the same as for previous tests, with the exception of the mandatory reactor shutdown for a sudden increase in gamma loop activity to 100 mr/hr. The loop settings are given in Tables I and II.

DISCHARGE PROCEDURE

The discharge procedure consists of pulling the test assembly out of the reactor into a train of casks consisting of a shipping cask, a cutter cask positioned over a waste cask, and three five-foot long lead rod casks. After withdrawal, the test element is severed from the lead rod by the hydraulic guillotine in the cutter cask, and is pushed back into the shipping cask. The lead rod is then fed into the guillotine and cut into short pieces which fall into the waste cask.

This procedure has been successfully executed with two irradiated test elements. Total outage time required was less than four hours for each element, with a maximum integrated radiation exposure to any operating personnel of 30 mrem. Due to the increased power generation of the proposed test, radiation levels are expected to increase by a factor of four. Total exposure, however, will still be well within control limits.

In the event the fuel element is frozen in the in-reactor tube, it will be necessary to remove the entire in-reactor assembly (loop tube and contents). This will be accomplished in a manner similar to that used in removing test assemblies from the H-1 Loop. The detailed procedure is given in the DR-Recirculating Gas Loop Operating Manual. A three-day outage is required for this procedure.

RESPONSIBILITY

Irradiation Processing Department - Irradiation Testing - FW Van Wormer and EC Bennett, Irradiation Testing, are responsible for operation of the facility and for any interruption of DR Reactor production caused by the facility.

Irradiation Processing Department - DR Processing - DR Processing is responsible for the operational safety and production continuity of DR Reactor.

E. C. Bennett

Irradiation Testing

E. C. Bennett/rch

[redacted]

REFERENCES:

1. General Atomic, Division of General Dynamics Corporation, Hanford Irradiation Request MGCR-HDR-3, GA-C-184, Secret.

2. Baars, R.E., Technical Aspects of MGCR-HDR-3, HW-67325, November 4, 1960; Secret.

3. Bunch, W.L., Anticipated Heat Generation Rate of MGCR-III Fuel Element as a Function of Enrichment. HW-65482, June 2, 1960; Secret.

4. Etherington, Harold, Nuclear Engineering Handbook; McGraw-Hill, New York, N.Y.; 1958 edition.

5. Moon, M.R., and G. E. Zima, Interim Temperature - Pressure Limits For The Inconel Process Tube, DR Gas Loop; HW-62915, December 23, 1959.

TABLE I

ALARM AND SCRAM SETTINGS

Tie In	Monitored Variable	Operating Value	Scram Trip	Alarm Trip
Heat exchanger water supply	Water pressure upstream of outlet orifice	70 psig	50 psig	60 psig
Loop gas pressure	Gas pressure upstream of test section	200-205 psig	170 psig	190 psig
Instrument air supply	Supply pressure to loop	100-120 psig	55 psig	80 psig
Primary loop gas flow	Pressure drop across venturi	45 to 100% of full flow(approximately 450 lbs/hr)	20 to 30 % less than operating value	10 % more than scram trip
Auxiliary loop gas flow	Pressure drop across venturi	Full flow	70% of full flow	80% of full flow
Compressor power supply	Voltage, 400 ~ side	220 V.	30 V.	
High radiation alarm	Gross gamma activity at two locations	15 mr/hr, max.	bypassed	95% full scale
Emergency systems gas supply	Gas pressure	200-250 psig	none	200 psig
Test assembly high pressure drop	Same		bypassed	

EMERGENCY SYSTEM: Orifices will be installed in the loop to provide initial emergency flows of 220 lbs/hr and to maintain a back pressure of one-half of the pressure in the emergency tanks.

[redacted]

TABLE II

Maximum Coolant Temperature

A. Process Tube Outlet Gas - 1200 F } (5) whichever is most limiting
B. Outer Tube Wall 1275 F

Maximum Pressure Settings

A.	Safety Valves		230 psig @ 72°F
B.	Rupture Discs		230 psig @ 72°F
C.	Pressure Regulating Valves # 364		220 psig @ 72°F
D.	Pressure Relief Valve DOV # 5		215 psig @ 72°F
E.	Gas Storage Supply Pressure Regulating Valves		325 psig @ 72°F
F.	Emergency Cooling System Supply Pressure	Max.	250 psig @ 72°F
		Min.	200 psig @ 72°F

[redacted]

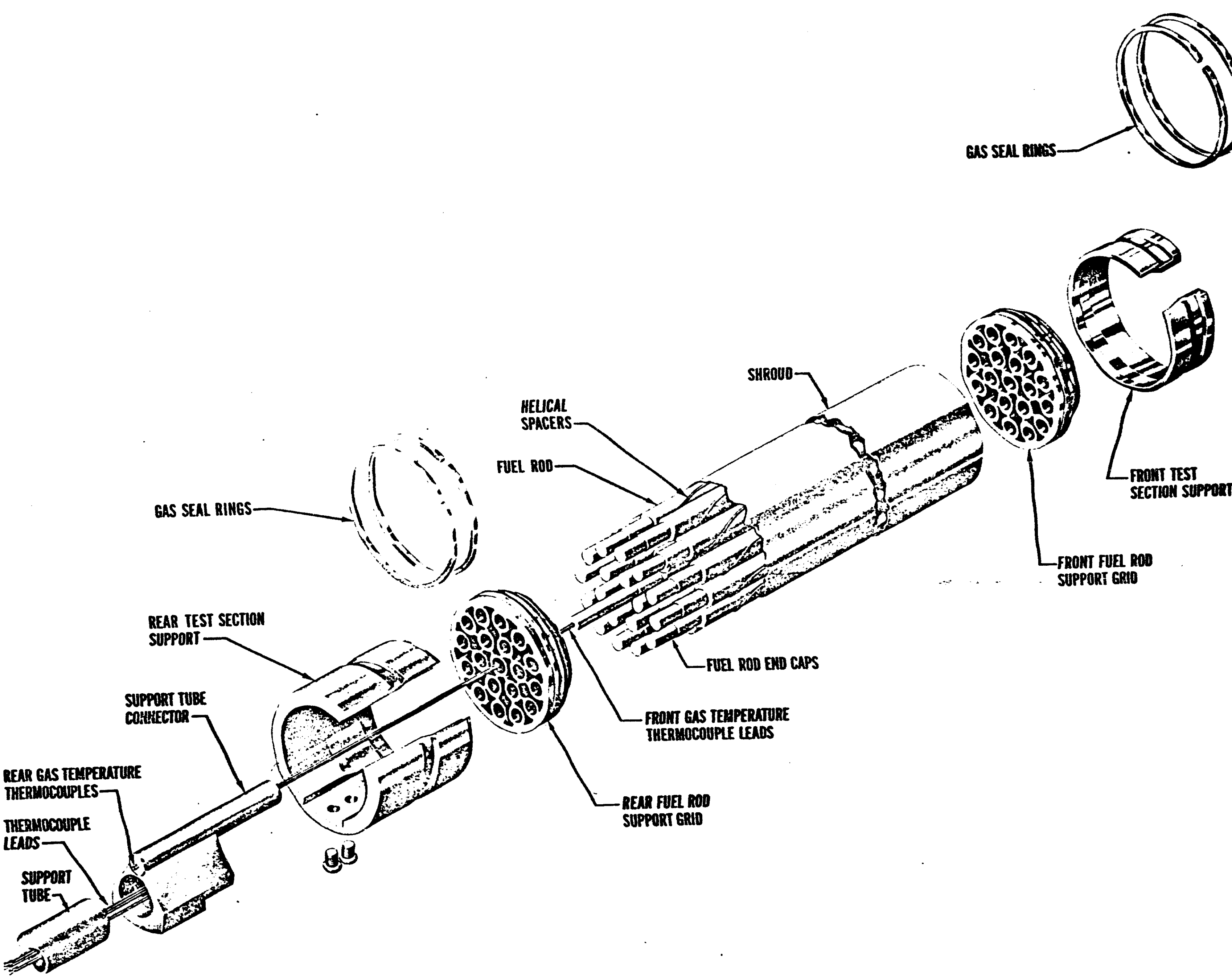
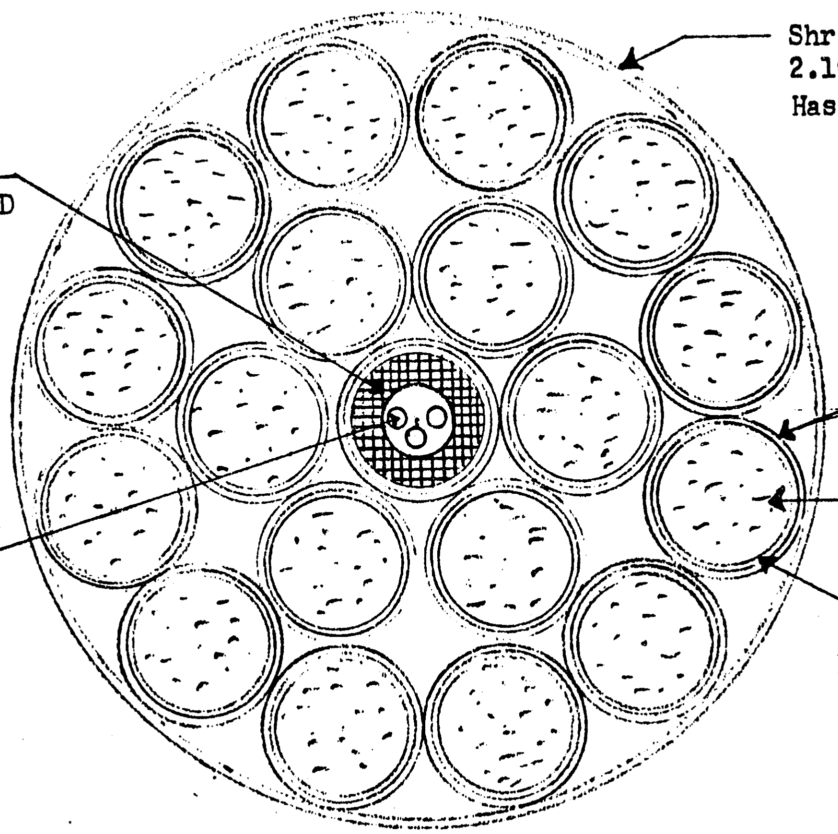


Figure 1

TEST ELEMENT ASSEMBLY

Figure 2

TEST ELEMENT CROSS SECTION



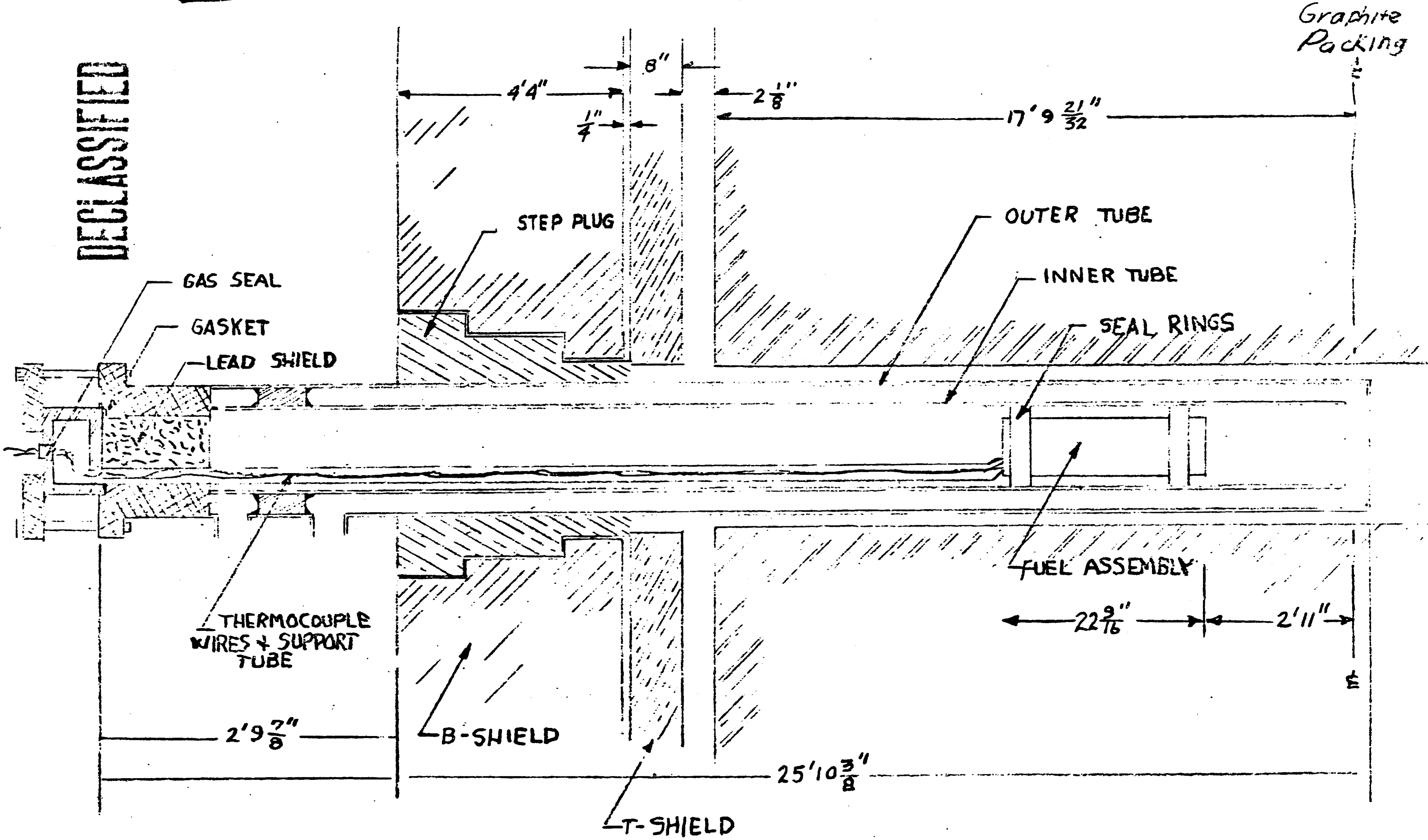


Figure 3

CROSS SECTION OF FUEL ASSEMBLY AND TUBE

Figure 4

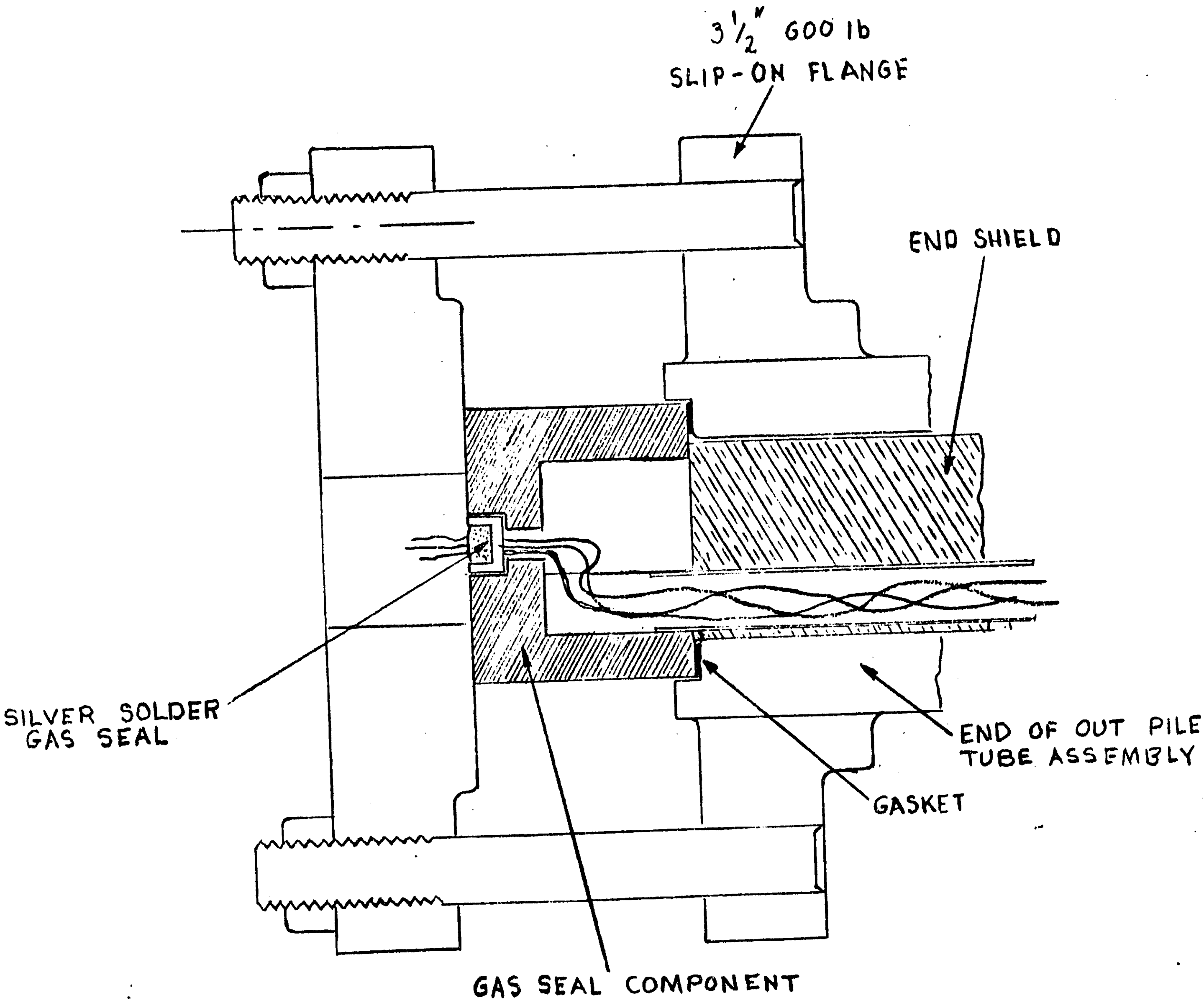


Figure 4
Gas Seal Details

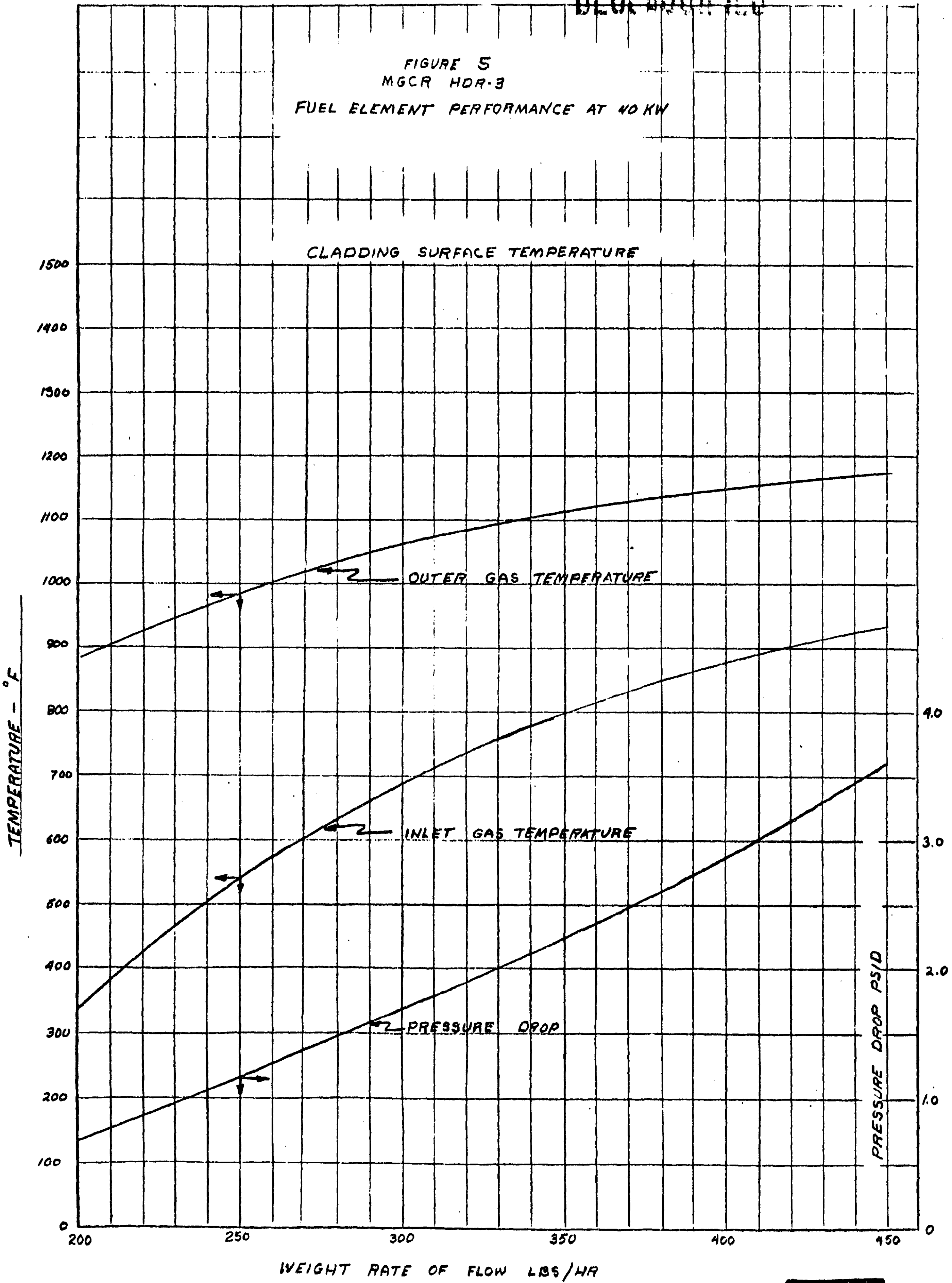


FIGURE 5
MGCR HDR-3
FUEL ELEMENT PERFORMANCE AT 40 KW
CLADDING SURFACE TEMPERATURE
OUTER GAS TEMPERATURE
INLET GAS TEMPERATURE
PRESSURE DROP
TEMPERATURE - °F
1500
1400
1300
1200
1100
1000
900
800
700
600
500
400
300
200
100
0
PRESSURE DROP PSID
4.0
3.0
2.0
1.0
0
200
250
300
350
400
450
WEIGHT RATE OF FLOW LBS/HR

[redacted]

APPROVALS

fwv C. G. Lewis ______ C. G. Lewis
Manager, Testing

R. W. Reid ______ R. W. Reid
Manager, Process Technology

R Wilson for ______ J. H. Brown
Manager, Process and Reactor Development

O. H. Greager ______ O. H. Greager
Manager, Research and Engineering

D S Lewis ______ D. S. Lewis
Manager, DR Processing

W. D. Richmond ______ W. D. Richmond
Manager, D-DR Reactors

E R Astley ______ E. R. Astley
Manager, Applied Reactor Engineering

O C Schroeder ______ O. C. Schroeder
Manager, Manufacturing

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