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OVERVIEW OF ROVER ENGINE TESTS FINAL REPORT

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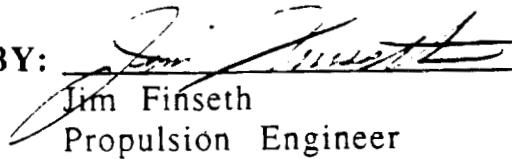
Sverdrup
CORPORATION

Prepared By:
J.L. FINSETH
Sverdrup Technology, Inc.
MSFC Group
620 Discovery Drive
Huntsville, Alabama 35806

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OVERVIEW OF ROVER ENGINE TESTS

PREPARED BY:


Jim Finseth
Propulsion Engineer

Sverdrup Technology, Inc.
MSFC Group
620 Discovery Drive
Huntsville, Alabama 35806

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SECTION 1 INTRODUCTION

This report was generated at the request of the Space Propulsion Branch at NASA's Marshall Space Flight Center, Alabama. The work which preceded the report generation included an extensive effort to identify, locate, and obtain the historical documents; although time-consuming, collection of the relevant documents was a success.

The purpose of this report is to summarize the results of nuclear rocket development activities from the time of the inception of the ROVER program in 1955 through the termination of activities on January 5, 1973. The amount of data generated by America's nuclear rocket program over many years along with the complexities and subtleties of such a propulsion system necessarily limit the scope of this document. This report discusses the nuclear reactor test configurations (non-cold-flow) along with the nuclear furnace demonstrated during this time frame. Included in the report are brief descriptions of the propulsion systems, test objectives, accomplishments, technical issues, and relevant test results for the various reactor tests. The level of detail reported for each test is primarily dependent upon the amount of data which were recovered from the archives, along with the thoroughness of the available final test reports. Furthermore, this document is specifically aimed at reporting performance data and their relationship to fuel element development with little or no emphasis on other (important) items such as control drums, nuclear safety, radiation heating, radiation environment, shielding design, reactivity (ambient or operating), structural analysis, test facilities, instrumentation, data acquisition, and other issues. Cold-flow tests (i.e. NRX A1) are not discussed in this report unless their results were particularly pertinent to critical problem resolution and/or key design changes. The RIFT program is not mentioned in detail due to the cancellation prior to completion. Where conflicting data were uncovered, I report details from what is believed to be the most "reliable" source of information.

An extensive list of references is included in the report which provides the reader with a valuable source for identifying key documents which were a product of the ROVER program.

The report is organized in four sections: 1) Introduction, 2) Fundamentals of Nuclear Thermal Propulsion, 3) Overview of ROVER Program, and 4) Details of the ROVER Nuclear Rocket Test Program. Section 1 provides background information, and Section 2 familiarizes the reader with the basics of nuclear propulsion in a simplified fashion. Section 3 provides a brief overview of America's nuclear rocket

program. Section 4 comprises the bulk of Volume 1 with details of each test as reported in the actual final test reports. This section is intended to give the reader a technical assessment of each test and is organized along a technological timetable. This report also contains a series of appendices containing supplemental information as well as some of the actual test data.

This report will, I hope, provide a comprehensive background for individuals seriously interested in the development of next-generation nuclear rocket engines. The level of detail provided is intended to convey the results of "almost-lost" test data for evaluation by today's propulsion engineers.

A note about references: It is virtually impossible to give proper credit to the original authors of the reports utilized in the preparation of this historical summary of the ROVER program. To assist the reader, the major source(s) of information used in discussion of each reactor test (Section 4) is acknowledged in brackets ([]) following the test subheading.

SECTION 2

FUNDAMENTALS OF NUCLEAR THERMAL PROPULSION

The advantage of a nuclear rocket is that it can achieve more than twice the specific impulse of the best chemical rockets. For a Mars mission, a 5000 MW engine would burn less than an hour to provide the necessary velocity for the mission. The major disadvantage of a nuclear engine is that its exhaust is radioactive, and hence it probably is useful only as an upper-stage engine, operating outside the earth's atmosphere. But the simplicity of design, and the fact that it can start, stop, and restart make it an attractive alternative to conventional chemical rocket engines. In addition, the nuclear engine can be started by using only energy generated by the system itself.

The ROVER test reactors utilized a solid core fission reactor. The basic concept employed a graphite-based reactor, loaded with highly enriched Uranium 235. Hydrogen was used as the coolant/propellant due to its low molecular weight. Early tests utilized gaseous hydrogen whereas liquid hydrogen was subsequently used for all tests conducted after 1961 (beginning with KIWI B1B in September 1962). The hydrogen propellant was passed through the fuel elements and the high temperature gas expelled through a converging-diverging nozzle, thus producing thrust. Since high performance requires high gas temperature, much of the development of nuclear rocket engines was focused on the goal of material and engine designs capable of achieving and withstanding prolonged exposure to high temperature gas. The heat exchanger was constructed of the active fissioning region, or core, of a nuclear reactor. A schematic of a nuclear rocket propulsion engine is shown in Figure 1.

To achieve a practical longevity, it is necessary to minimize hydrogen corrosion of the fuel and breakage of the core from vibration and thermal stress. Graphite is used in reactors designed to run at high temperatures because it is not a strong neutron absorber and it moderates neutrons leading to a reactor with a smaller critical mass of enriched uranium. Although graphite has excellent high-temperature strength, it reacts with hot hydrogen to form gaseous hydrocarbons and rapidly corrodes. One of the challenges was to develop a fuel coating which could withstand the severe environment of high pressure hot hydrogen over a lifetime of tens of hours without material degradation.

With the evolution of nuclear rocket development came changes to fuel elements (both material and configuration), feed systems, as well as nozzle design. This report discusses these changes as they evolved along with the accomplishments and setbacks associated with the technology progression.

SECTION 3 OVERVIEW OF THE ROVER PROGRAM

The United States embarked on a program to develop a nuclear rocket engine in 1955. This program was known as project ROVER and initiated at both the Los Alamos National Laboratory, then known as the Los Alamos Scientific Laboratory (LASL), as well as at the Lawrence Livermore Laboratory (LLL). In 1957 the Atomic Energy Commission (AEC) concluded that LASL should assume the role of lead laboratory for the experimental development of the nuclear rocket propulsion reactors and LLL would be redirected to pursue the development of the nuclear ram jet (Pluto Program). The Air Force and the Atomic Energy Commission were the initial sponsors and the responsibility for the Air Force portion passed to NASA on its creation in 1958. Initially nuclear rockets were considered as a potential backup for intercontinental ballistic missile propulsion but later proposed applications included both a lunar second stage as well as use in manned Mars flights.

The ROVER program basically consisted of four segments - KIWI, NERVA, PHOEBUS, and RIFT. The KIWI project was a series of non-flyable nuclear test reactors developed by LASL. This project concentrated its efforts on producing advanced hydrogen-cooled nuclear reactors. Eight KIWI reactors were built and tested during the period of 1959-1964. After the conclusion of the KIWI experiments, LASL concentrated its efforts on development of the higher power graphite PHOEBUS reactor technology.

The NERVA¹ project (which stood for Nuclear Engine for Rocket Vehicle Applications) began in 1961 with contracts to Westinghouse Astronuclear Laboratory (WANL) and Aerojet Nuclear Systems Division with the goal of a first generation nuclear rocket engine employing the best of KIWI reactor design. The ultimate goal of project NERVA was an engine that produces 890000 - 1112500 N (200000 - 250000 lb) of thrust. In 1968 budget pressures forced a cutback in NERVA plans to an engine which produces 333750 N (75000 lb) thrust. NERVA tests were carried out during the period of 1964-1969 at the Nuclear Rocket Development Station in Nevada meeting all technology development goals and culminating with the successful firing of a flight geometry engine labeled XE-Prime which achieved 244750 N (55000 lb) thrust. During 1970, engine system analyses continued and all the subsystems were being perfected to lead to a

¹ The NERVA project is often mistakenly used in the context that it was America's nuclear rocket program. In fact, the NERVA project was an advanced phase of the AEC-NASA ROVER program and will be used in this context throughout this report.

preliminary design review. The goal of extended life fuel elements with 60 cycles of use was achieved in electrically heated fuel tests. With the continuation of the Vietnam War, America's priorities were redirected and funding restrictions delayed perfecting tests and design review from 1971 to 1972. In February 1972, many NASA missions for the future were deferred or cancelled, and along with this the nuclear rocket program was eliminated. Work in Los Alamos continued for another year.

In addition to KIWI reactors, LASL was working on PHOEBUS as early as 1963. PHOEBUS was designed to produce higher power levels and longer duration operations than the KIWI reactors. PHOEBUS was being paralleled by alternate core concepts at the Argonne National Laboratory. Three PHOEBUS reactors were fired during the period of 1965-1968 with the last reactor achieving 890000 N (200000 lb) thrust and a power of 4100 MW.

The fourth segment of the ROVER program was RIFT (Reactor In-Flight Test). Management of RIFT was solely the responsibility of NASA, unlike the other ROVER projects which were managed jointly by AEC-NASA. The objectives of RIFT were to design, develop, fabricate, and flight-test a NERVA-powered vehicle as an upper stage for a Saturn-class launch vehicle; and the advanced technology effort would extend research and development leading to improved nuclear rocket engines. In 1961 it was thought that if a direct manned flight to the Moon were to be attempted, instead of using a tremendously large NOVA chemical rocket, it should be possible to use the Saturn V lower stages, but replace the S-IVB stage with a nuclear stage. The plan was to take the flight version of NERVA when it was ready and test it on a Saturn under the name RIFT. In 1961, Lockheed was selected to build the vehicle which would accept the Westinghouse/Aerojet NERVA engine. RIFT was to be built in the dirigible hangar at Sunnyvale, California, tested at Jackass Flats, Nevada, and launched at Cape Canaveral, Florida. The RIFT design had a dry structure mass of 19958 kg (44000 lb), with a propellant capacity of 70762 kg (156000 lb) and stage diameter of 10.06 m (396 in). The ejection of fuel elements from the reactor core which occurred during KIWI B1B and KIWI B4A tests caused the government to reassess the nuclear rocket program, including the RIFT project, subsequently leading to the cancellation of RIFT in December 1963. Although reinstatement of the program was often mentioned by NASA, this never materialized.

LASL also built a reactor named Pewee. This was a much smaller reactor than KIWI and PHOEBUS and was used to evaluate advanced fuel elements. The 500 MW PEWEE-1 ZrH reactor was tested in 1968 and achieved the highest peak fuel element gas exit temperature of all ROVER reactors,

equivalent to an ideal vacuum specific impulse of 901 seconds. Funding restrictions and environmental concerns caused cancellation of a Pewee 2 reactor test.

The final phase of the ROVER nuclear rocket technology program consisted of improvements to the reactor fuel elements. For evaluation of advanced fuel elements, the Nuclear Furnace was built. This was a nuclear fuels test assembly to be re-used, (but was not, due to program cancellation), not a rocket engine, and was operated during June-July 1972. Fuel elements were tested achieving a fuel exit gas temperature of 2450 K (4410 R) for 108.8 minutes. Although successful, in 1973 this work was phased out and nuclear engine research (and America's nuclear rocket program) ended. Figure 2 shows a comparison of projected endurance of several fuels as a function of coolant/propellant exit temperature.

A summary of the testing program for nuclear rockets is presented in Figure 3. The NERVA project would have led to the development of a flight engine had the program proceeded through a logical continuation. In fact an experimental, 5000 MW engine, NR-1, integrated the technologies from the various projects and had completed a preliminary design review at the time of program cancellation. A chronology of major nuclear rocket reactor tests is shown in Figure 4.

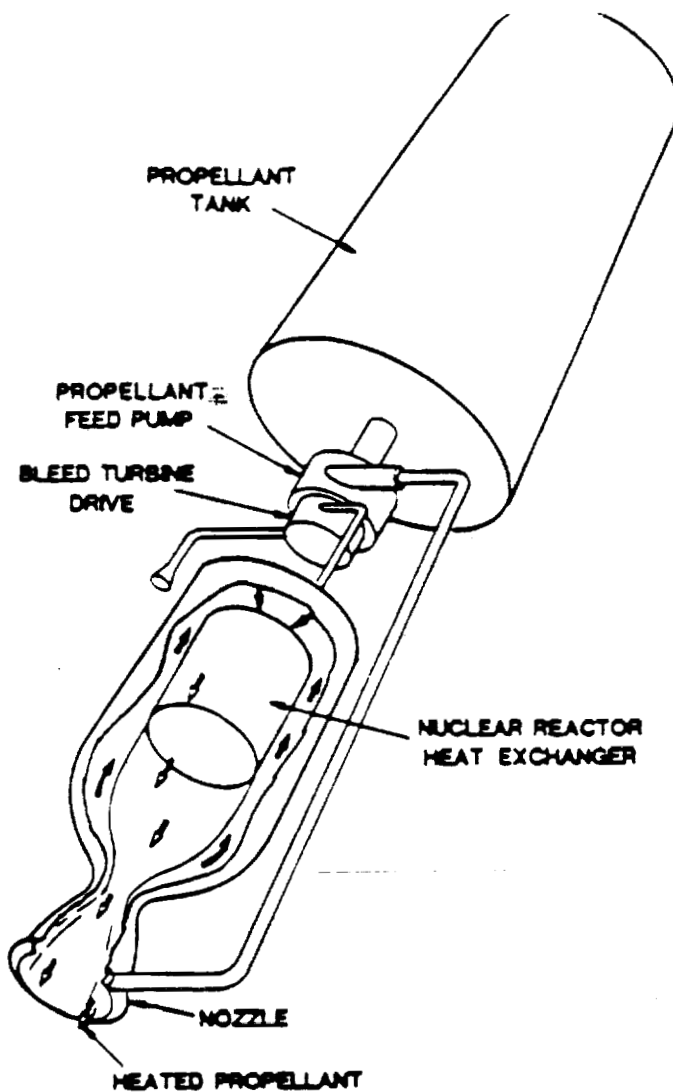


FIGURE 1. NUCLEAR ROCKET ENGINE SCHEMATIC.

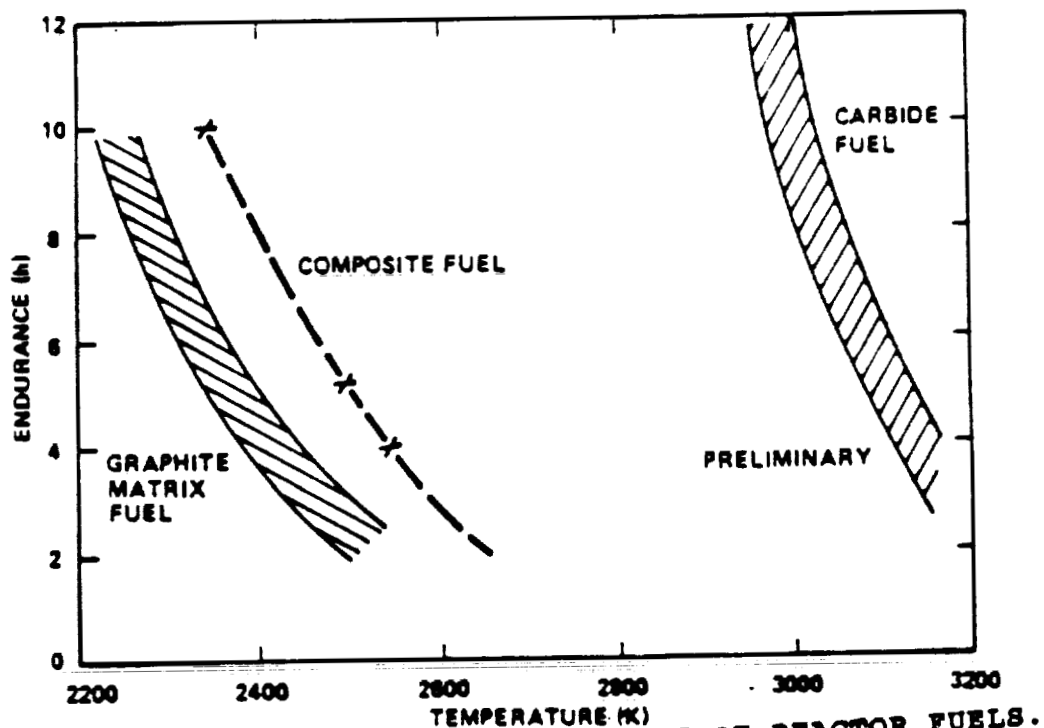


FIGURE 2. PROJECTED ENDURANCE OF REACTOR FUELS.

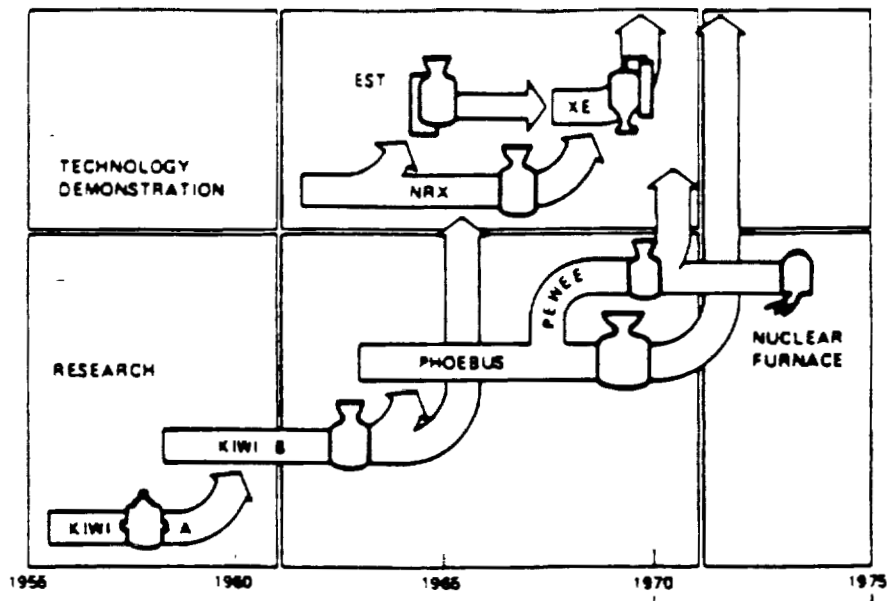


FIGURE 3. SUMMARY OF ROVER TEST PROGRAM.

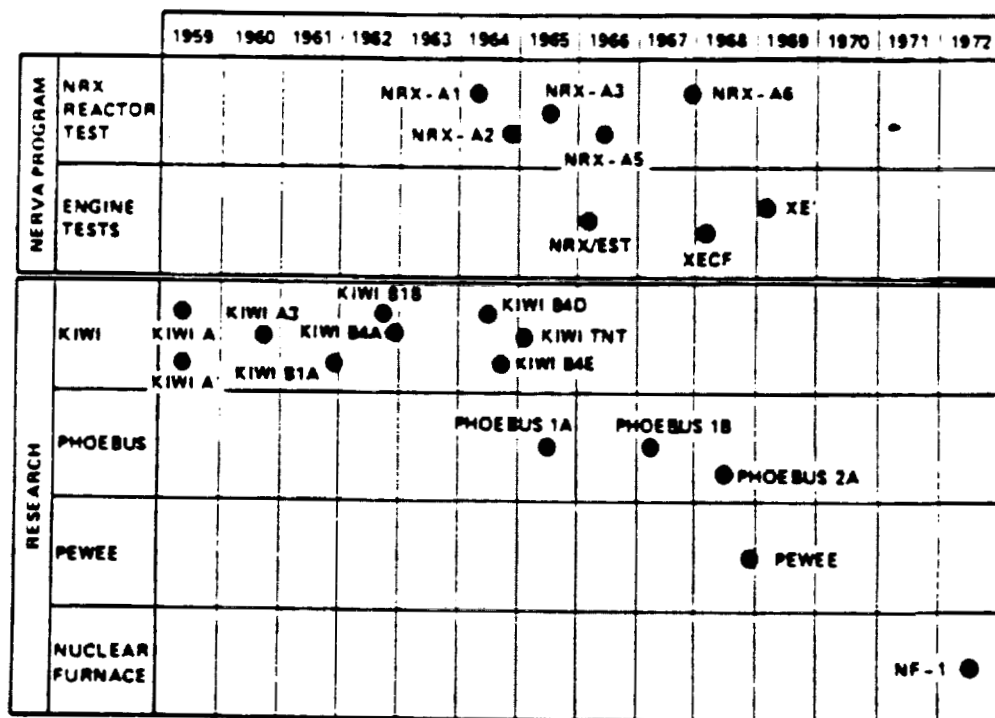


FIGURE 4. CHRONOLOGY OF NUCLEAR REACTOR TESTS.

SECTION 4

DETAILS OF THE ROVER NUCLEAR ROCKET TEST PROGRAM

This section discusses the following ROVER nuclear rocket engine tests: KIWI A, KIWI A', KIWI A3, KIWI B1A, KIWI B1B, KIWI B4A, KIWI B4D, KIWI B4E, PHOEBUS 1A, PHOEBUS 1B, PHOEBUS 2A, PEWEE 1, KIWI TNT, NRX A2, NRX A3, NRX/EST, NRX A5, NRX A6, XE-PRIME, and NUCLEAR FURNACE. The nuclear rocket program consisted of two major concurrent efforts. Therefore, for continuity reasons, this section is divided into 2 portions: Part I Research, and Part II Technology Demonstration. Each part reports some highlights of the nuclear reactor engine developments in a chronological, progressive manner.

In each case, much of the information is taken from the actual test reports with additional information found in supplemental reports, archived journal articles, past and present AIAA papers, and personal conversations with individuals who were involved in the ROVER program. When conflicting data have arisen, the information contained in the test reports is assumed to be most correct.

In some cases, the data retrieved are marginally legible. Nonetheless I have included these data since there are no substitutes and they may prove to be useful.

Some of the reactor tests were heavily focused on one or two specific technologies. This report attempts to report details of these significant achievements and necessarily may change focus with reactor tests. The final test reports significantly differ by originating agency; the LASL reports not only report the test results but also discuss the reactor configuration in detail, providing excellent background information, whereas the WANL reports focus heavily on the test results with limited background information on the reactor. The major accomplishments were so numerous over the course of the ROVER program that one can only scratch the surface when summarizing, but efforts were made to report the major highlights as best possible.

It should be understood by the reader that the early KIWI A reactors were primitive by today's standards but achieved early breakthroughs. The KIWI A evolved into the more sophisticated KIWI B engines. The successful KIWI B4E was the baseline for the NRX class of engines. Each successive NRX engine incorporated refined improvements, extended technology and system integration, culminating in the XE PRIME engine test. This section of the report presents information about each reactor test in a relatively balanced manner, however, future engine designers should focus on the features of the later NRX A6, XE PRIME, and PHOEBUS 2 reactors. The NR-1 reactor, which was cancelled shortly after preliminary design review, integrated the best

features of all reactors - KIWI, NRX, PHOEBUS, PEWEE - and would most logically be used as a baseline for next-generation nuclear rocket engines.

Specific Impulse Calculations.

Ideal vacuum specific impulse is reported throughout this document. The calculated value assumes an infinite nozzle expansion ratio, without losses, discharging into a perfect vacuum. Furthermore, the specific impulse value is based on the temperature of the hydrogen gas exiting the fuel elements - NOT the nozzle chamber temperature. No dissociation/recombination is assumed. With these assumptions, the calculation for ideal vacuum specific impulse, I_{sp} , using hydrogen gas becomes: $I_{sp} = 12.8(T_e)^{0.5}$ where T_e is the average fuel element exit gas temperature in degrees R.

PART I RESEARCH

KIWI A [1, 2]

The first four or five years of the ROVER program were spent laying the foundations in almost every area involved in nuclear rocket reactors. Efforts were focused on the first integral reactor test, KIWI A.

The first reactor test, KIWI A, was conducted on July 1, 1959 at the Nevada Test Site. The reactor was designed and built by Los Alamos and intended to produce 100 MW of power. The reactor achieved a power of 70 MW and operated at this level for 300 seconds. This test utilized gaseous hydrogen as the propellant with a flow rate of 3.2 kg/s (7 lb/s).

The converging, short diverging section nozzle was designed and fabricated by Rocketdyne. A double walled, nickel, water-cooled configuration was used. This nozzle was designed for sonic flow at the throat.

The reactor featured an 18 inch diameter core center body containing a central island of D_2O to reduce the amount of fissionable material required for criticality and also provided a low-temperature, low-pressure container for the reactor control rods that were cooled by circulating D_2O . Control rods were located in this island. This central island was surrounded by an array of four layers of UO_2 loaded graphite fuel plates (960 total) and one layer of unloaded graphite plates (240 total). The KIWI A was the only reactor which utilized elements in the form of plates. The fuel elements were retained and supported in graphite structures called whims. These whims were wheel-like structures with 12 wedge-shaped boxes of fuel plates fitted between their spokes, each box containing 20 fuel plates. The unloaded fuel plates were contained in a fifth whim which also served as an end reflector for the outlet end of the core. The resulting core size had a 83.8 cm (33 in) diameter and was 137.2 cm (54 in) in length. An annular graphite reflector, 43.2 cm (17 in) thick, surrounded the core. The entire reactor was encased in an aluminum pressure shell. A cut-away description of KIWI A is provided in Figure 5. Figure 6 shows the fifth whim of KIWI A during assembly. The reactor is shown in transit to the test cell in Figure 7.

The fuel particle size was 4 micrometers and the particle density was about 10.9 g/cm^3 . At high temperatures (1873-2273 K) during processing, the UO_2 reacted with the carbon surrounding it and was converted to UC_2 with evolution of CO and consequent loss of carbon from the element. The fuel melting temperature was 2683 K (4829 R), the melting temperature of the UC_2 -C eutectic. The fuel plates were molded and pressed at room temperature, then cured to 2723 K (4901 R). This was the only test where the plates had no coating to protect the carbon against hydrogen corrosion.

Higher fuel temperatures than expected were reached (up to 2900 K). These high temperatures were due to the graphite closure plate, located just above the D₂O island, shattering. This plate was ejected out the nozzle along with the graphite wool between the center island and the core. This plate was to contain the carbon wool insulation and to serve as a gas seal that prevented gas from bypassing the annular core into the central region. Figure 8 shows the closure plate and graphite wool configuration. Following the closure plate failure, the carbon wool surrounding the center island worked loose, briefly plugged the nozzle, and then blew out (visible to spectators). The plate failure allowed a significant amount of gas to flow radially inward through slots in the inside wall of the whims and into the central part of the core, bypassing the power-producing region of the core. The bypass flow is depicted in Figure 9. The test conditions required a prescribed average outlet temperature therefore demanding that the gas which did pass through the core had to be heated to a higher temperature. The resulting high fuel temperatures caused melting of the UC₂ fuel and high erosion of the graphite fuel plates.

The KIWI A underwent postmortem inspection. It was originally intended to be disassembled with the aid of an overhead manipulator, however delays in the manipulator availability made it necessary to undertake disassembly without it. The radiation level 0.91 m (3 ft) from the reactor core surface 9 days after the operation was approximately 10 rems/hour¹. Postmortem inspection revealed that significant cracking of the whim ribs had occurred. These cracks were believed to have been caused by thermal stresses resulting from the large radial temperature gradient across the whim wall and the large temperature differences between ribs and support shoulders. The whims also showed unexplained, appreciable weight changes. Three whims showed an increase in weight (on the order of 2%), one whim showed a decrease (roughly 2%), and the unloaded whim showed no change. When the core was examined it was immediately evident that corrosion much in excess of intention had occurred. This presented the opportunity to relate the amount of corrosion with temperature from point to point, provided a simplifying assumption was made that all the corrosion occurred during the full power run and at one temperature. It also appeared that significant migration of uranium to the surface of the hottest plates occurred and that this effect increased with plate temperature. No gradient of uranium concentration in the direction of gas flow was found.

¹ The current radiation dose limits are: general population 0.5 rem/year, radiation worker 5.0 rem/year (250 rem career), astronaut 50.0 rem/year (300 rem career).

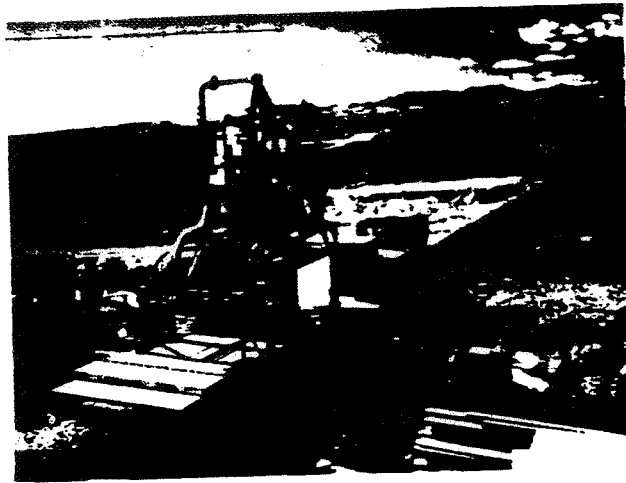


FIGURE 7. THE KIWI A IN TRANSIT TO THE TEST CELL.

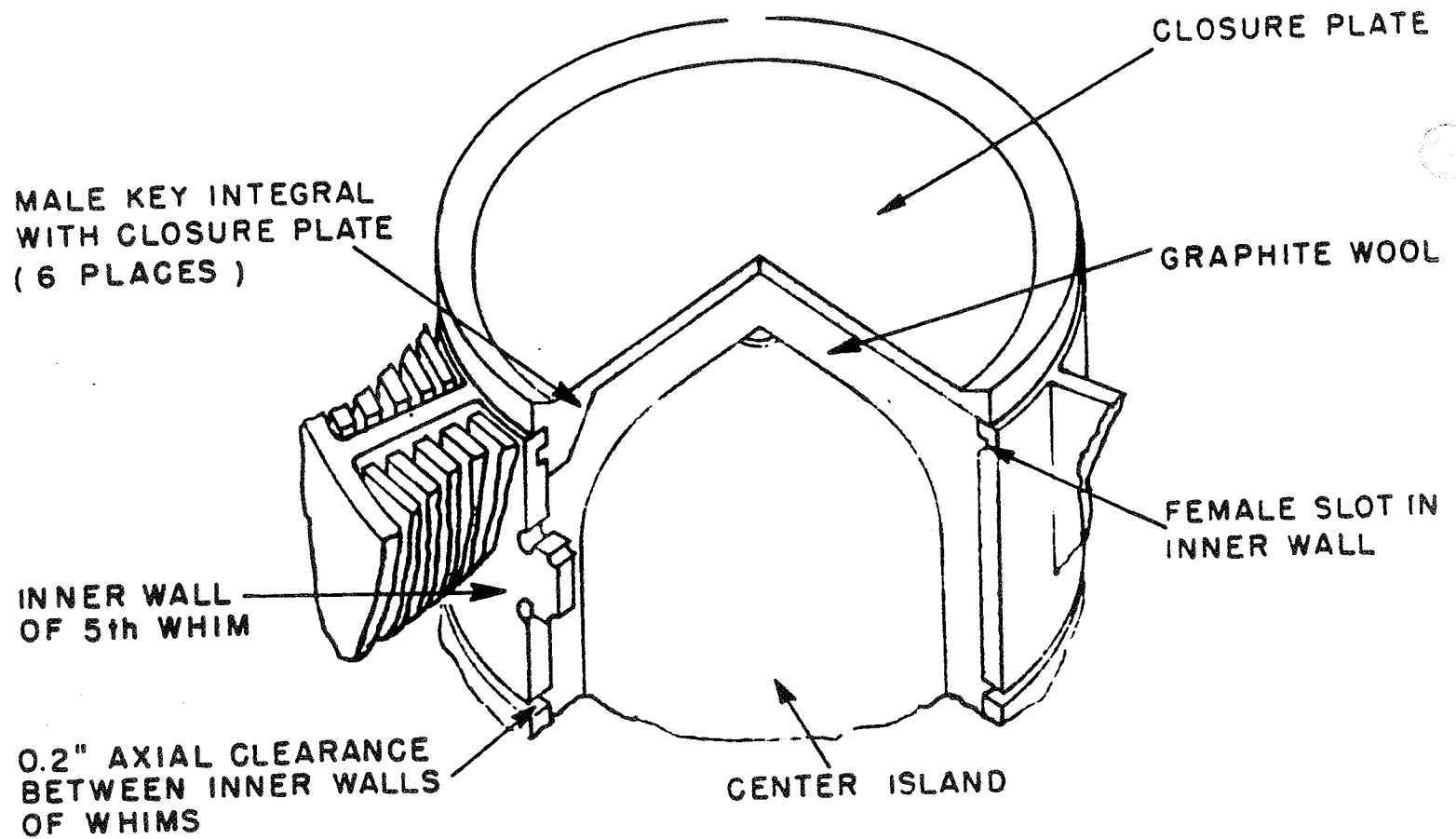


FIGURE 8. CLOSURE PLATE AND GRAPHITE WOOL CONFIGURATION.

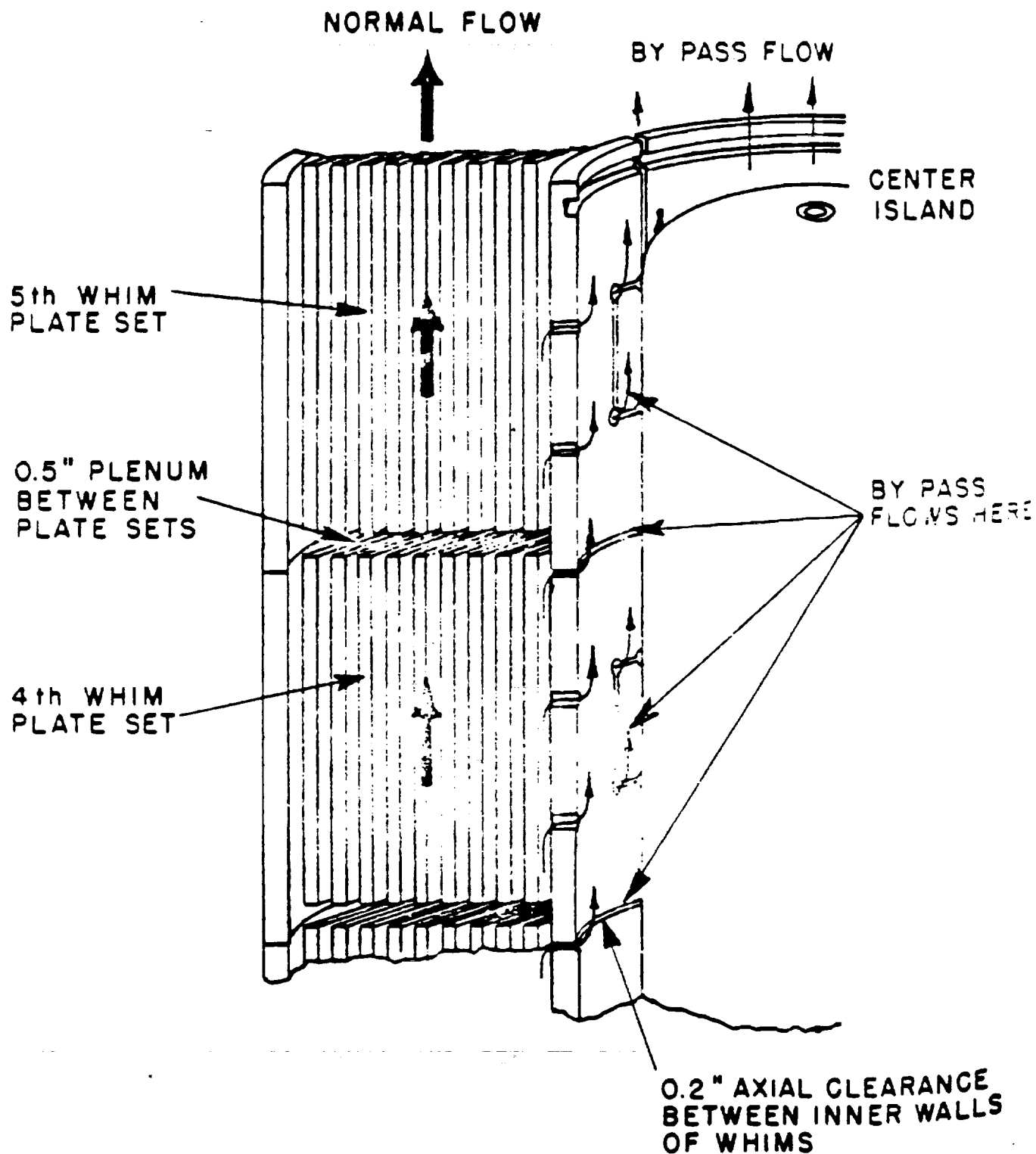


FIGURE 9. BYPASS FLOWS WITH LOSS OF CLOSURE PLATE AND GRAPHITE WOOL.

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The KIWI A experiment was successful at demonstrating the feasibility of a high temperature, gas cooled reactor for nuclear propulsion. In addition, it provided important reactor design and materials information.

KIWI A' [3]

The KIWI A' reactor incorporated several improvements over the KIWI A. Most significantly, it demonstrated an improved fuel element design during its 5 minute, 88 MW test on July 8, 1960. Figure 10 shows a sectioned view of the KIWI A' reactor.

The primary purpose of the KIWI A' was to bring the reactor to a power level of 100 MW and operate at this level for five minutes while maintaining a reactor exit gas temperature of approximately 2206 K¹ (3970 R) through adjustments in the propellant mass flow rate. Included within the primary purpose, were four underlying purposes:

1. To investigate the structural integrity of the core under designed operating conditions.
2. To determine the extent of fuel element and moderator corrosion caused by the hot coolant gas.
3. To determine the temperature coefficient of reactivity for the core.
4. To determine the response of the reactor core to sudden changes in flow and/or power.

Whereas the KIWI A used uncoated fuel plates, the KIWI A' contained UO₂-loaded fuel elements which were extruded and coated with NbC by a chemical vapor deposition (CVD) to reduce hydrogen corrosion. The fuel elements were cylinders with four coolant channels and were contained within graphite modules. The fuel cylinders were segmented in short lengths and six of them were stacked end-to-end in each of the seven holes of the graphite modules to make up a complete fuel module. Each fuel module was 137 cm (54 in) long containing 129.5 cm (51 in) long, 1.9 cm (0.75 in) diameter graphite tubes. Figure 11 shows details of the KIWI A' core design, and Figure 12 provides details of the fuel element assembly.

The first startup of the reactor resulted in an abort. The abort was caused by a lack of agreement between the linear and log power channels (the log channels were reading

¹ According to LAMS-2630 which was released after LAMS-2492 which reported a lower exit gas temperature (2178 K).

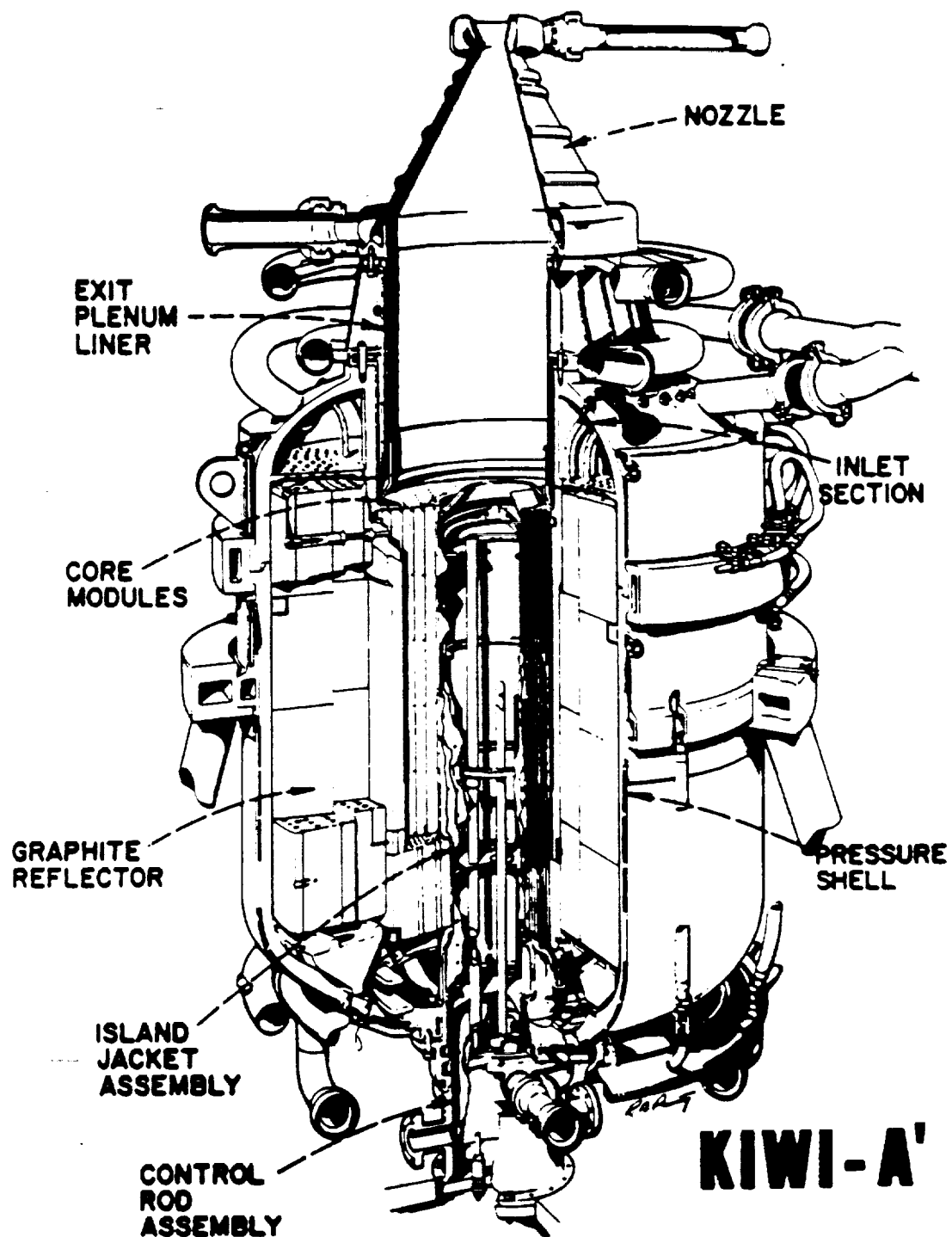


FIGURE 10. SECTIONED VIEW OF THE KIWI A'.

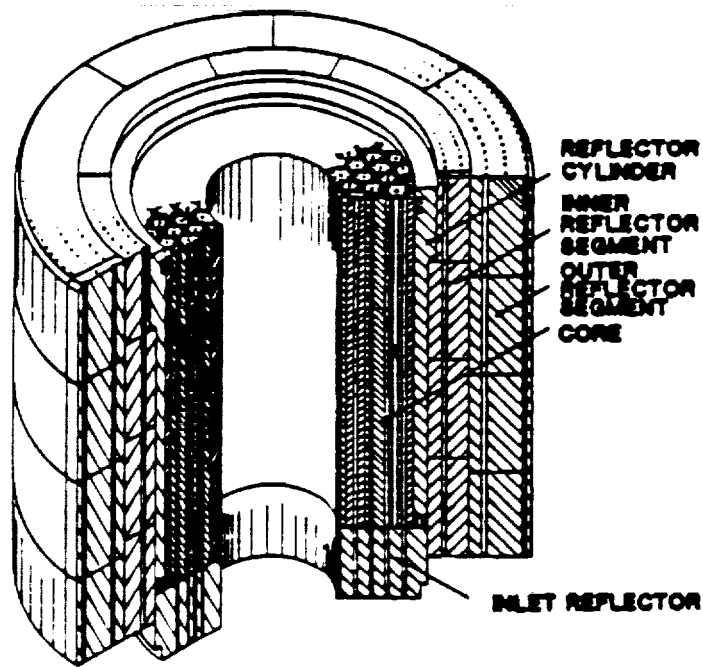


FIGURE 11. DETAILS OF KIWI A' CORE DESIGN. THE CENTRAL ISLAND HAS BEEN REMOVED.

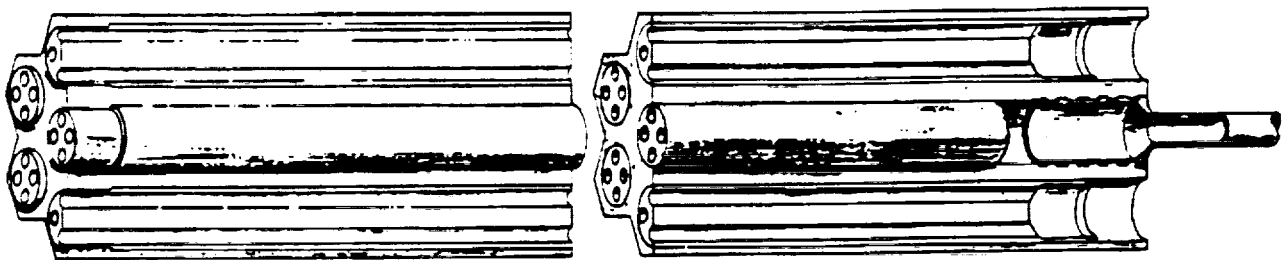


FIGURE 12. KIWI A' GRAPHITE MODULE-FUEL ELEMENT ASSEMBLY.

considerably higher than the linear channels), resulting in a reactor scram.

A second startup also ended in an aborted run. This time the methane flare system, used to ignite the hydrogen coolant, failed to light. The reactor was shutdown and it was found that a leaking relief valve caused the methane to leak to the atmosphere, prematurely emptying all the gas bottles.

The final startup was successful and the reactor attained a power of 88 MW. This power was sustained for 307 seconds with an average core exit gas temperature of 2178 K (3920 R). Thirty-six seconds before the end of the full power plateau, a sharp power perturbation occurred. At this time, the indicated power dropped 18.4 MW and recovered, all within two seconds. This perturbation had been caused by the loss of a core module segment and associated fuel elements, and was witnessed by observers as a shower of glowing fragments ejected from the nozzle. As power was reduced, two additional sharp power perturbations occurred. Again, each perturbation was caused by the loss of a core module segment.

The existence of a major structural weakness within the KIWI A' core was rather dramatically illustrated during the full power portion of the run by the three separate bursts of glowing fragments that were ejected from the nozzle. Each of these occurrences was the result of a separate core module failure in which a portion of the module was broken off and expelled through the nozzle.

After disassembling the core, it was found that the three failures had been caused by the formation of a transverse fracture across each of the affected modules. In each case, the transverse fracture had separated the downstream portion of the module from the remaining supported section; the separated section being ejected through the nozzle as the result of axial differential pressure existing across the module.

Inspection of the core revealed additional transverse cracks in 4 of the fuel modules and it seems likely that given sufficient time at full power, these modules also would have been ejected from the nozzle.

The majority of the fuel elements survived the full power run with little or no damage. However, approximately 2.5% of the fuel elements suffered moderate to severe thermal damage in the form of graphite corrosion and blistering of the CVD niobium coolant passage coatings. Twenty-eight fuel elements had one or more holes corroded through to the element outside surface, but only about half of these showed evidence of blistering in combination with the corrosion.

There were no instances of complete fuel element failure from corrosion, nor was there any evidence of the total obstruction of any fuel element coolant passage due to blistering.

Six 68.58 cm (27 in) long fuel elements, of the type that were used in the KIWI A3 reactor were placed in the core at various positions. The coolant passages of these elements had been fitted with niobium liners which had subsequently been converted to niobium carbide, as opposed to the CVD niobium coatings of the normal fuel elements. The fuel element ends were provided with a thin coat of tantalum carbide. Visual observations of these six fuel elements after the full power run showed that the niobium carbide liners held up very well, in fact they appeared as they had upon assembly. However, the elements ends showed some slight discoloration.

KIWI A3 [4, 5]

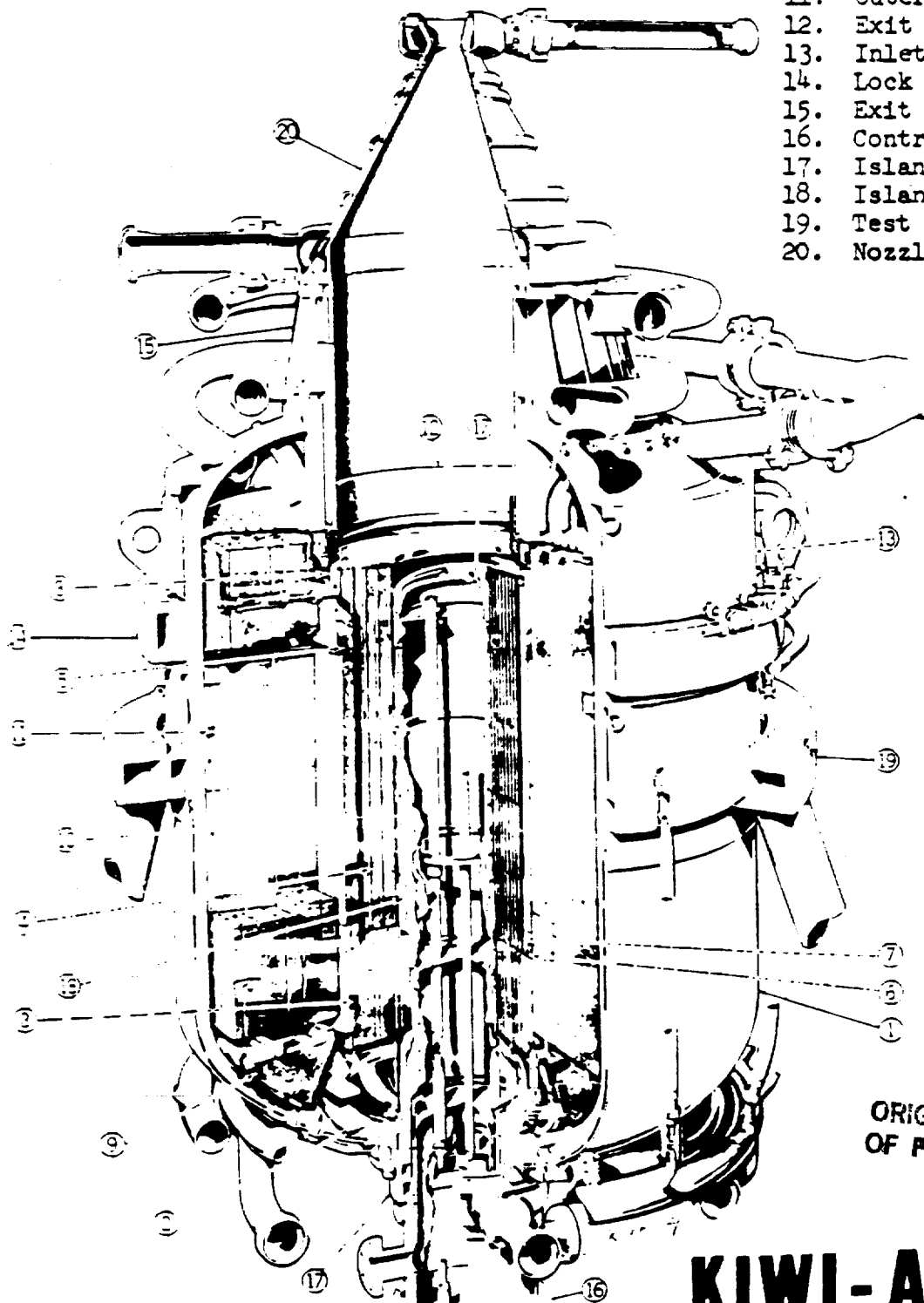
The final KIWI A series reactor, KIWI A3, was tested on October 19, 1960. This reactor test had been designed primarily as a further test of the apparently marginal KIWI A' modular core design. The reactor was operated at an average power level of 112.5 MW for 259 seconds. Figure 13 shows a cut-away view of the KIWI A3.

The primary objective of the KIWI A3 test was to operate the reactor assembly at a power level of 92 MW for 250 seconds while maintaining an exit gas temperature of 2173 K (3911 R) through adjustments in the coolant mass flow. Since the KIWI A3 core contained several modifications over that of the KIWI A' core, the above operating conditions were primarily specified to provide an evaluation of these modifications. Inherent in the primary objective of this experiment were the following more specific objectives:

1. To determine the structural integrity of the various core materials at designed reactor operating conditions.
2. To determine the effects of the hot flowing hydrogen coolant on the redesigned fuel elements of KIWI A3.
3. To determine the temperature coefficient of reactivity for the core.

The KIWI A3 core included several changes over the previous KIWI A' core that were noteworthy. Several different types of graphite had been employed in the core module fabrication in the hope that a comparison of their relative strengths could be obtained under operating conditions. Also, a much

1. Pressure Shell Assembly
2. Support Ring Assembly
3. Inlet Reflector
4. Core Modules
5. Reflector Cylinder
6. Reflector Ring
7. Core Entrance Seal
8. Core Exit Seal
9. Reflector Support
10. Inner Reflector
11. Outer Reflector
12. Exit Plenum Liner Shield
13. Inlet Section Assembly
14. Lock Ring
15. Exit Plenum Liner
16. Control Rod Assembly
17. Island Jacket Assembly
18. Island Shield
19. Test Stand
20. Nozzle



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KIWI-A3

FIGURE 13. SECTIONED VIEW OF THE KIWI A3.

more severe radiographic and visual inspection had been performed prior to the run to eliminate any modules with apparent structural flaws.

Whereas the KIWI A' used 22.86 cm (9 in) segmented fuel elements in the core, the KIWI A3 core contained long 68.58 cm (27 in) fuel elements. Additionally, instead of the carbided vapor-deposited niobium coolant passage coatings utilized for the KIWI A', the fuel elements of the KIWI A3 employed much more satisfactory carbided cylindrical niobium liners.

A full power run was first attempted on October 17, 1960, but was aborted due to adverse weather; the wind direction had been opposite that of the prevailing wind direction for which the fallout detection array had been positioned.

The October 19, 1960 test plan called for a 50 MW (half power) hold for 106 seconds to obtain cross-correlation data through a series of random-polarity power demand steps introduced into the power control system. The plan was to then increase power to 92 MW and maintain this power level for 250 seconds with an exit gas temperature of 2173 K (3912 R).

The actual run, however, deviated from the "planned" run. When the power reached 50 MW, the computed exit gas temperature was 1611 K (2900 R) and rising while the indicated mass flow rate was steady at 2.27 kg/s (5.0 lb/s) (determined later as 2.36 kg/s). Thirty seconds later, the computed exit gas temperature had risen to 1861 K (3350 R), well above the specified level of 1528 K (2750 R). The thermocouples were reading approximately 1806 K (3250 R). The reasons for this higher than specified exit gas temperature during the half power plateau were two-fold. First, the coolant mass flow rate was 7.8% lower than the specified level of 2.56 kg/s (5.64 lb/s), resulting in a higher exit gas temperature for the same power level. Second, the indicated neutronic power level was approximately 15% lower than the actual reactor thermal power as determined after the test. The combination of these two effects caused the exit gas temperature to rise approximately 22% above the specified level of 1528 K. The flow was increased to give the maximum overriding flow correction (10% of the indicated flow rate) which increased the flow rate to an indicated level of 2.45 kg/s (5.4 lb/s) (2.61 kg/s as determined later). This additional flow rate, due to the thermal saturation of the core materials as the half power plateau proceeded, accounted for an exit gas temperature reduction to only 1833 K (3300 R).

After 159 seconds at half power, the reactor power was increased to an indicated level of 90 MW. In order to stabilize the exit gas temperature at 2173 K (3912 R), the

flow rate was increased to 3.81 kg/s (8.4 lb/s). Throughout the full power plateau the reactor experienced several large power and temperature oscillations with a maximum peak-to-peak power fluctuation of 13 MW. Although the cause of these oscillations was not clear, the interaction of an operator watching a neutronic power meter and adjusting the drum positions through a manually operated potentiometer may have been sufficient to produce these oscillations.

It was later determined that through calibration errors in the neutronic power measuring system, the reactor was actually operated at an average power level of 112.5 MW for 259 seconds. Even though the reactor was operated 22% above the designed power level, there was sufficient reserve coolant flow capability to hold the average exit gas temperature at the specified level during the full power plateau.

As with the previous test, the core experienced structural damage indicating that tensile loads on graphite structures should be avoided. The postmortem inspection of the fuel elements showed that their appearance was similar to the previous test, with some fracture, blistering, and corrosion. The damage to the core, however, was to a considerably lesser degree for the KIWI A3 than found in the KIWI A'. Figure 14 shows the results of the KIWI A3 module visual inspection which revealed several cracked fuel elements. The results of the core module tensile test are shown in Figure 15.

As with the KIWI A', the carbon wool insulation, both between the core and the D₂O island and the core and the reflector, was found to be in very bad condition upon disassembly, indicating some coolant bypass flow in these regions during the full power run. Subsequent to the post-run disassembly of the core, the reflector cylinder was visually inspected for cracks, discolorations or other structural defects. No cracks were discovered and the minor chips and scratches observed were most likely a result of handling during disassembly.

KIWI B1A [6]

The KIWI B1A reactor was the first of a new series and the only reactor in the KIWI B test series to use gaseous hydrogen as the coolant. The KIWI B1A was tested on December 7, 1961 achieving a power of 225 MW. The reactor operated at full power for 36 seconds. The average hydrogen mass flow during the full power portion of the experiment was 9.1 kg/s (20 lb/s) and the reactor achieved the equivalent of an ideal vacuum specific impulse of 763 seconds. Figure 16 is a sectioned view of the KIWI B1A

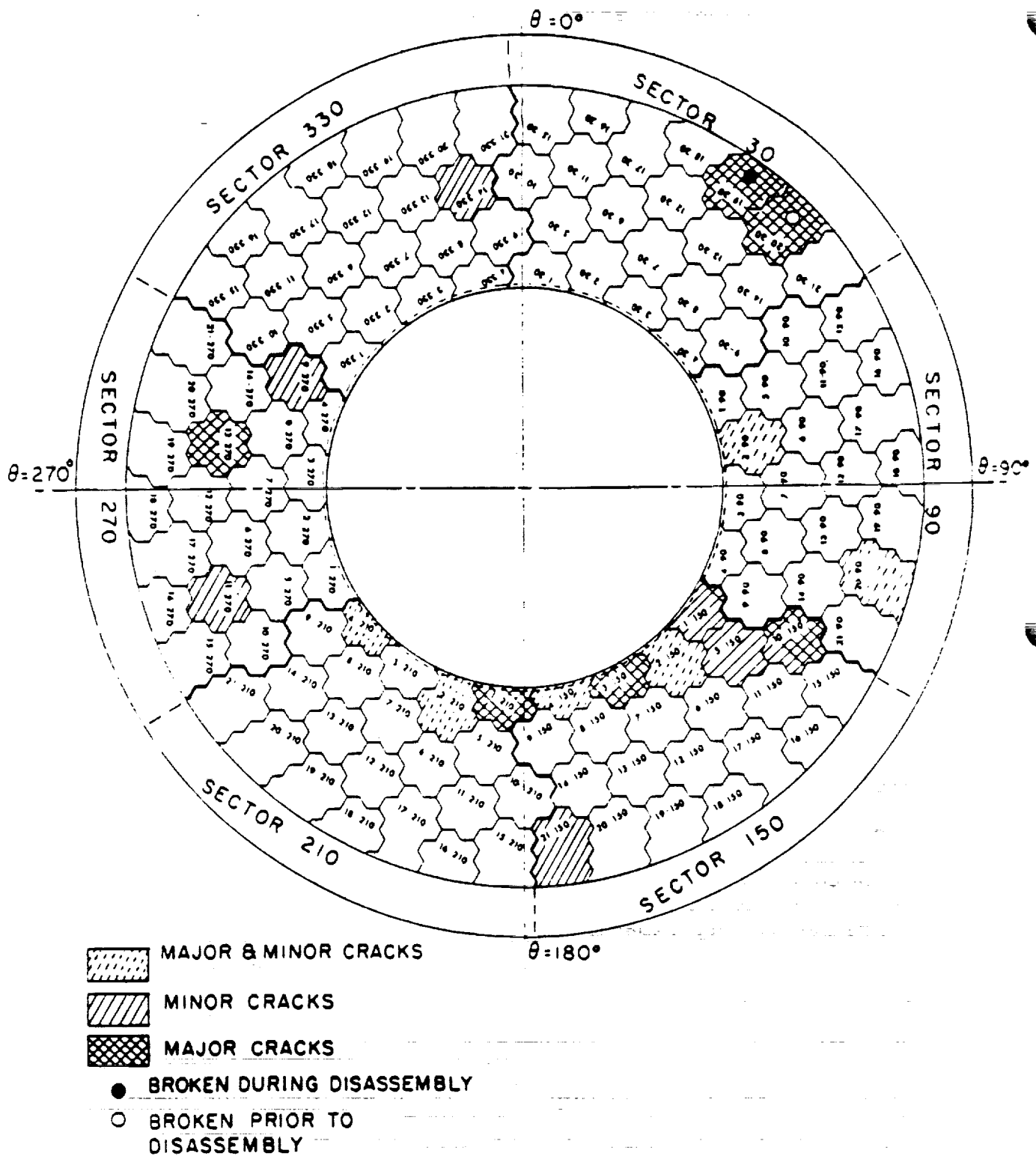


FIGURE 14. KIWI A3 MODULE VISUAL INSPECTION RESULTS.

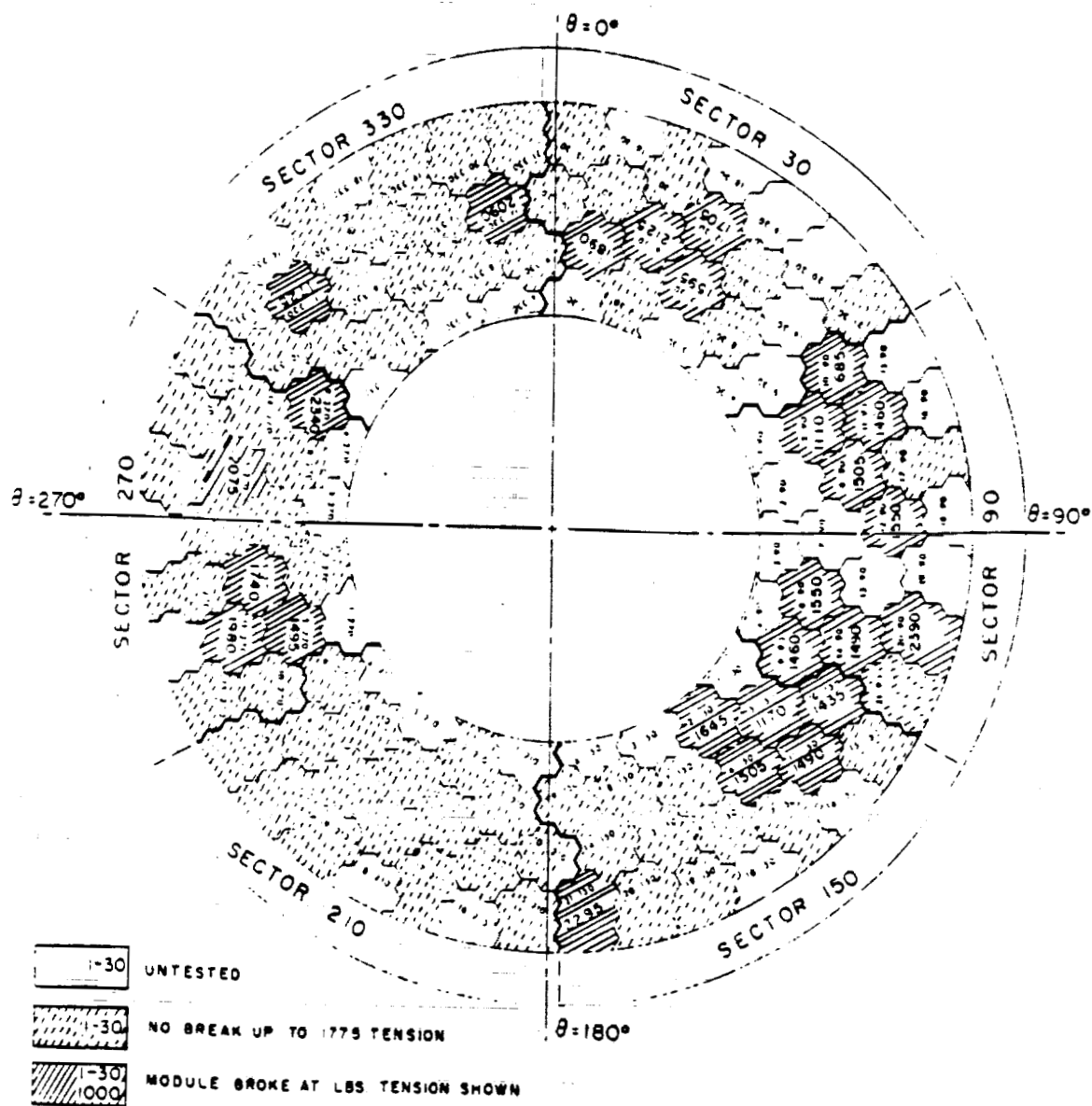


FIGURE 15. KIWI A3 CORE MODULE TENSILE TEST RESULTS.

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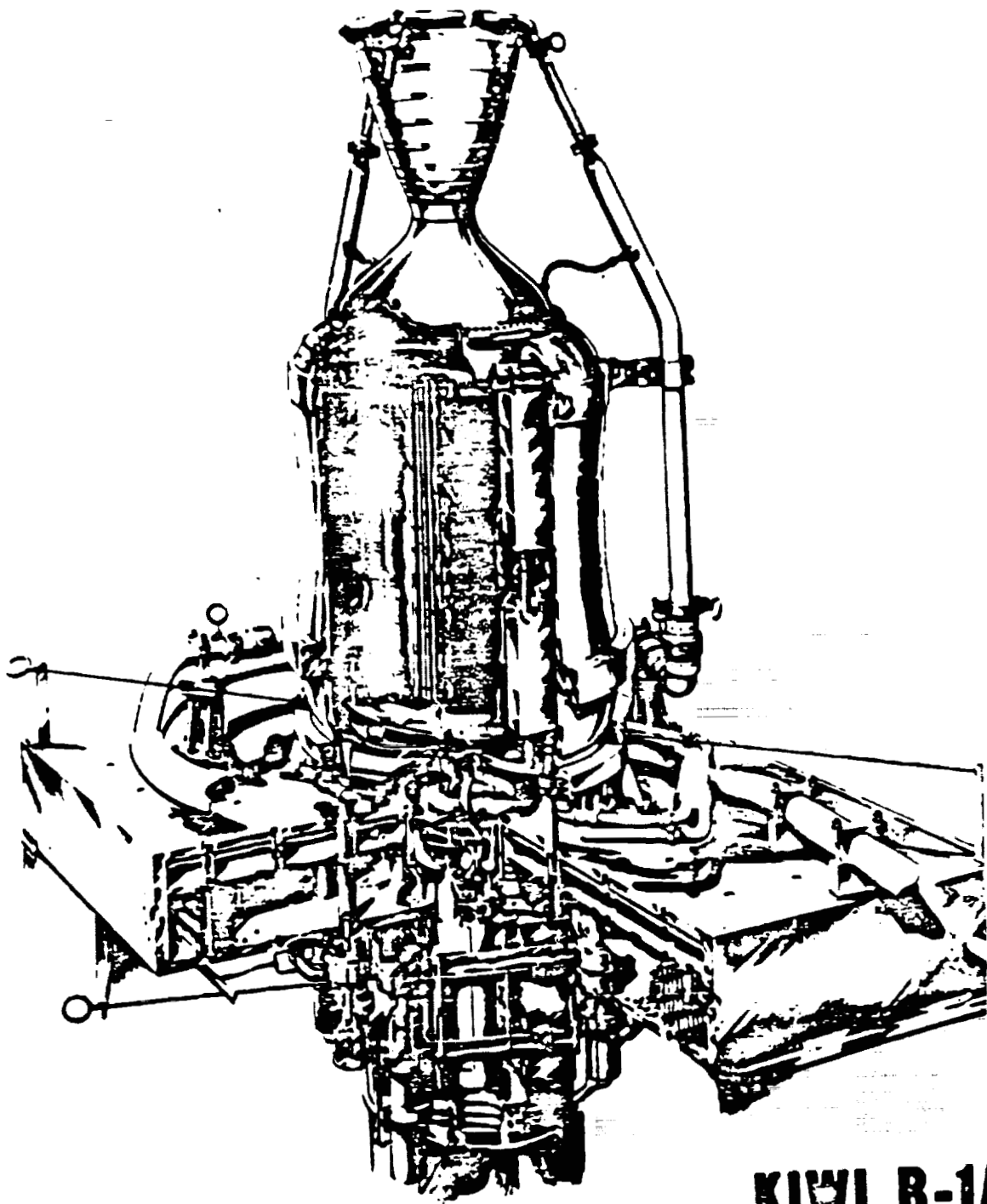


FIGURE 16. SECTIONED VIEW OF THE KIWI B1A.

The specific objectives of this test were to obtain early design verifications on the following items:

1. Module structural integrity.
2. Fuel element integrity.
3. Beryllium reflector and control vane system.
4. Graphite reflector cylinder and pyrographite insulation.
5. Regeneratively-cooled nozzle.
6. Closed-loop reactor control on measured exit gas temperature.
7. Core power and flow balancing.

Whereas the KIWI B1A was essentially a scaled-down version of subsequent reactor tests designed to operate with liquid hydrogen, the KIWI B series was ultimately designed for 1100 MW power, 29.9 kg/s (66 lb/s) hydrogen flow, use of a reflector control and a regeneratively cooled nozzle. The fundamental approach to achieving the tenfold increase in reactor power, while holding the core diameter and length approximately the same as the KIWI A reactor was to:

1. Eliminate the 45.72 cm (18 in) diameter central island.
2. Increase the number of fuel elements and coolant holes in each element.
3. Increase the density of the working fluid by increasing the design core exit pressure to approximately 3448 kPa (500 psia).
4. Change the reflector to an 20.3 cm (8 in) thick beryllium annular cylinder containing 12 boron coated control drums.

The KIWI B reactors were to use liquid hydrogen as a propellant and Rocketdyne was awarded a contract to design, fabricate, acceptance test, and deliver the nozzles required to support the planned KIWI B1 test series.

The reactor core of the KIWI B1A consisted of cylindrical UO_2 -loaded fuel elements with 7 coolant holes (as compared to four in the KIWI A3 fuel elements). As with the KIWI A3 reactor, the enriched uranium graphite fuel elements were contained in modified hexagonal graphite modules which supported the fuel elements against the core pressure drop

and provided additional neutron moderation. The modules in KIWI B1A were supported in tension from an aluminum plate at the inlet end of the core. Each of the interior modules contained seven two-piece fuel elements and was of a modified hexagonal shape, while the exterior modules contained from 5 to 9 fuel elements and deviated considerably from a hexagonal shape. These exterior modules afforded a transition from the resulting hexagonal core shape to a circular outer surface, as shown in Figure 17. The elements were 66 cm (26 in) long, coated with NbC (to prevent hydrogen corrosion) by a tube-cladding process, and were contained in graphite modules. The central island present in the KIWI A reactors was eliminated. The KIWI B1A utilized a beryllium reflector containing 12 control drums.

As with the KIWI A series, the KIWI B1A used gaseous hydrogen as a propellant. Although the KIWI B1A was originally designed for liquid hydrogen operation, the use of gaseous hydrogen would allow several months to be gained in facility availability. It was thought that the use of liquid hydrogen would cause power and temperature surge effects due to two-phase hydrogen flow entering the core during transition from initial hydrogen flow to full power flow rates. Additionally, gaseous hydrogen was used in place of liquid hydrogen due to an undefined core structural problem.

As originally planned, the reactor was to have been operated for 300 seconds (limited by hydrogen gas supply) at a nominal power level of 270 MW. However, as a result of the intentional opening of the temperature control loop and a considerable calibration error in the linear neutronics system, the reactor only reached a power level of 225 MW. Unfortunately, it was necessary to terminate the experiment soon after reaching the full-power plateau due to several large and potentially dangerous hydrogen fires near the nozzle flange area¹, the actual full-power duration being only approximately 36 seconds. The leaky seal between the nozzle and pressure vessel consisted of a soft aluminum o-ring placed in a groove in the interconnecting face of the pressure vessel. This o-ring had been substituted for the originally-planned aluminum "X" seal when it was found that the Rocketdyne nozzle was undersize and therefore too small

¹ At least two additional hydrogen fires were observed which were not associated with gas leaks across the nozzle joint. One of these fires (small) resulted from the burning through of a reflector pressure drop transducer line which appeared to have been caused by a jet of fire from one of the nozzle flange leaks. The second, and considerably larger fire, appeared to have originated near the lower part of the pressure vessel, on the side closest to the test cell face where it was shielded from camera view. The cause of this fire was unknown.

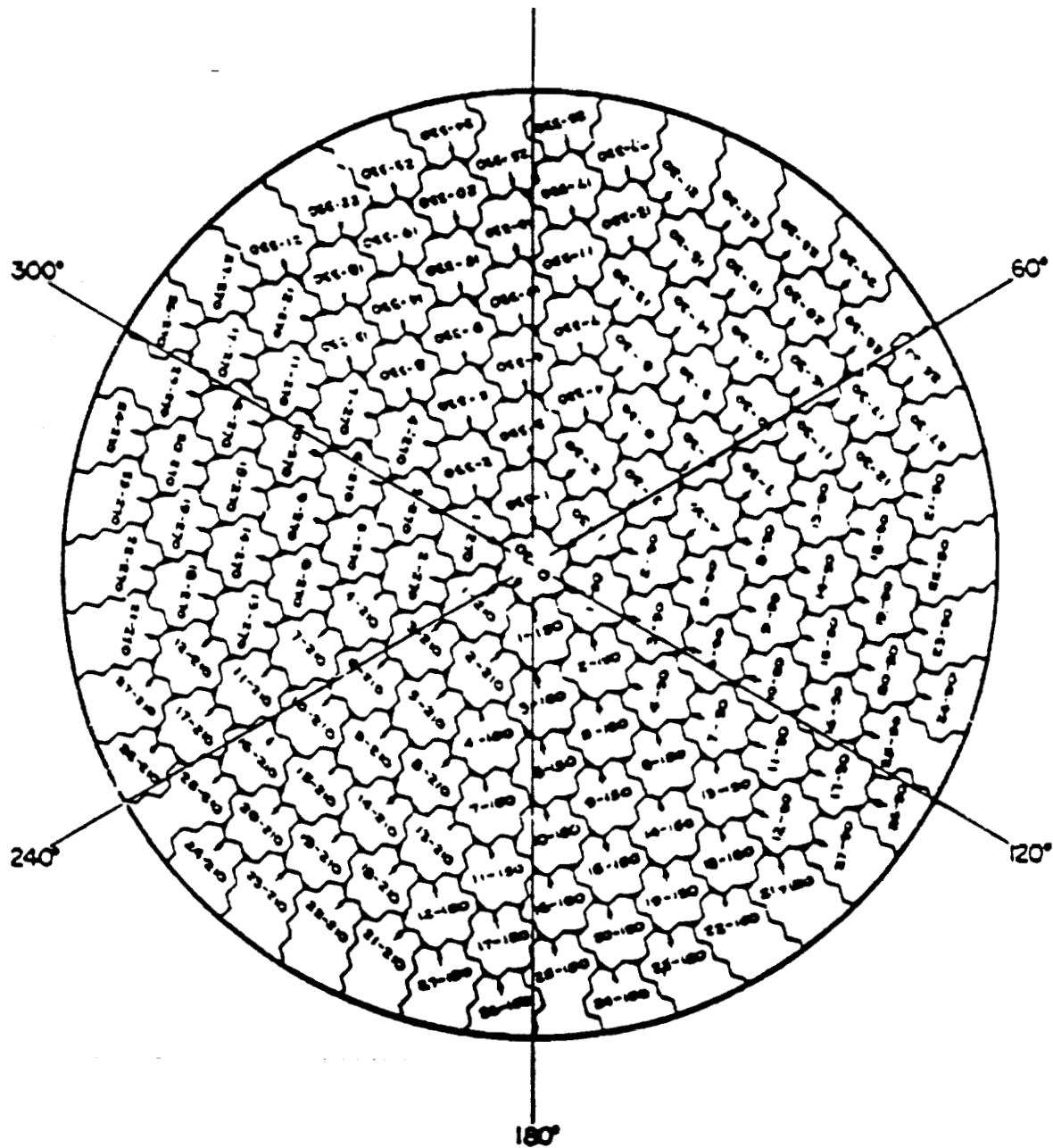


FIGURE 17. CORE CROSS SECTION SHOWING MODULE LOCATIONS AS VIEWED FROM THE EXIT END.

to accommodate the required "X" seal groove. At the time, this change caused no undue worry since soft aluminum o-rings had been used successfully on all previous KIWI reactor tests as well as on numerous Rocketdyne chemical simulation tests.

These hydrogen leaks appear to have resulted from the combined effect of the differential-pressure-induced flexure of the pressure vessel at full flow and the inability of the o-ring to seal the resulting gap between the nozzle and the pressure vessel. The bolts holding the nozzle to the pressure vessel had originally been torqued to 92.2 J (68 ft-lb). The counter-bored depressions in the nozzle flange that accepted the bolt heads were then filled with epoxy to prevent possible hydrogen leaks past the bolts. Following the hydrogen explosion on November 7, 1961 (discussed later), these bolts were not re-torqued because of the epoxy. A torque check during disassembly showed an average of only 27.1 J (20 ft-lb) required to break loose the bolts in the tightening direction. The o-ring appeared to be in good condition and none of the bolts appeared to have been bent.

Notwithstanding the brief duration of the full-power plateau at lower-than-design conditions, considerably more information was obtained from the KIWI B1A test than from any previous KIWI full power test. Almost every core thermocouple functioned properly during the full power run, providing information on the core thermal performance. The quality and amount of experimental information obtained from the highly-instrumented KIWI B1A core would indicate that this experiment was quite successful, even though the test conditions were not fully achieved.

The post-run inspection revealed that two of the seventy-two fuel modules which were visually inspected had large transverse cracks. The other modules examined showed very little or no effect of the run. It was concluded that these modules were cracked by thermal stress due to radial temperature gradients which were in turn caused by the bypass flow.

Abortive Full-Power Runs

The December 7, 1961 run actually represented the third attempt to operate the KIWI B1A reactor at full power. The full-power run had initially been scheduled for November 7th, but during pre-test operations on the morning of the test, an unfortunate and unlikely series of events culminated in a hydrogen explosion within the shed covering the reactor. This explosion resulted in extensive damage to the test car and the exposed instrumentation, but apparently only minor (but unknown) damage to the reactor core itself.

An examination of the core through the nozzle revealed two pyrographite tiles, which were originally located at an azimuthal angle of approximately 120 degrees, that had been dislodged from their correct positions on the inner surface of the core sleeve and were wedged between the core and the nozzle transition ring. The two pyrographite tiles were removed from the core¹. Core inspection revealed no additional damage, and the decision was made to operate the reactor as originally planned.

The full-power test of the KIWI B1A reactor was next attempted on December 6th. The experiment was begun but shut down early in the run profile just prior to the hold at 1056 K (1900 R). The reactor was shut down for two reasons:

1. A large amount of negative reactivity had been inserted in the system which required the drums to go almost all the way out to increase the reactor power. This negative reactivity contribution appeared to be considerably in excess of the expected negative temperature coefficient and permanent in nature. After the reactor was cooled down, it was again brought to cold delayed critical. The new drum positions indicated a loss of approximately \$1.59 in excess reactivity.
2. The majority of core diagnostic thermocouples had been connected backwards². This reversal of thermocouple polarity would have resulted in the loss of much significant data if the run had continued. During an overnight hold in test operations, the polarity of the core thermocouples was corrected and the full-power run re-scheduled for the next day (December 7th).

KIWI B1B [7]

The KIWI B1B reactor was the first to operate with liquid hydrogen and all subsequent nuclear rocket engines used liquid hydrogen. It was tested on September 1, 1962 and

¹ The reactor was not disassembled because of the high radiation level - three roentgen per hour outside the pressure vessel - resulting from a previous low-power test and because a delay to allow for disassembly would have interfered with facility work for the forthcoming liquid hydrogen tests.

² It was later found that the manufacturer of the tungsten/tungsten-26 rhenium core thermocouples had supplied incorrect information on the relative resistance of the two thermocouple wires. The connections of these thermocouples had been based on the measured resistances of the two legs.

operated at a power of 880 MW¹ for several seconds (full power rating was 1100 MW). Figure 18 is a cutaway drawing of the KIWI B1B reactor. A radial axial view of the reactor is shown in Figure 19.

The major objective of the KIWI B1B experiment was to investigate the cryogenic hydrogen start-up of the reactor. The objective was fully and successfully accomplished. The postulated neutronic control difficulties associated with local, unstable, high density liquid hydrogen entering the core did not materialize. The start-up was accomplished with a programmed power control, and no problems associated with this reactivity instability were encountered. The start-up transition was extremely smooth. This laid to rest the earlier concerns of the possible effects of two-phase hydrogen entering the reactor core at pressure below the critical pressure (1297 kPa).

Inherent in this primary objective were the more specific objectives:

1. To study the cryogenic startup of a KIWI B type of reactor.
2. To investigate the structural integrity of the modular B1 core design, both under conditions of rapid start-up using liquid hydrogen and at planned operating conditions of 2278 K (4100 R) and 31.8 kg/s (70 lb/s); a five fold increase in power density over KIWI B1A.
3. To study the thermal performance of the KIWI B1B reactor during rapid start-up and at planned operating conditions.
4. To obtain experimental verification of the analytically determined core temperature, power, and flow balancing at full power operating conditions.
5. To determine the effectiveness and structural integrity of the redesigned pyrographite core insulating system.
6. To investigate the stability of a liquid hydrogen cooled reactor during cooldown where the relative coolant reactivity contribution is considerably increased.

¹ The reactor power reached an indicated 1160 MW, but due to several pre-run calibration errors this was equivalent to an actual power level of 880 MW as determined by post-run radiochemistry.

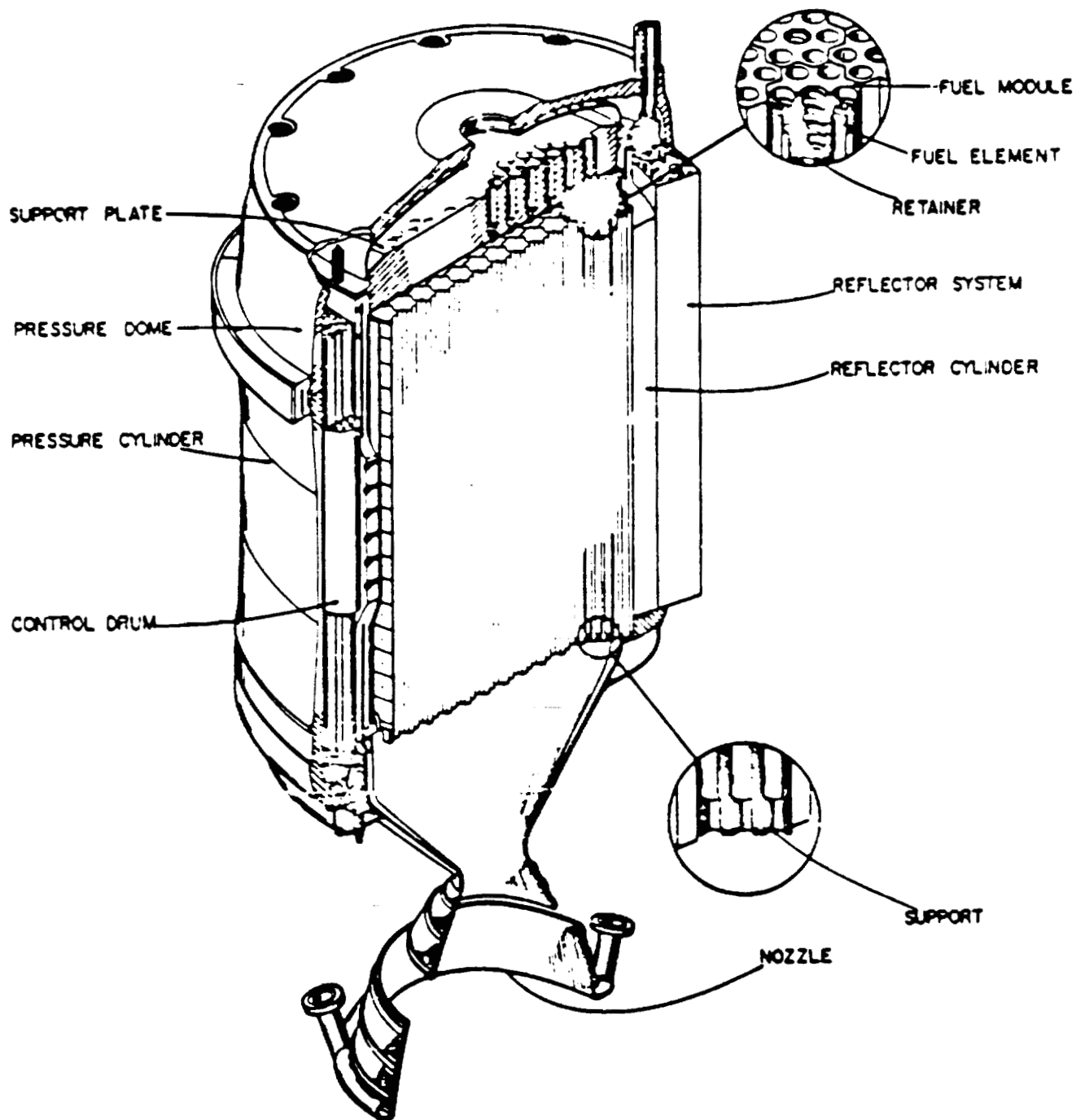


FIGURE 18. CUTAWAY DRAWING OF THE KIWI B1B.

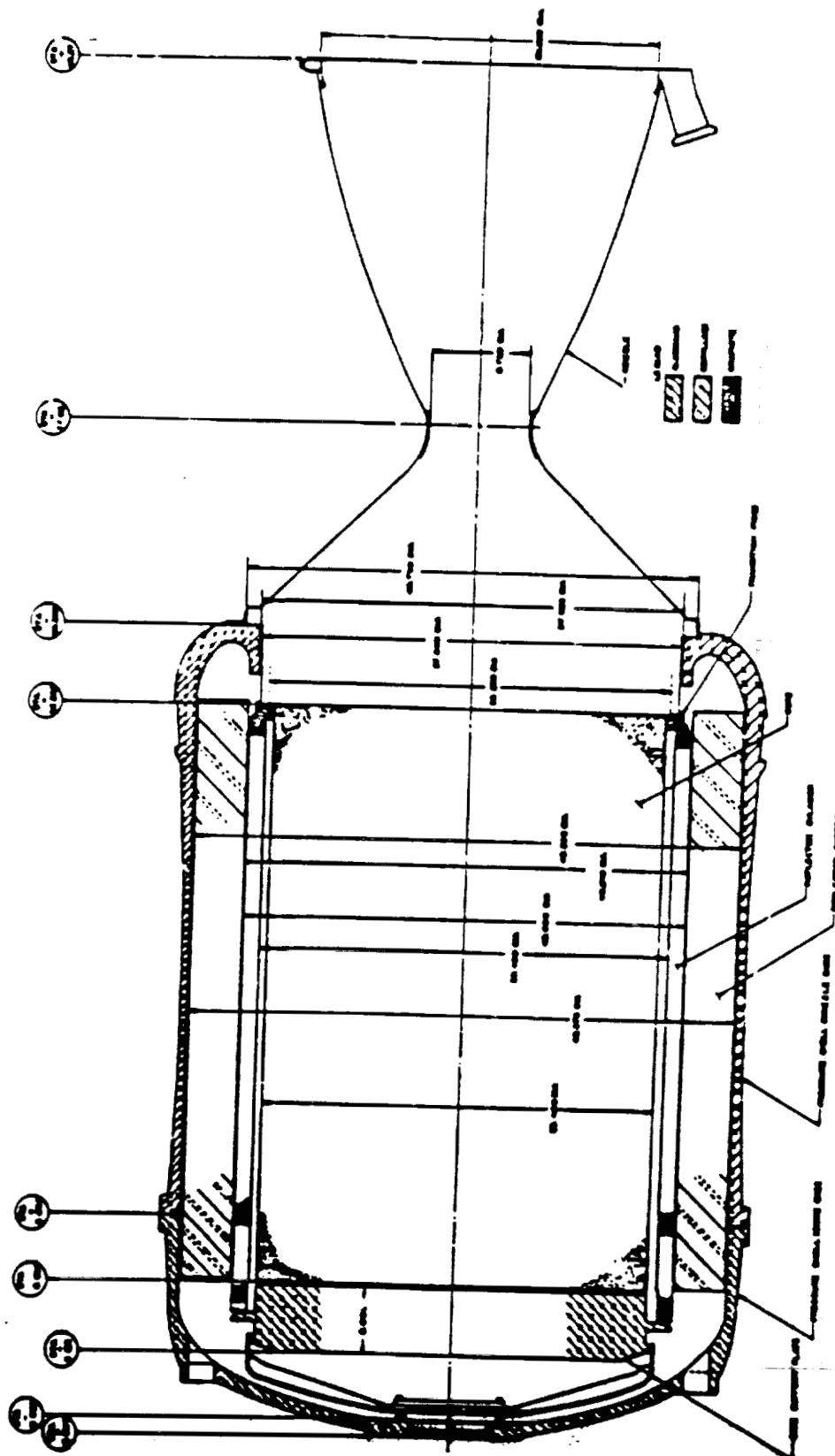


FIGURE 19. AXIAL VIEW OF THE KIWI B1B.

7. To obtain design verification of the liquid hydrogen-regeneratively-cooled nozzle fabricated by Rocketdyne.
8. To investigate the performance of the various control systems during cryogenic start-up and full power operation, particularly closed-loop reactor control on measured exit gas temperature.

Of these, objectives 1, 5, and 6 were essentially accomplished; objectives 2, 3, 7, and 8 were at least partially accomplished, and only objective 4 failed to be accomplished.

The reactor core for the KIWI B1B used the same type of fuel as was tested in the KIWI B1A. The core consisted of comparatively massive, hexagonal, graphite modules supported in tension from an aluminum support plate at the inlet end of the core. In general, each module contained seven internally-cooled, uranium-loaded graphite fuel elements which were supported against the core pressure drop load by the modules. Figure 20 shows a sectioned view of the module and included fuel elements. There were 1147 full length (127 cm) fuel elements in the KIWI B1B; each element contained seven coolant passages with a range of diameters between .381 and .427 cm (0.150 and 0.168 in) depending on core radial position and fuel-element loading¹.

Several significant changes from the KIWI B1A were:

1. The number of modules near the core periphery was increased.
2. The outer-most fuel elements were in a circular array.
3. Full-length fuel elements were used instead of the half-length elements used in KIWI B1A
4. The impedance rings on either side of the beryllium reflector were modified from their previous gas flow configuration to provide the desired reflector flow distribution at full power with cryogenic hydrogen flow.
5. The clearance between the core support plate and the graphite reflector cylinder was very critical - specified as 10 mils.

¹ There were two nominal fuel-element loadings of 240 and 400 mg ²³⁵U/cc.

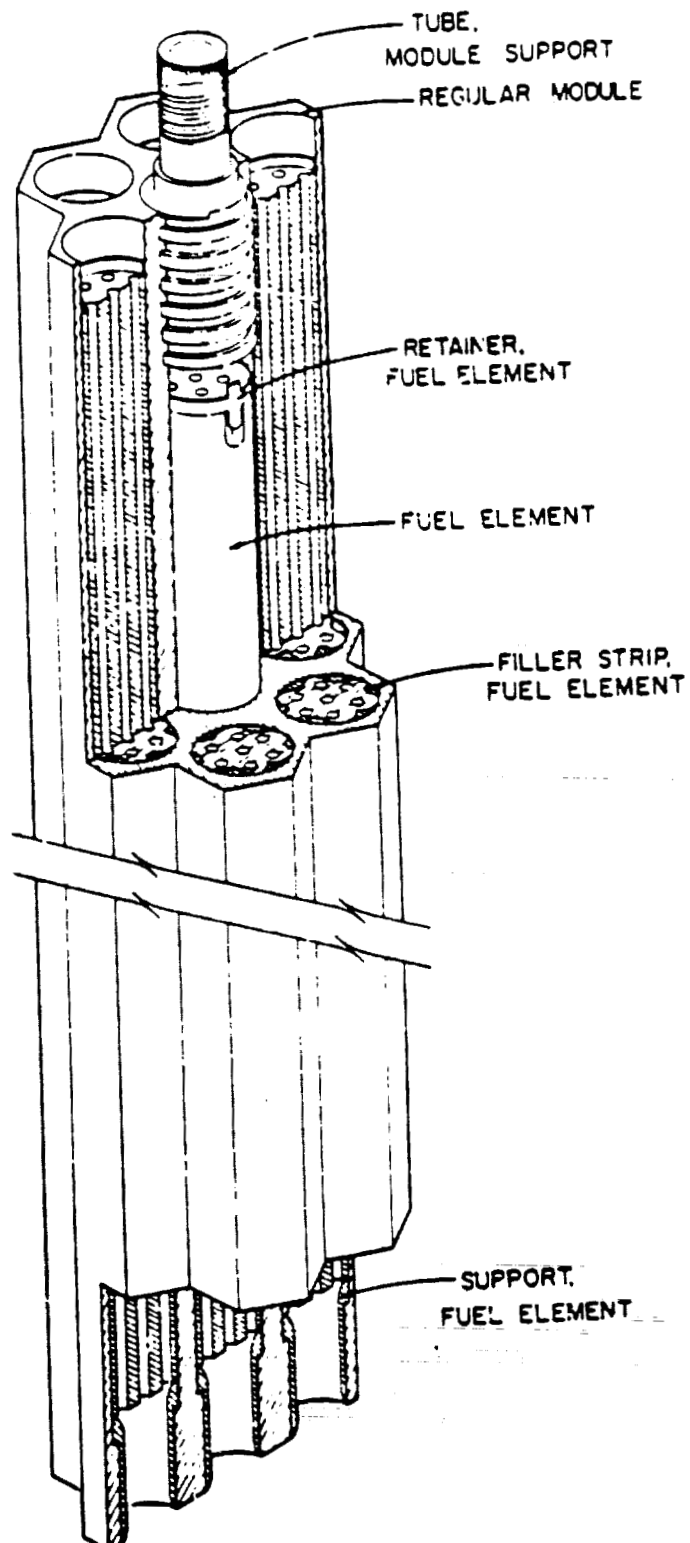


FIGURE 20. SECTIONED VIEW OF A REGULAR MODULE AND INCLUDED FUEL ELEMENT

6. The method of thermally insulating the reflector cylinder from the core was considerably different.
7. The graphite reflector cylinder was provided with axial coolant passages.
8. Numerous minor detail changes were made in the reflector system to account for the changed temperature environment.

During 18 seconds near the start of the 54 second rise to full power and prior to any apparent core damage (see below), the flow system went into severe oscillation. These oscillations were believed to have been generated in the liquid hydrogen feed system by a complex oscillatory interaction among the speed, specific speed and flow rate control systems under the influence of unchilled pipes. The evidence definitely points to the flow and pressure disturbances having been generated in the feed system and not in the reactor.

The KIWI B1B run was terminated after a few seconds at full power, due to flashes of light appearing in the nozzle exhaust. These flashes indicated that portions of several fuel elements were being ejected through the nozzle. In all, portions of 11 fuel modules were ejected from the core, all of which impacted against the convergent section before passing through the throat of the nozzle. A high percentage of the supplied hydrogen flow completely bypassed the reactor through several large holes in the nozzle coolant tubes. A small hydrogen fire was observed jetting out from an instrumentation tap in the nozzle inlet manifold.

Upon post-test inspection many broken fuel elements were found. Figure 21 shows the locations of the 11 ejected modules and the 50 modules found broken at disassembly. Several theories developed regarding the causes of the core module damage. The contending theories were that failure was caused by:

1. Rapid rate of rise of core temperatures during the initial startup.
2. Lateral vibrations of modules.
3. Flow and pressure oscillations.
4. Leakage flow of hydrogen between modules due to poor inlet-end sealing.
5. Liquid hydrogen.

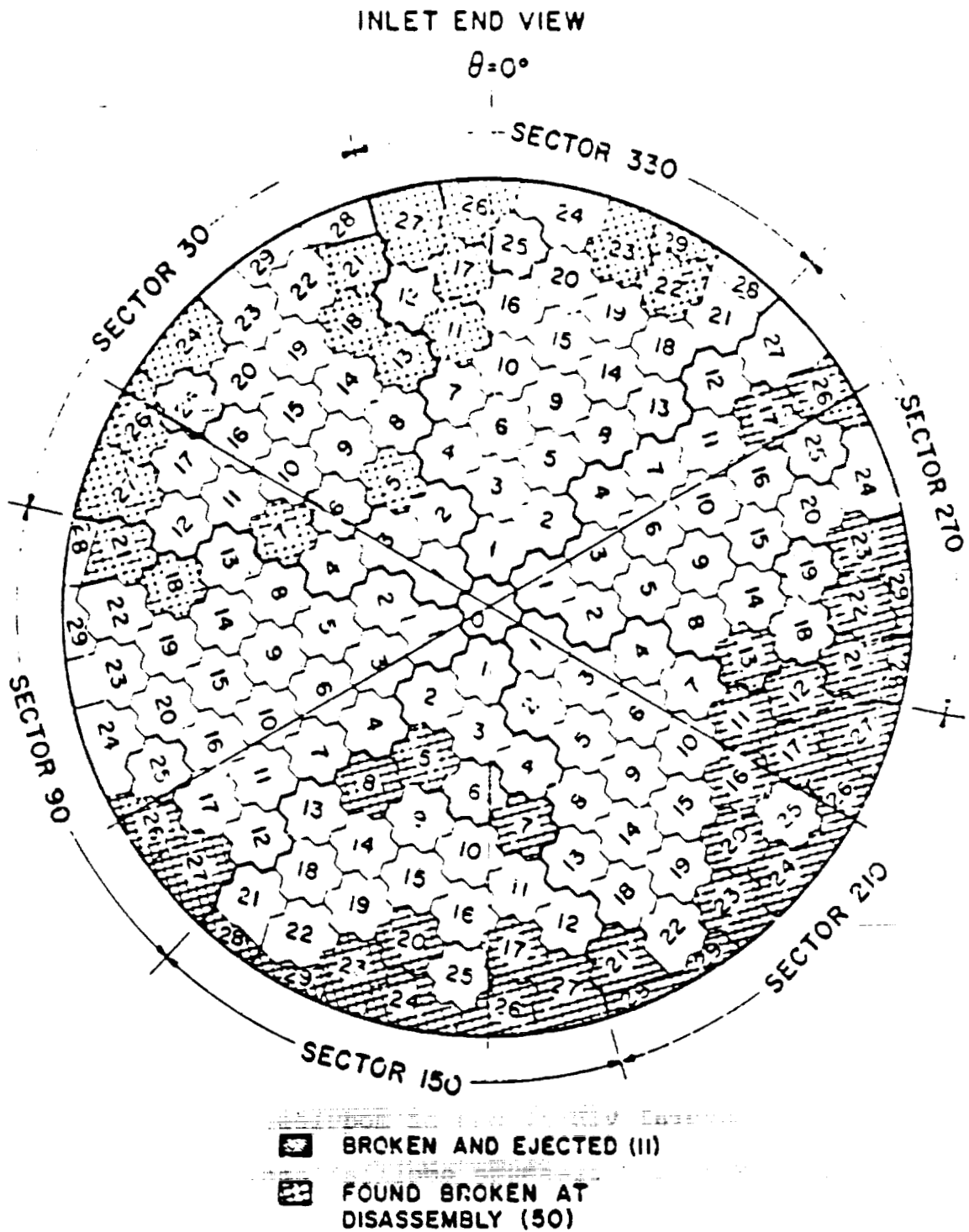


FIGURE 21. CORE INLET END VIEW SHOWING MODULES BROKEN AND EJECTED AND THOSE FOUND BROKEN AT DISASSEMBLY.

6. Some peculiar behavior or "sickness" of the module graphite in the reactor environment.
7. The (oxide) fuel elements may have expanded radially due to abnormally large expansion (caused by the back conversion to UC_2) combined with the thermal lag of the modules during the start-up program. The fuel element/module radial clearance could have then been exceeded, jamming or locking-up the fuel elements in the modules and resulting in module failure by transverse fracture due to excessive fuel-element-induced module tensile loads.

In summary: the main objective of the KIWI B1B test of starting up with liquid hydrogen was successfully achieved. Postulated problems of two-phase flow and concomitant reactivity and power excursions did not occur. The KIWI B1B reactor, however, did not perform satisfactorily and was essentially eliminated from serious consideration for use in the NERVA. This result was not particularly surprising but the test results made completely firm the already favored position of the B4 reactor concept.

KIWI B4A [8]

The KIWI B4A reactor was tested on November 30, 1962 and reached 450 MW (50% power level) when the run was terminated after just a few seconds. The termination was due to ejection of the core occurring with increasing frequency (evidenced by orange flashes in the burning hydrogen exiting from the nozzle). The KIWI B4A is shown in Figure 22 at the test cell.

The objectives of this test were:

1. Operation at 1100 MW, 2278 K (4100 R) exit gas temperature, and 31.75 kg/s (70 lb/s) propellant flow rate.
2. Evaluate the mechanical and neutronic design of the core and core support scheme.
3. Study the effects of using liquid hydrogen as the reactor coolant.

The reactor assembly cutaway is shown in Figure 23 and an axial view is presented in Figure 24.

With the goal of achieving higher power densities, a new extruded, hexagonal fuel rod with a flat-to-flat dimension of 1.905 cm (0.75 in) and a length of 132 cm (52 in) was developed. This reactor element shape became the standard

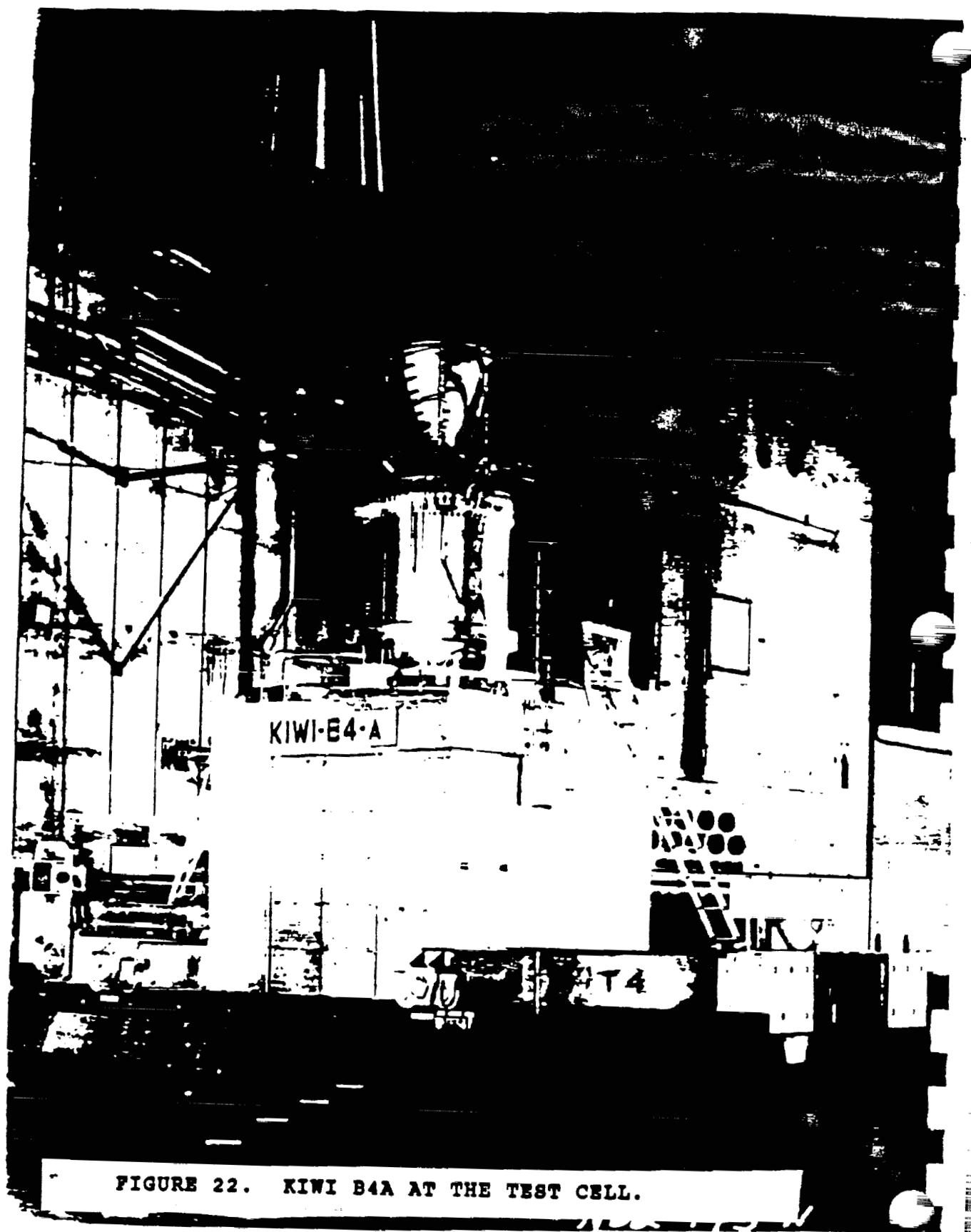


FIGURE 22. KIWI B4A AT THE TEST CELL.

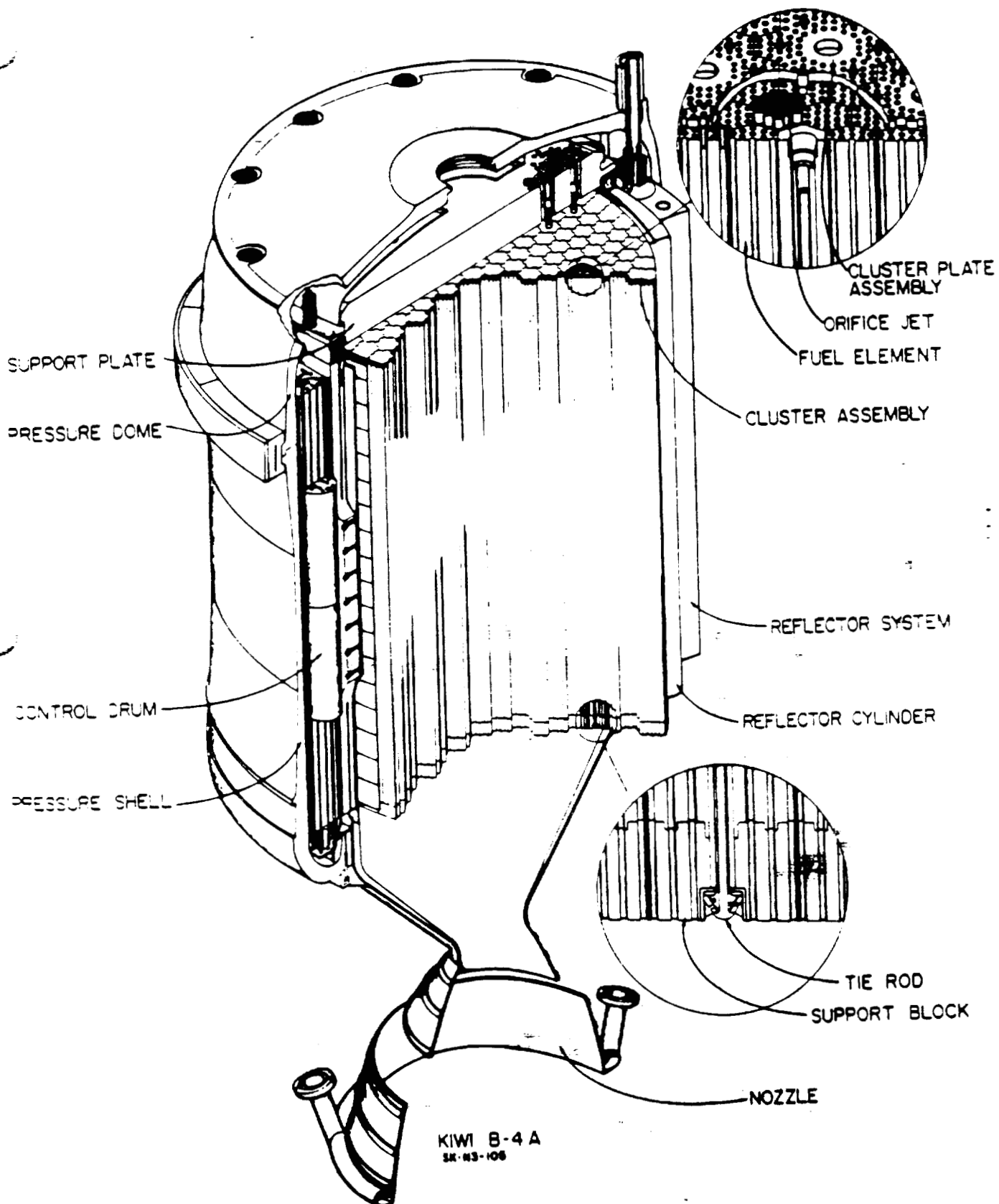


FIGURE 23. KIWI B4A REACTOR ASSEMBLY CUTAWAY.

for all subsequent reactor designs. These elements were still loaded with UO_2 and contained 19 coolant holes, each 0.239 cm (0.094 in) in diameter and located in a triangular array 0.442 cm (0.174 in) between centers. The coolant channels were NbC coated (0.0254 cm thick over a 91.4 cm length at the hot end of the elements) by the tube-cladding process. The exit 1.27 cm (0.5 in) of each element was also coated with niobium carbide. Five different nominal uranium loadings were used to flatten the radial power distribution. The new core of the KIWI B4A contained a closely packed assembly of 1644 of these fuel elements contained within 265 clusters. The cluster configuration consisted of the 6 fuel elements surrounding a center support element-tie rod assembly. Figure 25 is a drawing of the fuel element cluster.

The 7 (6 fuel and 1 central support) elements rested on a hot-end cluster niobium-carbide-coated graphite support block (provided with matching coolant passages). A matching thin perforated aluminum cluster plate was provided at the cluster inlet end for fuel element alignment. The cluster was held together axially by a stainless steel tie rod which acted as a tension member between the support block and the inlet-end aluminum support plate. The fuel element pressure drop load was transmitted to the support block and through the tie rod to the support plate, leaving the graphite fuel elements essentially in compression. The tie rod passed down through the center of the unloaded central element, but was thermally insulated from it by a pyrographite tube fitted against the I.D. surface of the central element. An annulus provided between the pyrographite tube and the tie rod allowed coolant gas flow (12% of the total core flow was bypassed for tie rod cooling).

An aluminum orifice-jet was inserted at the inlet end of each fuel element coolant channel, between the perforated aluminum cluster plate and the fuel element. The hole size for each orifice-jet was selected to control the coolant passage flow rate to provide a uniform core exit gas temperature under design conditions, essentially correcting for variations in radial power generation, fuel loading, and coolant channel flow diameter.

Twelve of the elements contained 16 holes, and thirty elements contained 12 holes. These elements were partial sized and used at the core periphery to transform the core's hexagonal shape to an approximate cylinder. One row of holes was removed from the 19 hole element to form 16 hole elements, and two rows to give 12 hole elements.

As the reactor was run, the first flame flash was observed with the power at 120 MW. Twenty-one seconds after the first flame flash, an intermediate power hold (250 MW) was started and sustained for 37 seconds. A second power hold

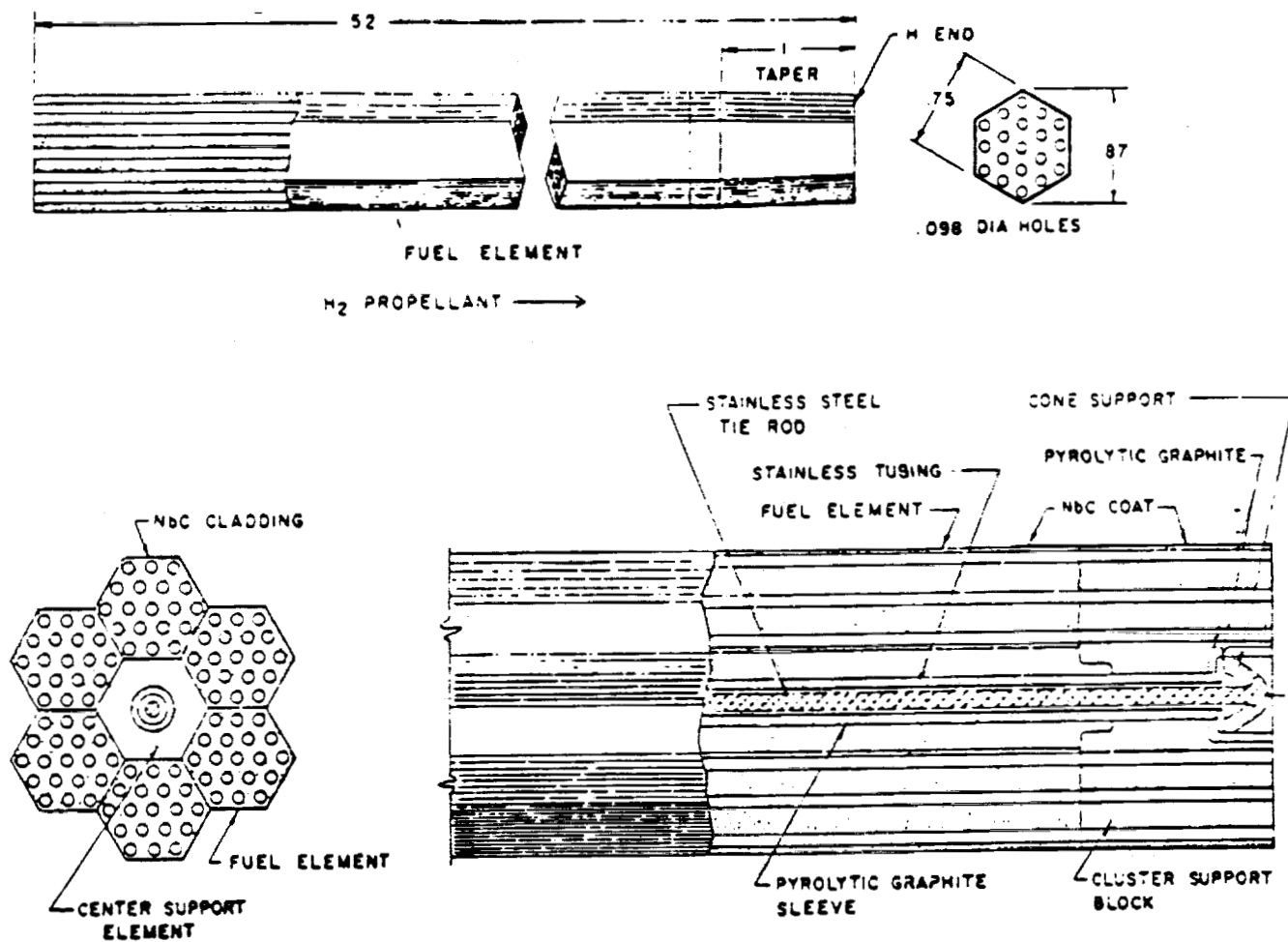


FIGURE 25. KIWI B4A FUEL ELEMENT CLUSTER.

was then initiated (210 MW) and sustained for 32 seconds. After this second power hold, the reactor power increased to a peak of 450 MW with intentions of reaching design goals. However, the flashes became so frequent that the test was terminated 13 seconds after the power started to rise. After the power had been reduced to 7.5 MW and flow rate was down to 5.9 kg/s (13 lb/s), an explosion took place inside the test car privy enclosure, blowing the privy doors open sufficiently to relieve the pressure. A cause for the explosion was determined but is beyond the scope of this report.

A study of the test data indicates that the test demonstrated a properly design liquid hydrogen startup program and adequate control system operation.

Postmortem examination revealed that 97% of the fuel elements were broken into varying lengths and number of pieces. The average cold end length was 98.3 cm (38.7 in). About 1/3 of the 45 whole elements retrieved had been pyrographite coated. 91% of the center support unloaded elements were broken but not as severely as the fuel elements. 80% of the unbroken center elements had been "shaved" (i.e. the O.D. surface was machined down at eight axial locations). Support blocks for the most part were only slightly damaged except for a few irregular blocks. 31 of the 36 insulating slats were found broken. About 20% of the fuel elements had one or more of their orifices partially or completely plugged with adhesive (glyptal). Nearly all thermocouples were found damaged and the thermocouples in turn had apparently caused some damage to fuel elements and cluster plates.

Although the core fuel elements were either ejected or broken, the loss of moderator graphite had been offset by an increase in neutron moderation, caused by the increasing density of the hydrogen in the core and reactor power build up had been sustained.

Post-test examination of the nozzle revealed two types of damage on the nozzle tubes. First, thermal buckling of the tubes occurred at the convergent end of the nozzle, where the tube diameter is a maximum. It was noted that such buckling had appeared in every nozzle which had undergone a hot firing, either on a reactor or chemically. It was also hoped that a new modified nozzle design, underway at Rocketdyne, would eliminate this problem through the doubling of the number of tubes in this region. The second type of damage which was found was denting of the tubes by solids in the hot gas stream. In the convergent section of the B4A nozzle there were relatively few of these dents and they were confined to the region near the throat. Much more extensive damage occurred in the divergent section of the

nozzle and was characteristic of previous runs in which material was ejected through the nozzle.

Although (apparently) relatively unalarmed by the post test nozzle examination, the possibility of nozzle failure due to vibration and fatigue was investigated. Unknown by ROVER members at the time of the conclusion of the KIWI B4A test was the fact that the Rocketdyne nozzle would develop severe problems in the later KIWI B4D test.

The disturbing results of the KIWI B1B and KIWI B4A caused the government to reassess the planned pace of the nuclear rocket program. Subsequent KIWI reactor tests were delayed until improvements could be incorporated in the core structure of KIWI reactors.

POST KIWI B4A TESTS [9]

During 1963-1964, several cold-flow tests of KIWI B type reactors were carried out to determine the cause and find solutions for the severe structural damage that occurred in the previous tests¹. The cold-flow reactor tests used fuel elements identical to the powered reactors excepts that they had no fissionable material and, therefore, produced no power. These tests were performed with gaseous nitrogen, helium, and hydrogen, and demonstrated that the structural core damage was due to flow-induced vibrations. It was found that a dynamic flow instability in the clearance gaps between adjacent fuel elements caused a severe vibration leading to fuel element fracture. Based on these test results, design changes were successful in eliminating core vibrations and demonstrated in four cold-flow tests. These changes were incorporated into the subsequent KIWI B4D reactor.

KIWI B4D [10]

The KIWI B4D incorporated post KIWI B4A design changes brought about to eliminate core vibrations and fracture. The KIWI B4D was tested at full power (990 MW) on May 13, 1964 with no evidence of any core vibration or ejection of fuel element fragments. However, the high power portion of the test was terminated after 64 seconds because of a hydrogen leak at the nozzle throat which resulted in an extensive fire around the reactor. Cutaway and axial view of the reactor are provided in Figures 26 and 27 respectively.

¹ Two tests were the KIWI-Pie experiment and the KIWI B4A-CF (cold flow) reactor mockup.

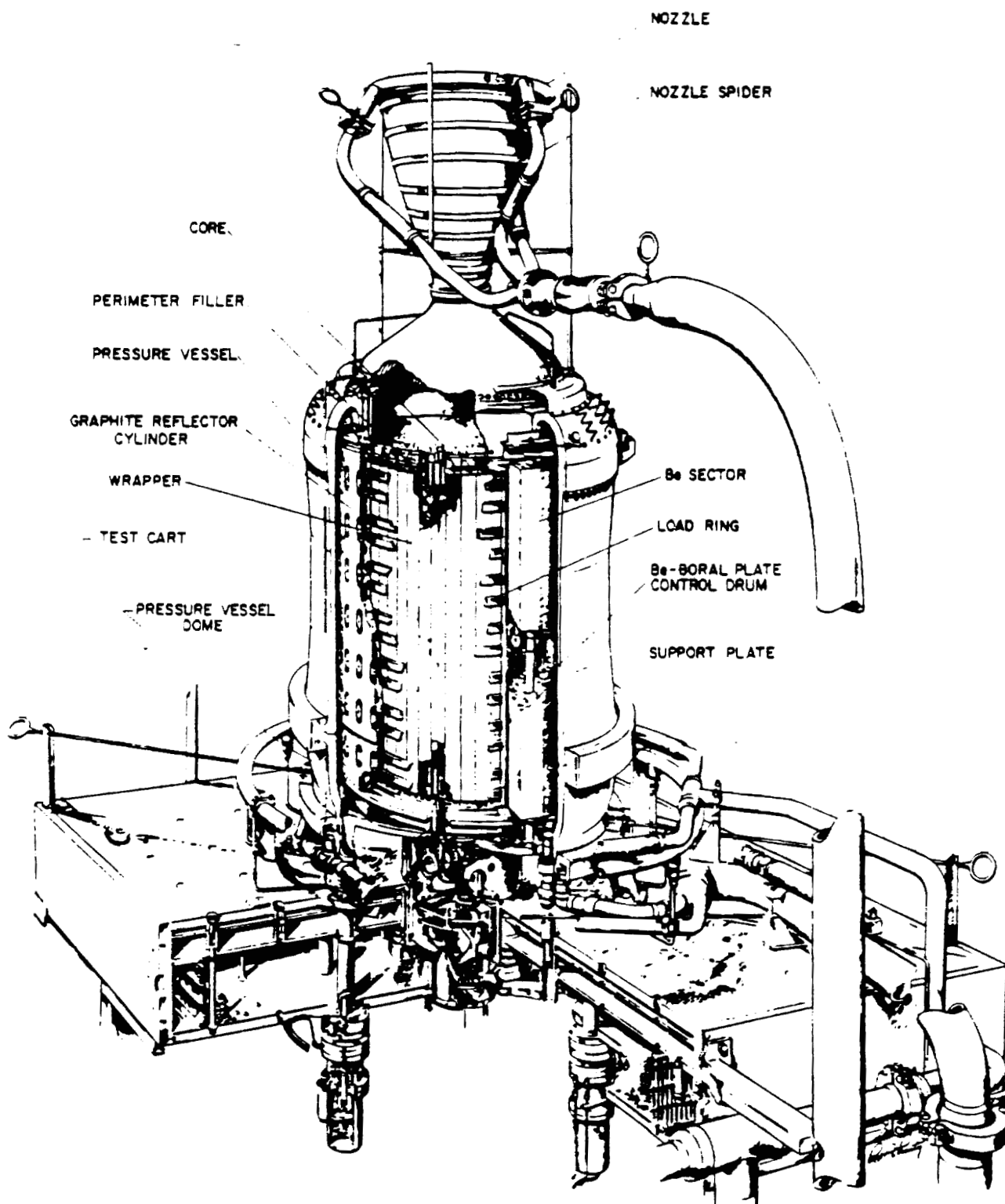


FIGURE 26. CUTAWAY VIEW OF KIWI B4D.



The objectives of the full power test were:

1. To investigate the structural integrity and dynamic stability of the B4D design under reactor operating conditions of full design flow rate and three-fourths of full design temperature and neutronic power.
2. To measure by means of test instrumentation and postmortem examination, the thermal, flow, and neutronic performance for comparison with design predictions.
3. To obtain information on the effects of operating time with the above environmental conditions on the overall test system including reactor, nozzle, feed system, and plumbing.

The secondary objectives of the KIWI B4D test were:

1. To obtain information on reactor cooldown using liquid hydrogen.
2. To measure the pump discharge pressure to reactor inlet pressure transfer function using cross-correlation techniques.
3. To perform an automatic startup from source power.

The core design and support system were similar to those of the KIWI B4A with four major modifications. These were:

1. A hot end seal.
2. Leaf springs for lateral core support.
3. Coolant flow slots in the periphery filler elements.
4. A flexible metal wrapper surrounding the core to prevent radial flow between the core and the expansion annulus outside the core.

The core contained 1542 fuel elements of two major types. The major portion of the core was comprised of uranium oxide-loaded elements but 212 elements were loaded with pyro-coated uranium carbide beads of the B4E design¹. There were 108 of the beaded elements located around the central cluster and 104 near the perimeter. The basic fuel element was 131.78 cm (51.88 in) in length. It had a

¹ The reason for this new fuel development is discussed in the KIWI B4E section.

hexagonal cross section of approximately 1.9 cm (0.75 in) across flats and contained 19 coolant holes. Forty-two of the outer elements were cut to the outline of the core perimeter and had 16 coolant channels. Twelve different oxide loadings and six different beaded element loadings were used in the core. The varied loadings were positioned in the core in a manner designed to even radial fission distribution and reduce sharp thermal peaking (thermal spike) on the core perimeter. For corresponding neutronic behavior, the beaded elements contained a slightly higher percentage of ^{235}U than their oxide equivalents. This increase was necessary to offset the effect of larger coolant channels and full length niobium carbide coating in the beaded elements.

This was the first time that a completely automatic start was accomplished for a nuclear rocket reactor. This technique brought the reactor from a sub-critical, shutdown condition to a pre-selected (pre-program) power level in a rapid and safe manner. The startup was achieved by programming the control rods out in an open-loop manner and then switching to closed power loop control at a pre-selected power level. Only one range of instrumentation with a fixed ion chamber position was required to change from source level to full power.

After 64 seconds at full power, the run was terminated due to several nozzle tubes rupturing causing a hydrogen leak at the nozzle throat section where the interstices between the coolant tubes and shell vented to the atmosphere. This lead to an extensive fire around the reactor. It was determined that the ruptures were caused by entrapment of liquid air flowing down the external nozzle tube walls, followed by ozone formation. This produced "micro explosions" between the tubes and the pressure shell causing local ruptures of the nozzle tubes. The nozzle tube ruptures are shown in Figure 28. With the exception of the coolant tube damage, the Rocketdyne nozzle performed generally as predicted during the full power test. A slight amount of tube-buckling occurred, as observed in previous reactor tests.

The reactor cooldown was performed using both hydrogen and nitrogen. At the end of the hot run gaseous hydrogen was initially used during cooldown. After about 2 minutes, gaseous nitrogen was used in place of the hydrogen. The gaseous nitrogen was continuous flow with step reductions over 606 seconds (3266 kg of nitrogen used). After this time, pulse flow gaseous nitrogen was used. Fifteen pulses of different durations (from 60-410 seconds) were made with gaseous nitrogen.

Upon post-test inspection the core was found to be in generally good condition. No broken fuel elements were found and no mechanical damage was noted on any core

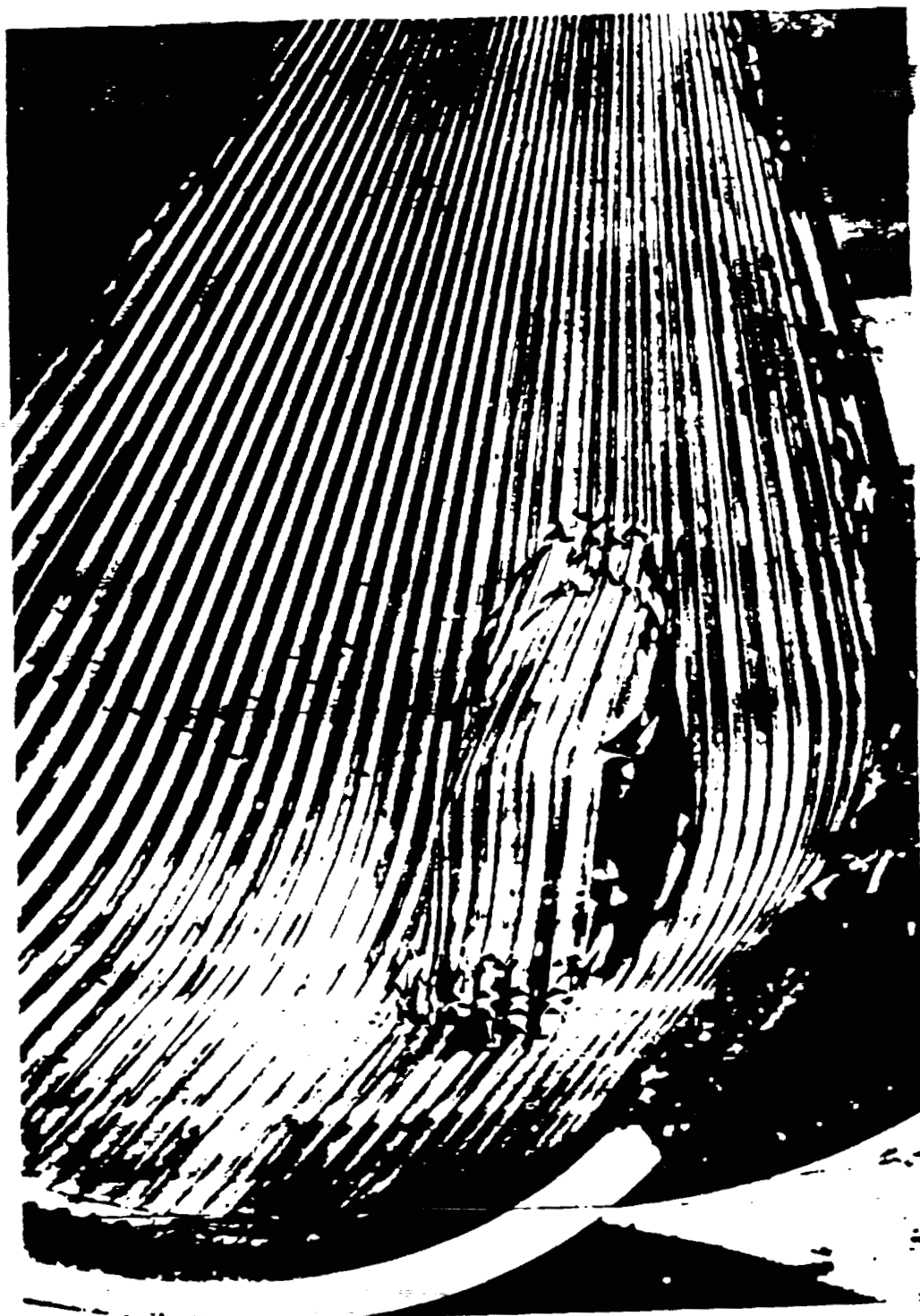


FIGURE 28. KIWI B4D NOZZLE TUBE RUPTURES.

component. Although corrosion was quite extensive and deep (20-25 mils) for the peripheral row of elements, most of the other elements appeared almost unused. About 50 cluster plate openings and/or jets contained foreign material (plastics, metal chips, wire, and glyptal). Three fuel element anomalies were found: One oxide element was found with a hole through the wall, two adjacent fuel elements indicated an apparent deposition on the end coat from the element interior, and one element showed cracks radiating away from the coolant holes in a circular pattern around the face exit end (crazed end coat).

KIWI B4E [11, 12, 13]

The KIWI B4E was first tested on August 28, 1964. The reactor operated for more than 12 minutes, with 8 minutes at full power (937 MW)¹. The fuel element exit gas temperature was held at 2222 K (4000 R) and the reactor propellant flow rate was 31.8 kg/s (70 lb/s). The reactor operation was smooth and "uneventful". The run time was limited by the quantity of available liquid hydrogen. Figures 29 and 30 provide reactor cutaway and axial views, respectively.

The reactor was restarted on September 10, 1964 and operated at 882 MW for 2.5 minutes. The core and fuel exit gas temperatures, as well as propellant flow rate, were approximately the same as for the August 28, 1964 run.

The primary objectives of the full power run were:

1. To operate the reactor at conditions near the core design point values of flow, temperature, and power.
2. To obtain information on the effects of operating time at the above conditions on the reactor and test system including nozzle, feed system, and plumbing.
3. To measure (by means of test instrumentation and postmortem examination) the thermal, flow, and neutronic performance for comparison with design predictions.

Secondary objectives were:

1. To close the temperature control loop using core material temperature thermocouples rather than core exit gas thermocouples.

¹ One reference, LA-3185-MS, claims the test occurred on August 27, 1964 and achieved a power of 907 MW.

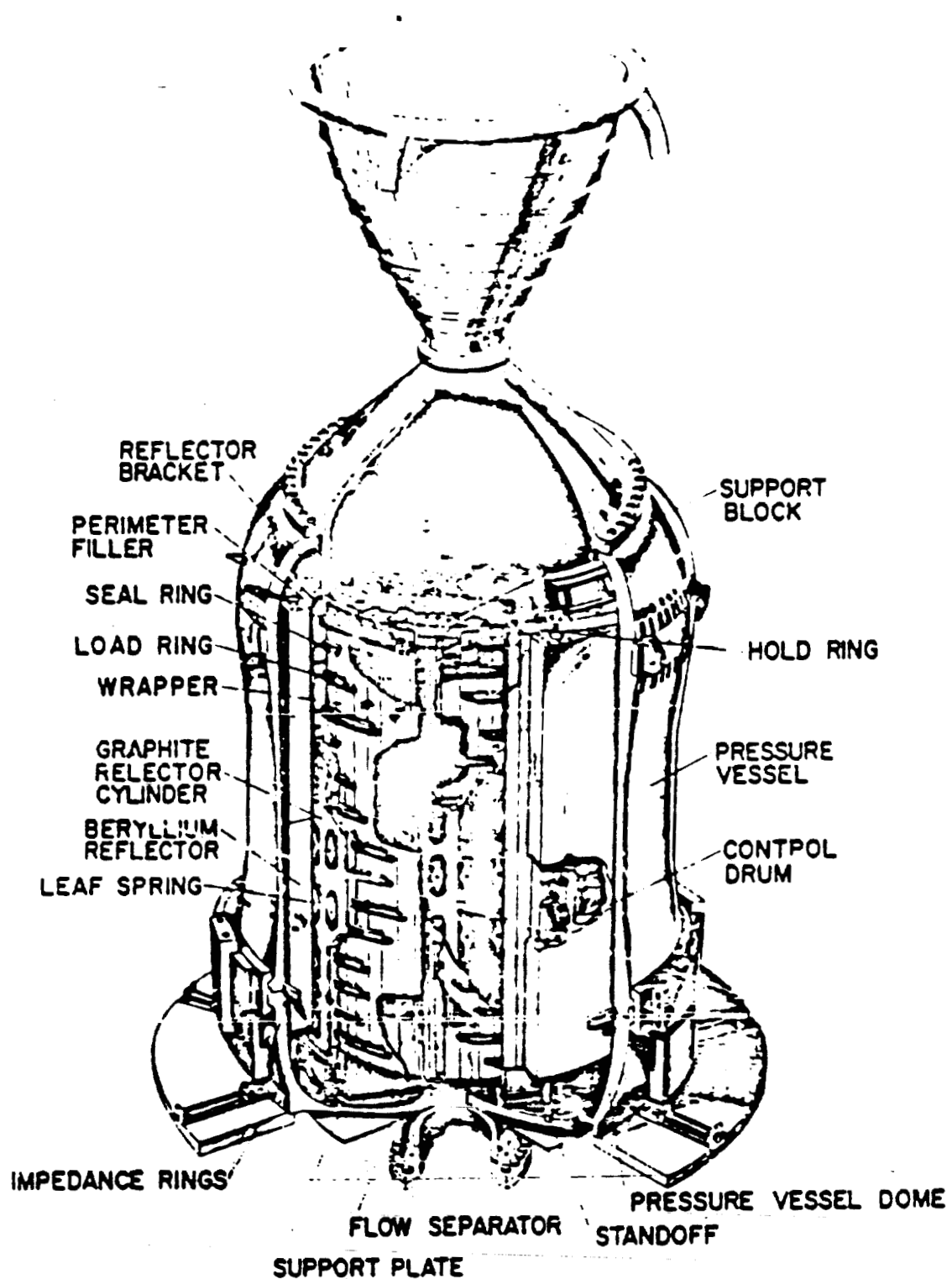
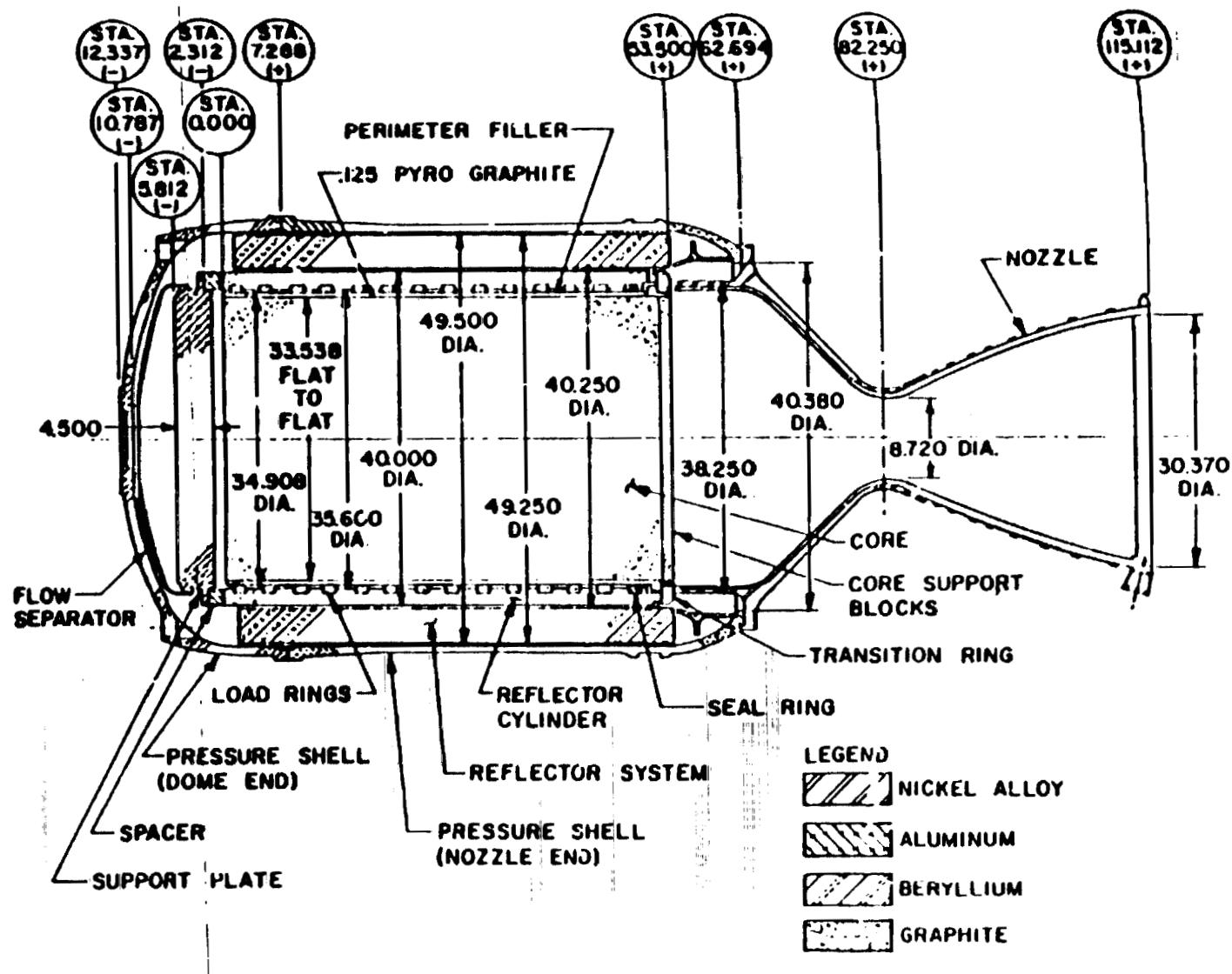


FIGURE 29. KIWI B4E REACTOR CUTAWAY.

FIGURE 30. SCHEMATIC VIEW OF KIMI BAE.



2. To measure the effect of hydrogen density on reactivity.
3. To obtain information on reactor cooldown using liquid hydrogen.
4. To measure the pump discharge pressure to reactor inlet pressure transfer function using a cross-correlation technique.

The basic design and components of the KIWI B4E were very similar to those of KIWI B4D. The following important changes were made in KIWI B4E

1. All fuel elements in the core were graphite loaded with pyrocarbon-coated uranium carbide beads.
2. The temperature control system for the reactor was revised to use the core material temperature at axial midplane as the control variable rather than the core exit gas temperature.
3. A reduction in core periphery coolant flow was made to reduce the degradation of reactor exit temperature by such bypass flow.
4. A Rocketdyne nozzle of the RN-6 design was used instead of the RN-2 design. An inverted shroud was mounted on this nozzle to prevent liquid air from entering the interstices between the nozzle shell and the coolant tubes. It was believed that the presence of liquid air in these regions caused or contributed to the damage experienced by the RN-2 nozzle on the KIWI B4D test.

The primary differences between the RN-6 nozzle and the RN-2 nozzle (used on all previous KIWI B reactors) were a change in the nozzle-pressure vessel flange and a coolant tube splice in the nozzle convergent section of the RN-6. A helium-purged shroud was provided on the B4E nozzle to prevent liquid air from entering the interstices and to prevent fires in the event of external-venting nozzle leaks. The shroud pressure was monitored during the test but did not rise. The helium-purged shroud successfully accomplished its goals.

The KIWI B4E was the first reactor to exclusively utilize coated UC_2 particles in place of UO_2 . The core consisted of 1500 full-length (132 cm), 19-hole, hexagonal fuel elements. An additional 42 fuel elements on the core periphery were reduced in size so as to contain only 16 coolant holes. This was done to provide a more satisfactory periphery geometry. The coolant holes were slightly less than 0.254

cm (0.10 in) in diameter and bores and exterior surfaces of the exit ends were NbC coated by the tube-cladding process. The carbide coating had a nominal thickness of 0.00508 cm (0.002 in). Ten different uranium loadings were used to flatten, in a rough way, the radial power distribution. Adjacent loadings differed by an increment of approximately 12%. The fuel element and cluster assembly are shown in Figures 31 and 32, respectively.

On September 10, 1964 the KIWI B4E was restarted. The decision to conduct this test was based upon the preliminary analyses of the test data from the August 28 run which indicated the reactor was in excellent condition and could be rerun. The reactor was restarted and run at near full power for approximately 2.5 minutes. Reactor shutdown was accomplished as planned. The capability to rerun reactors, as demonstrated by this experiment, came much earlier than anticipated, and was deemed a significant step towards the economical development of nuclear rockets.

During the KIWI B4E tests, there was no evidence of vibrations encountered in the KIWI B4A power test. The tests also confirmed the results of the KIWI B4D experiment; that the current KIWI B4 design was sound.

Post-test inspection of the core revealed that although it was considered to be in excellent post-run condition, visual observations of graphite core components indicated several areas of corrosion. Two areas of localized corrosion were element surfaces at the core periphery and on the exterior surfaces of elements adjacent to the center unloaded elements with thermocouple grooves. Five areas of general corrosion were: 1) element flats near the hot end at the undercut area, 2) element flats at the mid-range of the element length, 3) element ends, 4) uncoated bores of the center unloaded elements and support blocks, and 5) fuel element bores behind liner cracks.

Fourteen elements were found to have either corroded through or broken during the run or cooldown periods. Eleven of these were 16 hole peripheral elements, and three were 19 hole peripheral elements.

Details of coated UC₂ fuel particles

Prior to the KIWI B4E, UO₂ particles were used in the fuel. The major problem with oxide-loaded fuel elements was the so-called back-reaction. Micrometer-size UC₂ particles are extremely reactive and revert to oxide in the presence of air, particularly humid air. Thus oxide-carbide-oxide reaction occurred during each heating and storage cycle, including graphitizing, coating, and reactor operation; and each cycle caused loss of carbon by CO gas evolution and

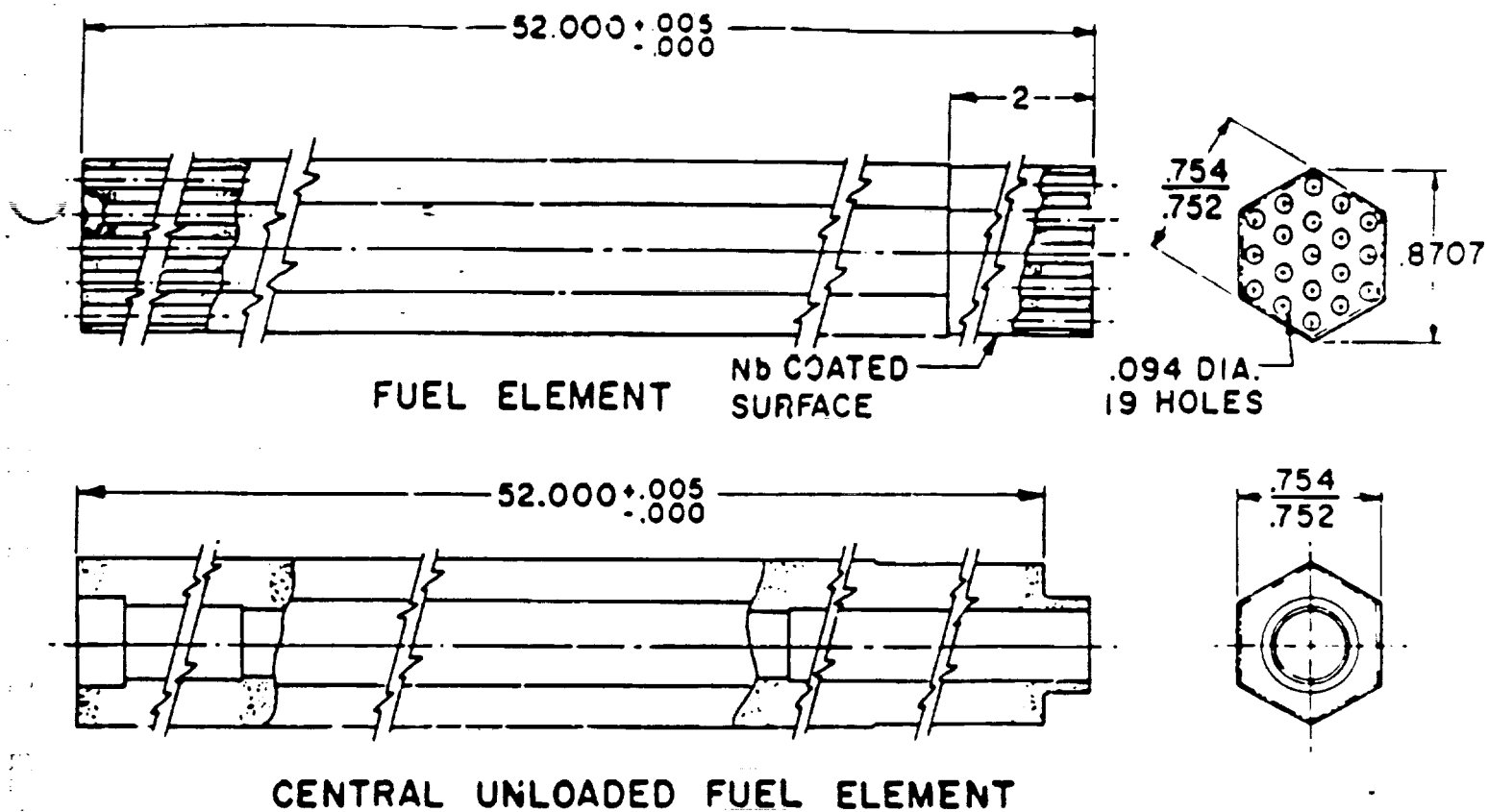


FIGURE 31. KIWI B4E FUEL ELEMENT.

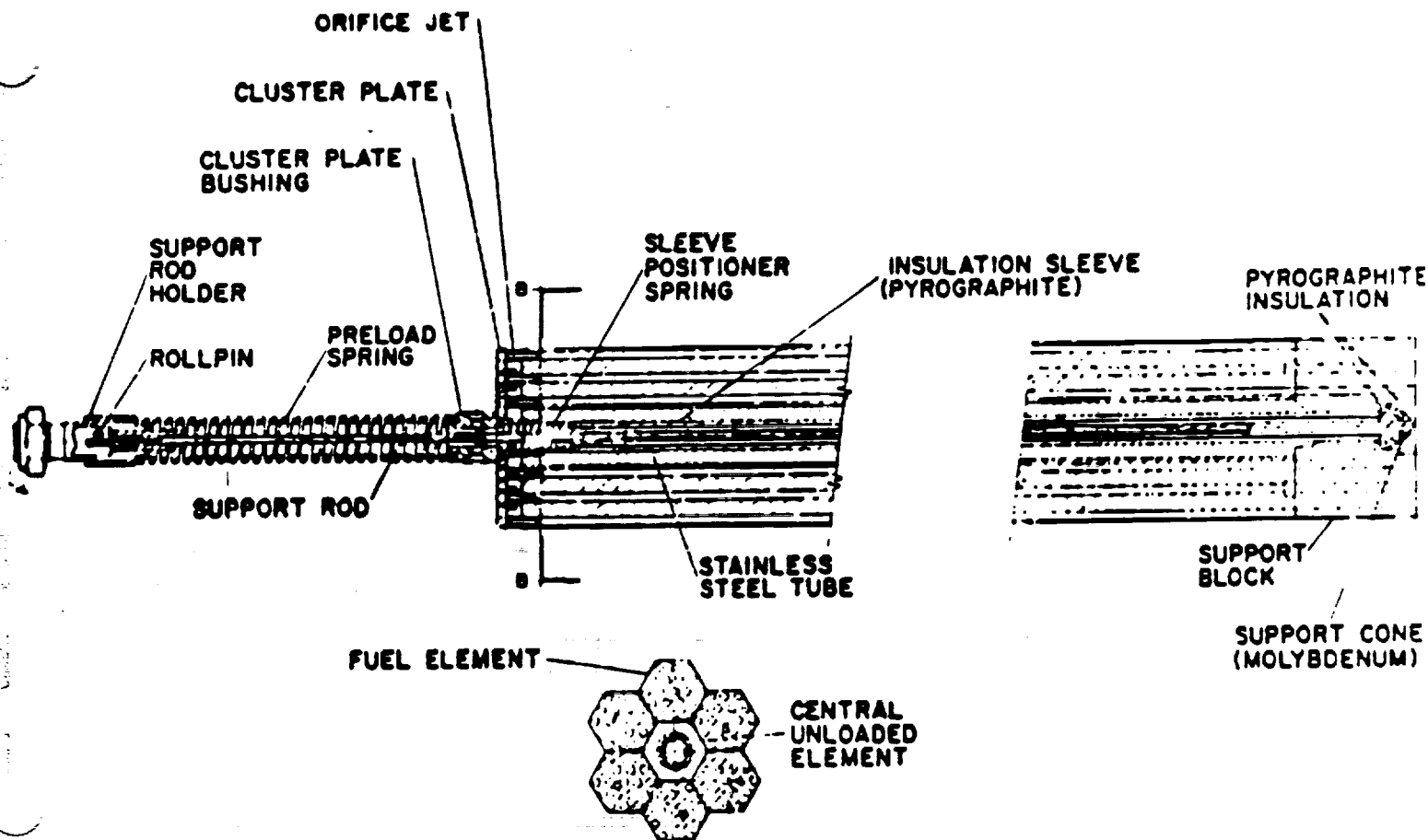


FIGURE 32. KIWI B4E CENTER ASSEMBLY.

degraded the element. Dimensional changes also were noted in stored elements. Oxidation of the UC_2 loading material caused the element to swell as much as 4% so that the final dimensions could not be controlled.

The solution to this problem was the introduction of UC_2 particles that were considerably larger, 50-150 micrometer diameter, and coated with 25 micrometers of pyrolytic graphite.

Additional Reactor Progress

Also in September, 1964 two KIWI reactors were positioned adjacent to each other to determine the influence of one reactor on another in a cluster. The results of this zero-power experiment verified that there is little interaction and that, from a nuclear standpoint, nuclear engines could be operated in clusters, similar to chemical engines. The ability to cluster nuclear rocket engines could provide great flexibility in the development of nuclear propulsion systems to meet the requirements for space missions and greatly increase reliability for these missions.

KIWI TNT [14, 15]

The KIWI TNT (Transient Nuclear Test), the last to carry the KIWI name, was not part of the regular KIWI series. It was a special flight safety test to study the behavior and effluent of a KIWI-type reactor undergoing sudden power surges or excursions as might happen if a chemical rocket booster aborted and dropped a non-critical nuclear reactor in the ocean where the water, being a good neutron moderator, would increase the likelihood of fissions and could make the reactor go critical very quickly¹. This test would also determine if the reactor could be "disassembled" in space after its mission was complete.

The KIWI type of reactor was designed to run normally at temperatures in excess of 2473 K (4451 R). Under accidental conditions temperatures were expected to be in the 4273-4723 K (7691 - 8501 R) range. Little was known about the physical properties and equation of state of graphite under the time-temperature-pressure conditions which are present in a large nuclear excursion. Furthermore, it had been impossible to achieve such conditions within the laboratory. The molecular species of the vapor produced from high temperature graphite systems

¹ Actually serious concern also existed if a nuclear engine would return to earth over a land-mass since it could fall into a reservoir or someone's swimming pool!

was not well known. Vapor forms ranging from C_1 to C_{10} had been observed in experiments.

The objectives, then, were:

1. To measure the reaction history and total number of fissions produced under a known reactivity and to compare these with theoretical prediction. The experimental results would improve the assumptions required in the calculational program for estimating accident situations.
2. To determine the mechanisms for energy release, i.e., distribution of fission energy between core heating and vaporization, and kinetic energies.
3. To determine the nature of the core break-up under such a transient and to determine the degree of core vaporization and the resulting particle size distribution. These data, although not directly applicable, would provide information on the feasibility of a nuclear destruct system.
4. To measure the release to the atmosphere and dispersion of the fission debris under known initial conditions to improve techniques for estimating and evaluating the release for other accident conditions.
5. To measure the radiation environment during and after the power transient.
6. To evaluate launch site damage and clean-up techniques required for such an accident.

The reactor used for the test was a modified KIWI core and had the same characteristics as the KIWI B4E. The KIWI core was comprised of uranium carbide loaded graphite elements surrounded by a graphite reflector cylinder and a beryllium reflector enclosed in an aluminum pressure shell. The fuel elements extruded particularly for the TNT reactor were uncoated with the exception of some elements which were rejects from propulsion reactor production lots (about 800) which were coated with niobium carbide in the 19 coolant holes and for approximately 2.54 cm (1 in) axially on the exterior surfaces of the hot end. Twelve different types of elements were used to assemble the TNT core. Reactivity control was provided by 12 rotary drums which were located in the beryllium reflector. No propellant was used.

The control drum actuators of a KIWI reactor were modified so that the control drum motion (poison withdrawal rates) was speeded up by a factor of 89 (from 45 deg/s to 4000 deg/s). The reactor was also special in the sense that with

the control drums fully inserted, the reactor was less than 1 subcritical. The excess reactivity required, relative to prompt critical, was \$6. Some of the reactor poisons normally present were removed to reduce the shutdown reactivity to \$0.60 relative to delayed critical and to provide \$8.40 excess reactivity relative to delayed critical. Specific modifications made to the actuators and hydraulic system were:

1. The area of the hydraulic ports into and out of the actuator was increased 50%.
2. The diameter of the hydraulic lines from manifold to control valves to actuators was increased.
3. Close-coupled hydraulic manifolds with accumulators were used.
4. Control valves with a capacity of 30 gpm were used instead of the normal 3.5 gpm capacity valves.
5. Hydraulic oil pressure was increased from 4826 kPa (700 psi) to 9308 kPa (1350 psi).
6. Delay circuitry was used in the firing control chassis to obtain the necessary simultaneity.

The entire reactor was mounted on a railroad car specially constructed for this test. The KIWI TNT test vehicle is shown schematically in Figure 33.

The KIWI-B type reactor was deliberately destroyed on January 12, 1965 at the Nuclear Rocket Development Station, Jackass Flats, Nevada, by placing it on a fast excursion by rotating the poison in the control drums as rapidly as possible. The test was successfully carried out, and essentially all objectives were met. The following measurements were made:

1. Reactivity time history.
2. Fission rate time history.
3. Total fissions.
4. Core temperatures.
5. Core pressures.
6. Core and reflector motion.
7. External pressures.
8. Radiation effects.

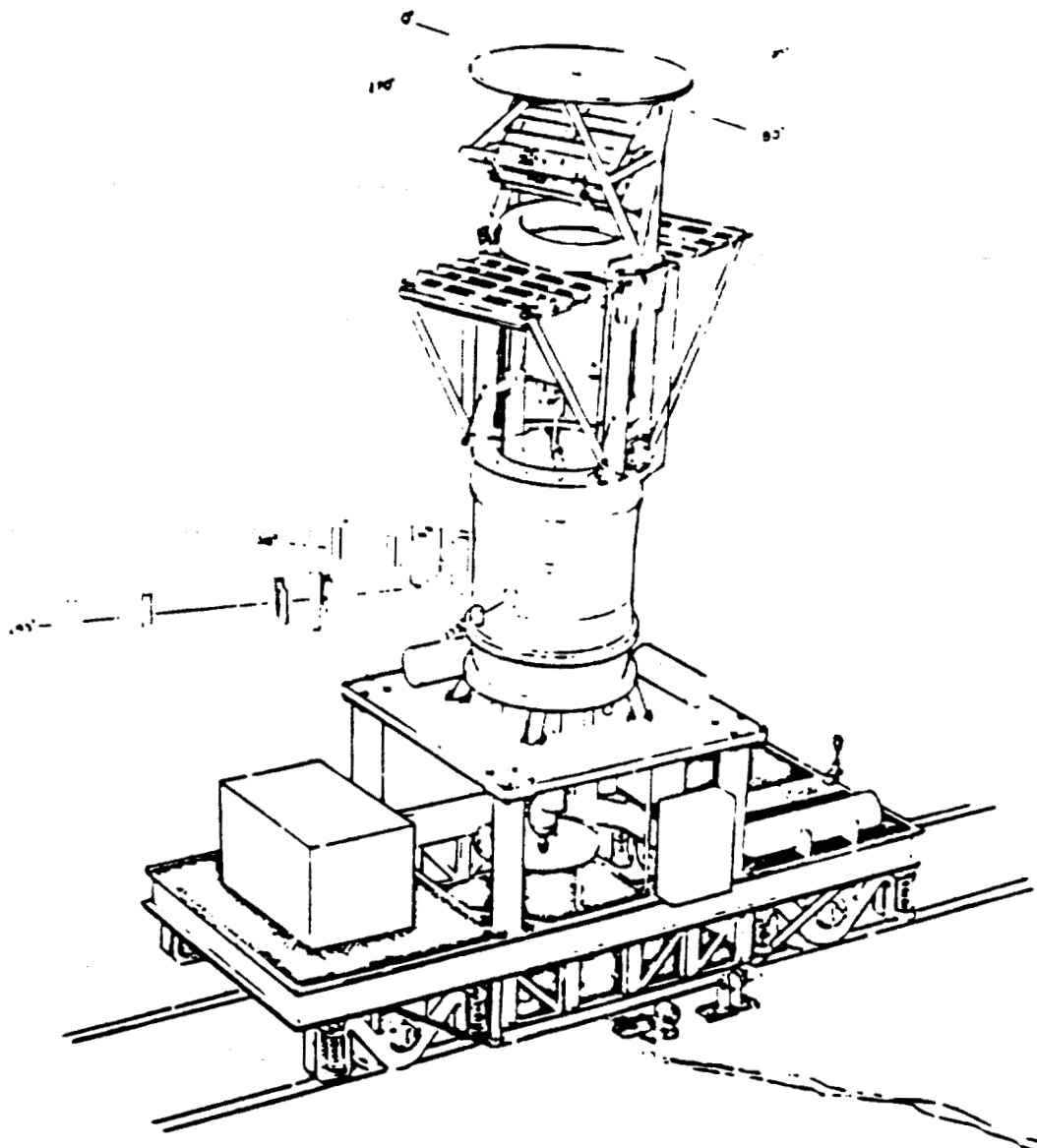


FIGURE 33. SCHEMATIC OF ASSEMBLED KIWI TNT TEST VEHICLE.

9. Cloud formation and composition. (Collected by two B-47-C aircraft).
10. Fragmentation and particle study.
11. Geographic distribution of debris.

The KIWI TNT test provided further empirical data on the nature of nuclear excursions in KIWI reactors. Some of the results were: 1) Core temperature measurements indicated a temperature of about 2167 K (3900 R); 2) Within a 7620 m (25000 ft) radius, only about 50% of the core material could be accounted for. The remainder presumably either burned in the air or was so fine as to be carried further downwind in the cloud; 3) It was estimated on the basis of the total energy which was produced by the excursion that only 5-15% of the core could have been vaporized; 4) The heaviest piece of debris found was a portion of the pressure vessel approximately 0.91 m (3 ft) square and weighing 67 kg (148 lb). It was located 229 m (750 ft) from the reactor. Another piece of the pressure vessel weighing 44 kg (98 lb) was found 457-533 m (1500-1750 ft) from the reactor; 5) The total number of fissions was determined to be approximately 3.1×10^{20} .

The experimental results from the KIWI TNT excursion provided the basic experimental information required for the general analysis of potential accidents of interest to the ROVER Flight Safety Program. A report, LA-3358-MS, "Safety Neutronics for ROVER Reactors", describes an analysis and application of the KIWI TNT results to potential nuclear rocket accidents. Note: the explosion was mechanical, not nuclear.

PHOEBUS 1A [16]

Essentially an extension of the KIWI project, the PHOEBUS class advanced graphite reactors were developed to increase the specific impulse, the power density in the core, and the power level. PHOEBUS 1A was tested on June 25, 1965 at full power (1090 MW) for 10.5 minutes. The intense radiation environment caused capacitance gauges to produce erroneous liquid hydrogen tank measurements and the supply was exhausted while the reactor was still in operation causing the core to overheat and become damaged. This course of events, however, was not related to any defect in the reactor.

The PHOEBUS project had a goal of a 5000 MW reactor, using the KIWI B4E as the starting point while incorporating new improved fuel elements as well as other detailed improvements to the reactor and nozzle. The power density increase was to be achieved mainly by enlarging the diameter

of the coolant flow channels in the fuel elements from 2.54 mm to 2.79 mm (0.10 - 0.11 in) to reduce thermal stress and core pressure drop. Because graphite is a good neutron moderator, as with KIWI, PHOEBUS was epithermal rather than a fast reactor.

The test objectives of the PHOEBUS 1A were:

1. To operate the reactor at design-point conditions of mass flow rate, temperature, and power for the maximum time allowed by the liquid hydrogen supply in order to evaluate (by postmortem examination) the relative merits of various design changes aimed at reducing corrosion.
2. To obtain data which could be used to predict the operation of the first full power restart.
3. To obtain data to be used in determining temperature, power, and pressure transfer functions in the reactor system.

The PHOEBUS 1A core consisted of 1534 full-length (132 cm), hexagonal fuel elements loaded with pyrolytic-graphite-coated UC_2 particles. Each fuel element contained 19 coolant holes, except for the 42 elements which were cut to the core contour at the periphery and contained 16 holes. The bores were NbC clad by the CVD process. Twenty-seven different uranium carbide loadings were used to help flatten the radial power distribution. The reactor and fuel element cluster are shown in Figures 34 and 35, respectively.

After startup of the PHOEBUS 1A, an intermediate power (565 MW, flow rate 26.8 kg/s) hold was achieved and held for about 1 minute. During this intermediate power hold, the nozzle chamber temperature was 1575 K (2835 R) and the fuel element temperature was 1700 K (3060 R). The power was increased to 1090 MW (flowrate 31.4 kg/s) with corresponding chamber and fuel temperatures 2278 and 2444 K (4100 and 4400 R), respectively. The full power hold duration was about 10.5 minutes before the dewars ran dry and the turbopump overspeeded, initiating an automatic reactor scram and flow shutdown.

Due to the premature shutdown, objective 1 was partially achieved. The second and third objective were successfully achieved. The inability to rerun the reactor for further duration testing prevented a comparison of hot-gas electrical testing of fuel elements with expected reactor environmental testing.

During the test, a leak occurred in the propellant ducting system of the reactor/test car complex. Ignition of the leak occurred and the fire continued throughout the full

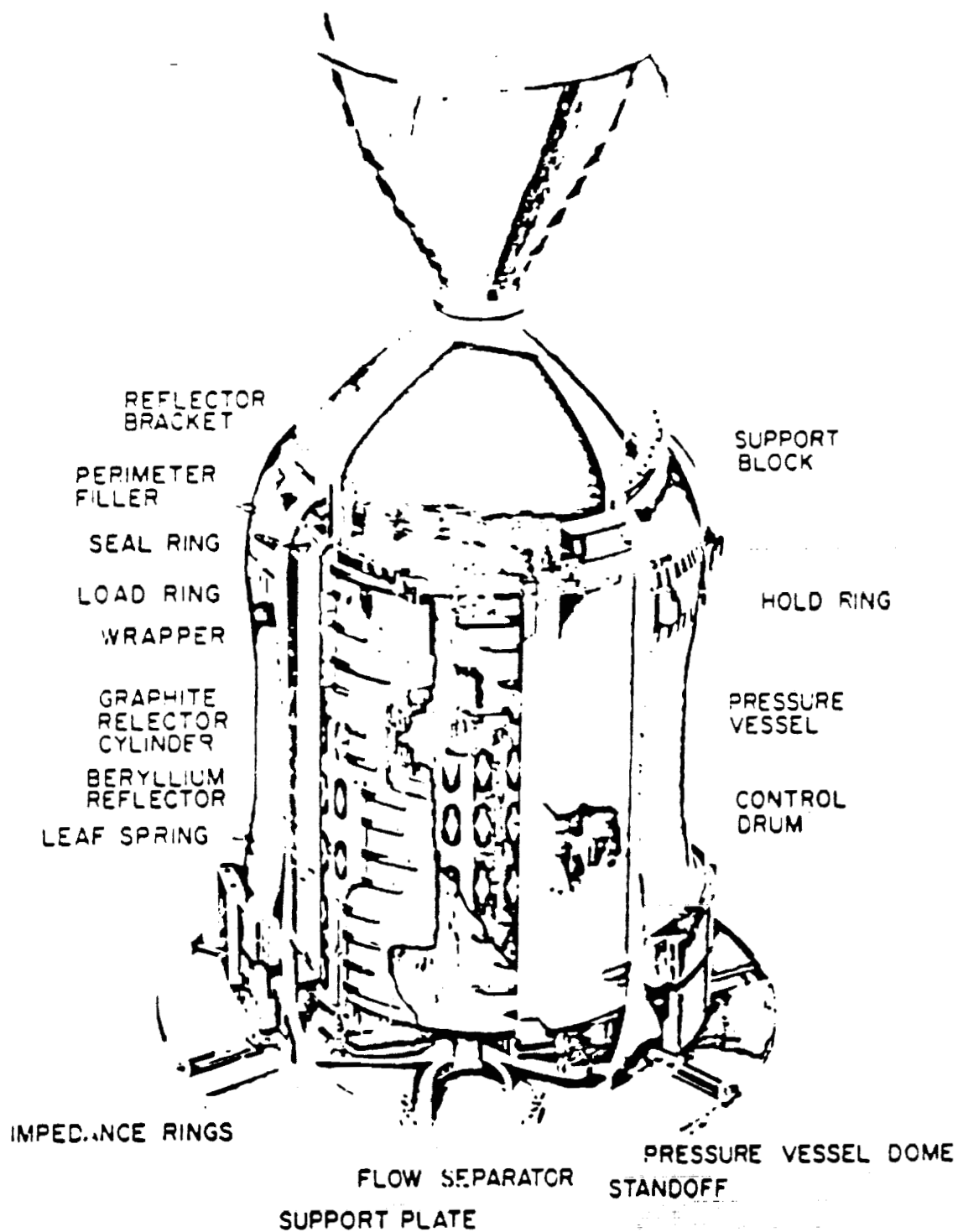


FIGURE 34. CUTAWAY VIEW OF PHOEBUS 1A.

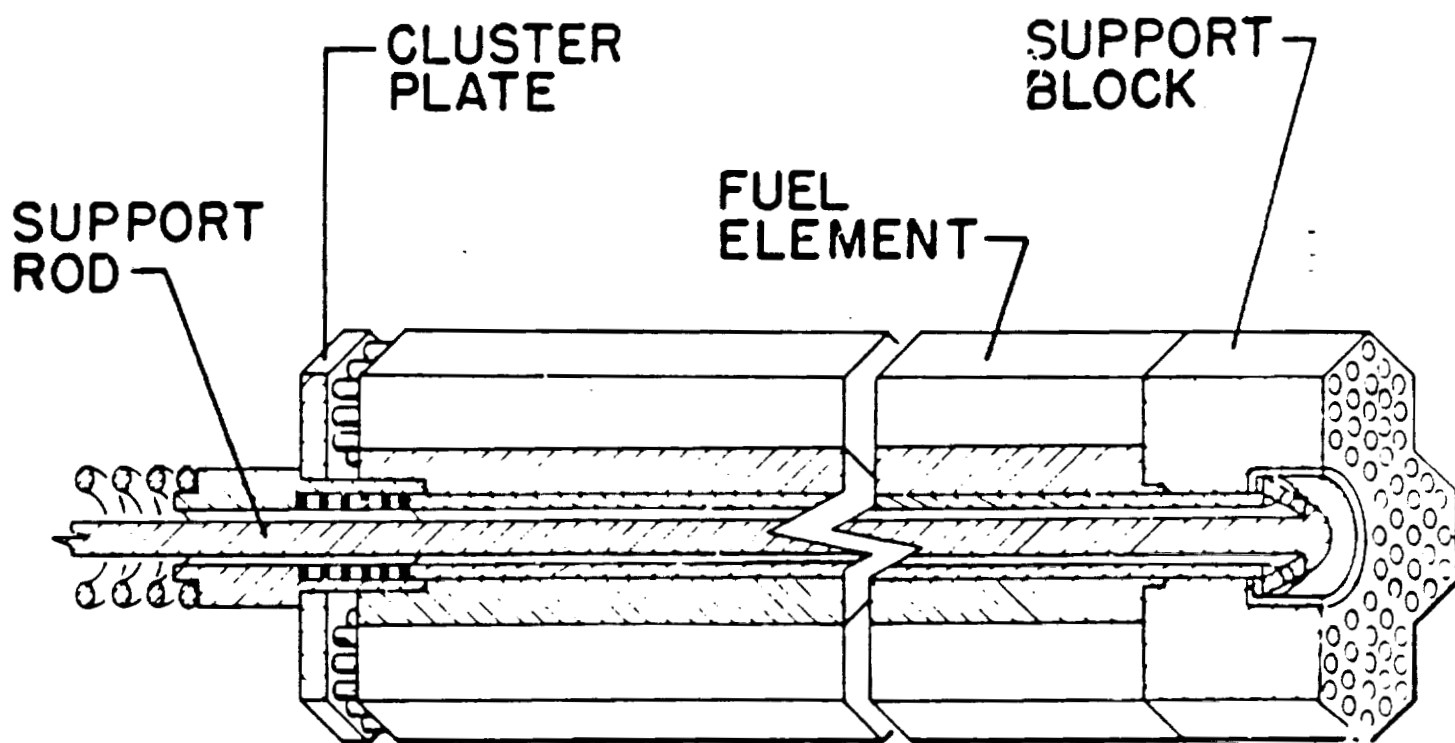


FIGURE 35. PHOEBUS 1A REGULAR FUEL ELEMENT CLUSTER.

power hold. This leak was found to have occurred due to an inadequate weld which caused a crack to develop in the weld between a clamp ring and fitting boss.

During reactor disassembly it was found that the reactor damage was confined almost exclusively to the core. The entire hot end of the core was fused together (except for portions of the badly disrupted central part) by a metallic-appearing melt which was probably melted stainless steel liners from center elements. In spite of the wide-spread damage, peripheral corrosion on the NbC coated sector was practically non-existent.

PHOEBUS 1B [17, 18]

The PHOEBUS 1B reactor was rated at 1500 MW and was tested at full power on February 23, 1967. The test duration was 46 minutes, of which 30 minutes were above 1250 MW with a maximum power of 1450 MW and gas temperature of 2444 K (4400 R) being achieved. It is also worthy to note that the reactor was run at an intermediate power of 588 MW for 2.5 minutes on February 10, 1967. Upon shutdown during this February 10th test, a power spike occurred (3500 MW).

The objectives of the PHOEBUS 1B test were:

1. Operate the reactor at a power of 1500 MW with an average fuel element exit gas temperature of 2500 K (4500 R) to obtain fuel element corrosion and thermal stress data in a reactor environment at fuel element power densities approaching those planned for the PHOEBUS 2A reactor.
2. Operate at full power for 30 minutes or until the control drums had turned 20 degrees.
3. Obtain information on bore corrosion and on external corrosion of the fuel elements; on the effectiveness of a molybdenum overcoating in reducing the mid-range corrosion in the bores; on the effectiveness of several design options in reducing corrosion at the core periphery; on the amount of corrosion at the hot end of the fuel elements and on support block corrosion; on the performance characteristics of six tie tube clusters; on the performance of fuel elements (49) having a reamed bore diameter of 0.28 cm (0.110 inch), similar to those to be used in the PHOEBUS 2A reactor; and, finally, to gain experience with exit gas thermocouples and thermocouples located in Bore 10 of the fuel elements.

The PHOEBUS 1B fuel elements were 132 cm (52 in) long and had a flat-to-flat dimension of 1.91 cm (0.752 in). They were equipped with 2.54 cm (1 in) long tips of unloaded graphite glued to the hot end of the elements and contained 19 bores for coolant flow. The bores, which had a reamed diameter of 0.254 cm (0.100 in), were coated with about 70 grams of NbC overcoated with 2 grams of Mo (the reason for this is discussed later). The reactor core contained 1498 elements.

Postmortem examination of the reactor focused heavily on the corrosion and mass loss of the "new" fuel elements. The reactor contained several varieties of elements and the significant results (with some background information) were:

1. The mid-range losses decreased with increasing Mo deposition, but the gross loss did not reflect this trend.
2. Corrosion loss was definitely dependent on the amount of NbC deposited on the elements; those with deposits of 70-90 grams performed poorly compared with those having coatings of 50-70 grams. The thinner NbC coating had a much better crack structure (large number of narrow cracks) and, consequently, withstood corrosive attack more easily.
3. Some fuel elements were coated with NbC on all six outer surfaces; the main purpose of this experiment was not whether corrosion would be reduced but rather if externally coated elements would bond together during a reactor run, and crack transversely or break. None of these elements showed any evidence of cracking or breaking due to lockup. The external NbC coating also reduced external corrosion to a degree that it was thought that external coatings would be adopted for all fuel elements in future LASL reactors. The corrosion was 0.7 g/element compared with 2 g/element for uncoated elements. Also, external NbC coatings appeared to drastically reduce groove corrosion (longitudinal face corrosion due to interstitial flow).
4. The 49 elements with 0.279 cm (0.110 in) coolant holes were compared with the elements with 0.254 cm (0.100 in) coolant holes. It was found that a) no direct correlation was found between the performance of such fuel elements and calculated thermal stress, core location, support method, and operation parameters; b) a direct correlation existed between coating-batch quality as measured by nondestructive testing techniques, and fuel

element mass loss; this loss was the main factor affecting the corrosion performance of elements with 0.279 cm (0.110 in) bores. The results suggest that these elements successfully withstood a tangential stress of about 11722 kPa (1700 psia) and a peak power of about 5980 MW/m³ (96 BTU/sec-in³).

5. An experiment was included in the PHOEBUS 1B reactor test to obtain as good a fit-up at the hot end of fuel elements as possible. Any unevenness in the external NbC coating at the hot end was removed by eloxing. Seven adjacent clusters of elements with eloxed hot ends were included in the reactor. The result was that eloxed hot ends were neither better nor worse than the normal hot ends. In fact, hot-end corrosion performance of the fuel elements was very good.

Numerous fuel elements were bonded together by pyrocarbon deposits bridging the external surfaces of the elements; about 27% of the 1498 elements were broken during removal from the reactor because of this bonding. Analysis of two samples of the airborne radioactive material produced by the reactor indicated a release of about 0.5% of the fission-product inventory from the core in the form of fission-product-bearing uranium fuel. Thermal diffusion of fission products accounted for an additional release of about 1%.

The PHOEBUS 1B reactor increased the average fuel element power density to 1 MW/element and the fuel elements demonstrated improved corrosion resistance. Additionally, the core exit pressure and hydrogen flow rate was increased over the PHOEBUS 1A.

In order to avoid core damage due to unexpected hydrogen depletion, as occurred during the PHOEBUS 1A test, an 30280 l (8000 gal), high pressure (5171 kPa) dewar was installed to provide an emergency supply of liquid hydrogen in the event of a failure in the primary propellant supply system.

Reason For Overcoating Fuel Elements With Mo

A major problem throughout the fuel development program was called midrange corrosion. It was the region where corrosion was greatest and was the central 1/3 of the core length (midband). The inlet end of the core had low corrosion rates because the temperatures were low. The fuel operated at much higher temperatures toward the nozzle chamber end of the core, but the fuel was processed during fabrication to accept the high-end temperatures. Also the neutron flux, and hence the power density, was low, resulting in low thermal stresses and consequently minimal cracking. There, mass loss was mostly due to carbon

diffusion through the carbide coating. However, in the midrange, the power density was high and the temperature was now appreciable, yet still much lower than that at which the fuel was processed; the carbide coatings would crack because of mismatched expansion coefficient of thermal expansion, and high mass losses would occur through the cracks. The Mo overcoat was used to help reduce this midband corrosion

The PHOEBUS 1B reactor test provided valuable information for the design and operation of future reactors and for the fabrication of corrosion-resistant fuel elements.

PHOEBUS 2A [19]

The PHOEBUS 2A was the most powerful nuclear rocket reactor ever built. The first power test was run at an intermediate level of nearly 2000 MW on June 8, 1968¹. The reactor was tested at full power on June 26, 1968, ran for 32 minutes with 12.5 minutes above 4000 MW, reaching a peak power of 4082 MW. A third run was performed on July 18, 1968 reaching a power of 1280 MW. A fourth and final test was performed also on July 18, 1968, with a power of approximately 3500 MW.

The overall test objectives of the PHOEBUS 2A were:

1. Demonstrate the capability of the reactor and test system to operate at 5000 MW and at a chamber temperature of 2500 K (4500 R).
2. Operate at the design point for a maximum of 20 minutes or until 15 degrees of control drum motion has taken place, whichever occurs first, to obtain endurance information on the fuel elements and the structural components.
3. Evaluate the structural and thermal flow performance of the new regeneratively cooled tie-tube core support setup.

Secondary objectives were:

1. Perform tie-tube flow mapping reactivity experiments including startup on tie-tube flow.
2. Perform controls dynamics experiments.
3. Obtain experience on the following types of temperature measurements: a) thermocouples in fuel elements, and b) fuel element exit gas

¹ Many "summarizing" references fail to mention this important June 8th test.

thermocouples with sheaths located in perimeter fillers.

Additional objectives of the intermediate power test were:

1. Determine the reactivity contributions of hydrogen in the tie-tube system with the reactor operating at low and intermediate power levels.
2. Obtain data on the dynamic behavior of the tie-tube system, the liquid hydrogen flow rate control system, and the temperature control system under various conditions of reactor load.
3. Operate the reactor at intermediate power levels to obtain data that will allow a performance evaluation of the reactor and of the facility system in preparation for the full power run, and verify the adequacy of the shutdown and cooldown system.

Additional objectives of the full power run were:

1. Operate the reactor at elevated power levels up to rated power.
2. Perform frequency response measurements at rated full power or at the maximum power level attained.
3. Obtain data on the dynamic behavior of the tie-tube system on liquid hydrogen flow rates, and on the temperature control system under various conditions of reactor load.
4. Determine temperature and hydrogen reactivity effects at a special neutronic power hold preceding the full power hold.

Additional objectives of the July 18th test were:

1. Complete the controls experiments that were omitted or compromised during the previous hot-fire tests.
2. Attempt to identify the parameter or component in the reactor system responsible for the reactivity loss. Two operational tools were used to gain more information on this subject; first, static incremental changes were included in the reactor run profiles during which the flow rates remained constant as core temperature was varied, and core temperature maintained constant while flow rate was varied; and second, sine-wave perturbations were introduced to the rod position and tie-tube

flow rate at a frequency high enough to preclude core temperature response to the perturbation.

3. Study experimentally the effects of an emergency shutdown from a hold point which was safe for the reactor and the nozzle.

Originally the PHOEBUS 2A was intended to be a prototype "optimum" thrust nuclear propulsion engine for ambitious planetary missions, with a design power of 5000 MW, a liquid hydrogen propellant flow rate of 129.3 kg/s (285 lb/s), and a core exit temperature of 2528 K (4550 R) to provide a nominal thrust of 1112.5 kN (250000 lb) and a specific impulse of 820 seconds. Figure 36 shows the size of PHOEBUS 2 relative to the KIWI and earlier PHOEBUS reactors.

Two major core design features, incorporated to optimize the propulsion system, distinguished PHOEBUS 2A from earlier reactors. First, the power density of the graphite fuel elements was increased by enlarging the diameter of the coolant flow channels from 0.254 cm (0.100 in) to a nominal 0.279 cm (0.110 in). This was due to the desire to decrease the pressure drop, and in turn affected the temperature drop. Secondly, the single pass cooling of the metal core support structure of earlier reactor designs was changed to two pass regenerative cooling by diverting about 10% of the liquid hydrogen to the core support and returning this coolant to the main flow through the core at the inlet of the fuel elements. This new coolant path eliminated the performance degradation associated with the single pass tie rod system by preventing the mixing of the lower temperature core support coolant with the core exit gas in the nozzle chamber. Theoretically, these two innovations permitted the attainment of 5000 MW with a 139.7 cm (55 in) core, compared to 1500 MW with a 88.9 cm (35 in) core of PHOEBUS 1B design. This represented a 3.3-fold gain in power, obtained with only a 2.7-fold increase in total core volume.

The PHOEBUS utilized 4789 fuel elements of the pyrolytic carbon coated UC₂ bead variety. The 19-hole fuel elements were similar in geometry and had the same external dimension as those of earlier reactors with the exception of the 0.279 cm (0.110 in) coolant holes which reduced the pressure drop in the core and the thermal stress in the fuel elements. The coolant holes were coated with NbC of tapered thickness and were overcoated with a layer of Mo to reduce corrosion of the graphite. In addition, some experimental elements were fabricated at the latest possible time for inclusion in

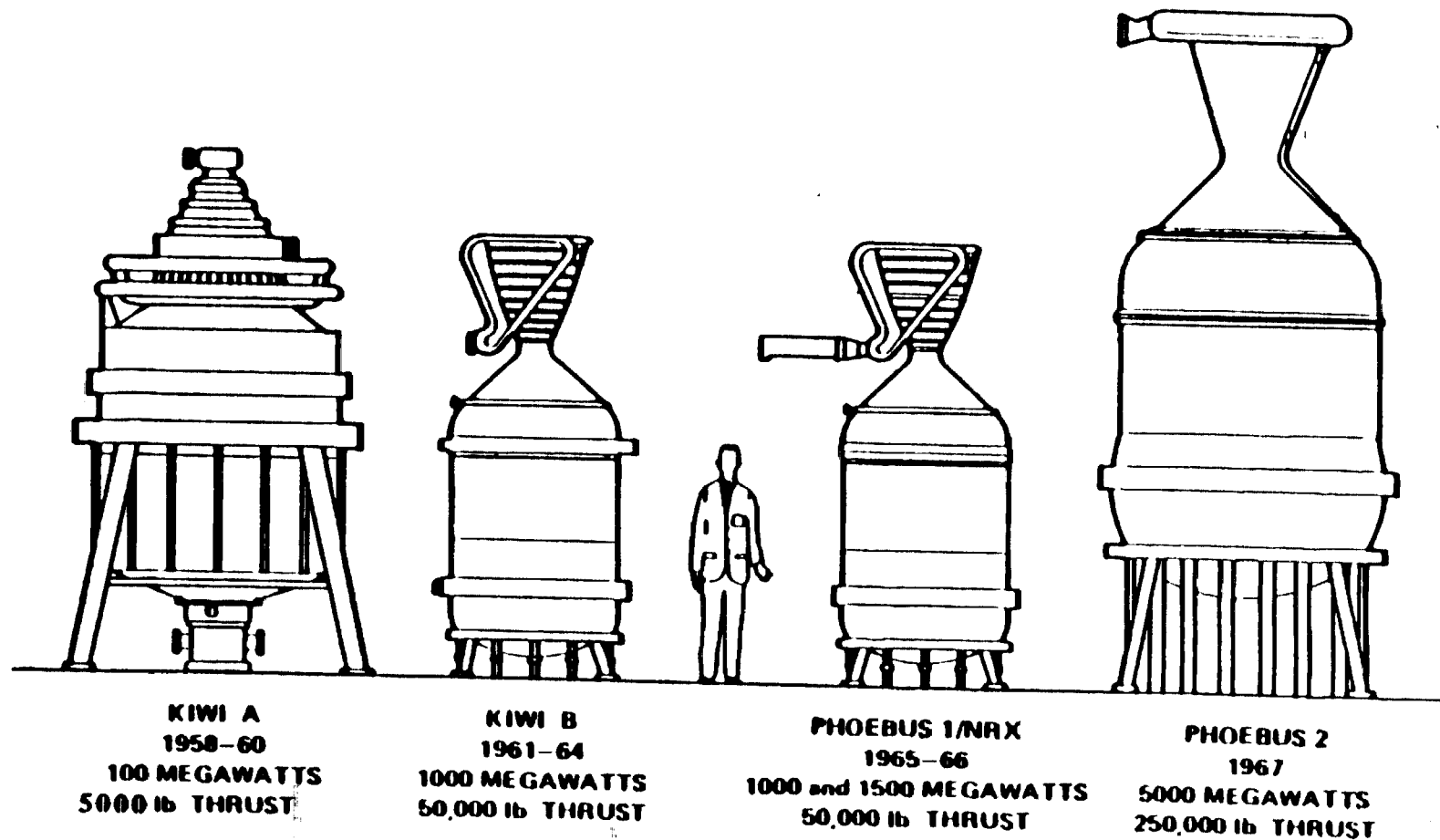


Figure 36

RELATIVE SIZE OF LOS ALAMOS REACTORS

the reactor (about mid 1967). These experimental elements were included to test the corrosion resistance of coatings applied by several new techniques¹.

The intermediate power test provided a useful benchmark to evaluate the operation of the reactor and of the facility systems. At the start of the intermediate power hold, reactor control temperatures and computed thermal power were much lower than predicted, while the total reactor flow rate was near the intermediate power operating point. A control drum trim was used to increase reactor control temperatures and thermal power to near intermediate power design conditions. The chamber temperature was 828 K (1490 R) instead of the planned 1667 K (3000 R), but the total flow rate (87.1 kg/s) was only about 1.36 kg/s (3 lb/s) lower than intended. During the intermediate power hold, the total flow rate was increased slightly and the control drums were repeatedly trimmed outward in an attempt to attain the desired operating point. However, due to the large amount of trim, which increased the chamber temperature to only 1472 K (2650 R), it was decided to shut down the reactor without performing the planned controls experiments until an extensive data analysis could be performed.

Some of the results of the June 8, 1968 intermediate power run were:

1. The reactor and its support systems performed excellently.
2. The reactor instrumentation gave a wealth of data and, with the exception of nozzle inlet thermocouples, reflector inlet thermocouples, and tie-tube load cells, performed exceptionally well.
3. The control drum position at the end of the intermediate power hold was 121.9 degrees instead of 92 degrees as predicted.
4. Core peripheral temperature peaking was much less than expected at the high control drum offsets.
5. The ratio of indicated power to true thermal power at all hot power holds was lower, by a factor 2.1, than it was at the initial condition hold.

¹ Introducing the NbC coating gases at the inlet end of the coolant passages; applying the coatings at lower temperature with gases containing CH₄; and by using a diffusion controlled duplex overcoating on the CH₄ coating to thicken the protective layer of the high temperature end of the fuel elements.

6. The delayed critical drum positions before and after the run were 115.4 and 113.8 degrees, respectively.

Since the control experiments planned during the intermediate power run were not made, they were performed during the full power run. The full power run was planned to attain a power of 5040 MW with a chamber temperature of 2500 K (4500 R).

The start of the high power run up to hold was normal, but temperatures were low. The control drum position required to obtain the desired temperature at the first hold was 139 rather than 121 degrees as predicted. Oscillations from 400-900 MW peak-to-peak at 0.3 Hz were experienced during early portions of the run. These were due to an instability in the flow control system, but the control system operated normally during later portions of the test.

The run lasted about 32 minutes. Temperatures at the clamp-band segments reached the red-line of 417 K (750 R), thus only 4082 MW power was achieved with a chamber temperature of 2256 K (4060 R), and total flow rate of 118.8 kg/s (262 lb/s).

Some of the significant results from the full power run performed on June 26, 1968 were:

1. The mechanical and thermal performance of the reactor was excellent.
2. The 18 degree difference between measured drum position and predicted position using data from the intermediate test indicated that either the reactivity offset was not repeatable or was extremely sensitive to small variations of some reactor parameter.
3. The drum excursions after reaching the holds were larger than expected to offset the reactivity effects of neutron precursor buildup.
4. Drum incremental worth at full power was calculated with 4 Hz drum sinusoidal perturbation data. However, because these data indicated that the worth was about 1/3 lower than the prediction, the sinusoidal worth measurement was repeated during the post-run delayed critical measurement to check the technique under conditions when the predictions were known to be correct. These measurements agreed with data on which the predictions were based.

5. The pre- and post-run delayed critical drum positions were 114.2 and 110.8 degrees, respectively.
6. The quality of the data was excellent; however, about 10% of the high-temperature instrumentation was lost.

The additional runs made on July 18, 1968 (1280 and 3430 MW) provided the following results during about 30 minutes of reactor operation:

1. Again the reactor performed excellently, until the flow was cut off during shutdown.
2. Again, control-drum motions could not be predicted with data from the previous run. However, the temperature control system very effectively compensated for this inability.
3. Good sinusoidal drum-worth data were obtained.
4. Flow oscillations similar to those observed during the intermediate power run were encountered.

In addition to major neutronics discrepancies which occurred, anomalies were observed in a) flow oscillations during startup, b) nitrogen-slush difficulty, c) nozzle bolt and shell temperatures, d) clamp segment temperature, and core temperature scaling inconsistencies. It is beyond the scope of this report to discuss these in further detail.

The postmortem examination of fuel elements revealed that the general production elements had mass losses of 10-13 grams/element. The losses of the experimental elements, however, were significantly lower, on the order of 6-10 grams/element with some elements losing less than 4 grams. The best performing fuel elements were WANL Batch 64018 which consisted of Union Carbide Corporation, Nuclear Division Plant Y-12 (hereon referred to simply as Y-12) elements made by a NbBr₅ coating process. It was also determined that Mo overcoating improves performance even on the best coatings.

The successful conclusion of the PHOEBUS 2A tests is a milestone in nuclear rocket reactor technology, particularly notable because it attained the highest power ever generated by a gas cooled reactor. The following were firmly established or demonstrated at the conclusion of PHOEBUS 2A testing:

1. The basic configuration of reactor core and fuel elements were very satisfactory.

2. Methods were definitely available to safely control reactor performance over a wide range of operating parameters.
3. The use of liquid hydrogen as a propellant is feasible at widely varying reactor operating conditions.
4. NbC coatings will protect the graphite fuel elements against corrosion by hot hydrogen for extended periods of time.
5. A metallic tie-tube core support, regeneratively cooled by liquid hydrogen, had been demonstrated which essentially eliminates core performance degradation.
6. Extensive experience, gained in test-facility operation and in the control of experimental reactors, provided a valuable basis for further testing and for the qualification of nuclear space engines.
7. Large rocket nozzles capable of withstanding very high heat fluxes and nuclear heating had been proved feasible.
8. A wide variety of engineering skills and techniques, of analytical methods, and of reactor instrumentation know-how had been acquired, adapted, applied, or developed to cope with the unique problems peculiar to the design, evaluation, testing, control, and analysis of a large nuclear propulsion engine for ambitious space missions.

At the conclusion of the PHOEBUS 2 testing, the Los Alamos team made specific recommendations directed strongly toward the continued study of the reactivity-neutronics phenomena observed. For the sake of preserving these important recommendations they are listed below:

1. An essential part of any future studies will be the accurate determination of reactivities, temperature distributions, and hydrogen distributions in the core and reflector for all power holds.
2. These parameters should be compared and cross-checked carefully to determine their functional relationship and thus to provide a better understanding of the reactivity discrepancies.

3. The possible effect of cold Be on neutron scattering, and the differences in liquid hydrogen scattering cross sections at low energies and at various relative concentrations of para- and ortho-hydrogen in the reflector should be investigated.
4. A particularly intriguing feature was the very low core-edge temperature peaking during large control drum motions. An attempt should be made to establish the cause of this phenomenon because it could eliminate, in future reactor designs, the "run-away" effects of carbon lost by corrosion at the core periphery, which increases with higher temperatures. These higher temperatures, in turn, are caused by increased power density at the periphery relative to the rest of the core when the control drums are rotated outward in response to enhanced carbon loss.

PEWEE 1 [20]

Due to the long lead time and expense of high power reactors, the PEWEE 1 was built to evaluate advanced fuel elements. The reactor was tested on three occasions, November 15 (check-out), November 21 (short duration, near full power) and December 4, 1968 (endurance, full-power). A state point schematic is shown in Figure 37 and lists the full-power design conditions for flow, temperature, and pressure.

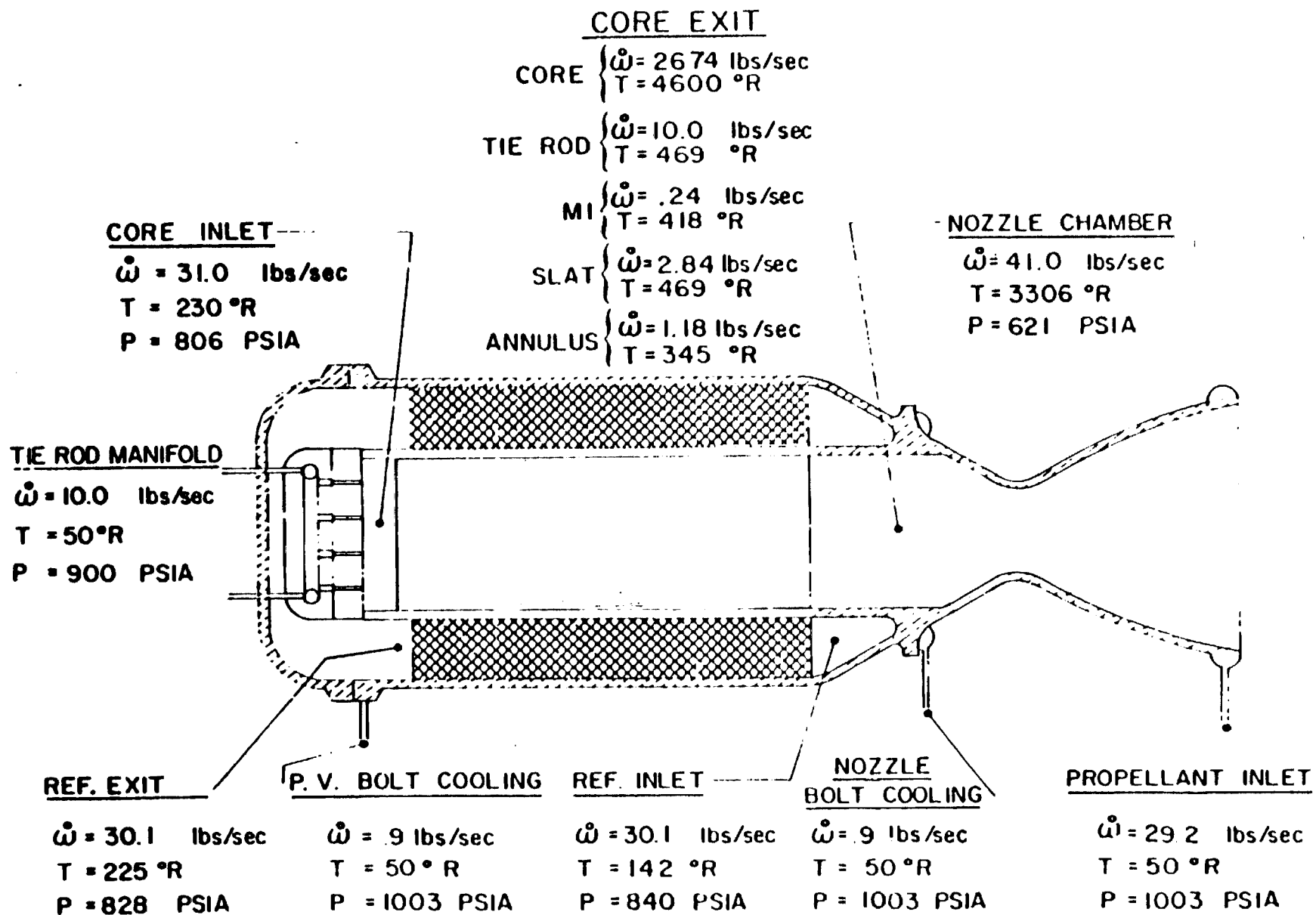
The objectives of the November 21st run were:

1. Operate all test and auxiliary systems to ensure they did not interact with the reactor control or safety systems.
2. Operate the reactor at near full power conditions for a short duration.
3. Perform mapping and control dynamics experiments.

The December 4th, endurance test objectives were:

1. Investigate the flow oscillations observed from the short duration run.
2. Demonstrate the capability of the reactor as a fuel element test bed.
3. Perform three 20 minute full power holds, with low power/low temperature cycles between these holds.

FIGURE 37. PEWEE-1 STATE POINT SCHEMATIC.



POWER~507 MW

PEWEE-1 STATE POINT

SK-N3-773

The PEWEE core contained 402 fuel elements. All elements had 0.279 cm (0.110 in) coolant channels and most were coated internally with NbC, but several were coated with ZrC instead; most had an overcoating of molybdenum. Of the 402 fuel elements 267 were fabricated by LASL, 124 at WANL, and 11 at Y-12. There were 390 nineteen hole elements and 12 twelve hole elements. The PEWEE fuel elements contained 27 different combinations of graphite matrix, coating process, hot-end tips, etc. A typical 19-hole fuel element is shown in Figure 38. Figure 39 shows a typical PEWEE 1 core pattern.

The basic design features of PEWEE 1 were similar to those of preceding ROVER program reactors, PHOEBUS 1 and 2. The fuel elements contained uranium in a graphite matrix and were held in place by support elements. There were, however, significant changes that distinguished PEWEE 1 from earlier reactors. The core diameter was reduced from 139.7 cm (55 in) (PHOEBUS 2) to 53.34 cm (21 in) to reduce the number of fuel elements. Sufficient reactivity with the smaller core was achieved by inserting sleeves of a hydrogenous moderator (zirconium hydride) around the tie rods in the support elements. The hydrogenous material moderated the core neutrons and reduced the critical mass of uranium in the core to 36.4 kg (80.2 lb).

Total hydrogen flow through the reactor was 18.6 kg/s (41 lb/s). The major portion of the hydrogen flowed successively through coolant tubes in the nozzle wall, through the reflector to the pressure vessel, through the support plate, and through the fuel and the core-periphery region in to the nozzle chamber. Small amounts of additional hydrogen joined the main stream after having cooled the nozzle bolts and the pressure vessel bolts. At design conditions, the hydrogen left the fuel elements at the temperature of 2556 K (4600 R).

Since PEWEE 1 was a test reactor for fuel elements, no attempt was made to maximize the specific impulse. A flow of 4.536 kg/s (10 lb/s), split from the main flow upstream of the reactor, was distributed by manifolds to the core support elements to cool these elements and their zirconium hydride moderator and support rods. This coolant discharged into the nozzle chamber where it mixed with the main propellant flow. The temperature of this mixed flow in the nozzle chamber was 1833 K (3300 R), at a flow rate of 18.6 kg/s (41 lb/s) and a pressure of 4275 kPa (620 psia).

The short duration run successfully achieved all of its test objectives. Ten power holds were performed. It achieved a power of 472 MW at an average fuel exit temperature of 2450 K (4410 R), and flow rate of approximately 18.1 kg/s (40 lb/s). Nearly every parameter was very close to predicted or desired values. The high power run was terminated

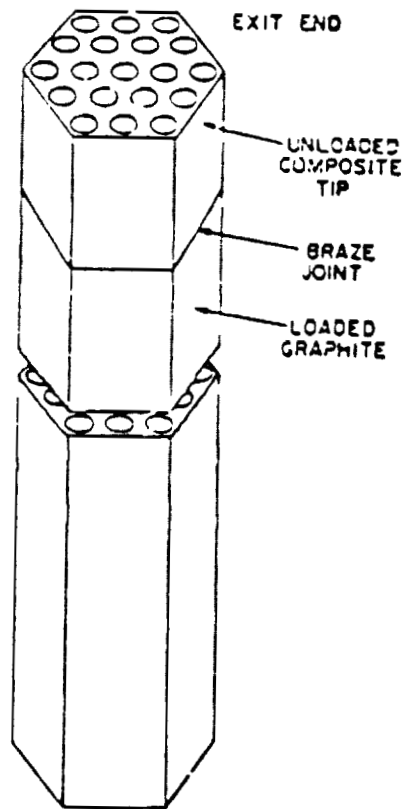


FIGURE 38. TYPICAL 19 HOLE FUEL ELEMENT.

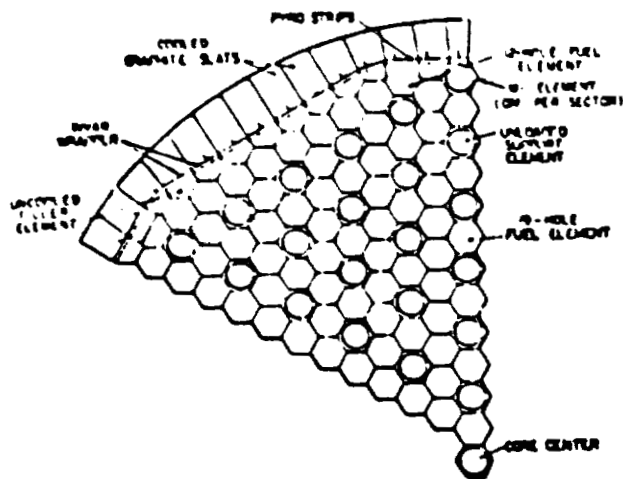


FIGURE 39. TYPICAL PEWEE 1 CORE PATTERN IN A 60 DEGREE SECTOR.

prematurely when an emergency shutdown occurred on a flow rate-over-turbopump RPM trip of the shutdown chain, just prior to reaching the programmed shutdown hold.

The reactor was programmed for the first full power run, with a short hold at 260 MW to evaluate reactor performance. The run profile was very smooth. At the full power hold, fuel exit temperatures were trimmed to the design condition of 2556 K (4600 R). A moderator mapping experiment was performed at this hold. After 20 minutes at full power, the reactor was programmed to hold at about 125 MW with a fuel exit temperature of about 1000 K (1800 R). Fuel exit temperatures were then further reduced to 667 K (1200 R). After a short hold, the reactor was programmed to the second full power hold.

No experiments were performed during the second full power hold. Fuel exit temperatures were trimmed as necessary to maintain 2556 K (4600 R). After 20 minutes, the reactor was again programmed to the low power/low temperature hold.

Following a short hold, the programmer was switched to initiate the third full temperature cycle. However just prior to reaching the third full power hold, flashes were observed in the reactor exhaust (indicative of core material ejection), and a program shutdown was ordered.

The PEWEE 1 tests achieved a thermal power of over 508 MW and demonstrated the capability of this reactor as a fuel element test bed. The PEWEE 1 set records in average power density and exit-gas temperature by operating at over 500 MW for 40 minutes at a coolant exit temperature of 2539 K (4570 R). The core average power density was 2340 MW/m³ (1.3 MW/element), 50% greater than required for the 1500 MW NERVA reactor. The peak average power density in the fuel was 5200 MW/m³. The PEWEE 1 also set a record for the highest peak equivalent ideal vacuum specific impulse, 901 seconds.

Postmortem inspection showed that the core support system and most fuel elements were structurally sound. Although showing numerous areas of damage, none of this damage was considered serious enough to indicate imminent failure of major structural components.

Postmortem inspection of the fuel elements revealed that the ZrC coated elements performed significantly better than the NbC elements in terms of fuel element mass loss. In addition, the hot end losses of the ZrC coated elements were only 50-75% as great as the standard NbC coated elements. NbC-ZrC duplex-coated elements were also tested and expected to outperform the ZrC coated elements. However, these elements did not live up to their expectations, with

performance slightly worse than the ZrC coated elements (but still better than the NbC elements).

NUCLEAR FURNACE, NF-1 [21, 22]

The NF-1 was devised to provide an inexpensive means of testing full-size nuclear rocket reactor fuel elements and other core component and was not meant to be a candidate rocket engine. The NF-1 was 10 times less powerful than PEWEE 1. The NF-1 was tested during the summer 1972 at the Nuclear Rocket Development Station at Jackass Flats, Nevada.

The Nuclear Furnace test had two major objectives:

1. To check out the operating characteristics of both the Nuclear Furnace and the effluent cleanup facility.
2. To operate the reactor at an average fuel element exit-gas temperature of 2444 K (4400 R) for at least 90 minutes.

The reactor consisted of two parts: a permanent, reusable portion that included the reflector and external structure; and a temporary, removable portion that consisted of the core assembly and associated components. This reusable test device would reduce both the time between reactor tests and the cost of testing. After completion of a test series, the core assembly would be removed and disassembled for examination, whereas the permanent structure would be retained for use with a new core. The axial view is shown in Figure 40 and a transverse view presented in Figure 41. Unfortunately, program cancellation did not allow the NF-1 to be reused.

Two fluids were required for reactor operation: 1) water, to moderate the core and to supply the injector, and 2) hydrogen propellant, to cool the fuel elements. Hydrogen flowed through the reactor at nominal design conditions of 1.7 kg/s (3.7 lb/s) for a power of 44 MW. The design water flow rate was 22.7 kg/s (50 lb/s) at all power levels. A state point schematic of the NF-1 is provided in Figure 42.

The water moderated beryllium reflected reactor contained 49 cells in which high temperature fuel elements could be tested. Neutronic control was provided by six rotatable drums in the reflector. The reactor core was a cylindrical aluminum can that contained 49 aluminum tubes. Each tube contained and supported one fuel element (or a cluster of small carbide elements) with associated insulation and support hardware. During operation, water flowed in a two-pass system between tubes while hot hydrogen gas flowed through the fuel elements. The hot hydrogen gas was

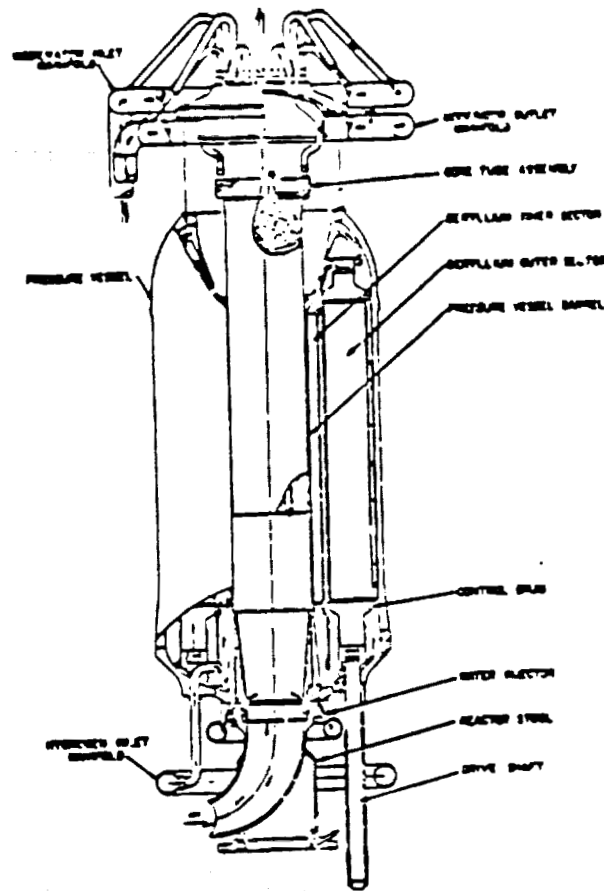


FIGURE 40. NUCLEAR FURNACE AXIAL VIEW.

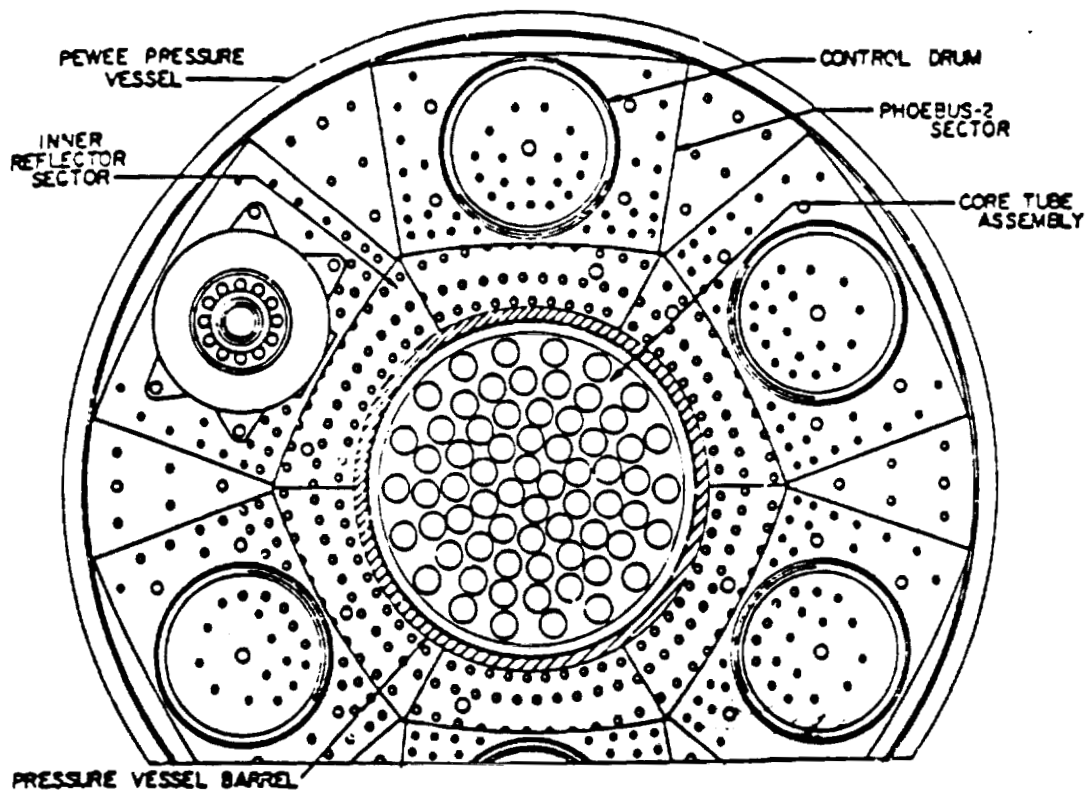


FIGURE 41. NUCLEAR FURNACE TRANSVERSE VIEW.

exhausted from the fuel elements and cooled by injecting water directly into the gas stream. The resulting mixture of steam and hydrogen was ducted to an effluent cleanup system.

Two classes of fuel were tested; 47 of the cells contained (UC-ZrC)C-carbon "composite" fuel cells and 2 cells contained a seven-element cluster of single-hole pure (U,Zr)C carbide fuel cells. Neither type had ever been tested in a nuclear propulsion reactor. Of the 47 composite cells, 24 cells were made in 1970 as part of the original NF-1 core and 23 cells were made in 1971 as replacements for some original cells. Component testing prior to the reactor run had indicated that the replacement elements had better corrosion and thermal stress resistance. Three types of carbide matrix elements were used in the reactor.

The composite elements had been under continuous development at LASL since 1967. The matrix of the element consisted of a continuous network of uranium-zirconium solid-solution carbide in conjunction with a continuous network of graphite. All surfaces, including the hydrogen coolant channels and the exterior surfaces were protected with an adherent ZrC coating. The hexagonal NF-1 composite fuel elements were 1.32 m (4.33 ft) long, measured 19.10 mm (0.75 in) from flat to flat, and had nineteen 2.5 mm (0.098 in) diameter longitudinal coolant channels. Figure 43 shows the NF-1 reactor cell containing a composite fuel element.

The use of the carbide elements had been studied at LASL since 1969. The possible advantage of carbide elements over composites is their higher temperature resistance: they could withstand exit-gas temperatures up to 3200 K (5760 R) for one to two hours; and at lower operating temperatures (2200-2700 K) could have life times up to 100 hours. The carbide-element matrix must be substoichiometric in carbon to obtain the highest melting point in the U-Zr-C phase system. Thus the fabrication process was designed to produce elements ranging in uranium content from 5-14 mol% (300-1300 kg/m³ loading) at total carbon-to-metal ratios of 0.85 to 0.93. The hexagonal carbide fuel elements in the NF-1 reactor were 639.4 mm (25.17 in) long, measured 5.512 mm (0.217 in) across the flats, and had a single coolant hole about 3.05 mm (0.120 in) in diameter. The elements were made by an extrusion process that left about 3 without free carbon in the elements after extrusion and heat treatment. The free carbon was subsequently removed by leaching with hydrogen gas. The overall carbon-to-metal ratio was reduced by impregnating the elements with zirconium using a chemical vapor impregnation. The carbide element is shown in Figure 44 and Figure 45 shows the NF-1 cell containing a bundle of these elements.

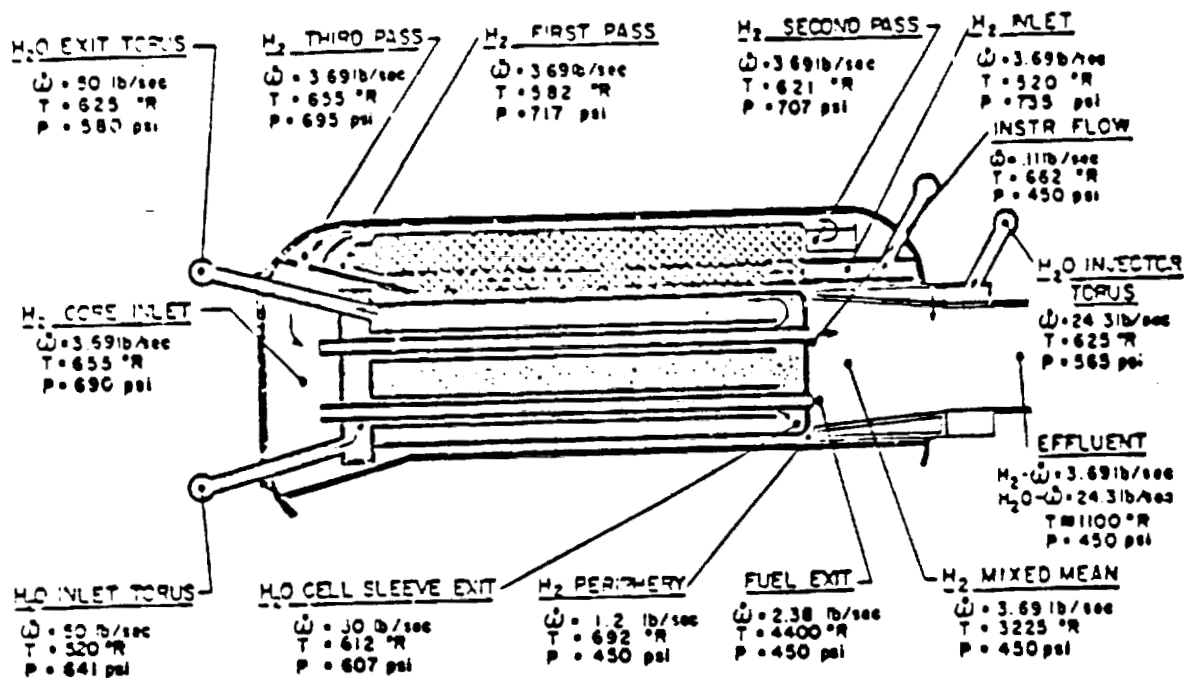


FIGURE 42. NUCLEAR FURNACE STATE POINT SCHEMATIC.

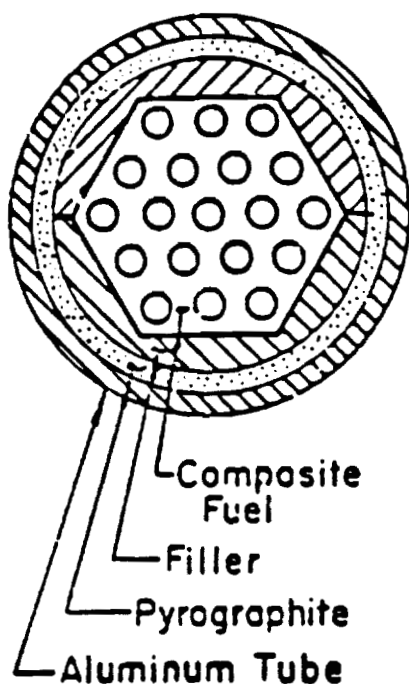


FIGURE 43. NF-1 REACTOR CELL CONTAINING (U,Zr)C-GRAPHITE (COMPOSITE) FUEL ELEMENT.

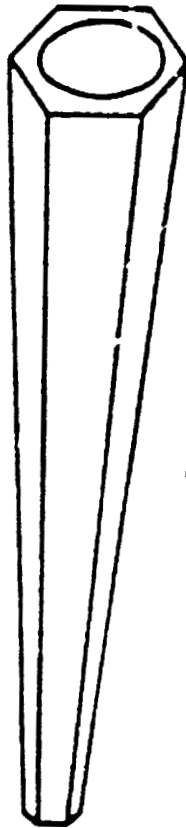


FIGURE 44. FULL LENGTH VIEW OF (U,Zr)C (CARBIDE) FUEL ELEMENT.

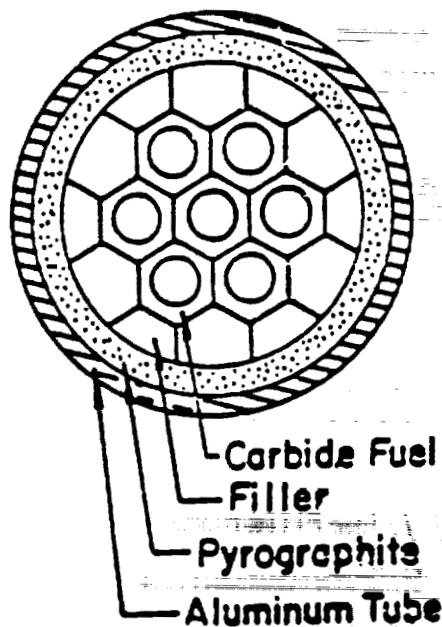


FIGURE 45. NF-1 REACTOR CELL CONTAINING (U,Zr)C (CARBIDE) FUEL ELEMENT

In all the NF-1 was operated 4 times at full power (44 MW, fuel exit gas temperature 2444 K) from June 29 through July 27, 1972 for a total of 108.8 minutes. The NF-1 operated 121.1 minutes with a fuel exit gas temperature above 2222 K (4000 R). A near record peak average power density in the fuel was achieved at 4500-5000 MW/m³ with matrix temperatures up to 2500 K (4500 R).

The composite fuel achieved better corrosion performance than was observed previously in the standard, graphite-matrix, PHOEBUS type fuel element. It was also found, however, that the composite fuel elements were susceptible to radiation damage. Mass losses were unexpectedly high in portions of the elements which had been damaged by radiation. This damage, apparently due to interaction of fission fragments with the graphite, degraded the thermal transport properties of the matrix, and the resulting temperature gradients caused extensive cracking of the coating. The basic conclusion was that the composite elements would perform satisfactorily for at least two hours in a nuclear propulsion reactor which heated hydrogen to the temperature region of 2500 to 2800 degrees K (4500 - 5040 R).

The carbide fuel elements cracked extensively (particularly near the center of the reactor where the peak average power density was 4500 MW/m³), as was expected, due to the low thermal conductivity. No evidence of fragmentation into millimeter size particles was seen. Improvements in strain-to-fracture properties of the matrix, and design changes to minimize temperature gradients, would make these elements useful at power densities of 3000 to 4000 MW/m³. Component tests have indicated that carbide elements would perform for many hours at temperatures of 2800 to 3100 K¹ (5040-5580 R). A proposed maximum operating temperature of 3200 K (5760 R) is equivalent to an ideal vacuum specific impulse of 971 seconds in a nuclear propulsion engine using hydrogen as a propellant (even higher if dissociation/recombination effects are taken into account).

Of significance is that a unique feature of these tests was the use of an Effluent Cleanup System (ECS) downstream of the reactor to remove fission products from the reactor effluent before release of the cleaned gas to the atmosphere. The ECS also operated quite successfully².

¹ Refer to Figure 2 which provides projected endurance of several fuels versus coolant exit temperature.

² Details are provided in "Nuclear Furnace-1 Test Report", Los Alamos Informal Report number LA-5189-MS, 1973.

PART II TECHNOLOGY DEMONSTRATION

NRX A2 [23, 24]

The NRX A2 closely resembled the KIWI B4E but had a distributed peripheral design, not a hot end seal. The NRX A2 was the first NERVA reactor tested at full power by WANL. The power test was conducted on September 24, 1964 and the reactor operated in the range of half to full power (1096 MW) for 6 minutes, with full power operation lasting 40 seconds. The reactor was restarted on October 15, 1964 to investigate the margin of control in the low flow, low power regime over a broad range of hydrogen density inlet conditions.

The major objectives of the power test were:

1. To provide significant information for verifying the steady-state design analysis for power operation.
2. To provide significant information which will aid in assessing the suitability of the reactor to operate at the steady-state power level and temperatures required for the reactor to be a component of an experimental engine system.

For the low power, low flow test, the major objectives were to demonstrate stable operation in the region of interest at transient startup and cooldown with liquid hydrogen.

Specific objectives were also defined for the power test. These specific objectives were:

Top Priority:

1. To evaluate the effects of the environmental conditions on the structural integrity of the test assembly and its components.
2. To evaluate the performance of the core assembly.
3. To evaluate the performance of the lateral support and seal system.
4. To evaluate the performance of the core axial support system.
5. To evaluate the performance of the outer reflector assembly.
6. To evaluate the performance of the control drum and actuation system.
7. To evaluate the overall reactivity characteristics.

Less Urgent

1. To evaluate the performance of the nozzle assembly.
2. To evaluate the performance of the pressure vessel assembly.
3. To evaluate the performance of the shield assembly design.
4. To evaluate the performance of the test assembly instrumentation.
5. To evaluate the performance of the nucleonic power control system.
6. To evaluate the performance of the propellant feed and control system.
7. To evaluate the performance of the nozzle chamber temperature control loop.
8. To evaluate the performance of the test car system.
9. To evaluate the radiological hazards associated with operation of the reactor.
10. To evaluate the thermal and nuclear environments surrounding the reactor.
11. To evaluate the performance of advanced NRX power control system components.
12. To evaluate the performance of a single range control system.
13. To evaluate transfer function of components and systems under various operating conditions.
14. To evaluate the performance of the in-core temperature control loop.
15. To evaluate the transient characteristics of the overall test systems.

For the low power, low flow test, the following specific objectives were set:

1. Demonstration of stability at low liquid hydrogen flow using dewar pressure.

2. Demonstration of suitability at constant power with flow variation.
3. Demonstration of stability at fixed control drum position with flow variation and resulting power change.
4. Achievement of a reactivity feedback value associated with liquid hydrogen at the core entrance.

The NRX-A program was conceived as a series of evolving nuclear rocket reactor designs, with the evolution of the design based on analytical results, component test results, and integral reactor test results. The program was an integrated development and test program intended to adapt and qualify the Los Alamos KIWI B-4 reactor concept for use in an experimental engine program.

The reactor contained 1626 fuel elements. The coolant flow path consisted of coolant entering the plenum beneath the outer reflector from the nozzle tubes, continuing up the outer reflector housing the control drums, then through the simulated shield, turning, passing through the fueled core, and discharging out the nozzle. Parallel flow paths cool the pressure vessel, dome, graphite reflector cylinder, lateral support parts, and the aluminum barrel surrounding the graphite reflector. The flow path is shown schematically in Figure 46.

The full power test consisted of holds at several power levels (51%, 84%, and 93-98% as tested). The flow rate at the higher power level was somewhat higher than planned because of compressibility effects and the precision of the venturi flowmeters. Early in the power profile, at about the 51% power hold, a number of small fires were noted on the test assembly. These were located in the vicinity of the dosimetry belly-band and the dome end closure. The fires were of a short duration and based upon post-run observations, appeared to be the result of partial melting of the belly-band and dosimeters and burning of pressure transducer insulation and epoxy potting compound. The inspection did not show any indication of significant hydrogen leakage.

The test duration was limited by the amount of hydrogen gas available to drive the turbopump. This test demonstrated an equivalent ideal vacuum specific impulse of 811 seconds.

From the standpoint of observations during the tests and early analysis of data, all the objectives of the power test were met. The nuclear, thermal, flow, and mechanical design analysis all predicted test parameters which were generally observed during the test. The power achieved during the

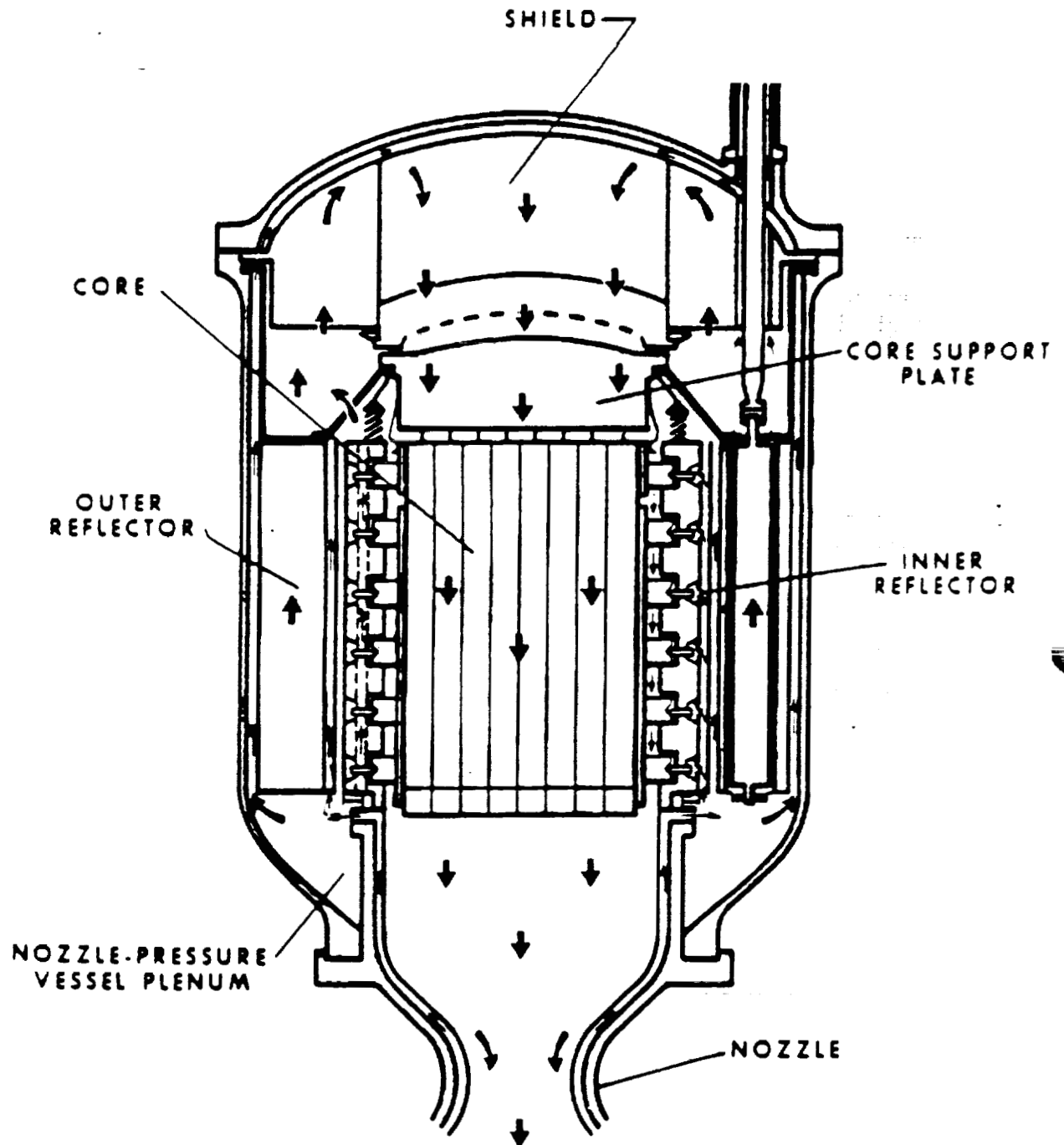


FIGURE 46. NRX A2 FLOW SCHEMATIC.

power holds was higher than planned. This pointed out certain shortcomings in the instrumentation system. These shortcomings occurred mainly in the in-core thermocouples at one particular station which were reading low and demanded increased power.

The low power mapping tests were performed for a period of approximately 20 minutes to investigate the margin of control in the low flow, low power regime over a broad range of hydrogen density inlet conditions. These tests covered an operating power range of 21-53 MW, with 2.27-5.9 kg/s (5-13 lb/s) hydrogen flow. No power or flow instabilities developed when the core inlet conditions were brought close to state properties of liquid hydrogen. Comparison of the measured data with predictions was reasonable considering the inaccuracy of the test results since most transducers were operating near the low end of their scale.

Postmortem inspection revealed incipient corrosion of the fuel elements. This indicated a potential problem for extended operating durations. There were no broken elements that could be attributed to the NRX A2 reactor test. The only serious element damage discovered that could not be conclusively identified with disassembly handling was transversely broken, unfueled, instrumented central elements. However it could not be definitely inferred that these elements broke during the power run since there was almost no corrosion at the breaks.

The most serious corrosion problem in the NRX A2 occurred along the fuel element flats at the core periphery towards the hot end of the core. This corrosion was the result of hydrogen leaking in through filler strips and pyro-tiles, and contacting fuel at elevated temperatures producing a characteristic striated pattern on the flats. In the few points where NbC coating was present on the periphery, corrosion was largely inhibited. It was decided, therefore, to coat the periphery of the NRX A3 core with NbC. In addition, the periphery fuel would be kept cooler by increased flow of coolant using appropriate orificing.

In conclusion, the NRX A2 power tests provided a sound basis and increased confidence for proceeding to the more stringent endurance and transient testing planned in the NRX A3.

NRX A3 [25, 26]

On April 23, 1965, the NRX A3 was operated for 8 minutes, with 3.5 minutes at full power (1093 MW). The test was terminated early due to a spurious overspeed trip of the turbopump. The reactor was restarted on May 20 and ran for 16 minutes, with 13 minutes at full power. A third and

final test was made on May 28 when it operated for 46 minutes in the low to medium power range to explore the limits of the reactor operating map.

The primary objectives of the NRX A3 were:

1. To operate at full power for fifteen minutes with margin for operation at full power for a period of five minutes after restart.
2. To shut down and cool down on liquid hydrogen.
3. To start up from a low power, low flow, steady-state operating condition and to shut down from a medium power level on liquid hydrogen flow control only.
4. To check the stability of certain new control concepts.
5. To determine the acceptability of design changes and modifications to this test article.
6. To verify the limits of the predicted steady-state power flow operating map up to medium power.

The NRX A3 reactor contained 1626 fueled elements, which made a 132 cm (52 in) long cylinder approximately 44 cm (17.3 in) in radius containing 172 kg (379 lb) of enriched uranium. Surrounding the core's cylindrical surface was a 5.3 cm (2.09 in) thick graphite barrel. Next is a beryllium reflector 11.7 cm (4.6 in) thick containing 12 beryllium control drums of radius 5.2 cm (2.05 in), each of which had a boron-aluminum poison vane that moved toward or away from the core center as control drums were rotated. Figure 47 provides a view of the reactor.

The first power run achieved 1093 MW for 3.5 minutes. An unplanned automatic shutdown occurred, which subsequently resulted in overheating of the core tie rod assembly. Data analyses indicated that the maximum average tie rod material temperature reached during the transient was approximately 1150 K (2070 R), with a corresponding maximum individual tie rod temperature of 1391 K (2503 R). A test limit of 667 K (1200 R) had been established for the average tie rod exit gas temperature. Tie rod liners reached a maximum temperature of 1556 K (2800 R). It was believed that a loose electrical connection in the turbine overspeed circuit was the cause of the shutdown.

A comprehensive review of the test data indicated that the reactor was not damaged, and a decision was made to continue the tests with a full power restart. During this second test, the reactor operated for 16 minutes, 13.1 minutes of

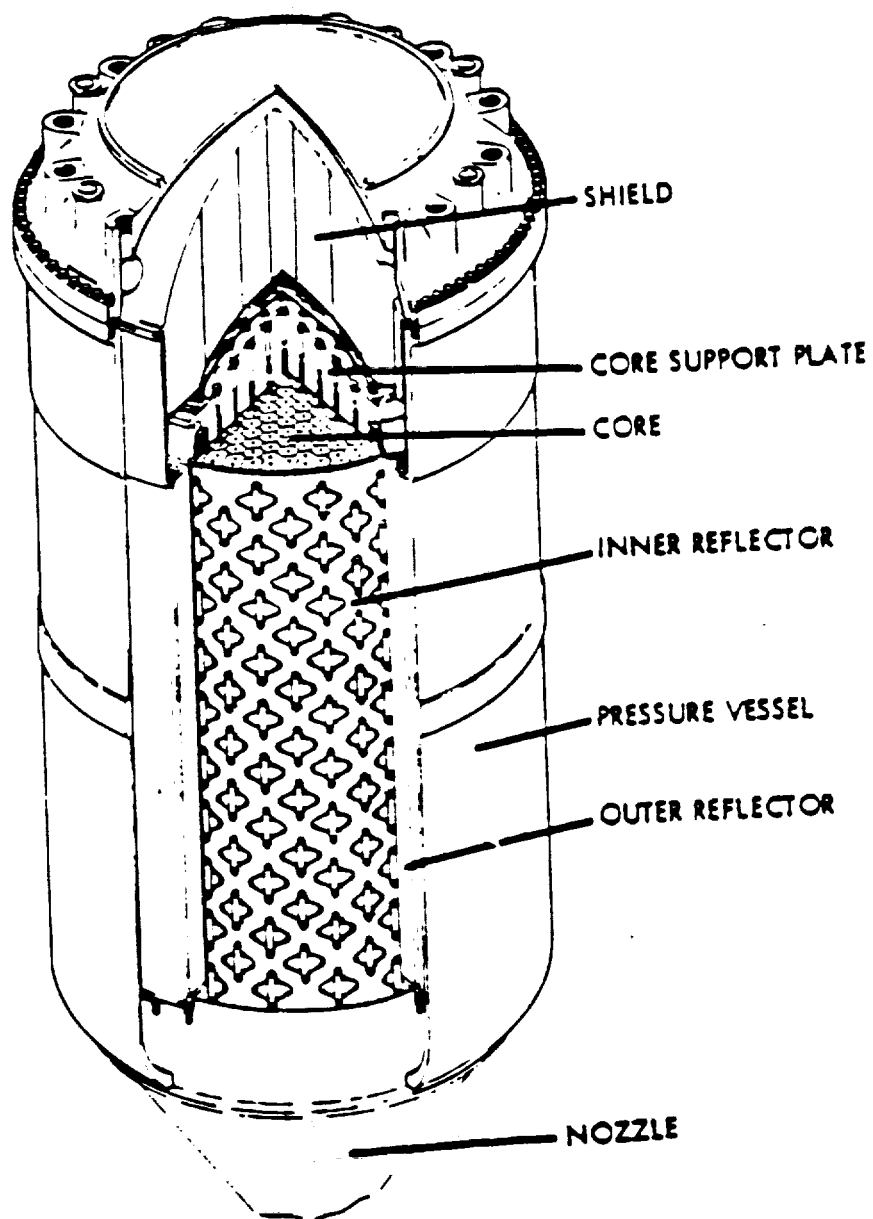


FIGURE 47. NRX A REACTOR.

which were at approximately 1072 MW. The run time was limited by the amount of available hydrogen. The equivalent ideal vacuum specific impulse achieved at the full power hold was at least 803 seconds¹, while the calculated thrust was 237630 N (53400 lb).

The reactor was started a third time, primarily for medium power mapping and controls tests. A significant feature of these tests was a fixed control drum position test in which the reactor power was controlled with the propellant flow rate only. The results were in excellent agreement with the predictions. This test showed definitely that the reactor was inherently stable on liquid hydrogen flow control only. Once the reactor was at stable low power, the reactor power could be controlled at the desired core exit temperature up to high power by increasing the turbopump speed, with the control drums used only as a fine trim of the core exit temperature.

Post-test disassembly and examinations confirmed analyses that the core structural system had not been damaged by the full-power shutdown transients, and that the restarts had not jeopardized reactor integrity or safety. The NRX A3 was the first reactor to use externally coated fuel elements in the outermost row of the core periphery; these fuel elements showed increased performance over that of the NRX A2 peripheral elements, pointing to the significant role that the NbC played in reducing surface corrosion in this area.

A total of 3301 pinholes were observed on 928 fuel elements from the NRX A3 core (58.2 percent of the elements examined). There was similarity between the axial distribution of corrosion pockets and pinholes indicating that pinhole formation was probably related to the formation of corrosion pockets. It was found that NbC coatings did not inhibit pinhole formation. A comparison was also made between elements graphitized in helium at Y-12 and those graphitized in vacuum at Cheswick. Twenty-three elements, approximately at the same core radius, were compared. These elements were all from the same coating batch. The elements graphitized in helium showed an average weight loss of 8.6 grams and 53% of them were pinholed. The elements graphitized in vacuum had an average weight loss of 10 grams and 12% of them were pinholed.

NRX/EST [27, 28, 29, 30]

The NRX/EST was run at intermediate power levels on February 3, and 11, 1966. A full power (1055 MW) run was performed on March 3, 1966 and engine duration tests were performed on

¹ Based on chamber temperature. The higher fuel exit temperature was not directly measured.

both March 16 and 25, 1966. In all, eleven starts were performed.

The significant objectives and major milestones achieved during the test series were:

1. Demonstration of the bootstrap startup capability of the engine system.
2. Evaluation of the effects of the test conditions on the structural integrity of the entire system.
3. Evaluation of engine system stability under transient and steady-state conditions.
4. Acquisition of data for improvement of analytical models.
5. Evaluation of capability of the nuclear system to operate under NRX/EST system conditions.
6. Evaluation of capability of the propellant feed system to operate under NRX/EST system conditions.
7. Evaluation of capability of the hot bleed thrust chamber assembly to operate under test conditions.
8. Evaluation of LH₂ pulse cooldown.
9. Evaluation of instrumentation performance.
10. Evaluation of alternative bootstrap startup operational methods.
11. Evaluation of the nuclear stability of the reactor system.
12. Evaluation of performance of the nuclear power control system.
13. Evaluation of the thermal and nuclear environments surrounding the reactor.
14. Evaluation of the performance of alternative control-system concepts.

The NRX/EST was the first NERVA "breadboard" power plant; the major engine components were connected in their flight functional relationship. The NRX/EST used the NRX A4 combined with the engine turbopump and other elements of a complete engine system. The NRX/EST engine system was comprised of a basic NRX A reactor subsystem, an engine propellant feed system, and a hot bleed port nozzle, all of which were installed on a NRC A-type test car. Included on

the test car was piping for the necessary auxiliary systems, such as normal and emergency cooldown, diagnostic and control instrumentation, and the necessary lines and valves for purging and venting the engine system. Also housed on the test car were the component direct shield and the privy roof-mounted shield. The NRX/EST engine system is shown in Figure 48.

The NRX/EST demonstration used all the principal components that would be used in a nuclear rocket engine. This engine utilized a hot gas bleed turbine drive. It also was the first nuclear reactor which used a "bootstrap" start-up; the engine was started using only the energy generated by the system itself. The bootstrap start up was demonstrated 10 times on the NRX/EST.

The NRX/EST reactor contained 1584 fuel elements which made a 132 cm (52 in) long cylinder of approximately 45 cm (17.7 in) radius containing 176 kg (388 lb) of enriched uranium. Hot end coatings were upgraded for the fuel elements. The NRX/EST pressure vessel was a cylinder 203.2 cm (80 in) in length and 127 cm (50 in) in diameter, with a 2:1 elliptical closure. The basic features of the reactor are the same as the NRX A3 (see Figure 47) with two minor exceptions: 1) the cylinder was provided with a seal gland located outboard of the closure-cylinder bolts and 2) the closure utilized a new configuration for the emergency cooldown ports and the diluent port.

The Aerojet-designed nozzle used was of the steel-jacketed U-tube type with a 10:1 expansion ratio and incorporated a bleed port in the convergent section. Except for the bleed port, variations in instrumentation, and installation of a flange-bolt coolant system, the nozzle was identical to the one used on NRX A3.

Thirteen coolant channels were interrupted by the access hole for the hot bleed port in the wall of the nozzle. Coolant flow through these channels was approximately 10% below the average for normal channels; therefore, the operating limits were established on the basis of the heat transfer capacity of these interrupted tubes.

The bleed port, which was located at a reinforced section of the nozzle jacket, served a dual function by providing access to the nozzle plenum for extraction of hot gases to drive the turbopump and by conditioning these hot gases to turbine inlet design requirements. The bleed port was constructed so that a thin-walled inner sleeve with an adequate radius at the entrance section was supported by a flanged structural member that provided an attachment point between the bleed port and nozzle and between the bleed port and the turbine inlet line. The inner sleeve was cooled by passing diluent for the hot gases through annular passages

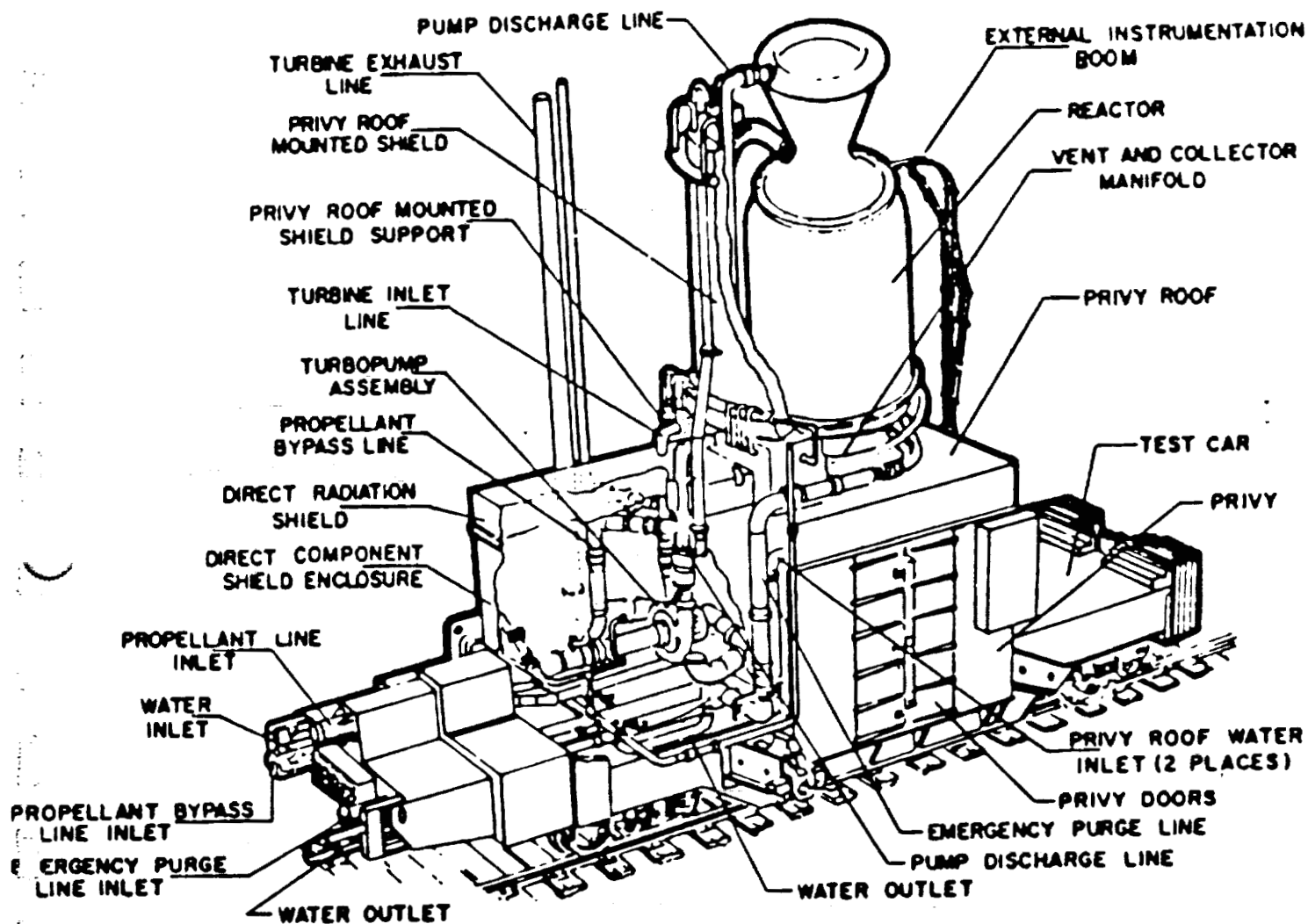


FIGURE 48. NRX/EST ENGINE SYSTEM.

on the back side of the sleeve. The NRX/EST was hot-fired during February 3 through March 25, 1966, and operated during 5 different days (11 start-ups and shutdowns, including 3 aborts) for a total of 1 hour and 56 minutes of which 29 minutes were at power levels in excess of 1000 MW and 30.3 minutes at chamber temperatures of 2056 K (3700 R) or greater.

On February 3, 1966 two intermediate power runs were performed which demonstrated a bootstrap startup. These runs produced a chamber temperature of 1389 K (2500 R) and chamber pressure of about 1724 kPa (250 psia). Three additional intermediate power runs were performed on February 11, 1966. Of these three, two were successful and one was aborted. The intermediate power runs allowed successful mapping of the constant chamber temperature line of the operating map.

A full power run was performed on March 3, 1966 in which the engine performed a bootstrap startup (at 483 kPa dewar pressure) and obtained a chamber temperature of about 2272 K (4090 R). Prior to the successful run, a bootstrap startup (at a dewar pressure of 241 kPa) was aborted when the operating limit of tie rod exit gas temperature was approached because of the liquid hydrogen flow rate lagging behind the programmed power ramp. The successful run operated at its design point for approximately 75 seconds when excessive turbopump displacement was indicated and the chamber temperature reduced to 1889 K (3400 R). The turbopump displacement was indicated at 1889 K (3400 R), the chamber temperature was further reduced (to 1167 K), where a normal shutdown was initiated.

It was concluded that an instrument had given erroneous readings and excessive pump displacement had not occurred. The system was again started (for a third time) and mapping operations were conducted for several chamber temperature ranges, the highest being performed over a period of 285 seconds at a chamber temperature of 1889 K (3400 R).

On March 16, 1966 the first engine duration test occurred. These tests had a goal of operating at full power for long (~ 15 minute) durations. The first test on this day was aborted because the reactor was supercritical by several degrees. This condition caused power to increase rapidly, and chamber pressure quickly rose, exceeding 345 kPa (50 psia) which resulted in shutdown. The second bootstrap startup was successful, and several chamber temperature holds were performed prior to reaching full power conditions. A total of 15.1 minutes was achieved at or above a 2056 K (3700 R) chamber temperature.

The final run (second engine duration test) was performed on March 25, 1966 with an objective of operating above a

chamber temperature of 2056 K (3700 R) for 13.5 minutes. This run was successful and achieved full power for 13.7 minutes. The chamber temperature was held between 2222 and 2306 K (4000 and 4150 R) for 820 seconds.

Two types of fluid flow oscillations were observed during the NRX/EST tests. One type occurred at intermediate pressures during startup and shutdown transients and at low power levels. The second type occurred immediately after the pump discharge valve was opened to initiate flow to the engine system. These test oscillations caused no operational difficulties.

Evaluation of the disassembled fuel elements indicated that considerable damage was sustained by peripheral elements and by elements located at the core center. Elements in the trough between the peripheral and central elements performed as well as the average NRX A3 element. Two predominant forms of damage were exhibited by fuel elements 1) external aft end corrosion attributed to element undercut depth (the NRX/EST incorporated a nozzle end undercut 1.905 cm (0.75 inch) long and up to 0.0069 cm (0.0027 in) deep which was coated with niobium carbide) and 2) mid-element internal bore corrosion. Additionally, a total of 528 elements were broken; 387 of these were broken in several places, and the remainder had only single breaks. Examination indicated that a major cause of element fracture was high localized pinhole density and formation of gross corrosion pockets that caused a general weakening of the elements.

The NRX/EST test series was a significant milestone in the development of a nuclear rocket engine. The hot bleed bootstrap principle of nuclear rocket engine operation was demonstrated for the first time, system stability under a number of control modes and over a wide operating range of pressure and temperature was demonstrated, the multiple restart capability of the engine system was demonstrated, and significant reactor engine operating endurance at rated conditions was demonstrated.

At the time of the NRX/EST test, the NERVA class engines were being designed for lunar missions as well as deep space probes to Mercury, Jupiter, Saturn, and beyond. At this time it was also believed that a nuclear Saturn V third stage would be operational by 1977-78 and that manned planetary exploration would be achieved in 1981-82.

According to Harold B. Finger, manager of the Space Nuclear Propulsion Office, "It looked great. It was the last major milestone in demonstrating the feasibility of nuclear engines." Finger also went on to say, "We could have an operational model ready by the 1970s, but this will depend

on when the U.S. wants to take on missions more ambitious than Apollo."¹

NRX A5 [31, 32, 33, 34]

The NRX A5 was operated on June 8, 1966 at full power (approximately 1120 MW) for 15.5 minutes and restarted on June 23 for 14.5 minutes at full power. The NRX A5 is shown in Figure 49 prior to testing.

The general objective of the NRX A5 test series was to operate at design conditions for a total time of 40 minutes. Secondary objectives were:

1. To evaluate reactor control concepts.
2. To startup from a low power, subcritical condition to near full power with liquid hydrogen flow control and constant drum position.

Specific test objectives for the NRX A5 were:

1. To evaluate the effects of the NRX A5 test conditions on the structural integrity of the test assembly and its components. Of prime interest was the extent of the in-core corrosion following the extended full power operation.
2. To evaluate NRX A5 hardware design modifications and experiments. Major design changes included: elimination of the aluminum barrel; full length pyrographite tile on filler strips; support blocks with modified washers; modified fuel element ends and control drum modifications to minimize boxing. The major experiments related to design changes included: Two 60 degree sectors of hot buffer periphery; brazed tips on fuel elements; fuel element bore coating profile variations.
3. To evaluate the nuclear performance of the test article. Of special interest was the change in reactivity because of increased burnup and in-core corrosion during the longer run time. Data obtained were to be used to determine reactivity changes during the test program, fission density distribution and type, and distribution of fission products generated and retained within the core.
4. To evaluate thermal and fluid flow performance of the test article.

¹ As quoted in Business Week, February 12, 1966.

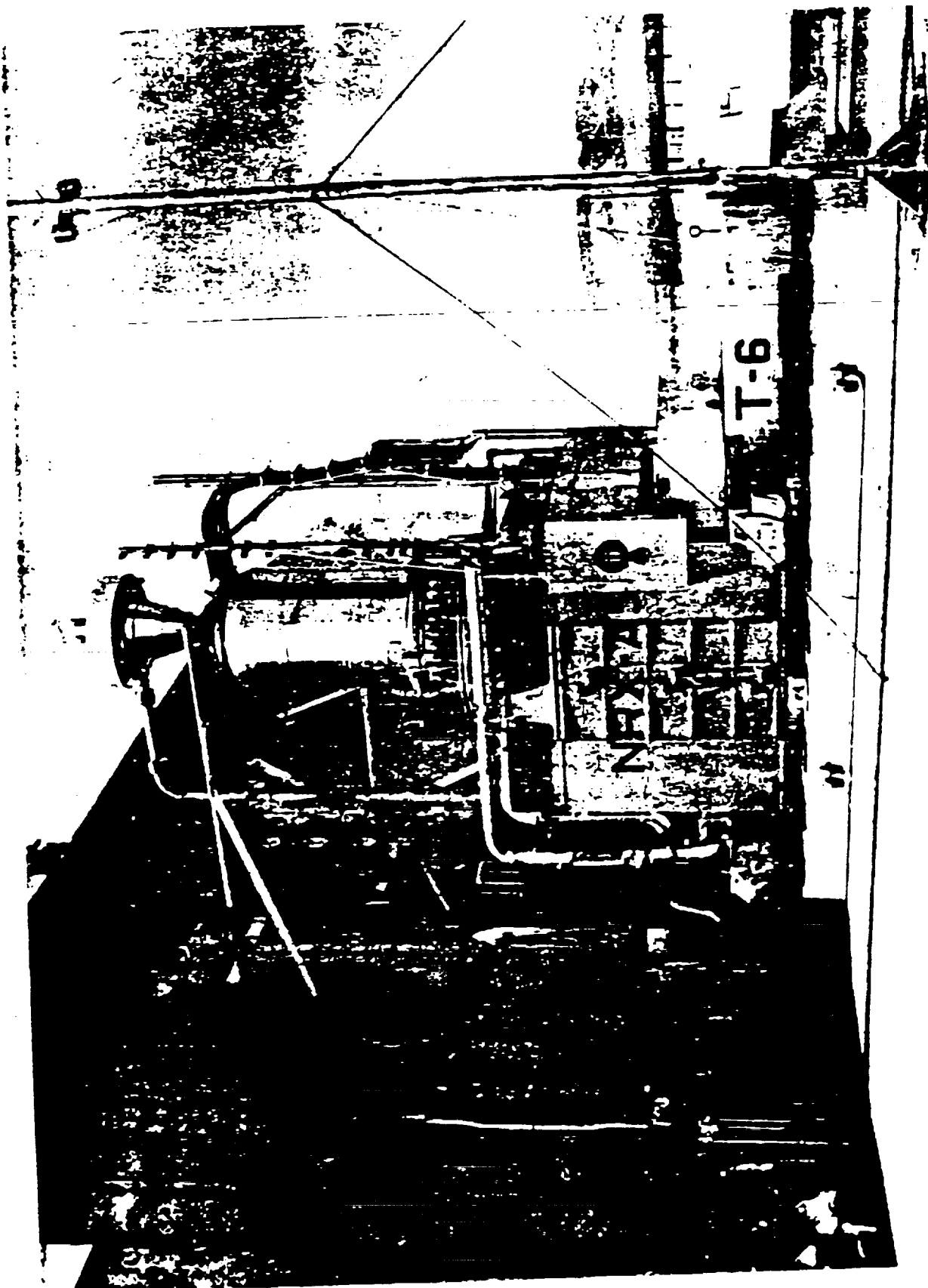


FIGURE 49. NRX A5 AT TEST CELL.

5. To evaluate further the adequacy and reliability of the control and diagnostic instrumentation.
6. To evaluate further the concept of performing a startup transient with constant drum position.
7. To evaluate alternate neutronic calibration techniques. An alternate technique considered was external wire and pellet irradiation.
8. To evaluate the performance of a "no-flux loop" temperature controller.

The NRX A5 reactor was similar to previous NRX A reactors. The core contained 1584 fuel elements and two rows of peripheral elements in the NRX A5 were externally coated with NbC in contrast to one row in the NRX/EST and NRX A3. The same general raw materials, extrusion technique, and NbC coating process parameters were used for the NRX A5 as for the NRX/EST. The NRX A5 nozzle was the same configuration as that used for the NRX A2 and A3 tests, with an expansion ratio of 10:1. The core contained fuel manufactured by both Y-12 and the Westinghouse Astrofuel Facility (WAFF).

The following are the differences between the NRX A5 and previous reactor designs:

1. Elimination of the aluminum barrel surrounding the outside of the inner graphite reflector.
2. Full length pyrographite tile on filler strips.
3. Support blocks with modified washers.
4. Modified fuel element ends.
5. Control drums modified to minimize bowling.
6. Reflector impedance ring relocated to the inlet between the inner graphite reflector and the Be reflector.
7. Tie rod material was changed from Inconel 750 (used in the NRX/EST) to Inconel 718.
8. Two sectors of hot buffer periphery were incorporated as an experiment.
9. Unfueled tips brazed to the hot ends of fuel elements, included as an experiment.
10. Fuel element bore coating profiles.

11. Two fuel elements with molybdenum overcoated bores, included as an experiment.
12. Six skirtless support blocks, included as an experiment.

The first power run consisted of a 1083 K (1950 R) chamber temperature hold for 130 seconds and proceeded with a chamber temperature ramp to 2167 K (3900 R). Upon reaching full power, power oscillations occurred (mentioned later) and lasted about 25 seconds. The run continued under full power conditions for about 15.4 minutes with chamber temperature above 2056 K (3700 R).

The second full power run achieved a power of about 1050 MW with a chamber temperature of over 2222 K (4000 R) over a period of 14.5 minutes. The run was terminated prematurely when the loss in reactivity became excessive.

The accumulated run time for two full power tests was 22.4 minutes above 2222 K (4000 R) nozzle chamber temperature and 30.1 minutes above 2056 K (3700 R) nozzle chamber temperature. The test was terminated by the control drum position limit of 145 degrees before completing 40 minutes of run time at design conditions. This control drum position limit corresponded to 2.2 dollars of corrosion reactivity loss. A fixed control drum start-up from 30 KW to near design power was successfully accomplished. The "no-flux loop" temperature controller also functioned satisfactorily during the test series.

During the first test, power oscillations occurred 20 seconds after nominal full power conditions were attained. These oscillations are very evident on the power and control drum position curves. The oscillations lasted for about 25 seconds, and ended just before control was switched from the "no-flux loop" temperature control to control drum position control. Post-test analysis indicated that the signal from one of the temperature measurements had become noisy and was causing positive temperature spikes in the average. These then caused spikes in the feedback to the "no-flux loop" temperature controller. This thermocouple was automatically rejected 15 seconds after the oscillations ended, and the "no-flux loop" controller functioned properly during the remainder of the test. After the test, the controller was modified to prevent the recurrence of such oscillations.

Following each power run the reactor was "pulse" cooled with liquid nitrogen. Approximately 20 pulses were required to complete each cooldown. The average flowrate during a pulse was about 2.3 kg/s (5 lb/s).

The most significant operations and accomplishments of the NRX A5 test series were:

1. The test assembly was operated for 29.6 minutes at, or above, chamber temperatures of 2111 K (3800 R) and for 22.4 minutes at, or above, chamber temperatures of 2222 K (4000 R).
2. Operation of a new eight decade neutronic system was demonstrated.
3. The reactor was checked out and operated at rated conditions using a temperature and control system without the neutronics power control as an inner loop.
4. The acceptability of a startup from low power to near rated conditions using programmed LH₂ flow with drums in a fixed position was demonstrated.
5. The initial criticality of the reactor was performed after all poison wires were removed.

The post-test evaluation of the fuel elements showed that there was a significant weight loss difference between the Y-12 and WAFF fuel elements. The Y-12 average weight loss was 16.0 grams/element, compared with 36.9 grams/element for the WAFF elements. Also, 9.7% of the Y-12 elements were broken, whereas 70% of the WAFF elements were found broken. It was found that the elements which experienced high weight loss were characteristic of the elements which were broken. The final test report, however, does not discuss the differences between the Y-12 and WAFF elements. Reference 32 reveals that the average NbC bore coating thickness was 1.49 and 1.77 mils for the Y-12 and WAFF elements, respectively. This reference provides additional technical data regarding the fuel element composition.

An improvement in peripheral fuel element performance over the NRX/EST test was found and attributed to the external surface coating of NbC on these elements. However, the fraction of broken elements in the NRX A5 was 13% higher than the NRX/EST.

The two molybdenum bore coated fuel elements both were broken upon disassembly. These elements showed significantly lower weight losses than non-molybdenum coated elements adjacent to them. Also a comparison of corrosion on axially sliced elements from the same core region indicated that molybdenum coating may have been beneficial in reducing midband corrosion. It should be stressed that the extremely small sampling of molybdenum coated elements, and the uncertainties with regard to local reactor environment on corrosion behavior preclude full assessment of molybdenum overcoating performance.

NRX A6 [35, 36]

The NRX A6 was successfully tested on December 15, 1967 following an aborted run made on December 7, 1967. The chamber pressure and propellant flow rate were nominally 4089 kPa (593 psia) and 32.7 kg/s (72 lb/s), respectively.

The primary operational objective was to accomplish a full power (1120 MW) run to a predetermined loss of reactivity, or for a time of 60 minutes. Secondary (experimental) objectives were:

1. Evaluation of the effects of rated conditions duration testing, in fulfillment of the prime objective, on the structural integrity of the test assembly.
2. Evaluation of the capability of experimental fuel elements to withstand the effects of rated conditions duration testing.
3. Evaluation of the performance of NRX A6 design changes and hardware modifications.
4. Experimental evaluation of the effects of an aft-supported reactor on the pressure vessel and nozzle.
5. Evaluation of the nuclear performance of the test article.
6. Evaluation of the thermal and fluid flow performance of the test article.
7. Further evaluation of the performance of the control instrumentation.
8. Evaluation of the performance of improved resistance temperature transducers, accelerometers and a control pressure transducer for use on future test programs
9. Further evaluation of the performance of the NRX control systems.
10. To obtain accurate evaluation of the decay heat after extended operation at rated conditions.

The NRX A6 reactor had the same general configuration as the NRX A2 through NRX A5 reactors, consisting of a fueled graphite core, surrounded by a beryllium reflector assembly and enclosed in an aluminum pressure vessel. The reactor configuration is shown in Figure 50. The principal

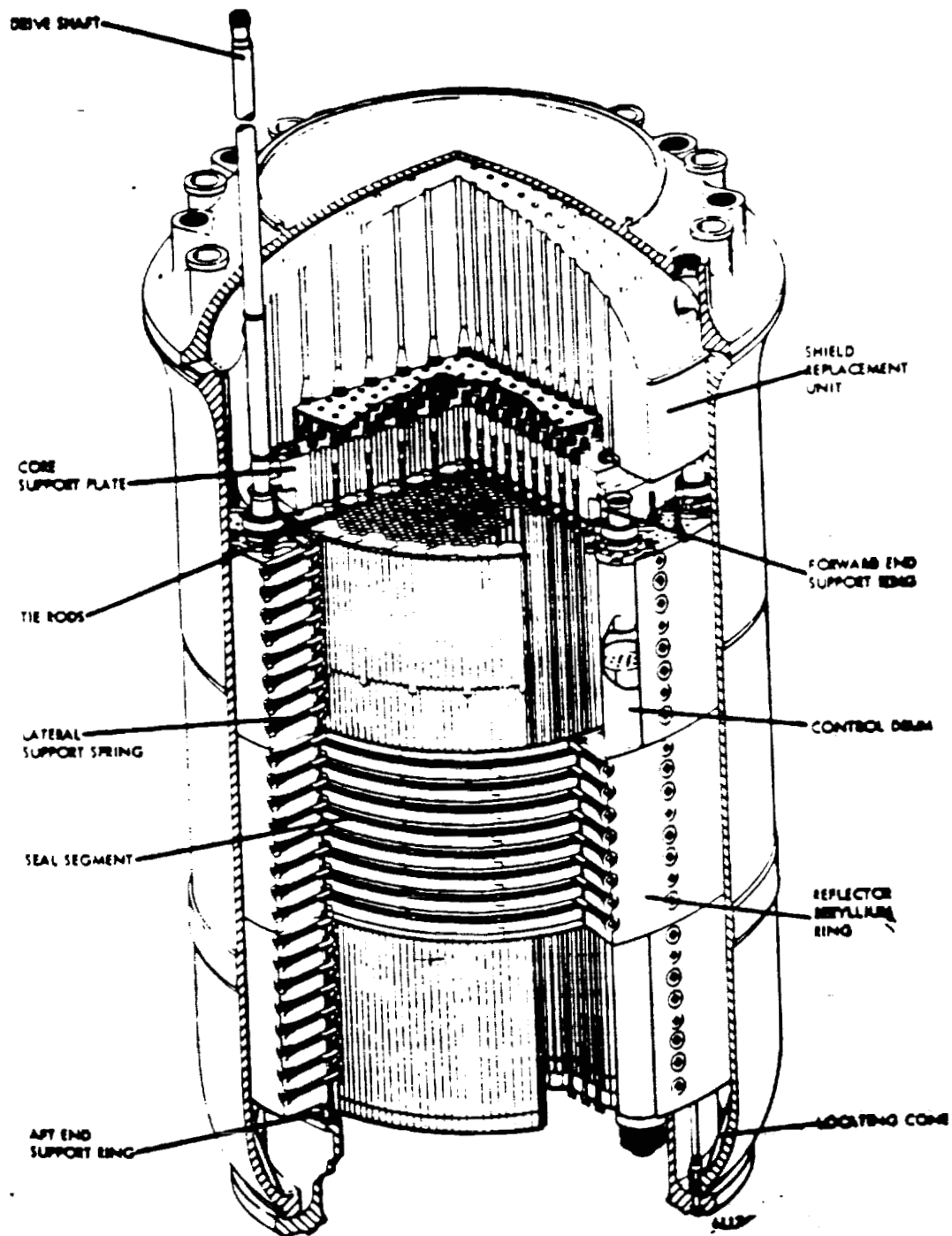


FIGURE 50. NRX A6 SCHEMATIC.

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differences in the NRX A6 design from previous reactors was the elimination of the graphite inner reflector and consequent modifications in core periphery and core lateral support systems, and that the reactor was supported from the aft (nozzle) end. These design differences resulted from two of the NRX A6 design objectives:

1. That the reactor have a basic structure applicable to reactors of greater power density and size.
2. That the core periphery and lateral support system design lead to the reduction of core corrosion.

The NRX A6 core was nominally made up of clusters of six fuel elements and a central, unfueled element. Each cluster was supported axially by means of a tie rod attached to the support plate. The tie rod passed through the central element and was connected to a support block at the aft end of the cluster. Irregular cluster assemblies were used at the core periphery. Partial fueled elements and filler strips completed the cylindrical shape of the core at its periphery.

Changes from NRX A5 in the design of the regular fuel elements consisted of: 1) changes in fuel loading, 2) fuel element coatings, and 3) fuel raw materials and processing. The core had fourteen loading zones which required eleven separate fuel loadings ranging from 132.4 grams/element to 23.9 grams/element. The increased number of loading zones was designed to provide minimum variation in power density across the core, thereby reducing pressure differences between bores to decrease pinholing. NbC channel coating thickness was decreased to improve NbC adherence and crack distribution, and molybdenum overcoating was applied on fuel element channel bores to reduce midband corrosion. New improved requirements (i.e. stricter tolerances across flats, increased flexure strength, etc.) were added to the fuel element specifications to make the elements more uniform.

The run which was attempted on December 7th was initiated by autostart. After 75 seconds at a 18.1 kg/s (40 lb/s) 301 MW hold, a shutdown occurred. It was determined that the shutdown was caused by electrical transients from the mode switching relays, which were coupled into the drum position averaging amplifier and appeared as a transient decrease in average drum position. This signal then caused a minimum drum position shutdown. A fix was made by installing a filter on the drums average position amplifier.

The full power run which was made on December 15, 1967 was successful. The full power hold lasted 60 minutes at or above 2278 K (4100 R) chamber temperature and 1125 MW thermal power. The nominal conditions reached during the

hold were 1130 MW thermal power, 1250 MW neutronic power, 4089 kPa (593 psia) chamber pressure, 2300 K (4140 R) chamber temperature, and 32.7 kg/s (72.0 lb/s) flow rate. During the hold, chamber temperature indicated a lower temperature than predicted, but provided a consistent feedback signal adequate for maintaining reactor temperatures at desired levels. It was postulated that the low chamber temperature thermocouple readings were caused by gas stratification or non-turbulent flow near the nozzle wall where the thermocouples were located.

The operation of all cooldown systems followed the planned shutdown and was satisfactory. LN₂ pulse cooling was completed after 75.3 hours. A post-run criticality test was performed on December 19, with LN₂ and GN₂ used to cool the reactor until reflector material and core inlet temperatures were near ambient.

Post-test examination of the NRX A6 revealed severe axial cracks (both on internal and external surfaces) in the reflector assembly. This was attributed to a 200 degree temperature spike at the end of the test. The NRX A6 was the first reactor to employ three annular beryllium rings assembled (stacked) into a single reflector. Previous reactors had utilized a two-reflector system, an inner graphite cylinder, and an outer beryllium cylinder made up of 12 full-length segments. The NRX A6 was functionally adequate for the NRX A6 tests; the reflector performance did not limit the reactor performance. It was determined that the cracks were due to thermal stresses in the reflector ring. The cracks occurred about two minutes before the end of the full power run; at this time the increased thermal stress exceeded the strength of the beryllium. The thermal stress increase and the fracture toughness decrease were both due to irradiation effects on the beryllium material. The NRX A6 was the first reactor which exhibited reflector cracks, except for NRX A1 which was a cold-flow test only.

The performance of the NRX A6 fuel elements was characterized by inter-element bonding, mild surface corrosion, low pinhole densities, lower midband weight losses, and higher hot end weight losses relative to the NRX A5. The weight loss was 13.1 grams/element¹, midband corrosion was 2.3 grams/element², and gross hot end weight loss was 10.9 grams/element³.

There were several experimental fuel elements in the NRX A6 core. These elements had matrix-additives (25 v/o NbC, 5.0 v/o and Nb Resinate) and various bore coating features. With respect to pinholing, surface corrosion, and aft-end

¹ Compared with 27.0 grams/element for the NRX A5.

² Compared with 25.8 grams/element for the NRX A5.

³ Compared with 8.5 grams/element for the NRX A5.

integrity, the performance of these experimental elements was similar to that of other NRX A6 elements.

In conclusion, the total run time of 62 minutes above a nozzle chamber temperature of 2278 K (4100 R) more than doubled the full power and temperature endurance of previous reactors with a reduction of 75-80% in the fuel element time rate of corrosion compared with that observed in the NRX/EST and NRX A5 reactors. The increase in core corrosion performance is attributed to the combination of improved fuel element coating techniques, across flats dimensional control, attention to coefficient of thermal expansion, flattened core power distribution, and changes in core interstitial pressure distribution. All NRX A6 test objectives were achieved.

XE-PRIME [37]

The XE-PRIME was fired from December 4, 1968 through September 11, 1969, with 24 separate startups. The engine rating was 1140 MW, 3861 kPa (560 psia) chamber pressure, and 2272 K (4090 R) chamber temperature. The XE-PRIME used NRX A5 type fuel.

The major objectives of the test series were:

1. Operate the hot-bleed-cycle engine in a flight-type configuration (i.e., a close-coupled propellant feed system) at rated conditions.
2. Conduct engine start-ups without the use of nuclear instrumentation.
3. Conduct engine start-ups using different control logic sequences.
4. Start and restart the engine from a variety of different initial conditions including: 1) different core, reflector, and pump material temperatures, 2) different source power levels, and 3) different pump-inlet fluid conditions.
5. Demonstrate liquid hydrogen pulse cooling using run-tank flow.
6. Demonstrate the Engine Test Stand 1 (ETS-1) design concept.
7. Obtain experimental data on low-temperature reactivity effects and low-flow operation.
8. Remotely remove a "hot" engine from the test stand and perform remote disassembly at Engine

Maintenance, Assembly and Disassembly (E-MAD)
facility.

The detailed objectives were:

1. Perform a reactor thermal calibration at 1 MW.
2. Perform nuclear autostart to 100 KW.
3. Measure S-1 shield worth.
4. Measure integral worth of drums 1 and 7.
5. Irradiate intermediate power dosimetry.
6. Checkout operation of the steam generator system.
7. Checkout repressurization system.
8. Checkout control drums with hydrogen as the actuation gas.
9. Checkout the turbine power control valve with hydrogen as an actuation gas.
10. Checkout the cryotrap performance using hydrogen.
11. Checkout the modification of the test stand cooling system.
12. Verify that no scram-producing interactions exist between the engine and facility systems.
13. Checkout the system under LH₂ flow conditions.
14. Verify valve sequencing and instrumentation performance.
15. Verify control system temperature loop closure.
16. Confirm predictions of initial portions of XE startup.
17. Demonstrate satisfactory startup to intermediate power, followed by programmed control using the state-programmed temperature control loop.
18. Verify neutronic system calibration at intermediate power.
19. Obtain steady-state engine data: a) rated power, b) intermediate power, c) low power.

20. Obtain transfer function measurements: a) rated power, b) intermediate power, c) low power.
21. Demonstrate engine operation with run tank topping: a) rated power, b) intermediate power.
22. Demonstrate duct operation at: a) rated engine flow conditions, b) intermediate flow conditions, c) low flow conditions.
23. Demonstrate satisfactory operation during an emergency flow shutdown.
24. Demonstrate satisfactory startup to rated conditions.
25. Demonstrate satisfactory engine restart with nuclear autostart to temperature loop closure, from cooldown conditions.
26. Demonstrate "normal shutdown" and LH₂ pulse cooldown.
27. Obtain final post EP-5 xenon measurements.
28. Obtain preliminary data on the dry temperature autostart system on run tank flow only.
29. Obtain preliminary data on the wet temperature autostart system for various run tank pressures and power levels on run tank pressure only.
30. Investigate startup with closed loop temperature control and low chamber pressure demand.
31. Investigate restart with closed loop temperature control and low chamber pressure demand.
32. Demonstrate a wet temperature autostart and bootstrap from initially ambient engine conditions.
33. Demonstrate restart with a low run tank pressure, low chamber temperature and pressure demands and closed loop temperature control.
34. Demonstrate a low run tank pressure, low P_C/T_C demands and closed loop temperature control.
35. Demonstrate a dry temperature autostart from initially ambient engine conditions.

36. Investigate controller stability under low P_C/T_C conditions with the turbopump assembly (TPA) operating.
37. Obtain temperature autostart data for restart with engine conditions off ambient (hot core, cold reflector, delay neutron power level in the 100 KW range).
38. Obtain dry temperature autostart data for restart with engine condition off ambient (cold core, cold reflector).
39. Obtain open loop startup characteristics of the engine.
40. Investigate the ability of the engine system to restart with low T_C/P_C demands on low run tank pressure (138 kPa).
41. Investigate the effect of an early pump discharge shutoff valve (PDSV) opening on a dry temperature autostart with a cold core (cold ramp autostart).
42. Obtain autostart data on sensitivity to drum profile.
43. Obtain information on the chilldown characteristics of the engine system with a low tank pressure (172 kPa) and no steam generator operation.
44. Obtain data on the: a) reflector thermal reactivity coefficient, b) core thermal reactivity coefficient, c) drum worth at reduced reflector temperatures, d) measurement of hydrogen worth.
45. Obtain mapping data using constant drum rate (on/off temperature controller) and constant turbine power control valve (TPCV) rate.
46. Obtain open loop startup data under immediate restart conditions with very high source power level.
47. Investigate the effect of very high core temperature on the wet temperature autostart method of startup.
48. Obtain data on the engine shutdown characteristics as a function of TPCV position.
49. Demonstrate engine startup using the high specific impulse program.

50. Provide engine data to determine preconditioning an operating technique which will result in a predictable elapsed time for startup with uncertainties in prebootstrap conditioning and critical angle.
51. Determine the accuracy with which the critical drum angle can be determined without using nuclear instrumentation, from various initial conditions.
52. Determine the effect of variations in drum program terminate temperature.
53. Determine the effect of initial engine condition, various drum exponential angles, points of drum program terminate and PDSV delay time on dry temperature autostart sequence up to point of bootstrap.
54. Investigate laminar flow instability at NERVA maximum and minimum core inlet temperature during cooldown.
55. Provide information on engine temperature asymmetries associated with low flow rates of GH_2 and LH_2 .
56. Demonstrate and evaluate start/restart over a wide range of initial conditions: a) at or near ambient engine temperatures, b) cold core and reflector, c) hot core with cold reflector, d) ambient core with cold reflector, e) source power low, f) source power high, g) miss critical position on drum program high, h) miss critical position on drum program low, i) nominal drum program.
57. Demonstrate and evaluate alternate startup schemes: a) nuclear power control, b) temperature autostarts; wet, dry, damp, c) open loop.
58. Evaluate critical position measurement without nuclear instrumentation.

The XE-PRIME was the final in a series of reactor and engine development test assemblies and the first nuclear rocket engine to be tested with components in a flight-type close-coupled arrangement. Reactor assemblies NRX A3, NRX A5, and NRX A6 were very similar to XE-PRIME in basic design of the reactor and nozzle but utilized an independent facility liquid hydrogen feed system. The NRX/EST test article also was very similar in basic design to the XE-PRIME, including the hot-bleed TPA engine cycle. The operating

characteristics of NRX/EST and XE-PRIME correlate directly, except for start-up and shutdown transients.

The XE-PRIME engine system shown in Figure 51 was a close-coupled nuclear rocket engine designed for ground test development using liquid hydrogen as a propellant. There were two engine modules. The lower module contained the reactor and pressure vessel assembly, nozzle, lower thrust structure, external engine shield, control drum actuators and lower module instrumentation. The upper module consisted of the upper thrust structure which housed the TPA, lines, valves, and upper module instrumentation. The upper module was designed to be remotely replaced should a major module component fail. All fluid lines and electrical wires which passed from the upper to the lower module had remote connectors. The test stand adapter (TSA) provided the necessary transition from the engine to the test stand. This unit housed the remote connectors which were used to attach the engine to the TSA, remote line disconnects, and all lines and instrumentation cables which led from the facility to the engine. The propellant shutoff valve and supporting instrumentation were also housed in the TSA.

The engine was designed to produce a nominal thrust of 246663 N (55430 lb) with the reactor operating at a power level of approximately 1140 MW, chamber temperature of 2272 K (4090 R), chamber pressure of 3861 kPa (560 psia), nozzle flow rate of 31.8 kg/s (70.0 lb/s), and total flow rate of 35.8 kg/s (79.0 lb/s) (including 0.4536 kg/s diverted for the cooldown system). The engine had an overall specific impulse of 710 seconds at rated conditions. It was 6.9 m (227 in) in length, 2.59 m (102 in) in diameter, and weighed approximately 18144 kg (40000 lb) as a test article. The engine operating map is provided in Figure 52.

The XE-PRIME main nozzle, shown in Figure 53, was a convergent-divergent shape with a half angle convergence of 45 degrees, a half angle divergence of 17.5 degrees, and an exhaust expansion ratio of 10:1. The nozzle consisted of a bundle of stainless steel U-tubes supported by a stainless steel jacket. Flow passages were formed by inserting the legs of the U-tubes into slots in the jacket and then brazing them in place. Hydrogen from the pump discharge line entered the nozzle assembly through a torus inlet manifold at the aft end of the nozzle, flowed through the tubes, and emerged radially from the forward end of the tubes and flowed into the reflector inlet plenum. A portion of the coolant from the inlet manifold was diverted through three external tubes to cool the nozzle flange and bolts that attached the nozzle to the pressure vessel. A bleed port was located in the convergent section of the nozzle to divert hot hydrogen gas to the turbine. Diluent gas was tapped from the pressure vessel dome and routed through a 3 inch diameter diluent line to an annular passage in the

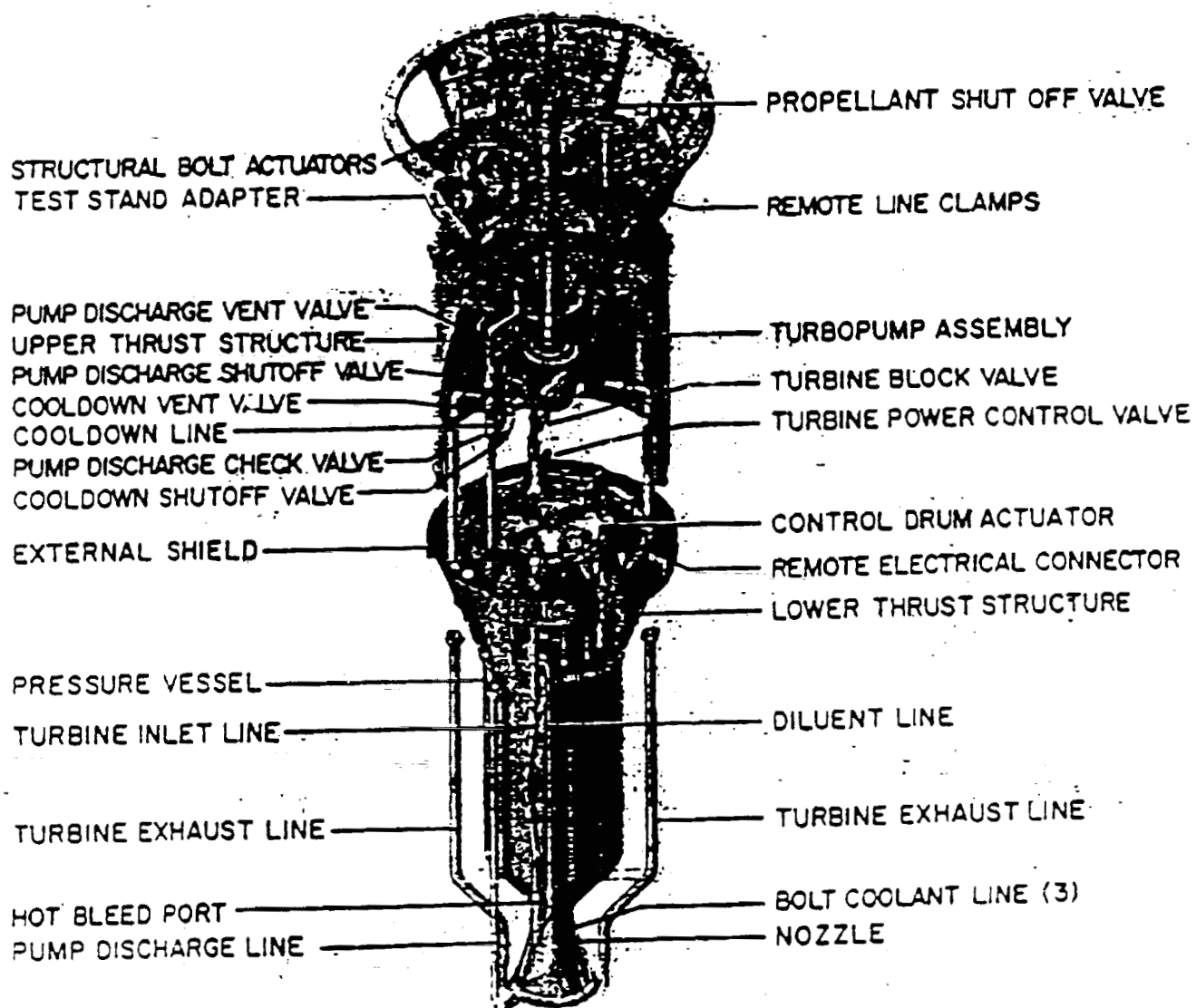


FIGURE 51. XE-PRIME ENGINE CONCEPT.

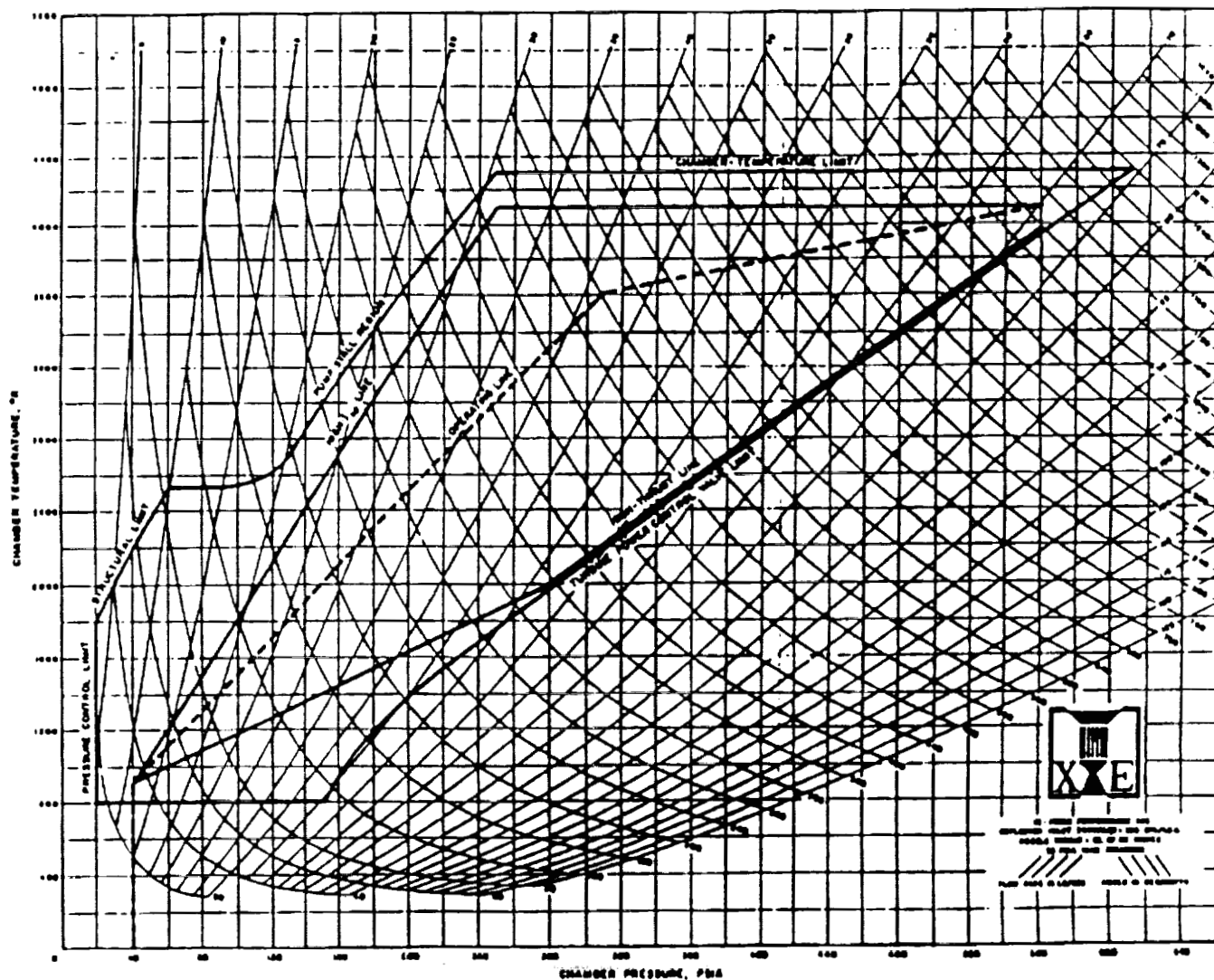


FIGURE 52. XE-PRIME OPERATING MAP.

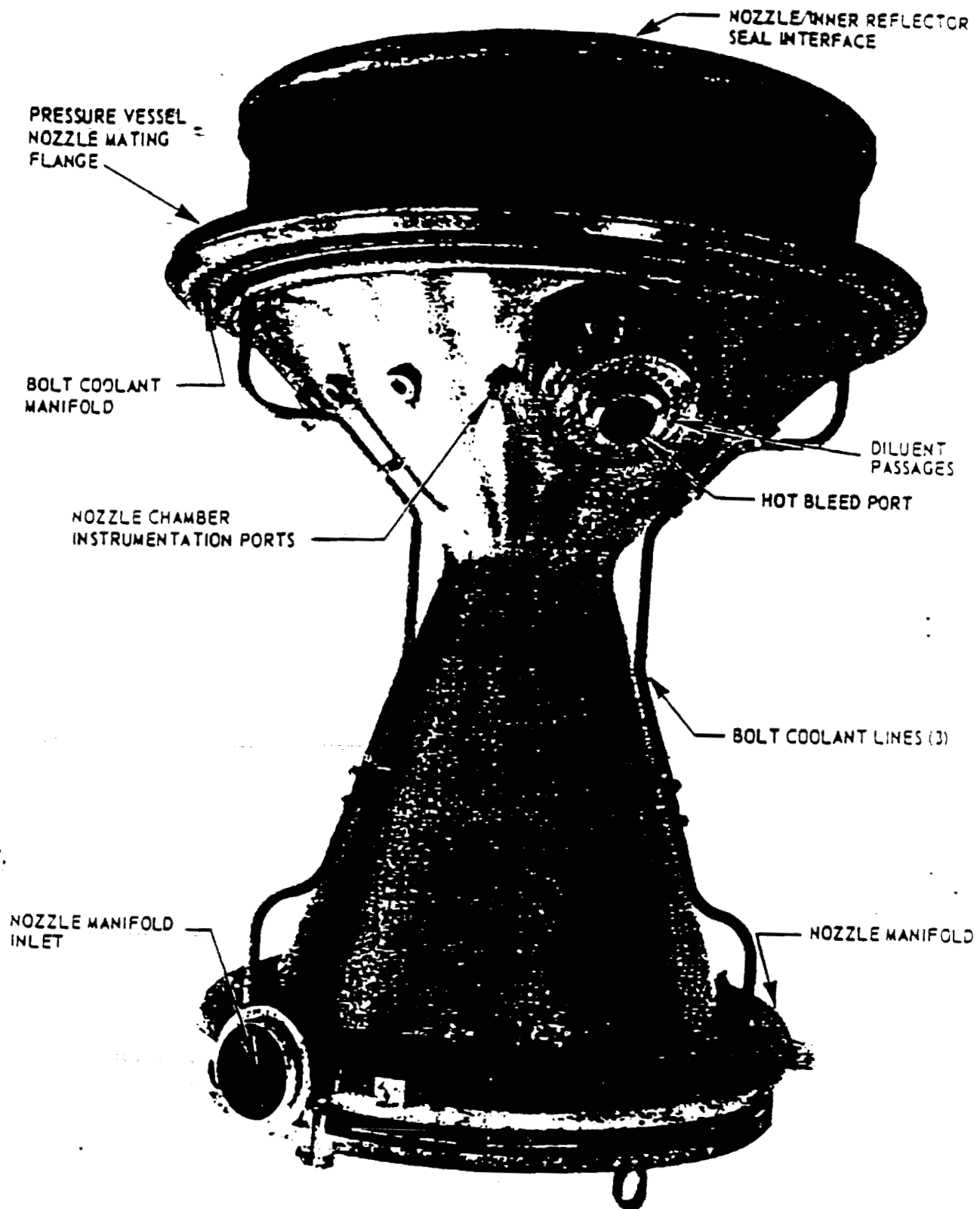


FIGURE 53. ENGINE NOZZLE.

turbine inlet line. The diluent gas cooled the elbow of the turbine inlet line and the hot bleed port. The diluent was then injected into the hot gas stream from the nozzle chamber, and the mixed gas exited from the port into the turbine inlet line which carried the gas to the turbine.

The XE-PRIME engine control system (ECS) provided several modes of automatic operation, as well as various manual modes of operation. The purpose of the multiple modes of control was to obtain performance information for guidance in the development of the NERVA engine, and to obtain additional information for confirming and improving methods which were used to analytically model the engine. The control drums regulated reactor power while the TPCV regulated the gas flowing to the turbine. Normally, the objective was to obtain desired engine chamber temperature and pressure conditions. However, there were interactions between the two control parameters which made them interdependent in terms of controlling chamber temperature and pressure. In the automatic modes of control, these interacting effects were automatically regulated to maintain the desired operating condition. In manual control, operator action was required to maintain control parameters at the desired operating point.

The ECS provided the following operating modes: 1) manual drum control, 2) reactor power level control, 3) chamber temperature control, 4) manual TPCV control, 5) chamber pressure control, and 6) program control. In addition, control of startup and shutdown operations was provided. Startup and shutdown could be accomplished either manually (with the operator supplied with feedback information from console meters) or automatically. Startup could be made on nuclear power, or on temperature without the use of nuclear instrumentation. Further details are beyond the scope of this report but can be found in Reference 37.

The XE-PRIME test program consisted of 40 runs grouped into ten Experimental Programs (EPs) which began December 4, 1968 and ended September 11, 1969. The engine was down-fired at the Nuclear Reactor Development Station. The test stand provided a reduced atmospheric pressure (about 6.9 kPa or 18288 m altitude) around the engine to partially simulate space conditions. The engine was successfully started 24 times, 15 of which were from initial conditions or used control logic never before attempted. The engine operated at essentially full power (1140 MW, 3861 kPa chamber pressure, 2272 K chamber temperature) for 3.5 minutes on June 11, 1969.

Highlights of the ten experimental plans (EPs) are as follows:

EP-1 was conducted in four parts: EP-1 and SL-2, December 4 and 6, 1968; EP-1A and EP-1B, February 20 and 27, 1969. The test assembly was removed and reinstalled in the ETS-1 between the December and February runs as a precautionary measure during the Benham¹ event. EP-1 consisted of attaining initial criticality at an average drum bank position of 99.8 degrees and: 1) determined the worth of the S-1 side shield; 2) achieved power calibration of the test stand control system (TSCS) shield mounted neutronic detectors and engine mounted neutron detectors (EMND); 3) performed control system preliminary verification; 4) insured that the individual control drum worths were adequate for power testing and verified that the worth of each drum was approximately equal to that of each of the other drums; 5) activated the low and intermediate level dosimetry; and 6) verified acceptability of modifications to the steam delivery system.

EP-2A, conducted March 20, 1969, consisted of three runs which duplicated the initial portions of the engine startup procedure. The runs checked out the systems under liquid hydrogen flow conditions, and verified valve sequencing, instrumentation performance, and control system temperature loop closure.

An attempt to perform EP-3 April 3, 1969 was terminated when the turbine block valve (TBV) could not be fully closed. The valve was removed for an investigation which indicated that foreign particles were the cause. A different valve was installed and EP-IIIC was conducted April 17, 1969. Two runs to intermediate power levels in preparation for full power testing were initiated. Run 1 was successful and was terminated as planned with a simulated loss of flow and emergency cooldown. In the second run, there was a late reopening of the TBV, resulting in a very rapid bootstrap with excessive hydrogen reactivity insertion. The startup was terminated by a fixed programmed power scram.

After two attempts (EP-5A and -5B) in which the TPA failed to rotate, a new TPA with increased bearing coolant labyrinth clearance was installed and the full power test (EP-5C) was conducted June 11, 1969. The engine was operated essentially at full power for 3.5 minutes. During the full power hold, transfer function measurements and run tank topping were performed. Following normal shutdown, liquid hydrogen pulse cooling was demonstrated for three pulses. Prior to reaching full power, excessive low frequency TPA vibrations were reported by the Test Diagnostic Center (TDC) at the 1793 kPa/1722 K (260

¹ Nuclear weapons test areas are located adjacent to Jackass Flats. Benham is the name of a weapons test. The test assembly was removed from ETS-1 to prevent damage due to terrestrial accelerations.

psia/3100 R) hold and a reduction in operating conditions was ordered. During the retreat, TDC verified that the amplitude of the vibration was anomalous. The run was then continued with the 1722 K (3100 R) hold eliminated. Also a TPCV override occurred 5 seconds after the ramp from 2827 kPa/2111 K (410 psia/3800 R) to full power was initiated, and full power conditions were established in chamber temperature control and TPCV position control.

EP-4A, conducted June 26, 1969, consisted of four startup runs. Runs 1 and 2 were preliminary dry temperature and wet temperature autostarts where bootstrap was prevented by keeping the TBV closed. Runs 3 and 4 were closed temperature loop start and restart with a 276 kPa (40 psia) pressure null point. Previous tests had used 414 kPa (60 psia) as the null point. The temperature loop was closed prior to TPCV opening by not activating "start engine". Dry temperature and wet temperature autostart logic was satisfactory and no problems were encountered while in chamber temperature control prior to bootstrap.

EP-6A, conducted July 10, 1969, consisted of six startup runs. Closed temperature loop restarts were satisfactorily achieved with run tank pressures of 172 and 159 kPa (25 and 23 psia). Wet temperature autostarts were performed from ambient, initial engine conditions and from a hot core and cold reflector initial condition. Dry temperature autostarts were performed from ambient conditions and from a cold core and cold reflector condition. Mapping operations at nominal chamber temperature and pressure conditions of 944 K/414 kPa (1700 R/60 psia), 611 K/414 kPa (1100 R/60 psia), 578 K/345 kPa (1040 R/50 psia), and 444 K/276 kPa (800 R/40 psia) were also performed.

EP-7A, conducted August 24, 1969, consisted of four startup runs. The first was an open loop startup and was successful although three TPCV overrides occurred as the valve was ramped open. This was followed by three damp autostarts (i.e., data acquisition system (DAS) control logic with flow initiated at "start reactor") with different start conditions: i.e., cold core, nominal drum program; ambient core nominal drum program; and ambient core nominal minus 10 degrees drum program. Bootstrap was not achieved during a startup attempt with a run tank pressure of 138 kPa (20 psia) and a DAS with a cold core and delayed flow was aborted by a maximum drum position scram before reactor startup was completed. The EP was terminated by a malfunction of a steam generator after 48 minutes of operation.

EP-8A, conducted August 13, 1969, included an engine chilldown test physics experiment and five startups. Engine chilldown characteristics from ambient temperatures to 33 K (60 R) at the reflector inlet were measured using only 172

kPa (25 psia) tank flow. Reactivity data with reflector and core temperatures below ambient (i.e., approximately 83-111 K) and the hydrogen worth were obtained. Two open loop startups were performed. The first was from ambient initial conditions to 1667 K (3000 R), followed by demonstration of "on-off" temperature controls, open loop mapping, and shutdown. The second open loop startup from hot core and cold reflector initial conditions to 944 K (1700 R) was satisfactory though the drum program was anomalous (i.e., there were two exponentials rather than one). Drum transfer function measurements were made at 414 kPa (60 psia) 556 K (1000 R). A damp temperature autostart was made from ambient initial conditions to 944 K (1700 R), using core material temperatures for control feedback and with a drum program of nominal +10 degrees; however, this nominal figure was determined in a different manner than that used for EP-7. Two wet temperature autostarts completed this EP. The first was from initial warm core ($T_C=578$ K) and high source power (2.77 MW) conditions. The second was from hot core ($T_C=700$ K) and low source power (87.5 KW) conditions. Low power TPCV mapping with in core temperature control feedback was performed just before the last run was terminated.

EP-9A, conducted August 28, 1969, consisted of two startups in the wet temperature autostart mode along the high specific impulse program line to 2068 kPa/2233 K (300 psia/4020 R). Three attempts to start were aborted by period scrams. The aborted starts were preceded by an inadvertent severe cooling of the engine system by liquid hydrogen flow from the cooldown system. The core was warmed both before the third attempt and the successful startup. When the drum program was initiated for the successful start, T_C was 250 K (450 R) and decreasing, reflector inlet was 23 K (42 R), and source power was 600 W. These conditions represent the coldest initial engine system condition from which a successful temperature autostart was made. During the first run, the 2068 kPa/2272 K (300 psia/4090 R) hold was maintained in program control and run tank topping performed. For the second run, the hold was made with a fixed TPCV and the cooldown system remained "on-line" during shutdown controller operation.

EP-10A, conducted September 11, 1969, consisted of six dry temperature autostart tests and a laminar flow test with liquid hydrogen and gaseous hydrogen. The steam generator system (SGS) was not operated and the TBV was closed to prevent bootstrap. These tests provided information to evaluate: 1) the effect of performing dry temperature autostarts from various initial conditions over a range of drum exponential ramp settings relative to the critical drum bank position, and with various temperature criteria for drum program termination; 2) the ability to delay engine chilldown and bootstrapping following a reactor startup in the dry temperature autostart mode; 3) the ability to

startup the reactor and precondition the entire engine to the point of initiating bootstrap and then hold at that point; and 4) whether critical drum angle can be estimated without nuclear instrumentation. The laminar flow test was done in two parts: 1) with liquid hydrogen at 0.771 kg/s (1.7 lb/s) for 21 minutes at power levels of 2.8, 8.5, 11.1, and 14 MW; and 2) with gaseous hydrogen at flow rates between 0.068 and 1.27 kg/s (0.15 and 2.8 lb/s) and power levels between 0.4 and 8 MW. Considerable fluid temperature asymmetries were noted through the reactor system during liquid hydrogen flow. The asymmetries were worse during decreasing temperature transients. Very little asymmetry was present during gaseous hydrogen flow. Instability conditions were not apparent, although localized instability, that did not express itself in the individual measurements, could have existed. Also, equilibrium conditions were never achieved; thus, an evaluation of flow stability under extended steady-state conditions was not possible.

The significant results of the test series were:

1. The engine was successfully started 24 times, 15 of which were from initial conditions or used control logic never before attempted.
2. Start-up test results showed that bootstrap characteristics can be controlled over a wide range of chamber temperature. For 15 tests, the time from TPCV first-motion until pressure-null was 12.0 ± 1.7 seconds. Thirteen of these were within 11.1 ± 0.8 seconds although chamber temperature at initiation of bootstrap differed by as much as 278 K (500 R).
3. Temperature-autostart bootstraps were successfully conducted over a drum exponential set point range of 19.5 (+11 to -8.5 degrees from critical).
4. Test results indicated that the autostart equipment can be used to determine the approximate critical drum angle.
5. Successful engine start-up was achieved at a run-tank pressure as low as 159 kPa (23 psia) with a back pressure of 55 kPa (8 psia).
6. The physics tests conducted during EP-8A showed:
1) there is a small but clear dependence of dry drum-worth on core-reflector thermal conditions (drum worth increases with increasing core and reflector temperatures); and 2) the reflector reactivity coefficient is negative to core-reflector temperatures of 56 K (100 R) (a strong

dependence on core temperatures: at elevated core temperatures (~ 556 K), the reflector effect becomes negligible).

7. Several different control modes were used: 1) programmed control; 2) independent chamber-pressure and temperature control; 3) state-point temperature control; 4) "on-off" temperature control; 5) TPCV-position control; 6) drum-position control; and 7) power control. In each case, control was sufficiently positive and precise to obtain planned engine conditions. During EP-9A, the ability of the engine and control system to repeat a programmed transient operation from extremely different initial conditions was demonstrated.
8. Results of start-up tests suggest a possible operating sequence to provide a more constant start-up time. The technique is to precondition and hold the engine in a ready-to-bootstrap condition, requiring only the initiation of power to the turbine to initiate bootstrap.
9. In EP-9A, period scrams occurred during attempts to perform a wet-temperature autostart with an initial cold reactor and a low-source power level. In addition, positive total reactivity conditions existed with the drums fully in following two scrams. These results indicate that with low-source power and cold-reactor conditions, liquid hydrogen flow to the engine should be terminated very quickly following a scram during start-up. Also, when liquid hydrogen flow is initiated to a cold engine, special restrictions are necessary to prevent exceeding a given shutdown value.
10. Chamber-pressure oscillations observed during shutdown from moderate power indicated that shutdown-controller design was not optimum: i.e., pump tailoff and turbine power control valve (TPCV) reset were too rapid.
11. Data from the low-flow tests of EP-10A showed greater system temperature asymmetries with liquid hydrogen than with gaseous hydrogen.
12. Run-tank topping during engine operation was demonstrated to be a practical method of extending the test-duration capability of the ETS-1 facility. Results indicate that topping produces a small increase in pump-inlet fluid temperature which results in a slight increase in engine-system flow impedance.

13. Performance of the aerodynamic duct throughout the test program was essentially as predicted. There was no evidence of "buzzing" or other undesirable interaction during duct "pull-in" or "drop-out", even though several steady-state engine hold periods were conducted at conditions near the duct "pull-in" point.

Post test examination of the fuel elements showed that performance was generally good, with moderate weight losses, low pinhole densities, light surface corrosion, no interelement bonding, and very little corrosion weakening. Weight losses and bore corrosion in the hot end region were somewhat higher than predicted, with losses, bore corrosion, and pinhole densities tending to be highest in Y-12 bore coated elements. Bore corrosion was predominantly by ring corrosion with some corrosion pocket formation indicating influence of cyclic and low power testing. Some minimal hydrolysis damage could have occurred in the Y-12 bore coated elements. Mild "notch" pattern corrosion occurred on some peripheral fuel elements. Element fracture due to corrosion weakening occurred predominantly on peripheral elements. Some corrosion of the exit faces of fuel elements occurred due to "sticking" to support blocks. It is important to remember that the XE-PRIME was not a fuel test reactor, and used NRX A5 type fuel - not the current fuel type.

ACRONYMS ASSOCIATED WITH XE-PRIME

DAS	- Data acquisition system.
ECS	- Engine control system.
EMAD	- Engine Maintenance, Assembly, and Disassembly facility.
EMND	- Engine mounted neutron detectors.
ETS-1	- Engine Test Stand -1
PDSV	- Pump discharge shutoff valve.
SGS	- Steam generator system.
TBV	- Turbine block valve.
TDC	- Test Diagnostic Center.
TPA	- Turbopump assembly.
TPCV	- Turbine power control valve.
TSA	- Test stand adapter.
TSCS	- Test stand control system.

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36. "NRX-A6 REACTOR TEST ANALYSIS REPORT", Westinghouse Astronuclear Laboratory WANL-TNR-223, 1968.
37. "XE-PRIME ENGINE FINAL REPORT", Aerojet General RN-S-510, 1970.

*At the time this report was generated, most of these documents could be obtained from the National Technical Information Service (NTIS), Springfield, VA 22161, (1-800-336-4700).

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APPENDIX A SOME CLOSING REMARKS

Each reactor test had particular objectives, problems areas, and technical accomplishments. Major advancements were made in a variety of areas too numerous to list. Just a few of the areas where knowledge was gained were:¹

1. Fuel Development

Reactor fuel elements and coatings to resist hydrogen corrosion were successfully developed. It was found that NbC was effective at reducing hydrogen corrosion and most ROVER reactors utilized NbC coated fuel elements. Later in the program, fuel coated with ZrC was found to perform superior to NbC coated fuel and was used in the PEWEE-1 reactor. It was also found that an overcoating with molybdenum was effective in preventing midband corrosion. Late fuel developments included uncoated (U,Zr)C composite and pure (U,Zr)C carbide fuel elements which were tested in the NF-1. A third advanced fuel, referred to as high-CTE graphite matrix fuel, was developed and intended for the NF-2 test which, unfortunately, did not take place. Thus the program terminated with three promising fuels at hand.

2. Two Phase Flow

Two phase hydrogen flow did not pose any significant problems during the ROVER program. The postulated neutronic control difficulties associated with local, unstable, high density liquid hydrogen entering the core did not occur.

3. Engine Clustering

Limited testing of engine clustering of KIWI class engines showed no significant neutronic interactions. However, only very limited documentation was reviewed, and the interested reader should further explore this area.

4. Radiation Effects On Instrumentation

The importance of understanding the effect of radiation on instrumentation was clearly demonstrated when the PHOEBUS 1A capacitance gauges gave erroneous hydrogen level readings. As reported, the reactor overheated when propellant was exhausted - while the gauges

¹ This section is my no means intended to summarize or identify all major areas of advancement achieved by the ROVER program. The areas identified were at the request of Mr. Harold Gerrish of MSFC.

indicated that significant hydrogen was still available.

5. Flow Instability and Vibration

The ejection of core from early KIWI B reactors was found to be due to flow-induced vibration. This emphasizes the important role that cold flow testing (reactor tests containing fuel elements identical to the power reactor except that they contain no fissionable material) plays in the development of operational nuclear rocket engines.

6. Control Drums

Although no control drum failures occurred during the reactor tests, this was one area of concern. During one startup of the NRX/EST reactor, control drum actuators experienced high torque readings as a result of control drum bowing which caused the control drums to rub against the reflector sector bores. This was due to high (361 K) initial beryllium temperature.

The reader is also referred to the PHOEBUS 2A section of this document which reports recommendations made by Los Alamos personnel at the conclusion of the PHOEBUS 2A reactor test.

APPENDIX B LIMITED GLOSSARY OF TERMS

Bootstrap: Self starting reactor - i.e. after setting the control drums and opening the propellant valve, the engine starts itself.

\$ (Dollar): A measure of the reactivity of a reactor system equal to one delayed neutron fraction. The amount of reactivity necessary to make a reactor prompt critical is used to define the unit known as the dollar. A dollar is not an absolute unit, but varies from fuel to fuel.

Epithermal Reactor: A reactor that operates in the neutron energy range between a thermal reactor and a fast reactor, 1 eV - 100 keV for ^{235}U .

Fast Reactor: A reactor characterized by average neutron energies above 100 keV for ^{235}U .

Hydrogen Worth: The difference in reactivity of a reactor system when hydrogen is present as opposed to the system without hydrogen present.

Pinholing: A form of fuel element bore corrosion evidenced by the appearance of pinholes extending normal to the direction of axial flow within the fuel element. Typically these pinholes would extend from the element bore to the element surface.

Reactivity: The deviation of the core multiplication from unity.

$$\text{Reactivity} = \frac{k-1}{k}$$

where k = ratio of rate of neutron production to rate of neutron loss (absorption plus capture).

Scram: Inserting negative reactivity very rapidly to force the reactor subcritical in case of an emergency.

Specific Impulse: The ratio of engine thrust to propellant weight flow rate. Usually expressed (although not technically correct) in units of seconds.

APPENDIX C SUMMARY OF REACTOR RUNS

This section summarizes the key parameters for each reactor test. Interpretation of the test results, however, is not always objective and will vary from individual to individual. For instance, the time at reactor full power is difficult to determine because what does one consider full power - 90%, 95%, or some other percentage of rated power? Not only did power, flow rate, and temperatures vary with time for a "steady" condition, but many runs were tested over a range of parameters. Added complexity arises because reactor test information is not consistent from report to report (written during the same time period). Even worse are so called reactor test summary results written years after the test was completed¹.

What is offered in this section is my interpretation of the test results from reports closest to the test source. This is presented in Table C-1. The reader should consider much of the information as good approximations rather than absolute values. Deviations from other reports are expected. The ideal vacuum specific impulse values reported are based on the propellant temperature at the fuel exit and are, therefore, the theoretical maximum assuming non dissociated hydrogen.

Also included in this section is a useful table, identified as Table C-2, taken from Westinghouse Astronuclear Laboratory report WANL-TME-1788. Time and resources did not permit a critical review of this table and the results may deviate from my own. Nonetheless, it is felt that the reader will benefit from its inclusion.

¹ This author has encountered gross errors contained within summary tables being circulated at the time this report was written.

TEST DATE	REACTOR	NUMBER OF STARTS	AVG FULL POWER (MW)	TIME AT FULL POWER (s)	AVG PROPELLANT TEMPERATURE (K) CHAMBER FUEL EXIT	CHAMBER PRESSURE (kPa)	FLOW RATE (kg/s)	IDEAL VACUUM SPECIFIC IMPULSE (s)
- RESEARCH REACTORS -								
7/1959	KIWI A	1	70	300	1778	?	3.2	724
7/1960	KIWI A'	1	88	307	2206	1125	3.0	807
10/1960	KIWI A3	1	112.5	259	2172	1415	3.8	800
12/1961	KIWI B1A	1	225	36	1972	974	9.1	763
9/1962	KIWI B1B	1	880	SEVERAL	>2278	2413	34.5	>820
11/1962	KIWI B4A	1	450	SEVERAL	1556	1814	19.0	677
5/1964	KIWI B4D	1	915 990 PEAK	64	2006 2156 PEAK	2378 2539 PEAK	3606	31.1 837 865 PEAK
8/1964	KIWI B4E	2	937	480	1972	2356	3427	31.0 834
6/1965	PHOEBUS 1A	1	1090	630	2278	2444	3772	31.4 849
2/1967	PHOEBUS 1B	2	1290 1450 PEAK	1800	2094 2244 PEAK	2306 2456 PEAK	5075	38.1 825 851 PEAK
6-7/1968	PHOEBUS 2A	4	4082	744	2256	2283	3827	119.0 821
11-12/1968	PEWEE 1	3	503	2400	1803	2539 2750 PEAK	4344	18.8 865 901 PEAK
6/1972	NF-1	5	44	6528		2444	1.7	(849)
- DEVELOPMENT REACTORS -								
9-10/1964	NRX A2	2	1096	40	2119	2229	4006	34.3 811
4-5/1965	NRX A3	3	1093	990	2189	>2400	3930	33.3 >841
2-3/1966	NRX/EST	11	1144	830	2292	>2400	4047	39.3 >841
6/1966	NRX A5	2	1120	580	2287	>2400	4047	32.6 >841
11-12/1967	NRX A6	2	1199	3623	2406	2558	4151	32.7 869
3/1969	XE-PRIME	28	1137	1680	2267	>2400	3806	32.8 >841

TABLE C-1. REACTOR TEST SUMMARY

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TABLE C-2. ROVER PROGRAM P

TEST DATE	TEST DESIGNATION	TIME AT CHAMBER TEMPERATURE						MAXIMUM TEMPERATURE	TIME AT MAXIMUM TEMPERATURE	200 MW AND OVER
		1900°R AND OVER	2500°R AND OVER	3400°R AND OVER	3800°R AND OVER	4000°R AND OVER	4100°R AND OVER			
13 MAY 1964	KIWI-B4D,EPIV	210 SEC	165 SEC	60 SEC	-	-	-	3830°R	20 SEC	190 SEC
		3.5 MIN	2.8 MIN	1.0 MIN	-	-	-	-	-	3.2 MIN
28 AUG 1964	KIWI-B4E,EPV	607 SEC	592 SEC	514 SEC	-	-	-	3780°R	5 SEC	625 SEC
11 SEPT 1964	KIWI-B4E,EPVI	237 SEC	225 SEC	164 SEC	-	-	-	3610°R	95 SEC	230 SEC
	KIWI-B4E, TOTAL	844 SEC	817 SEC	678 SEC	-	-	-	-	-	855 SEC
		14.1 MIN	13.6 MIN	11.3 MIN	-	-	-	-	-	14.2 MIN
24 SEPT 1964	NRX-A2, EPIV	391 SEC	355 SEC	207 SEC	-	-	-	3750°R	40 SEC	390 SEC
		6.5 MIN	5.9 MIN	3.4 MIN	-	-	-	-	-	6.5 MIN
23 APRIL 1965	NRX-A3,EPIV	418 SEC	403 SEC	260 SEC	205 SEC	5 SEC	-	4040°R	3 SEC	428 SEC
20 MAY 1965	NRX-A3, EPV	994 SEC	964 SEC	837 SEC	781 SEC	-	-	3838°R	5 SEC	1002 SEC
28 MAY 1965	NRX-A3, EPVI	1726 SEC	665 SEC	15 SEC	-	-	-	3700°R	2 SEC	810 SEC
	NRX-A3, TOTAL	3138 SEC	2032 SEC	1112 SEC	986 SEC	5 SEC	-	-	-	2240 SEC
		52.4 MIN	33.9 MIN	18.6 MIN	16.4 MIN	0.1 MIN	-	-	-	37.4 MIN
3 FEB 1966	NRX/EST,EPIII, RUN 1	580 SEC	82 SEC	-	-	-	-	2576°R	4 SEC	688 SEC
3 FEB 1966	NRX/EST,EPIII, RUN 2	726 SEC	-	-	-	-	-	2070°R	80 SEC	707 SEC
11 FEB 1966	NRX/EST,EPIC, RUN 1	471 SEC	-	-	-	-	-	2014°R	3 SEC	478 SEC
11 FEB 1966	NRX/EST,EPIC, RUN 3	944 SEC	80 SEC	-	-	-	-	2584°R	20 SEC	987 SEC
3 MAR 1966	NRX/EST,EPIII, RUN 2	292 SEC	280 SEC	215 SEC	84 SEC	75 SEC	-	4125°R	5 SEC	295 SEC
3 MAR 1966	NRX/EST,EPIII, RUN 3	839 SEC	802 SEC	504 SEC	-	-	-	3650°R	3 SEC	843 SEC
16 MAR 1966	NRX/EST, EPIV, RUN 2	1050 SEC	986 SEC	913 SEC	821 SEC	110 SEC	-	4075°R	5 SEC	1045 SEC
25 MAR 1966	NRX/EST, EPIVA, RUN 1	918 SEC	882 SEC	825 SEC	820 SEC	816 SEC	-	4075°R	450 SEC	915 SEC
	NRX/EST TOTAL	5820 SEC	3112 SEC	2457 SEC	1726 SEC	1001 SEC	-	-	-	5955 SEC
		97.0 MIN	51.9 MIN	41.0 MIN	28.6 MIN	16.7 MIN	-	-	-	99.2 MIN
8 JUNE 1966	NRX-A5,EPIII	1174 SEC	1018 SEC	980 SEC	899 SEC	473 SEC	-	4115°R	7 SEC	1173 SEC
23 JUNE 1966	NRX-A5, EPIV	989 SEC	962 SEC	893 SEC	878 SEC	873 SEC	-	4200°R	7 SEC	992 SEC
	NRX-A5 TOTAL	2163 SEC	1980 SEC	1873 SEC	1777 SEC	1346 SEC	-	-	-	2165 SEC
		36.0 MIN	33.0 MIN	31.2 MIN	29.6 MIN	22.4 MIN	-	-	-	36.1 MIN
25 JUNE 1965	PHOEBUS 1A,EPIV	732 SEC	724 SEC	660 SEC	655 SEC	651 SEC	-	4260°R	5 SEC	750 SEC
		12.2 MIN	12.1 MIN	11.0 MIN	10.9 MIN	10.8 MIN	-	-	-	12.5 MIN
10 FEB 1967	PHOEBUS 1B,EPIII	242 SEC	153 SEC	-	-	-	-	2700°R	80 SEC	271 SEC
23 FEB 1967	PHOEBUS 1B, EPIV	2379 SEC	2155 SEC	1850 SEC	1260 SEC	400 SEC	-	4125°R	5 SEC	2432 SEC
	PHOEBUS 1B, TOTAL	2621 SEC	2308 SEC	1850 SEC	1260 SEC	400 SEC	-	-	-	2703 SEC
		43.7 MIN	38.5 MIN	30.8 MIN	21.0 MIN	6.7 MIN	-	-	-	45.1 MIN
7 DEC 1967	NRX-A6, EPIII ABORT	82 SEC	-	-	-	-	-	2250°R	6 SEC	82 SEC
15 DEC 1967	NRX-A6, EPIIIA	4254 SEC	3839 SEC	3797 SEC	3775 SEC	3764 SEC	3717 SEC	4330°R	10 SEC	4255 SEC
	NRX-A6 TOTAL	4336 SEC	3839 SEC	3797 SEC	3775 SEC	3764 SEC	3717 SEC	-	-	4337 SEC
		72.3 MIN	64.0 MIN	63.3 MIN	62.9 MIN	62.7 MIN	62.0 MIN	-	-	72.3 MIN

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TIME AT THERMAL POWER

500 MW AND OVER	800 MW AND OVER	1000 MW AND OVER	1100 MW AND OVER	1400 MW AND OVER	MAXIMUM POWER	TIME AT MAXIMUM POWER	INTEGRATED THERMAL POWER	EQUIVALENT RATED POWER RUN TIME	REACTIVITY LOSS
100 SEC	70 SEC	2 SEC	-	-	1020 MW	2 SEC	1.04x10 ⁶ W-SEC	1.4 MIN	-
1.7 MIN	1.2 MIN	-	-	-	-	-	-	-	-
607 SEC	528 SEC	-	-	-	937 MW	10 SEC	6.30x10 ⁶ W-SEC	10.0 MIN	-
218 SEC	179 SEC	-	-	-	937 MW	95 SEC	1.92x10 ⁷ W-SEC	3.0 MIN	-
825 SEC	707 SEC	-	-	-	-	-	8.22x10 ⁶ W-SEC	13.0 MIN	-
13.8 MIN	11.8 MIN	-	-	-	-	-	-	-	-
372 SEC	293 SEC	160 SEC	32 SEC	-	1100 MW	32 SEC	3.81x10 ⁶ W-SEC	5.6 MIN	19.2 %
6.2 MIN	4.9 MIN	2.7 SEC	0.5 MIN	-	-	-	-	-	-
408 SEC	267 SEC	219 SEC	214 SEC	-	1165 MW	2 SEC	3.52x10 ⁶ W-SEC	5.2 MIN	22.3 %
949 SEC	834 SEC	790 SEC	10 SEC	-	1122 MW	2 SEC	9.62x10 ⁶ W-SEC	14.3 MIN	48.2 %
95 SEC	-	-	-	-	500 MW	95 SEC	4.87x10 ⁶ W-SEC	7.2 MIN	23.1 %
1452 SEC	1101 SEC	1009 SEC	224 SEC	-	-	-	18.01x10 ⁶ W-SEC	26.7 MIN	93.6 %
24.2 MIN	18.3 MIN	16.8 MIN	3.7 MIN	-	-	-	-	-	BASED ON WGT LOSS ANALYSIS
-	-	-	-	-	442 MW	2 SEC	1.85x10 ⁶ W-SEC	2.7 MIN	6.8 % (TOTAL FOR EP-118)
-	-	-	-	-	241 MW	5 SEC	1.63x10 ⁶ W-SEC	2.4 MIN	
-	-	-	-	-	249 MW	3 SEC	1.47x10 ⁶ W-SEC	2.2 MIN	2.5 % (TOTAL FOR EP-119)
-	-	-	-	-	368 MW	4 SEC	3.84x10 ⁶ W-SEC	5.7 MIN	
246 SEC	215 SEC	89 SEC	15 SEC	-	1130 MW	2 SEC	2.59x10 ⁶ W-SEC	3.8 MIN	2.3 % (TOTAL FOR EP-118)
587 SEC	148 SEC	54 SEC	-	-	1055 MW	42 SEC	5.89x10 ⁶ W-SEC	8.7 MIN	
892 SEC	826 SEC	459 SEC	13 SEC	-	1150 MW	2 SEC	9.66x10 ⁶ W-SEC	14.2 MIN	48.7 %
863 SEC	838 SEC	822 SEC	802 SEC	-	1170 MW	3 SEC	9.83x10 ⁶ W-SEC	14.5 MIN	252.4 %
2588 SEC	2027 SEC	1424 SEC	830 SEC	-	-	-	36.8x10 ⁶ W-SEC	54.2 MIN	312.7 %
43.1 MIN	33.8 MIN	23.7 MIN	13.8 MIN	-	-	-	-	-	(300.8 % BASED ON DRUM ANGLE FROM 99.2° TO 147.5°)
1008 SEC	979 SEC	965 SEC	17 SEC	-	1115 MW	2 SEC	12.0x10 ⁶ W-SEC	17.7 MIN	22.5 %
958 SEC	922 SEC	880 SEC	563 SEC	-	1140 MW	3 SEC	10.0x10 ⁶ W-SEC	14.7 MIN	213.3 %
1966 SEC	1901 SEC	1845 SEC	580 SEC	-	-	-	22.0x10 ⁶ W-SEC	32.4 MIN	235.8 %
32.8 MIN	31.7 MIN	30.8 MIN	9.7 MIN	-	-	-	-	-	-
725 SEC	661 SEC	653 SEC	-	-	1090 MW	8 SEC	7.2x10 ⁶ W-SEC	10.8 MIN	-
12.1 MIN	11.0 MIN	10.9 MIN	-	-	-	-	-	-	-
174 SEC	-	-	-	-	650 MW	60 SEC	1.5x10 ⁶ W-SEC	1.7 MIN	-
2265 SEC	2019 SEC	1912 SEC	1862 SEC	990 SEC	1500 MW	45 SEC	29.5x10 ⁶ W-SEC	32.8 MIN	-
39 SEC	2019 SEC	1912 SEC	1862 SEC	990 SEC	-	-	31.0x10 ⁶ W-SEC	34.5 MIN	-
6 MIN	33.6 MIN	31.9 MIN	31.0 MIN	16.5 MIN	-	-	-	-	-
-	-	-	-	-	330 MW	2 SEC	0.3x10 ⁶ W-SEC	0.4 MIN	-
3825 SEC	3787 SEC	3766 SEC	3623 SEC	-	1180 MW	10 SEC	44.6x10 ⁶ W-SEC	66.4 MIN	70 %
3825 SEC	3787 SEC	3766 SEC	3623 SEC	-	-	-	44.8x10 ⁶ W-SEC	66.8 MIN	70 %
52.3 MIN	52.1 MIN	52.8 MIN	50.4 MIN	-	-	-	-	-	-

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APPENDIX D REACTOR TEST DATA

This section presents actual test data as published in the reference reports. The key parameters which are presented are power, propellant flowrate, fuel exit gas temperature, chamber pressure, and nozzle chamber pressure and/or core exit pressure. In some cases, not all data were available for a given test (and for some runs of a test series no data at all could be located). Also, legibility is poor in certain instances but it is felt that the data should be presented in their "original" form. It is indeed fortunate that these data still exists today, over 30 years since the first nuclear reactor test, and hopefully they will be useful to those individuals working towards development of "second generation" nuclear thermal rocket engines.

Time and resources did not allow narration of the data, but comprehension of the data should be relatively straightforward based on the text in the body of the report as well as the tables contained in Appendix C.

This Appendix is organized as follows:

Page	Contents
D-2	KIWI A TEST DATA
D-5	KIWI A PRIME TEST DATA
D-8	KIWI A3 TEST DATA
D-10	KIWI B1A TEST DATA
D-14	KIWI B1B TEST DATA
D-21	KIWI B4A TEST DATA
D-27	KIWI B4D TEST DATA
D-33	KIWI B4E TEST DATA
D-38	PHOEBUS 1A TEST DATA
D-41	PHOEBUS 1B TEST DATA
D-46	PHOEBUS 2A TEST DATA
D-50	PEWEE 1 TEST DATA
D-52	NUCLEAR FURNACE NF1 TEST DATA
D-55	NRX A2 TEST DATA
D-60	NRX A3 TEST DATA
D-67	NRX/EST TEST DATA
D-90	NRX A5 TEST DATA
D-95	NRX A6 TEST DATA
D-102	XE PRIME TEST DATA

KIWI A
TEST DATA

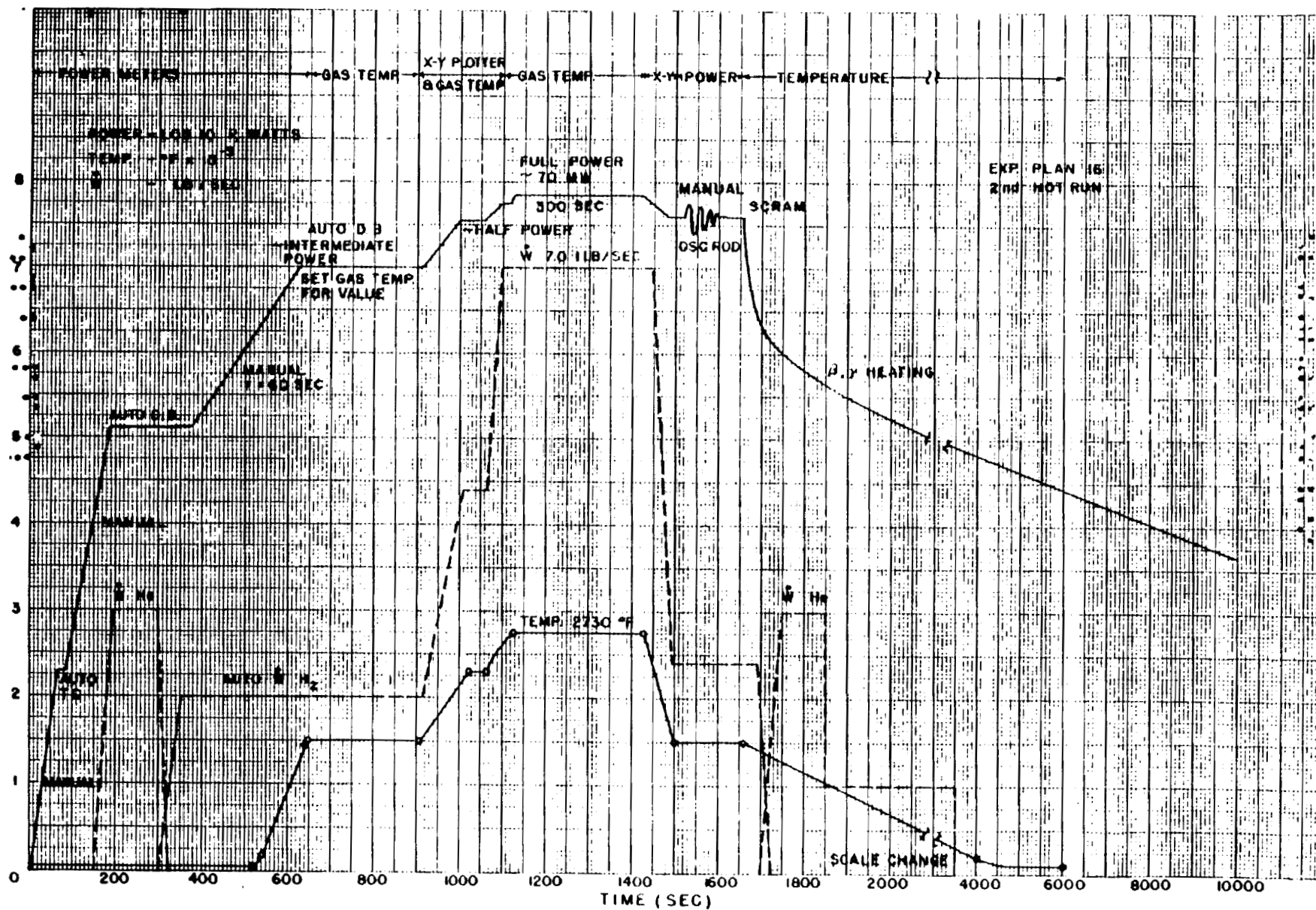


Figure D-1. KIWI A Planned Full Power Run

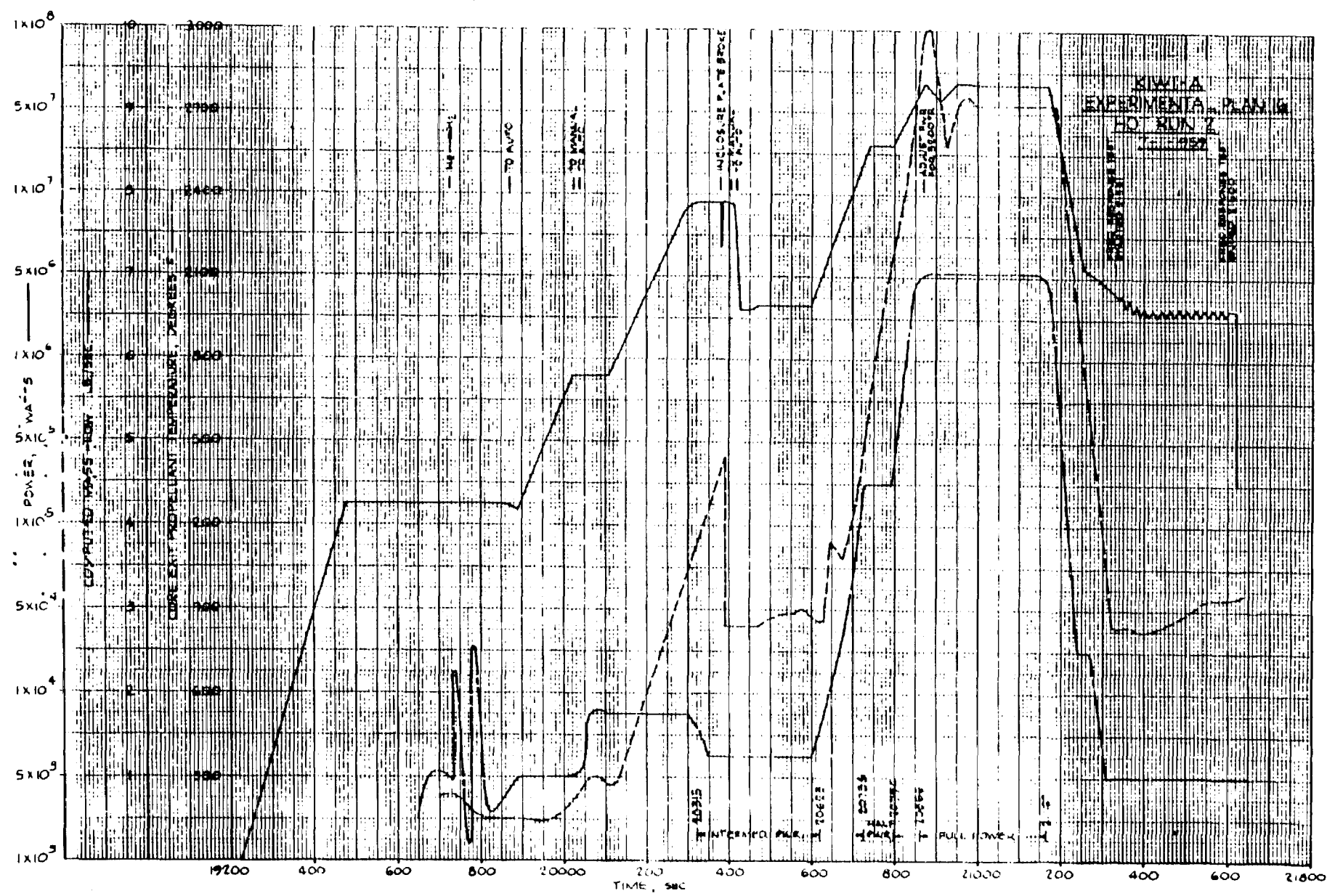


Figure D-2. KIWI A Actual Full Power Run

KIWI A PRIME

TEST DATA

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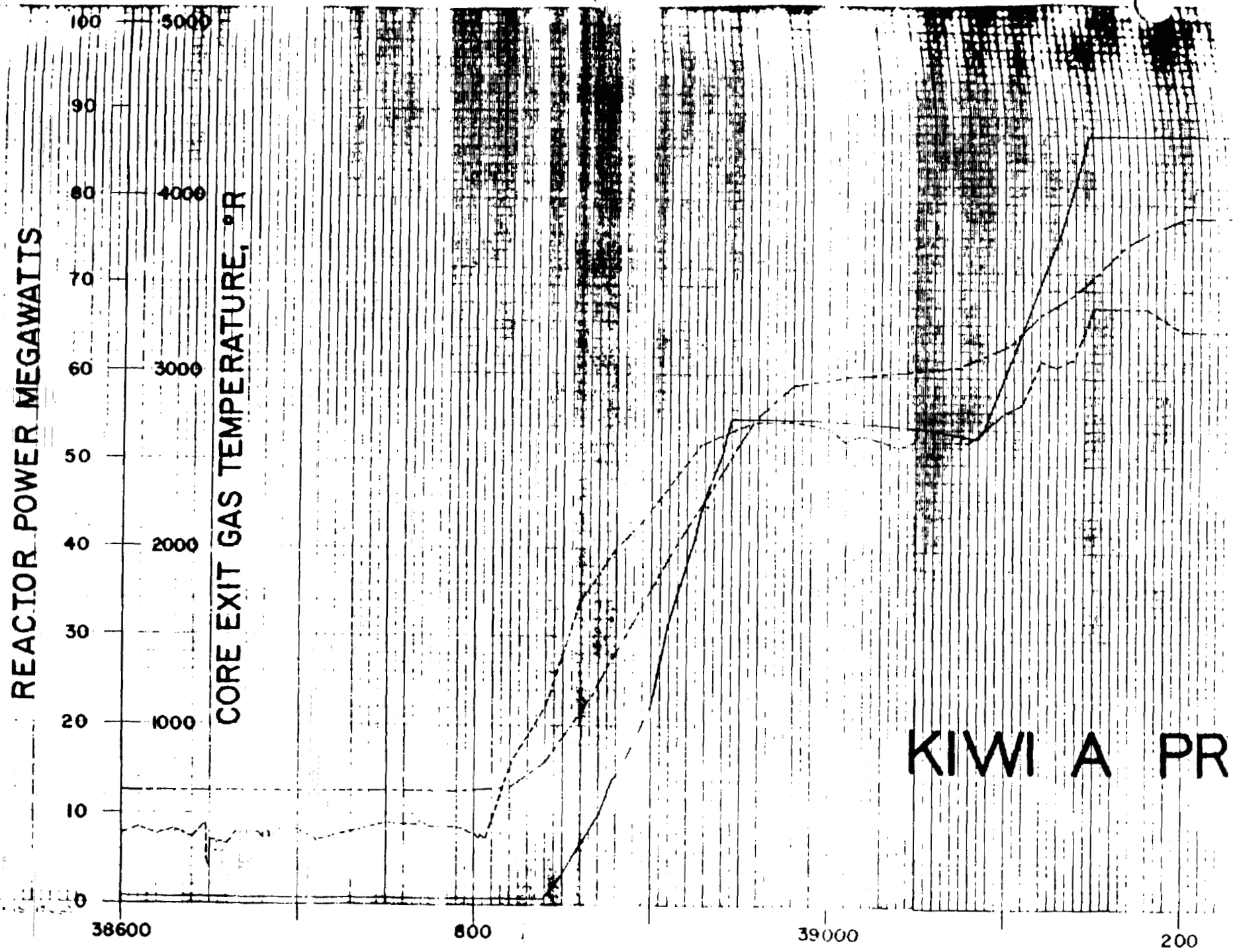


Figure D-3A. KIWI A Prime Full Power Run

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HYDROGEN MASS FLOW RATE LBS/SEC

POWER
TEMPERATURE
MASS FLOW

J-19-NTS-156

TIME FULL POWER RUN

TIME (SEC.)

400

600

800

40000

Figure D-3B. KIWI A Prim Full Power Run

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KIWI A3
TEST DATA

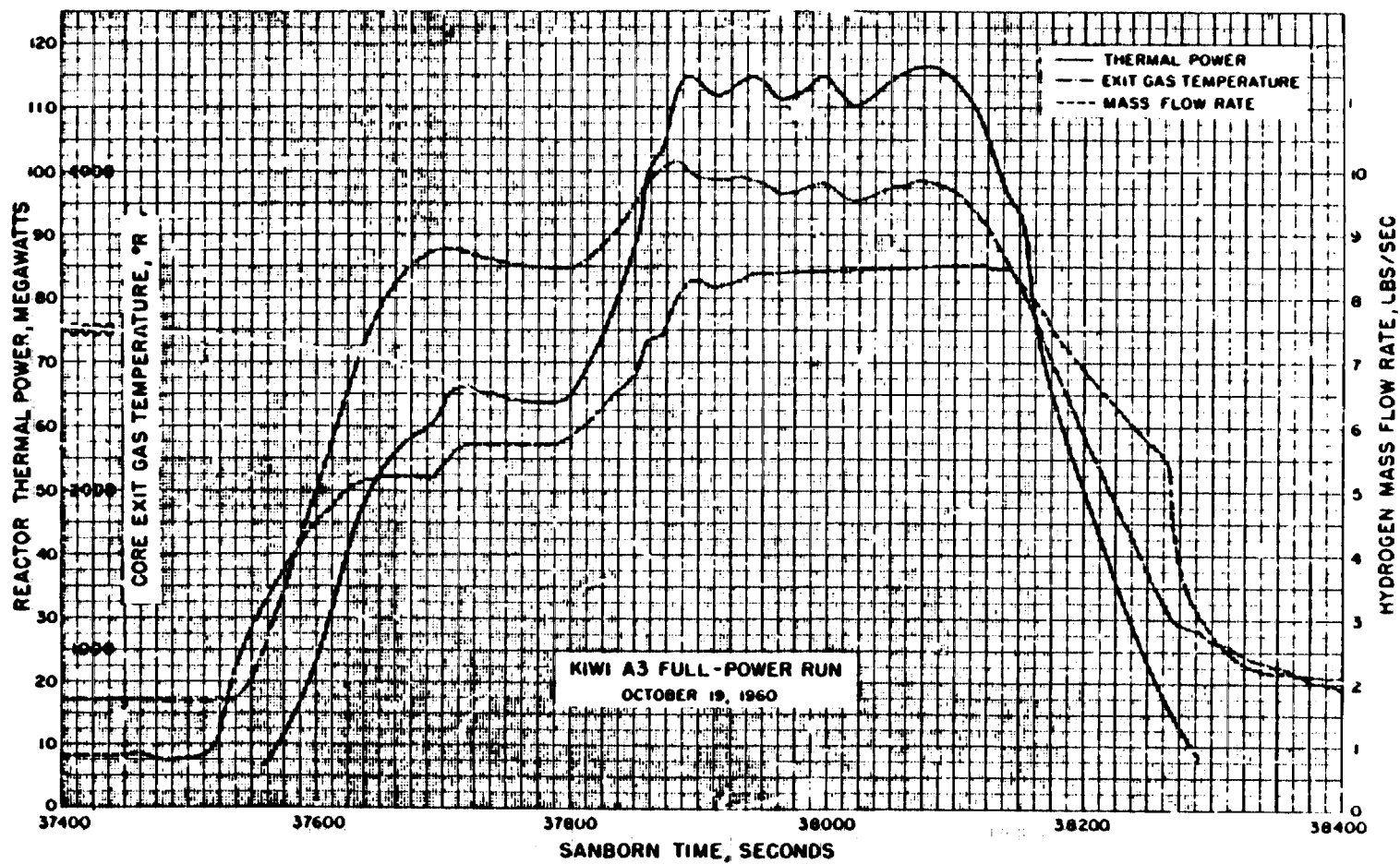
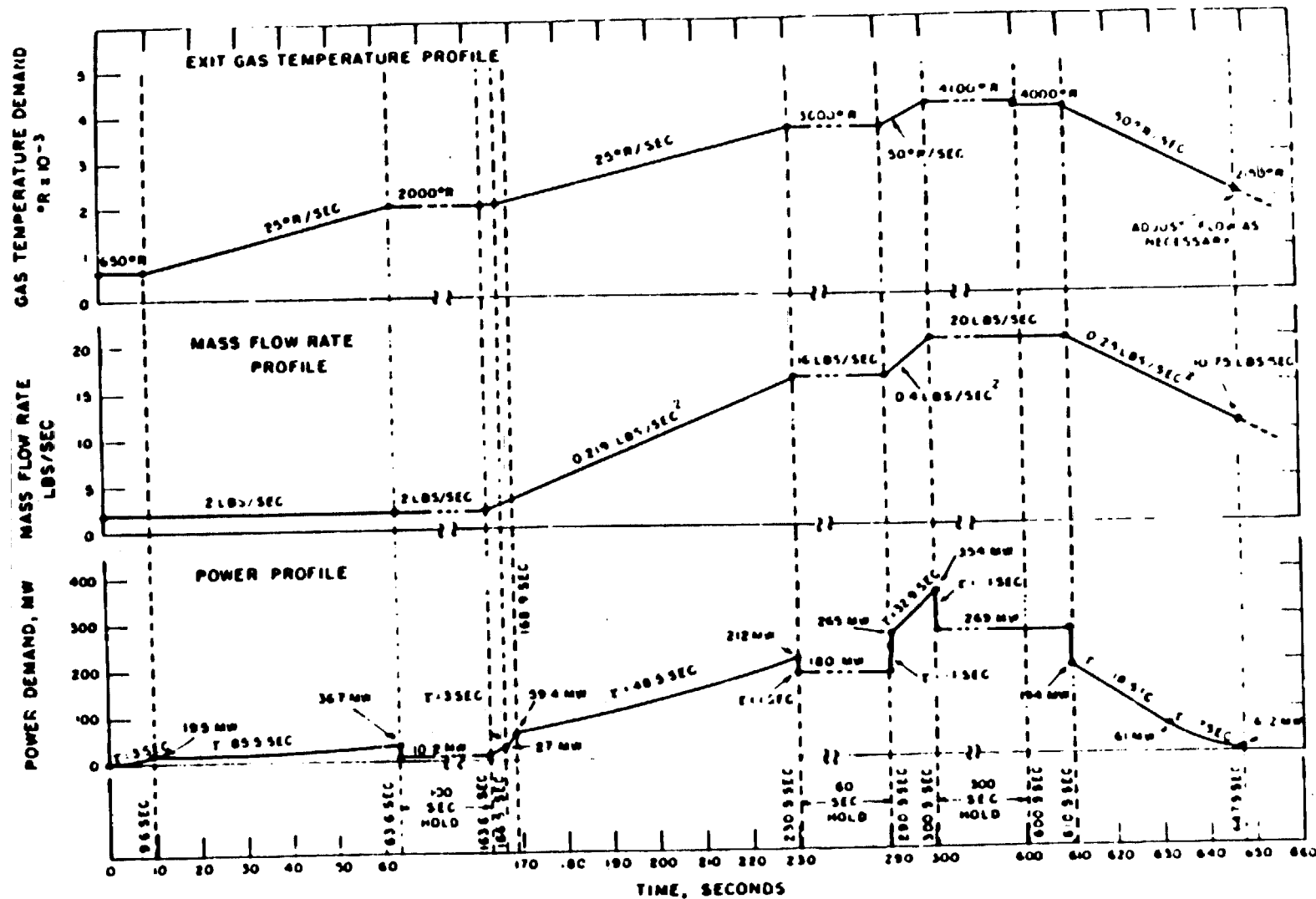


Figure D-4

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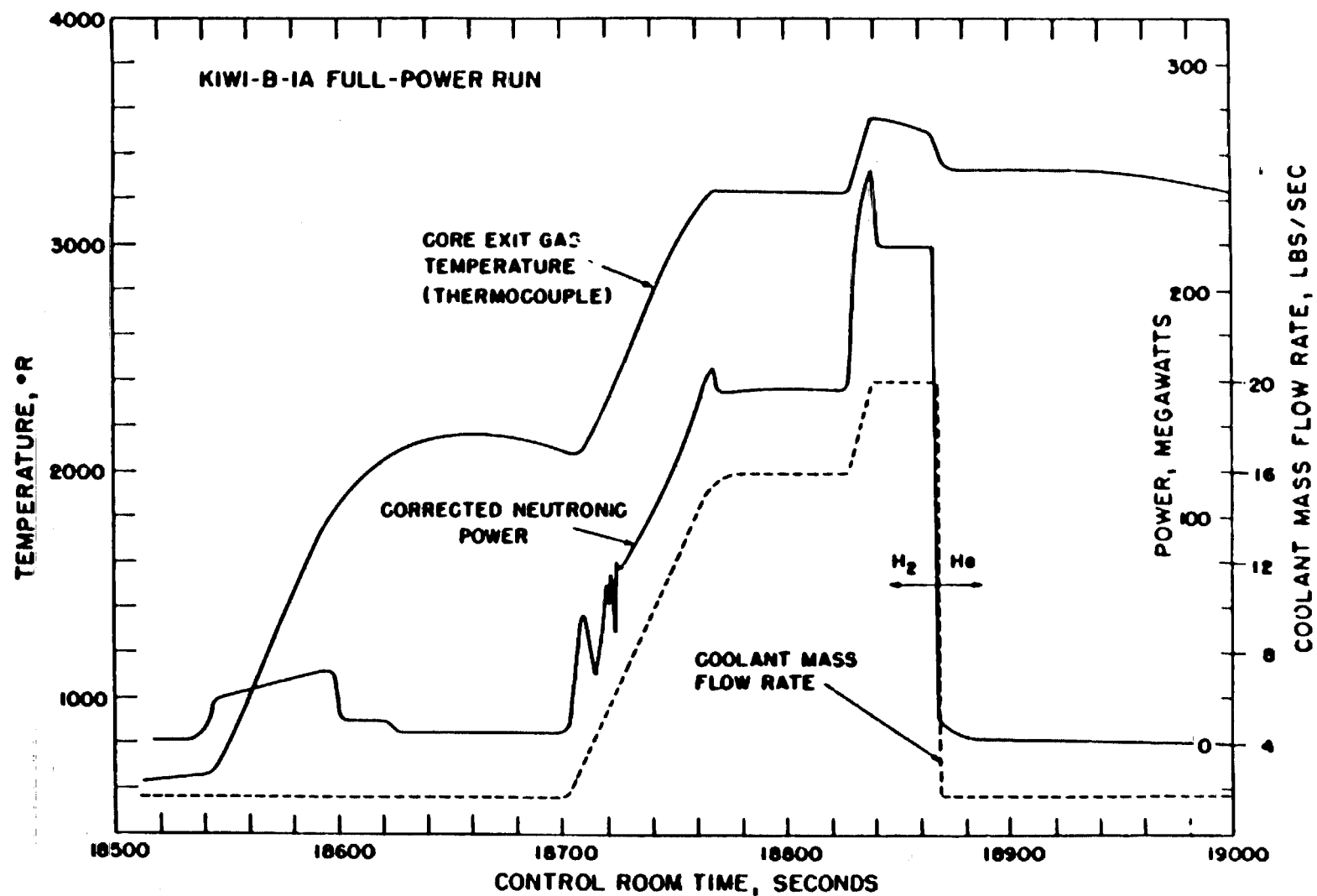
KIWI B1A

TEST DATA



Planned Run Profile for the Full-Power Test of Kiwi-B-1A

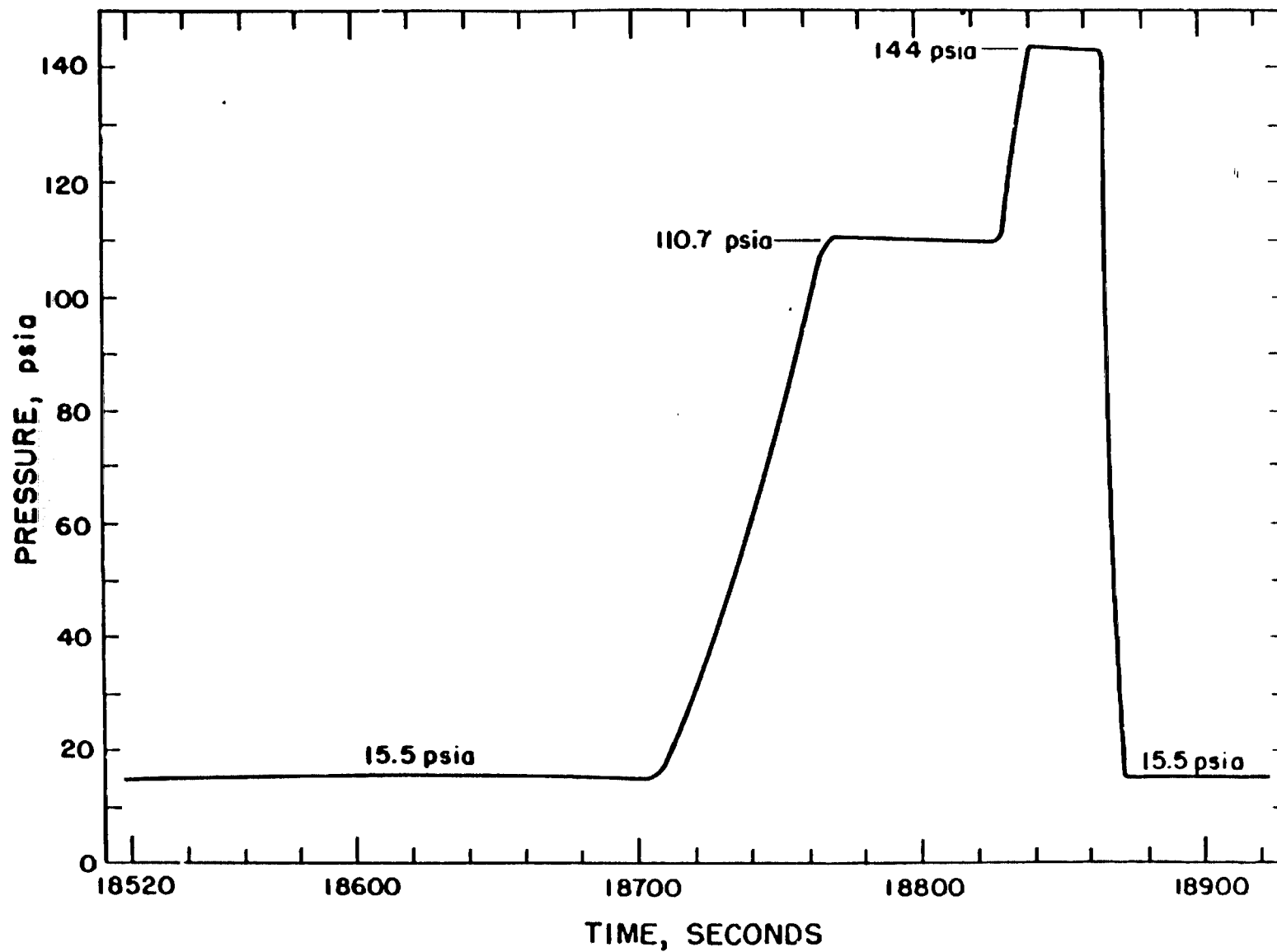
Figure D-5



Power, Exit Gas Temperature, and Mass Flow Rate as Measured for Kiwi-B-1A

Figure D-6

D-13

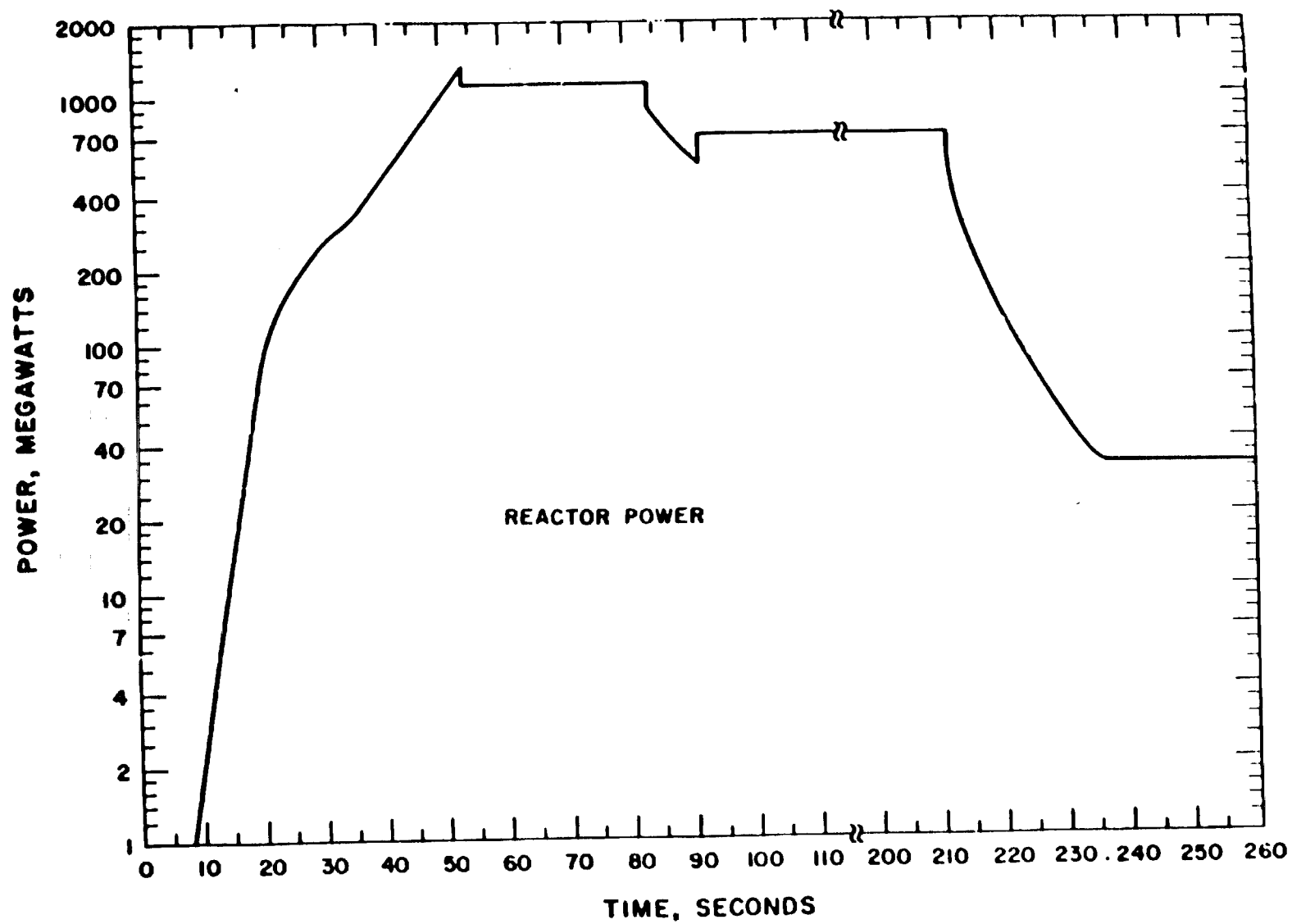


KIWI BIA Time Variation of Core Exit Pressure

Figure D-7

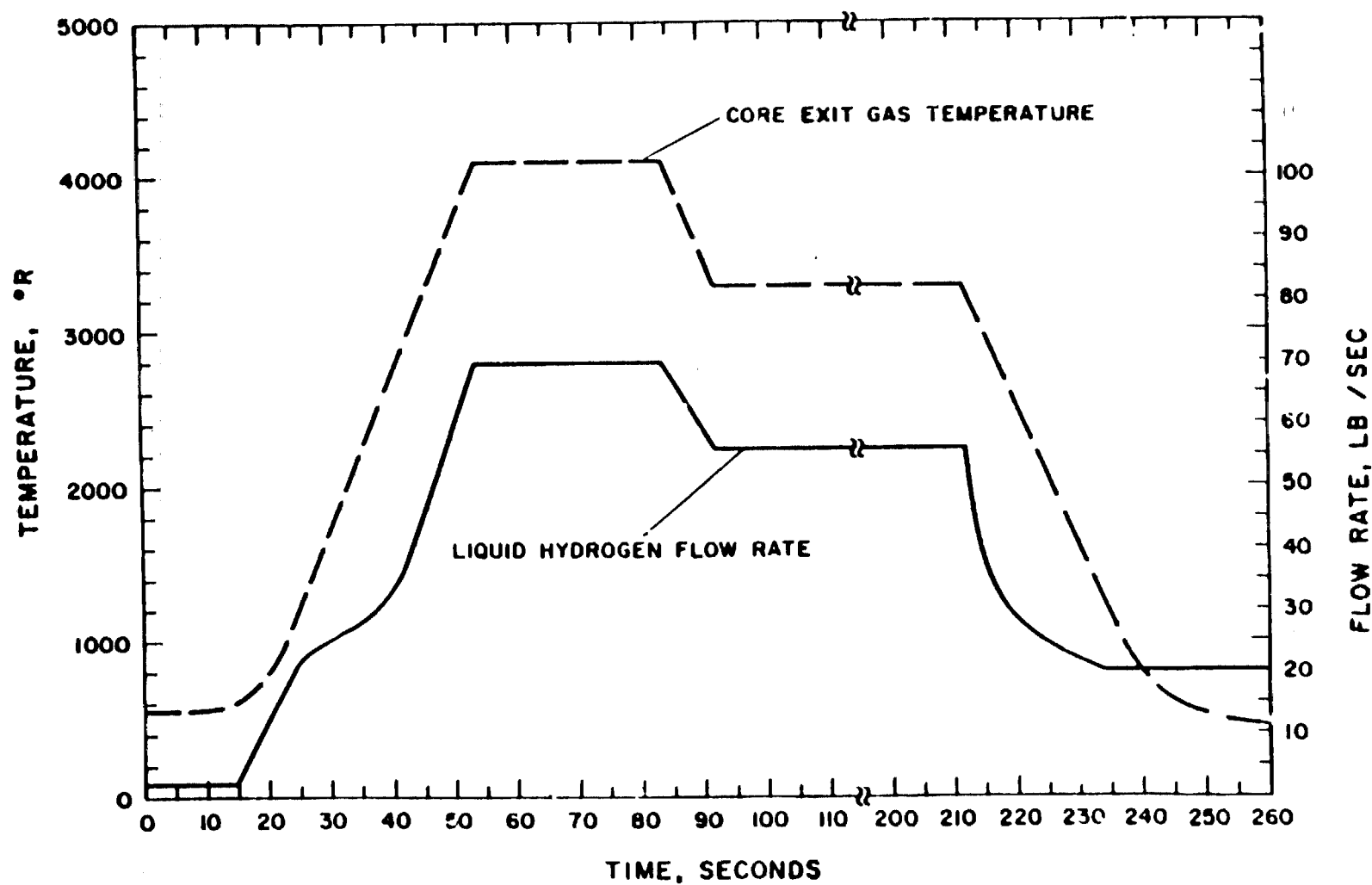
KIWI B1B

TEST DATA



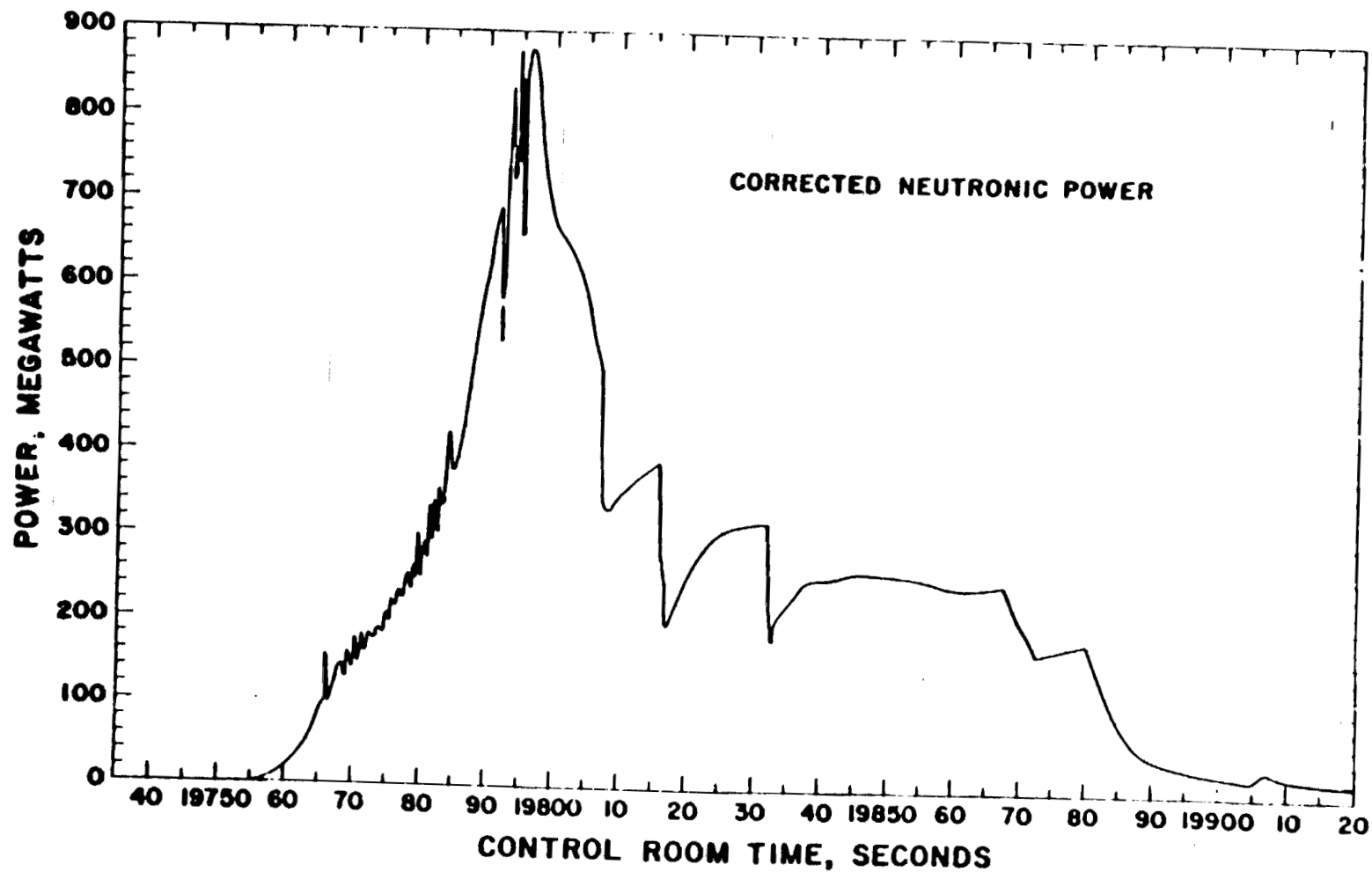
KIWI BIB Planned Run Profile -- Reactor Power.

Figure D-8



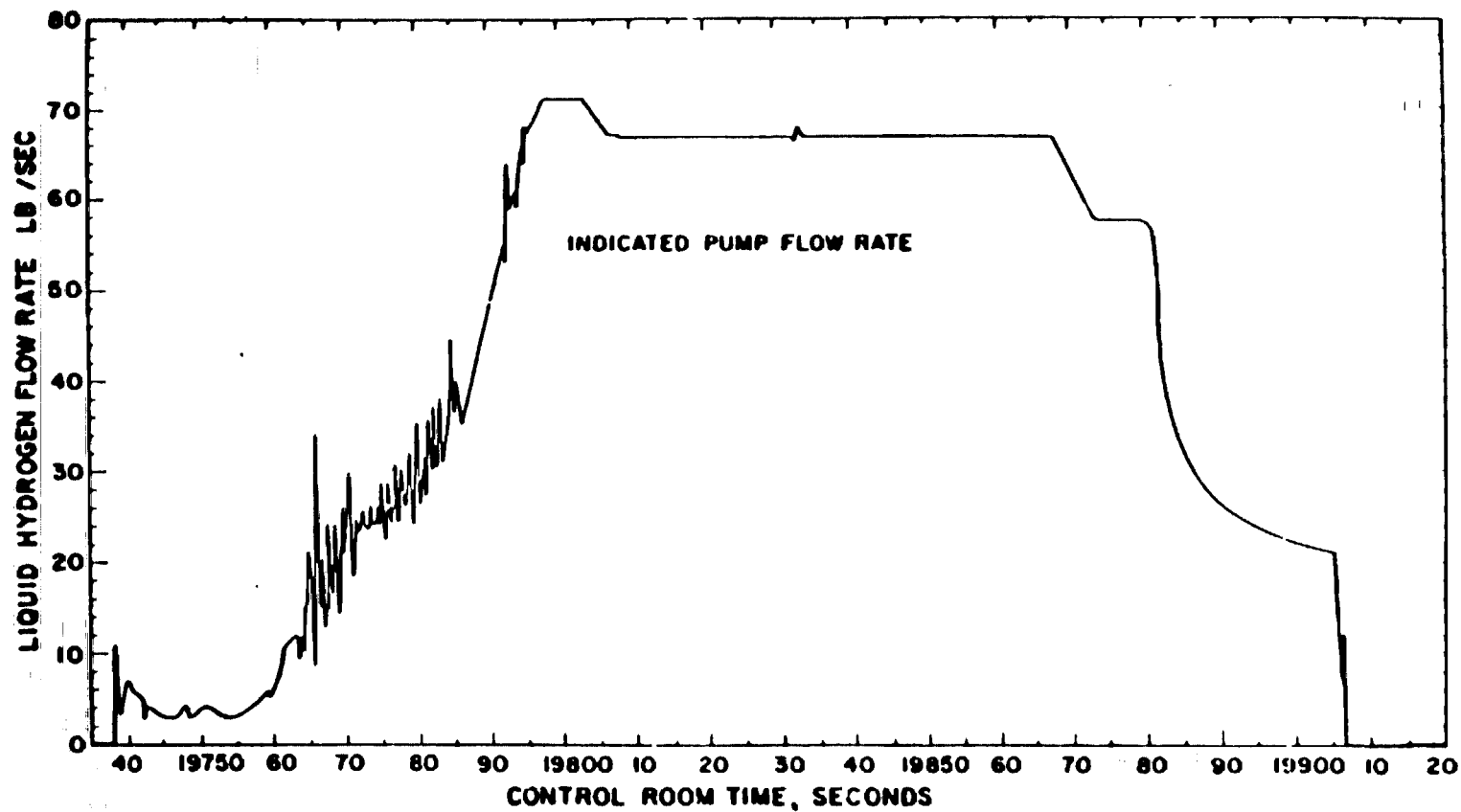
KIWI B1B Planned Run Profile -- Core Exit Gas Temperature and Liquid Hydrogen Flow Rate.

Figure D-9



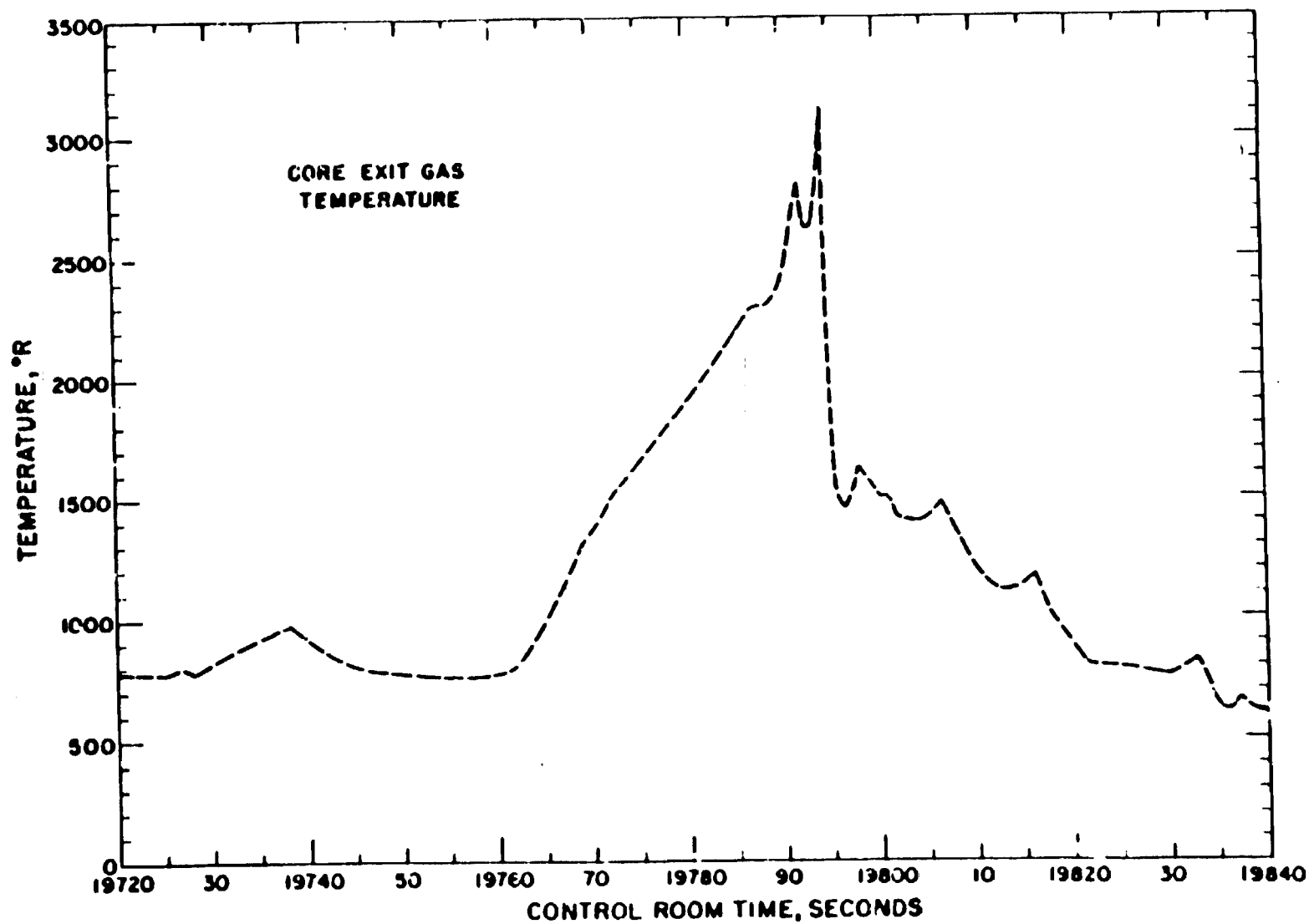
KIWI B1B Full-Power Run -- Corrected Neutronic Power.

Figure D-10



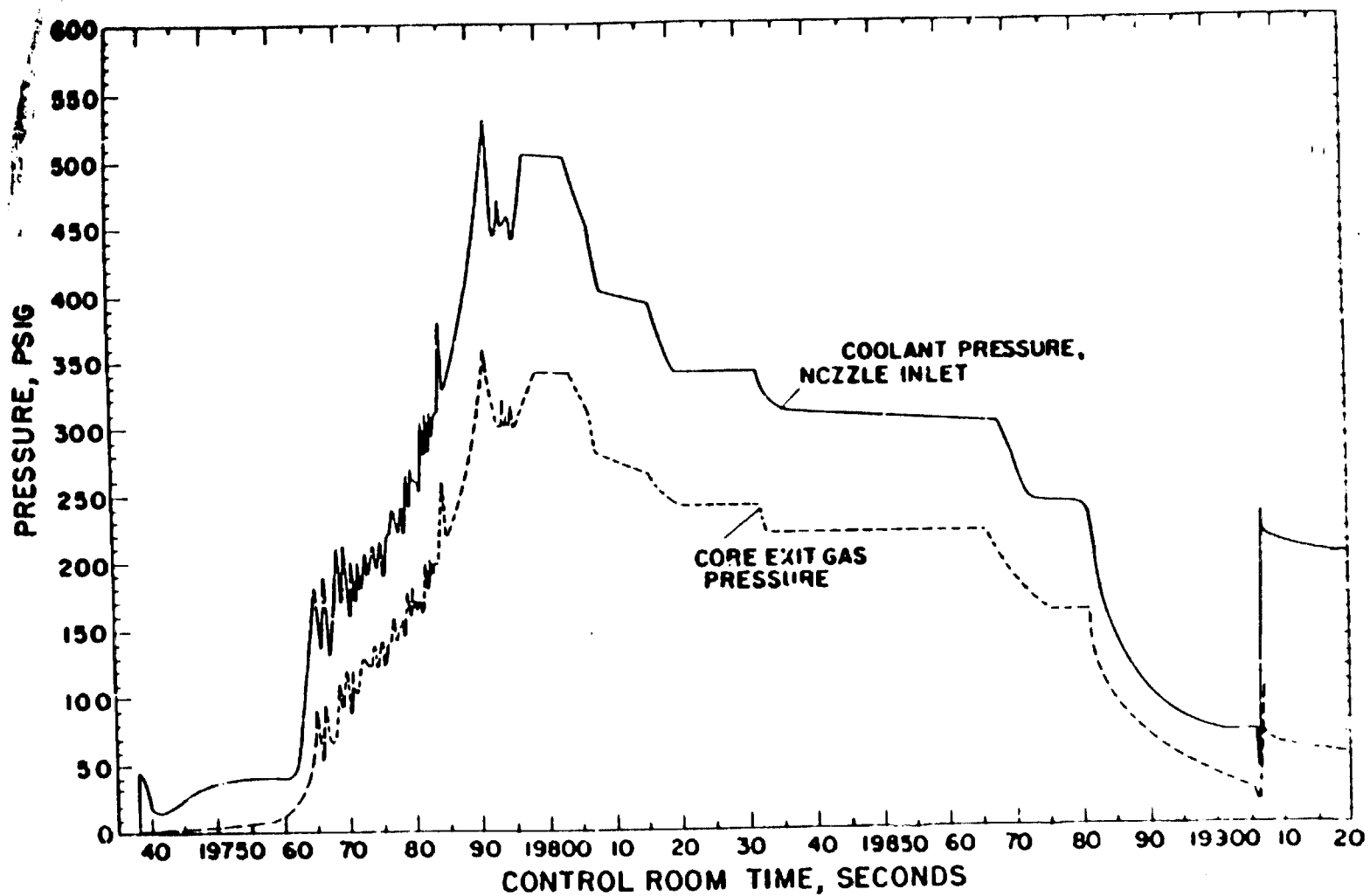
KIWI B1B Full-Power Run -- Indicated Coolant Pump Flow Rate.

Figure D-11



KIWI BIB Full-Power Run -- Measured Core Exit Gas Temperature (Temperatures Beyond 19840 Are Not Valid Due to Excessive Core and Nozzle Bypass Flow).

Figure D-12

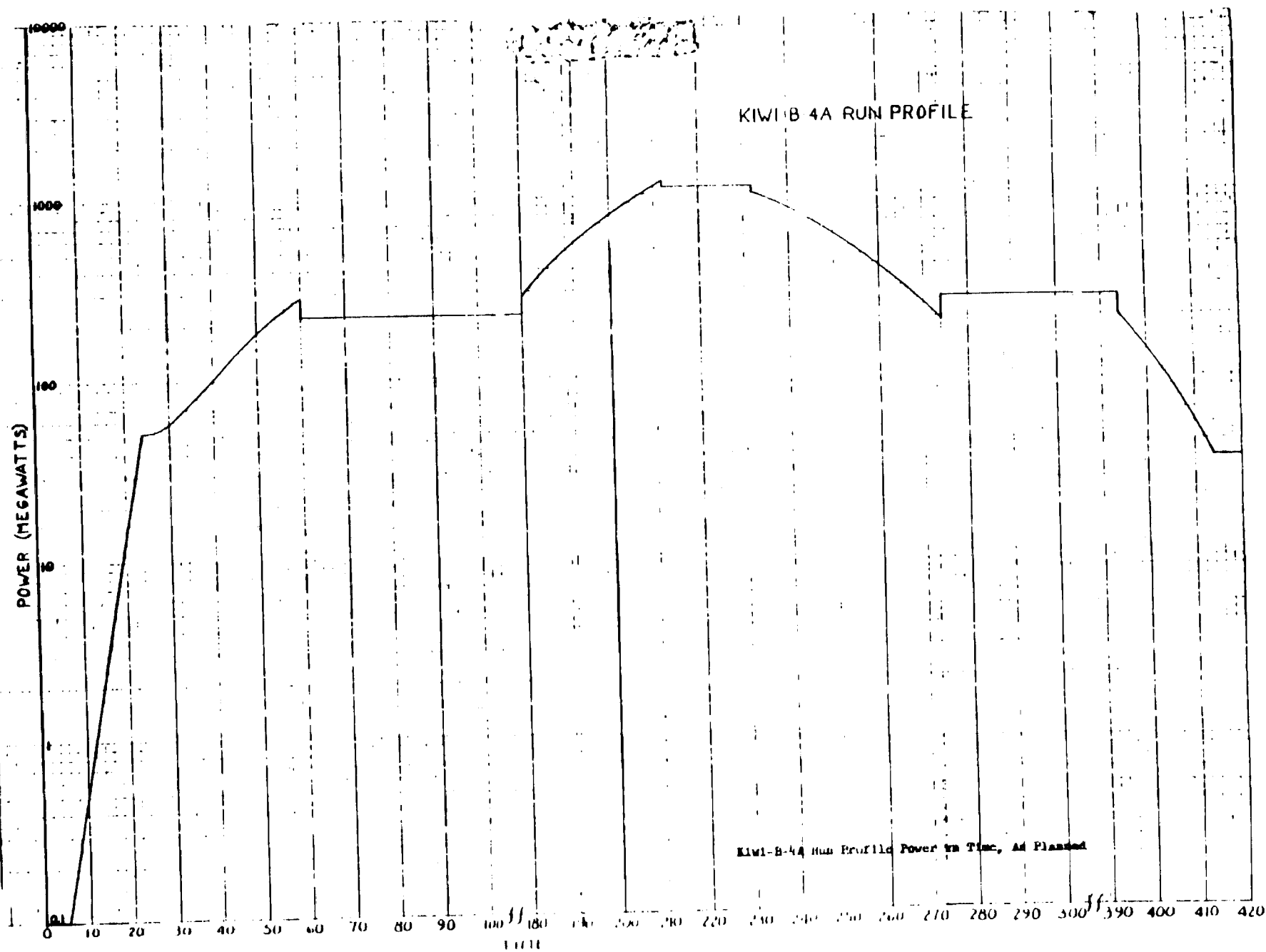


KIWI BIB Full-Power Run -- Nozzle Coolant Inlet and Core Exit Gas Pressures.

Figure D-13

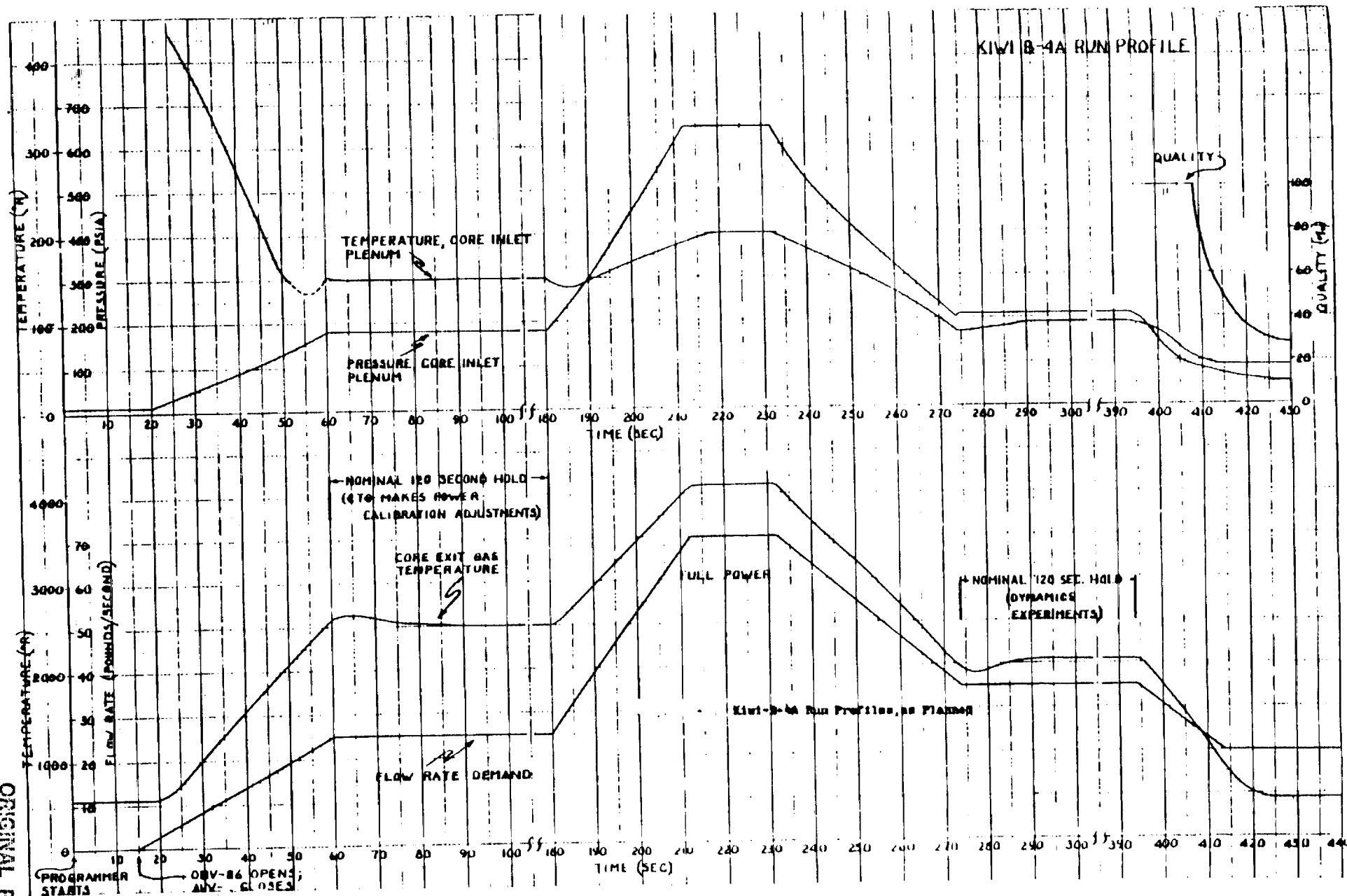
KIWI B4A

TEST DATA



KIWI B4A Run Profile Power vs. Time, As Planned

Figure D-14

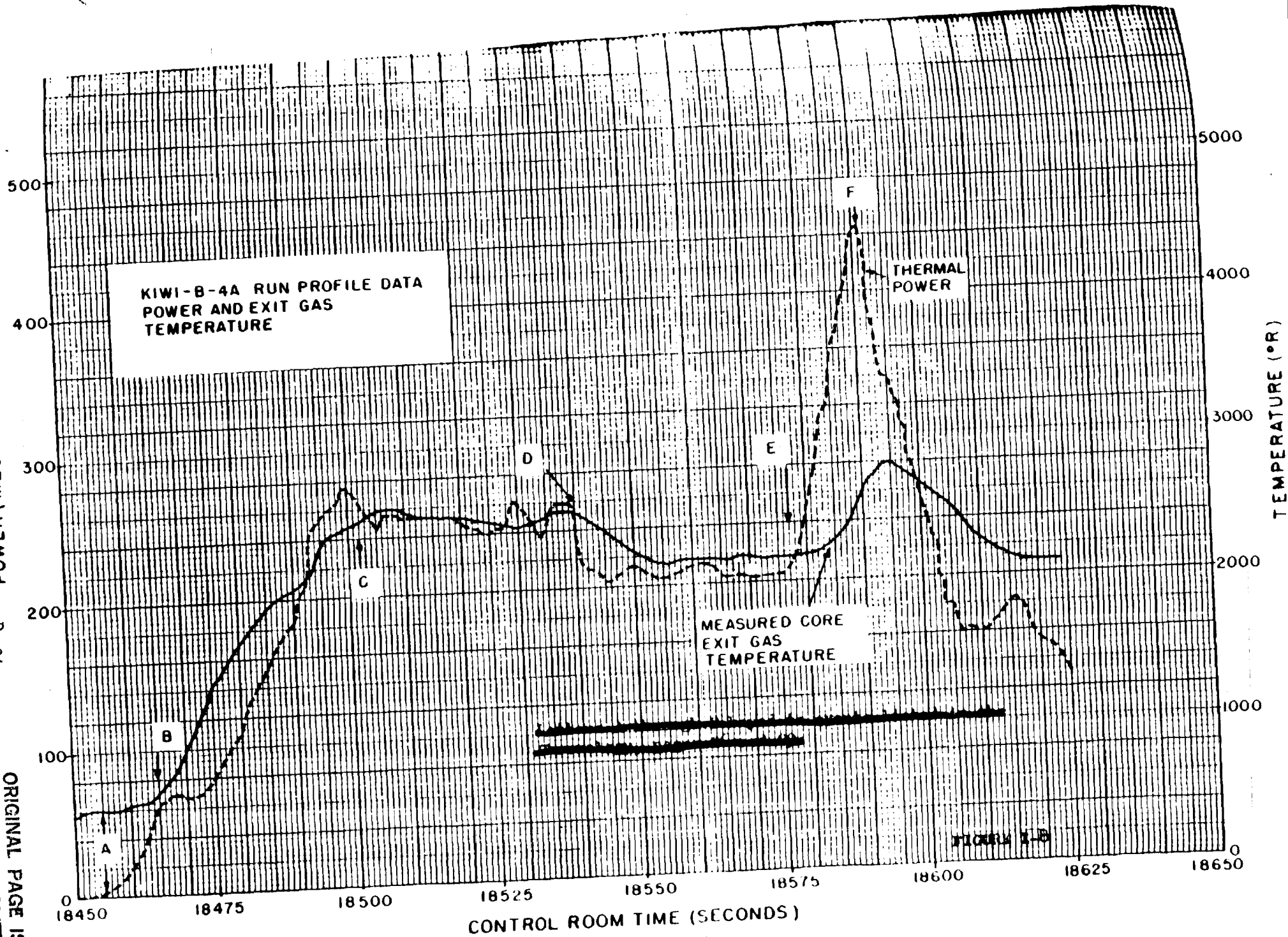


KIWI B4A Run Profiles, As Planned

Figure D-15

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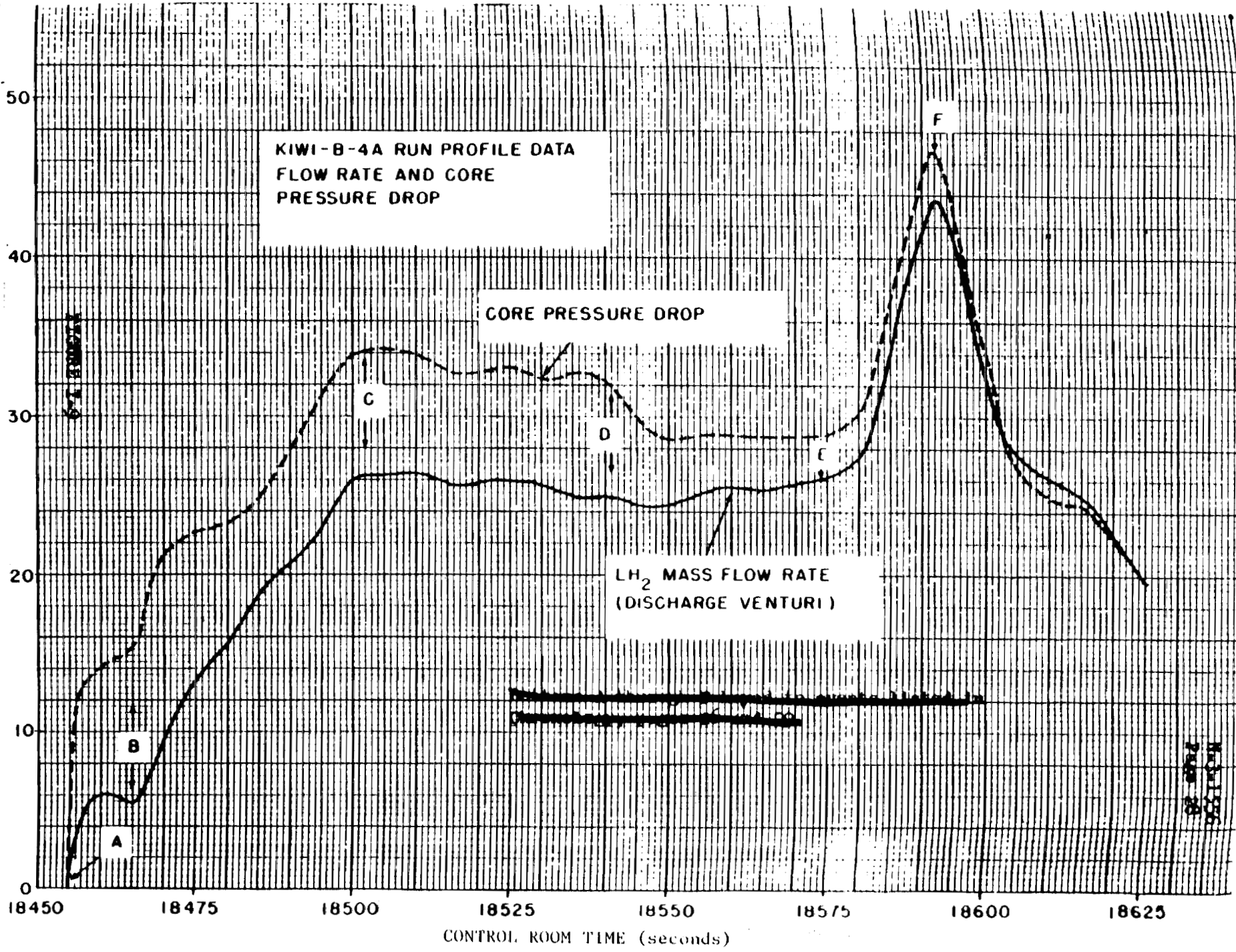
D-24



LH₂ MASS FLOW RATE (LBS/SEC)

D-25

CORE PRESSURE DROP (PSI)



Model 100
Page 25

Fig. D-17

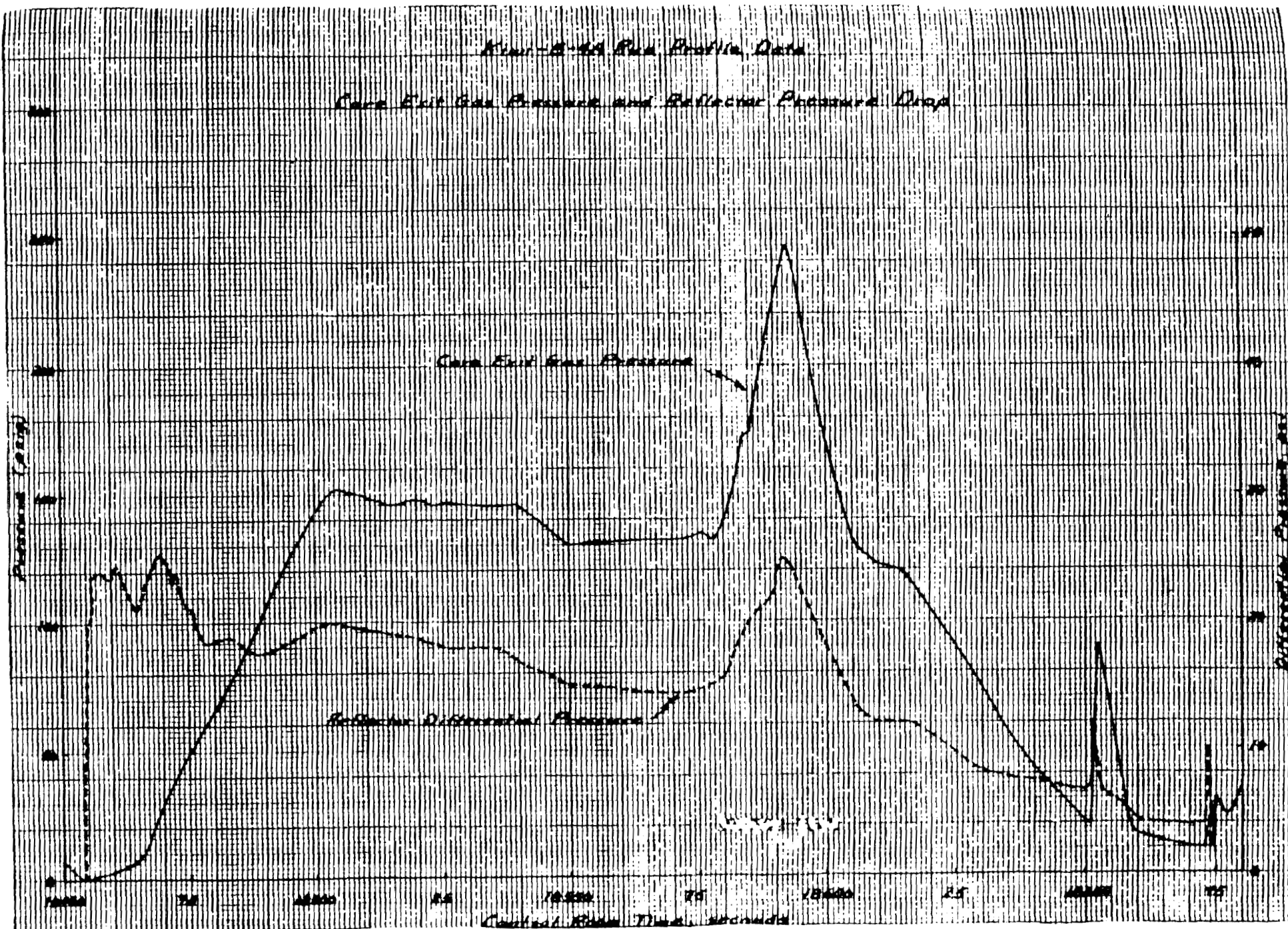
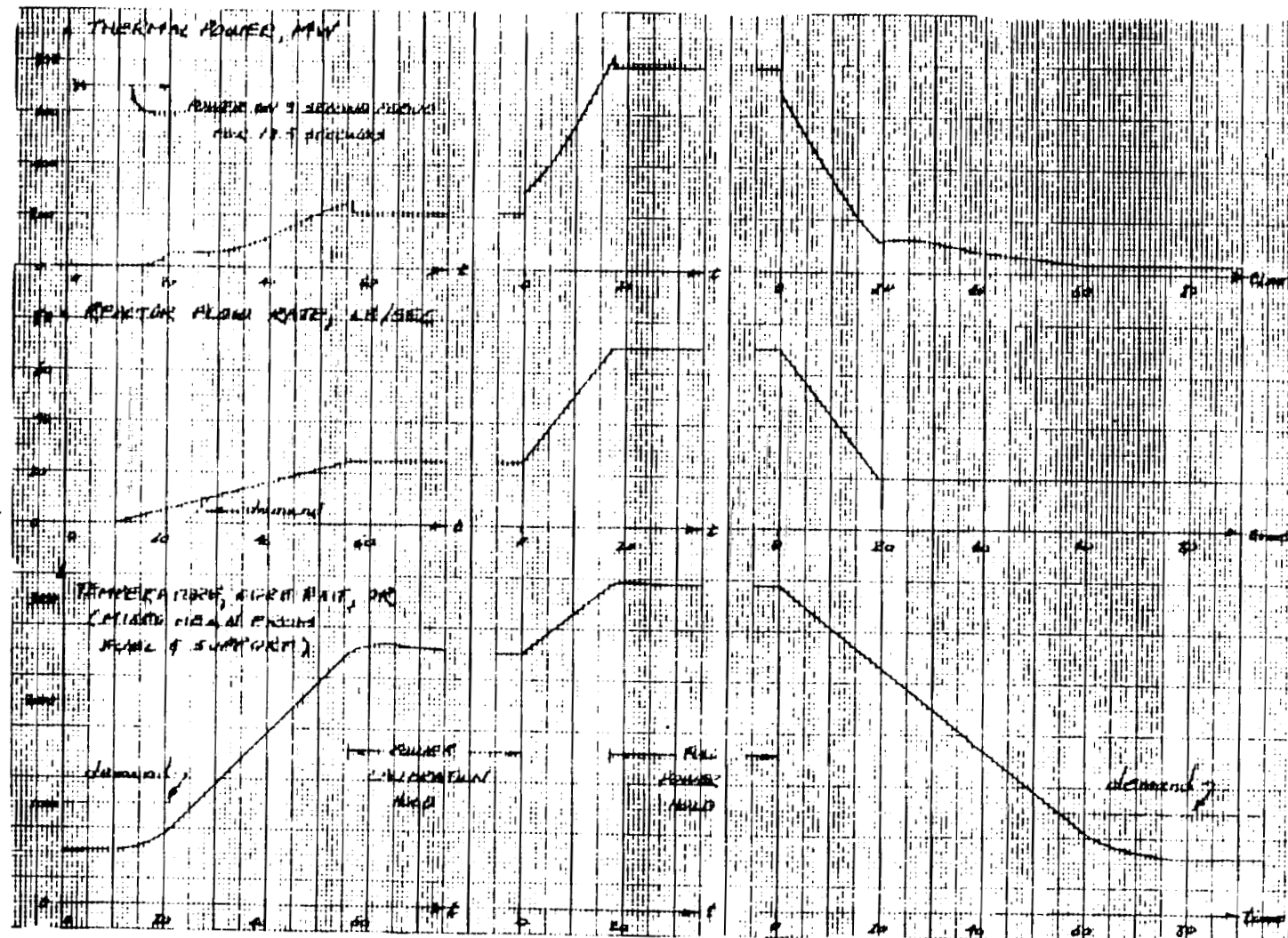


Figure D-18

KIWI B4D

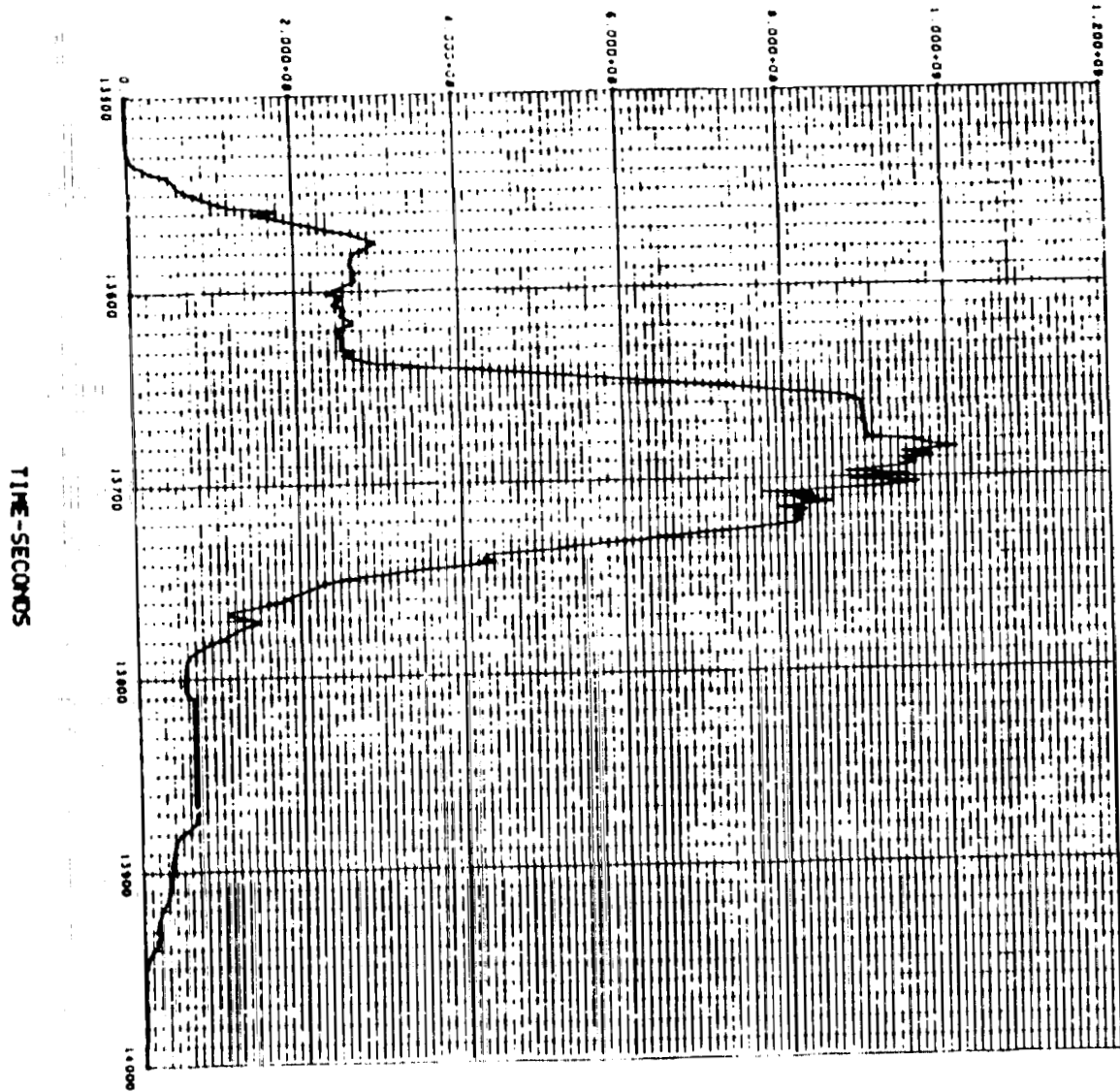
TEST DATA



KIWI B4D Planned Run Profiles

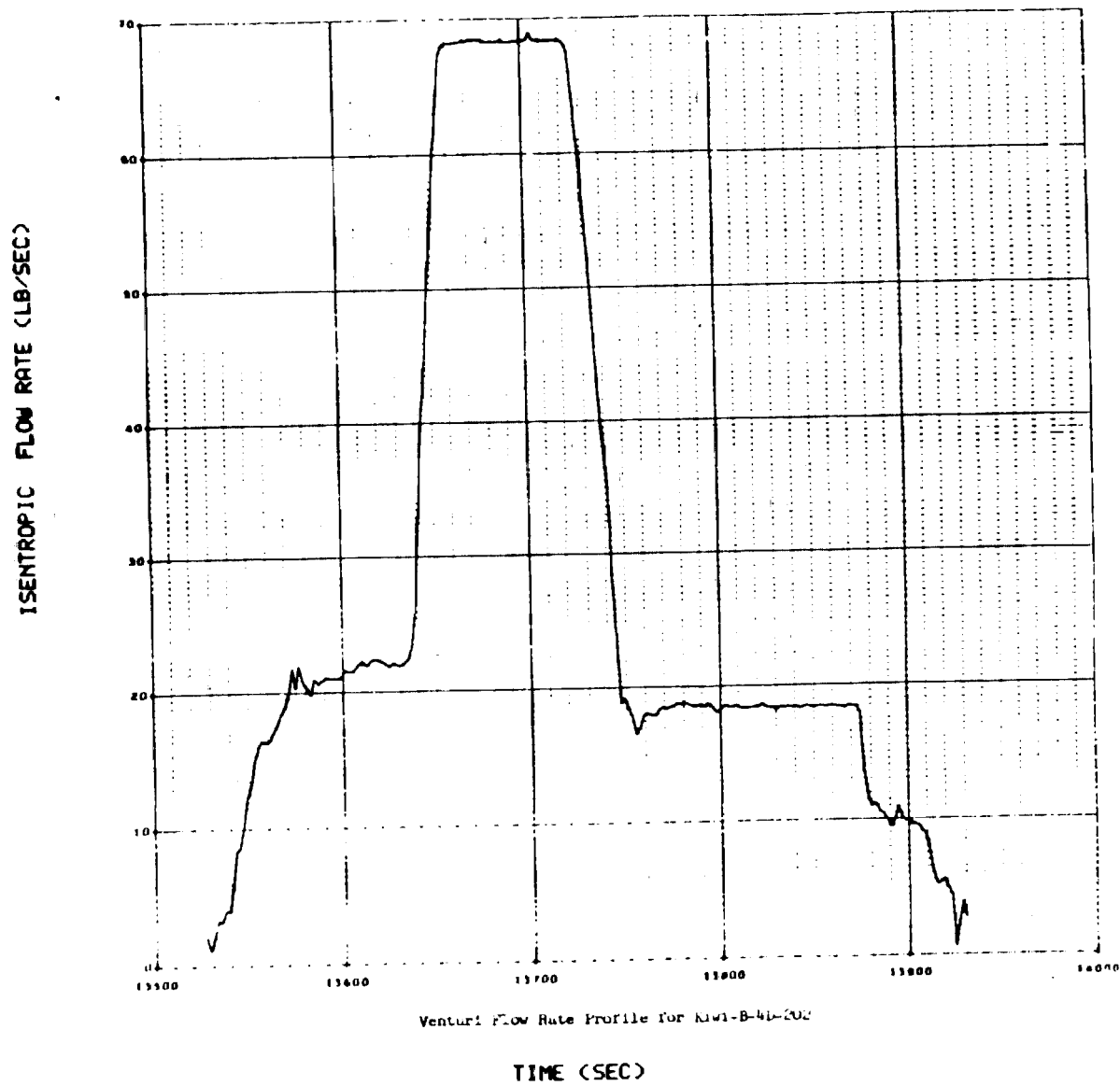
Figure D-19

POWER-WATTS
RADIOCHEMISTRY SCALED LINEAR NO. 2



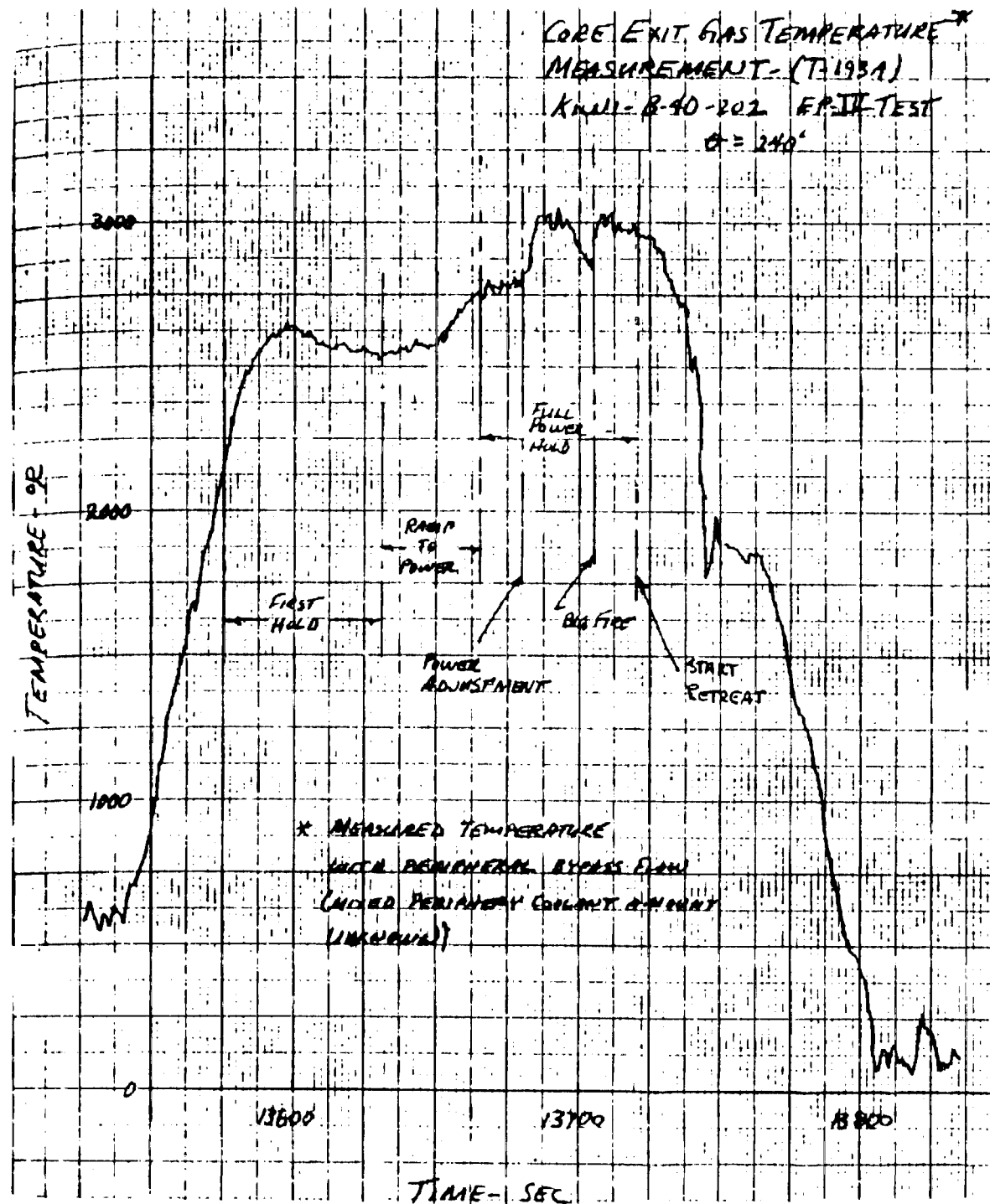
KIWI B4D Time Profile of Power Generation

Figure D-20

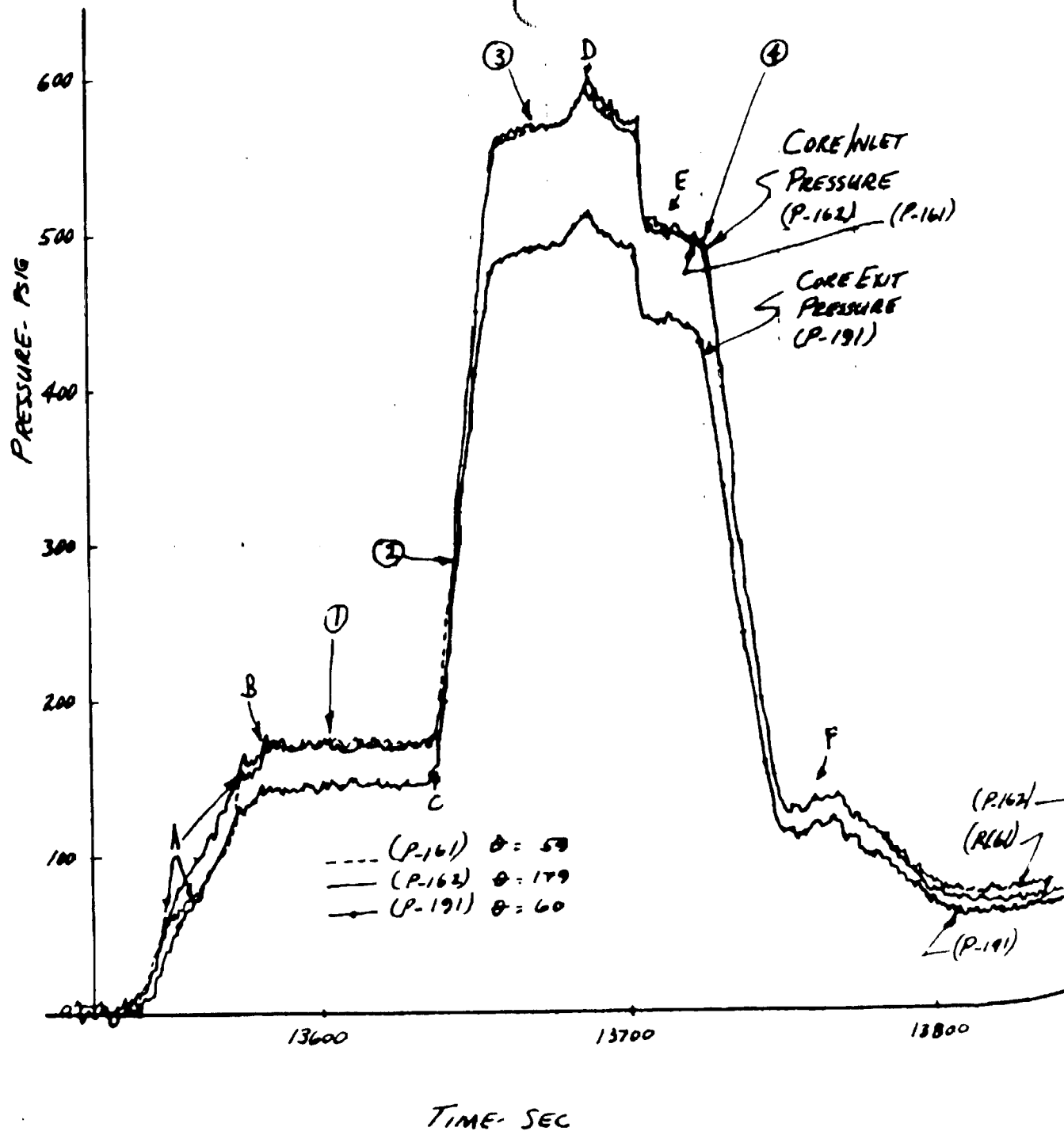


KIWI B4D LH2 Flow Rate Profile

Figure D-21



KIWI B4D Representative Measured Core Exit Gas Temperature Profile

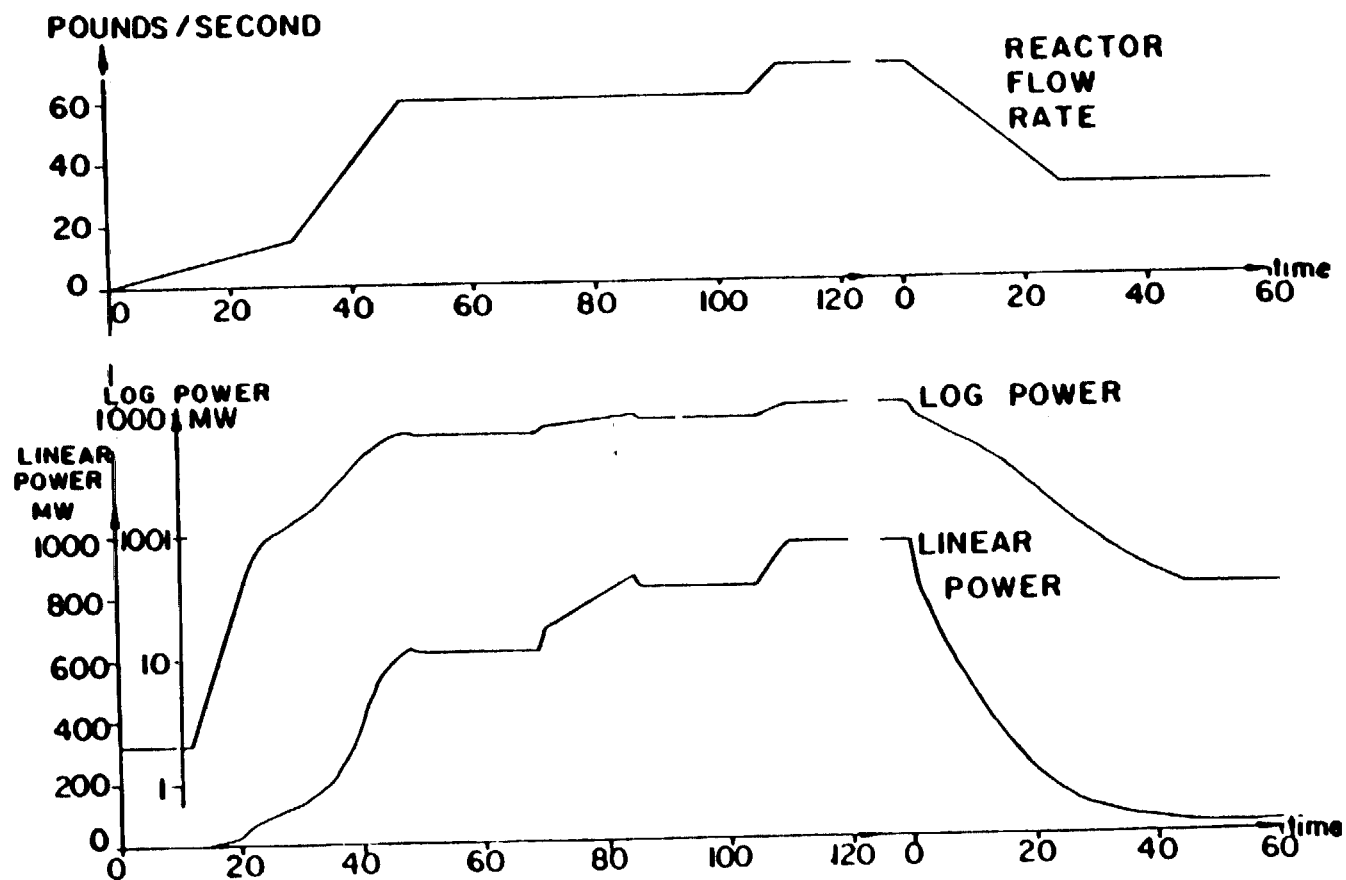


KIWI B4D Core Inlet and Core Exit Pressure Versus Time

KIWI B4E

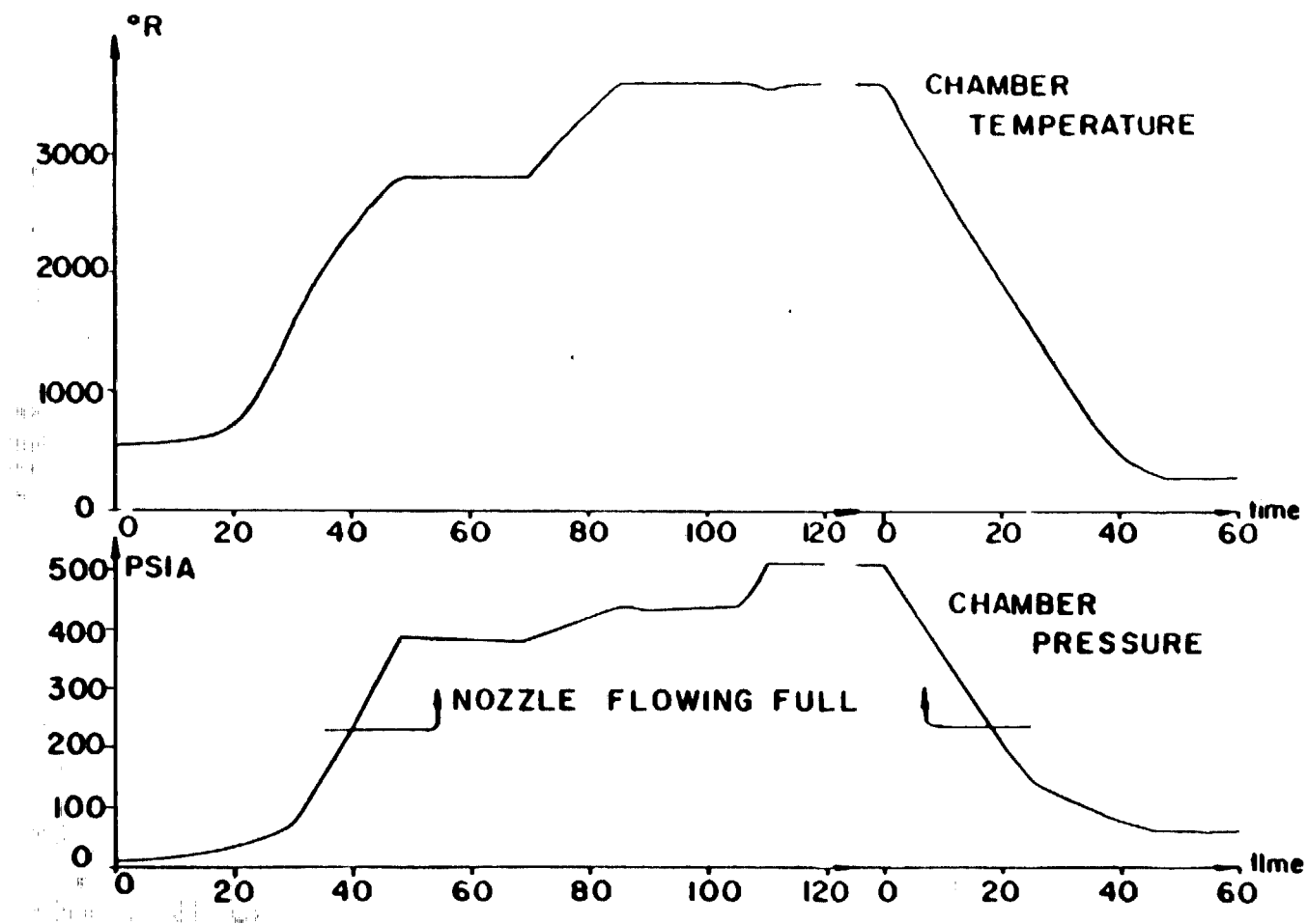
TEST DATA

NOTE: Restart Data Unavailable

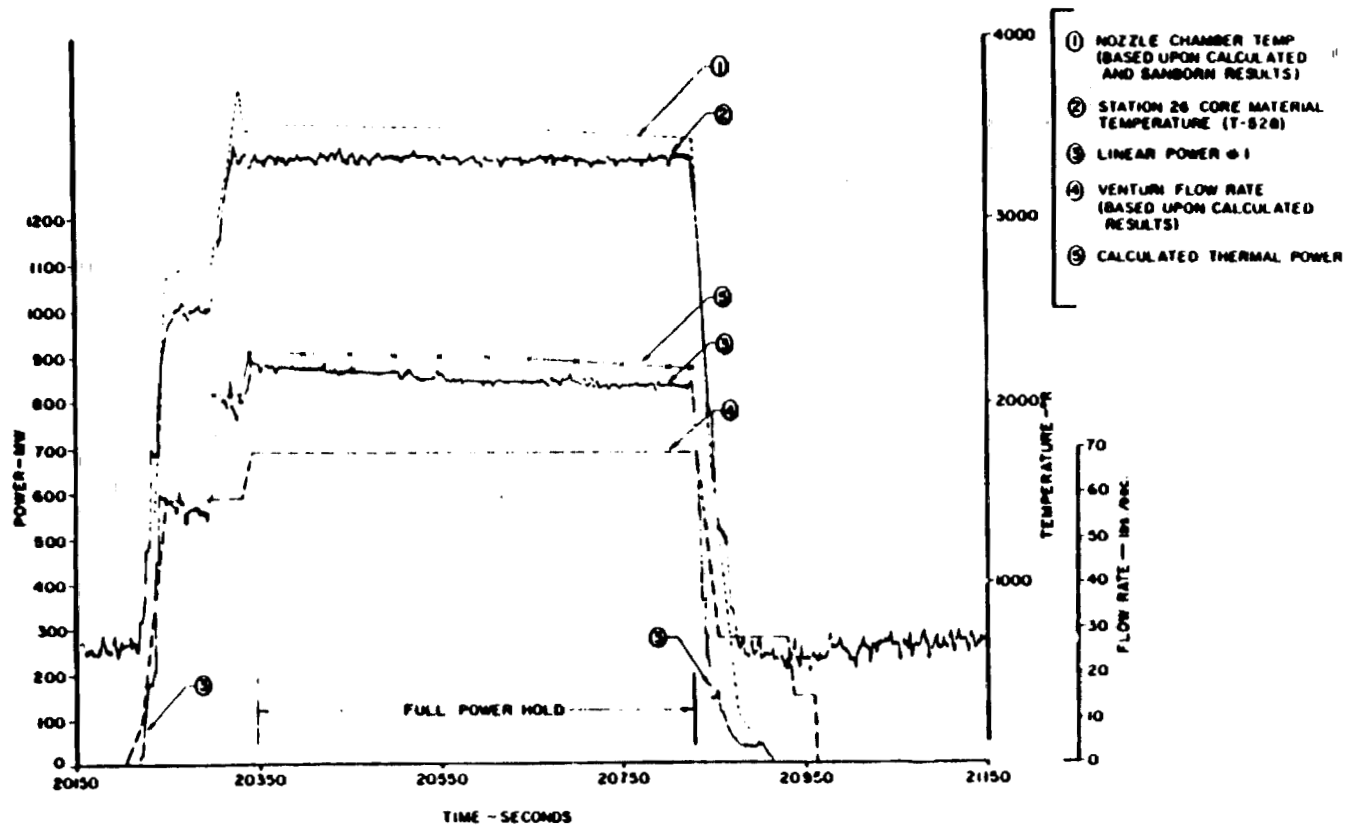


KIWI B4E Planned Flow Rate and Power Profiles

Figure D-24

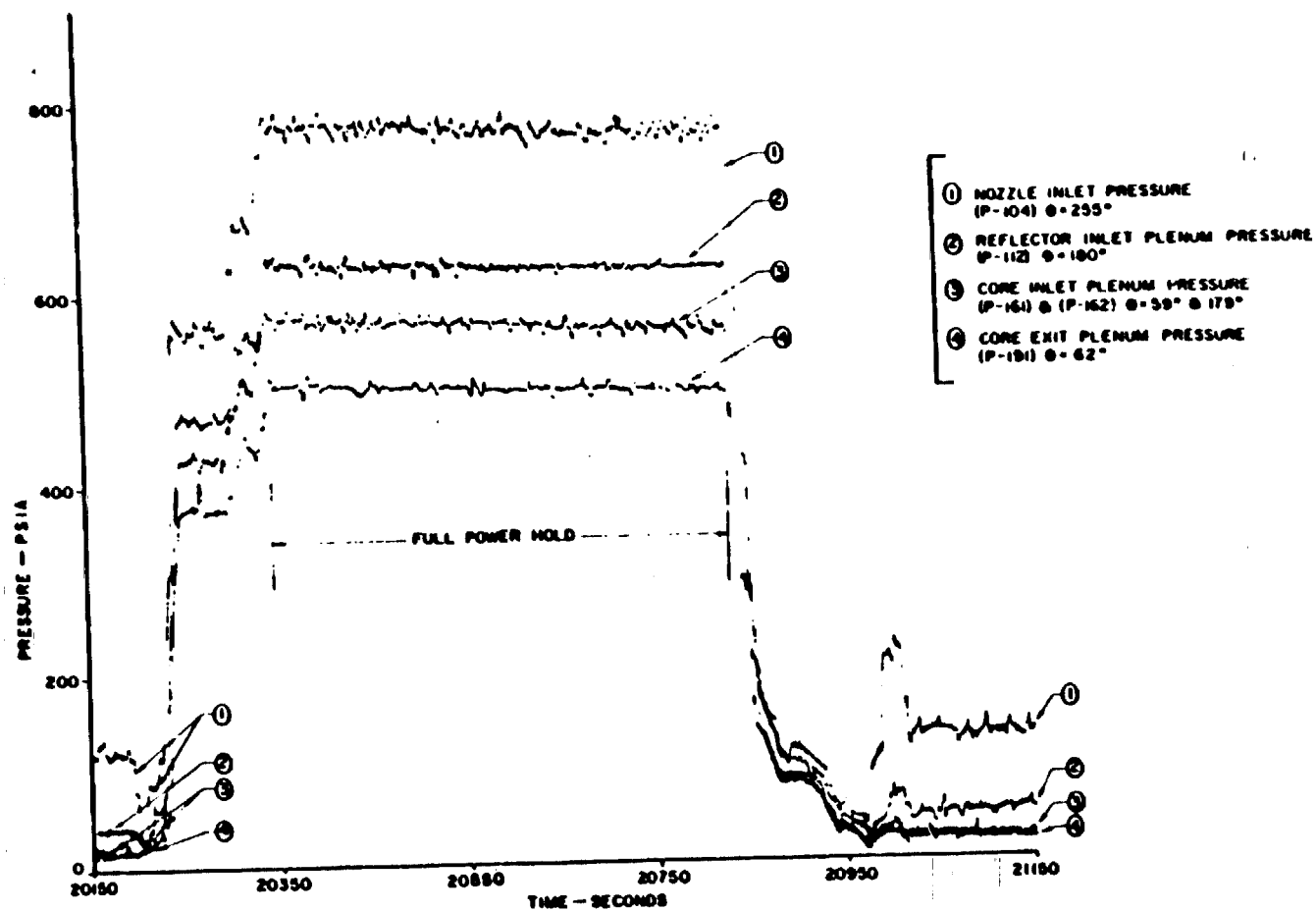


KIWI B4E PLANNED NOZZLE CHAMBER TEMPERATURE AND PRESSURE PROFILES



KIWI B4E TIME PROFILES OF PERTINENT SYSTEM PARAMETERS

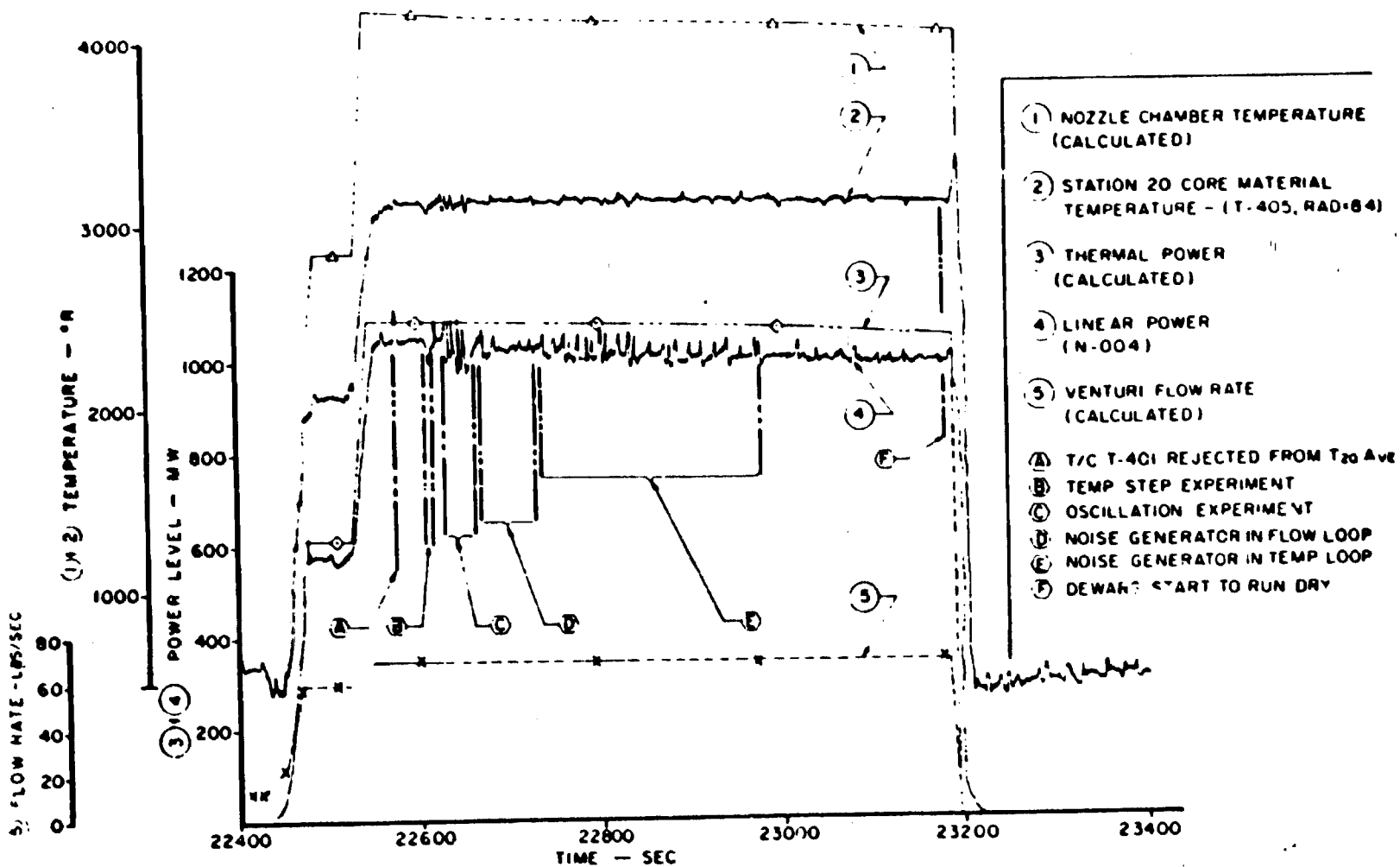
Figure D-26



KIWI R4E TIME PROFILES OF PERTINENT SYSTEM PARAMETERS

PHOEBUS 1A

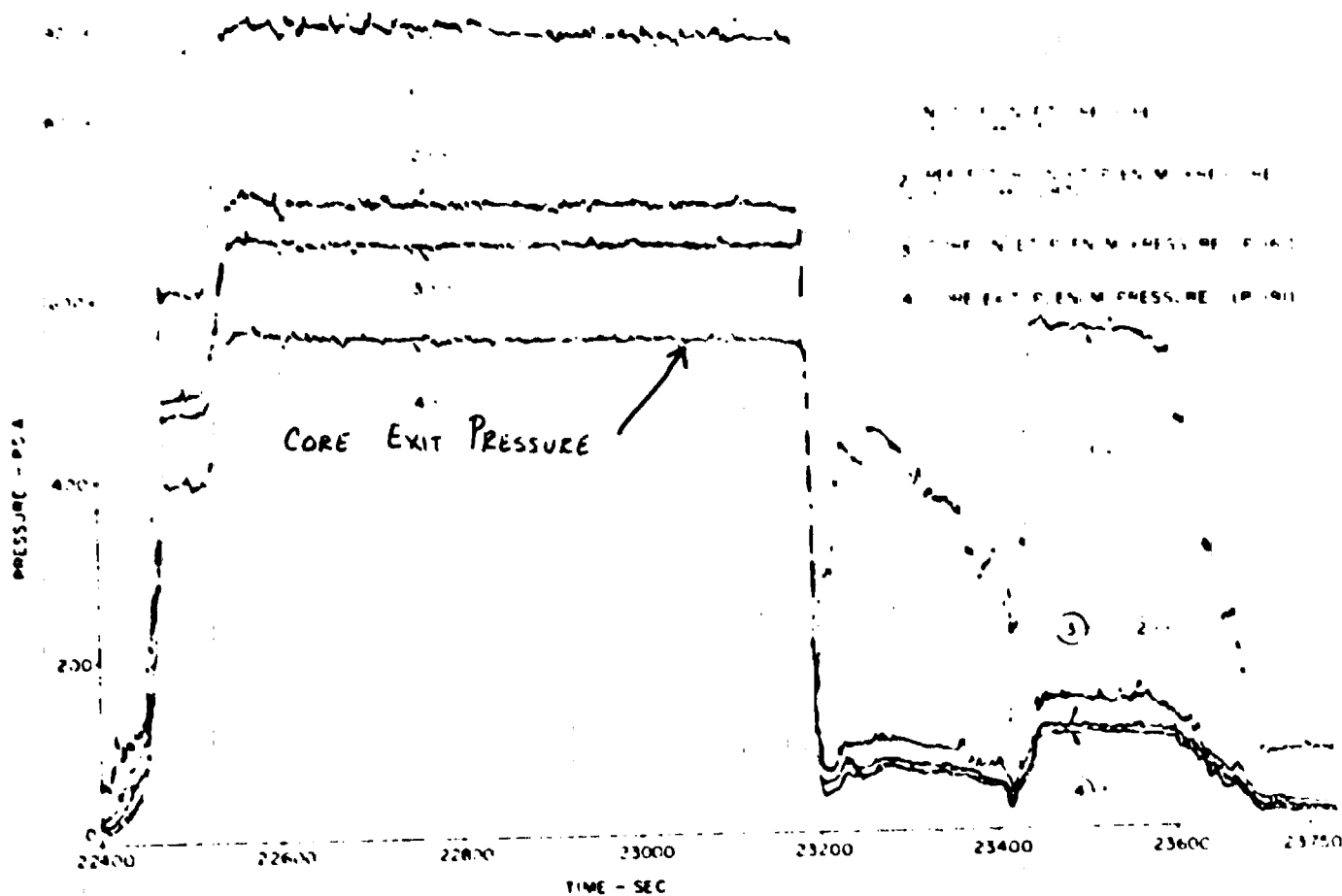
TEST DATA



PHOEBUS 1A

Time Profiles of Venturi Flow Rate, Thermal and Linear Powers, and Nozzle Chamber and Station 20 Core Material Temperatures

Figure D-28



PHOEBUS 1A
Time Profiles of Nozzle Inlet, Reflector Inlet, Core Inlet and
Core Exit Pressures

Figure D-29

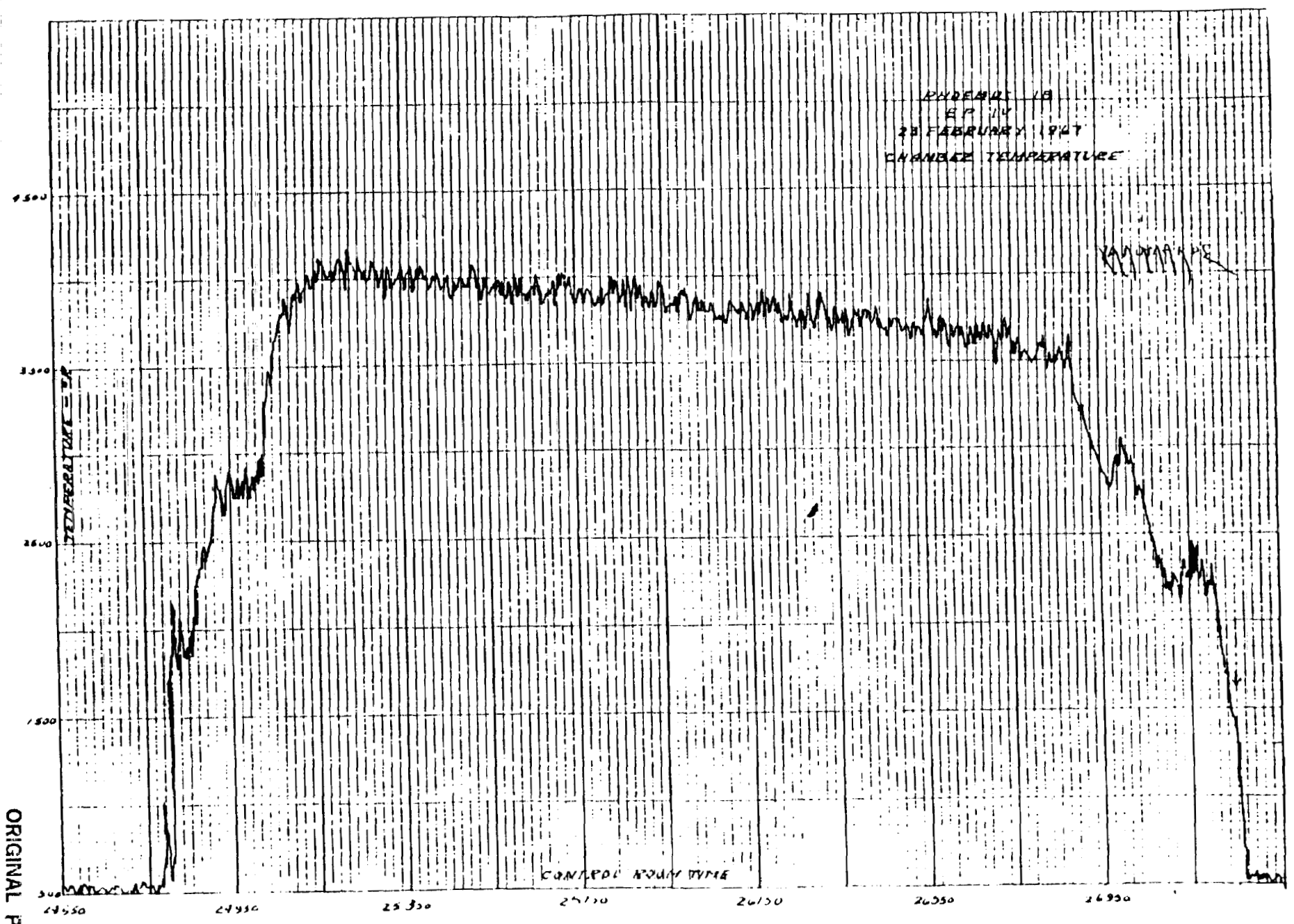
PHOEBUS 1B

TEST DATA

NOTE: 10FEB1967 Run Data Unavailable

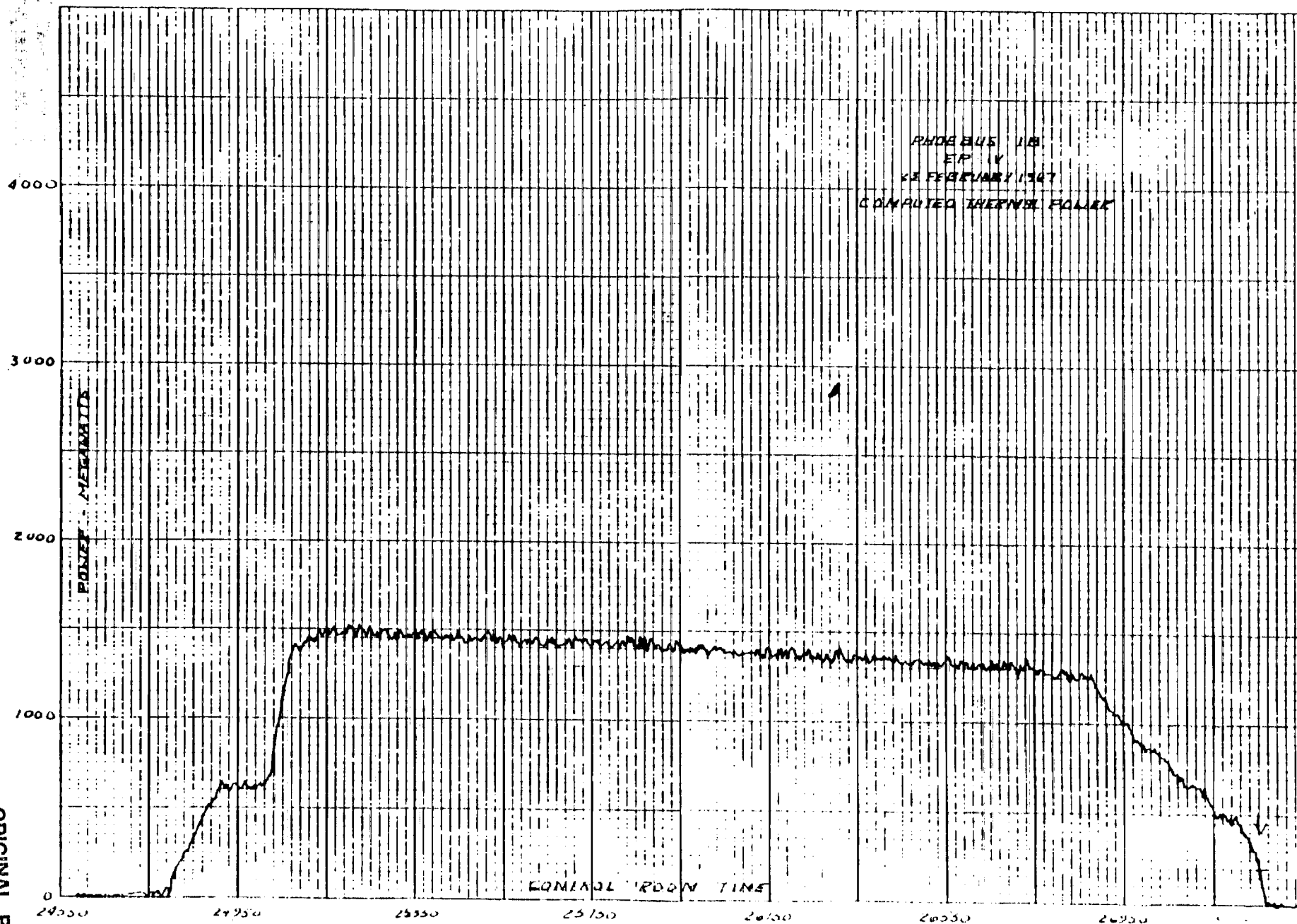
D-42

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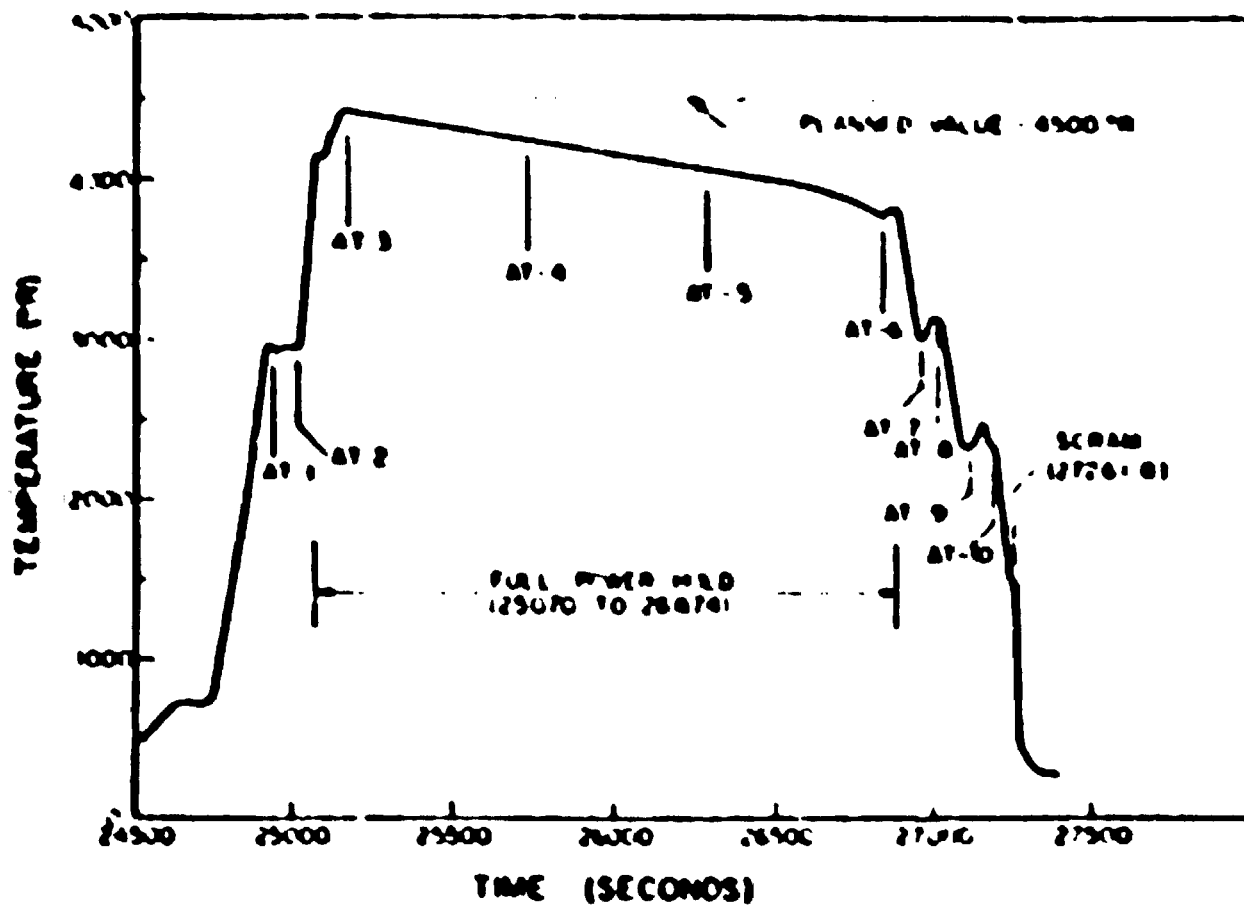


PHOBUS 1B Chamber Temperature

Figure D-30

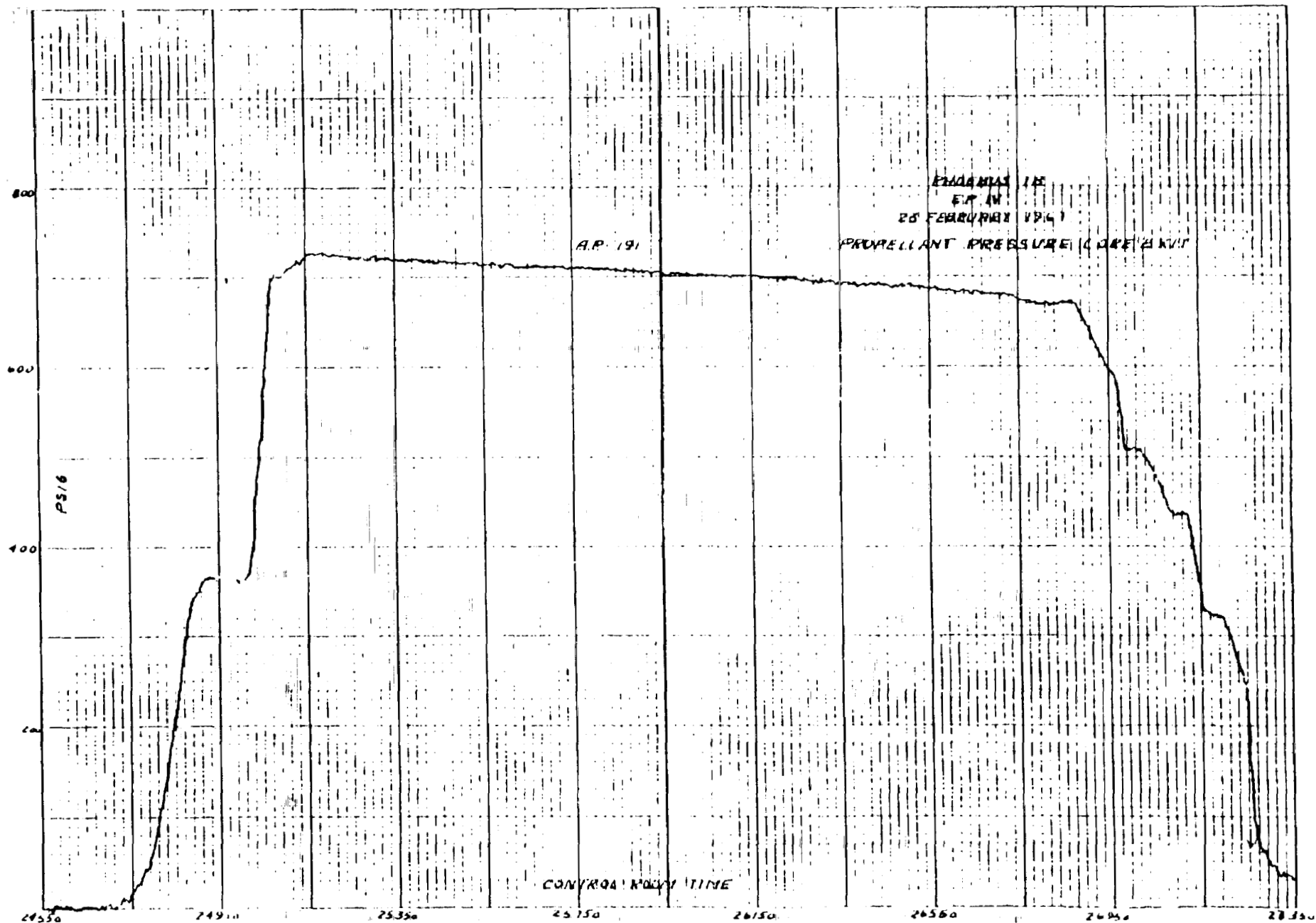


PHOEBUS 1B Computed Thermal Power



PHOEBUS 1B
 Time Variation During Full-Power Experiment
 of Average Fuel-Element Exit Hydrogen Tem-
 peratures.

Figure D-32

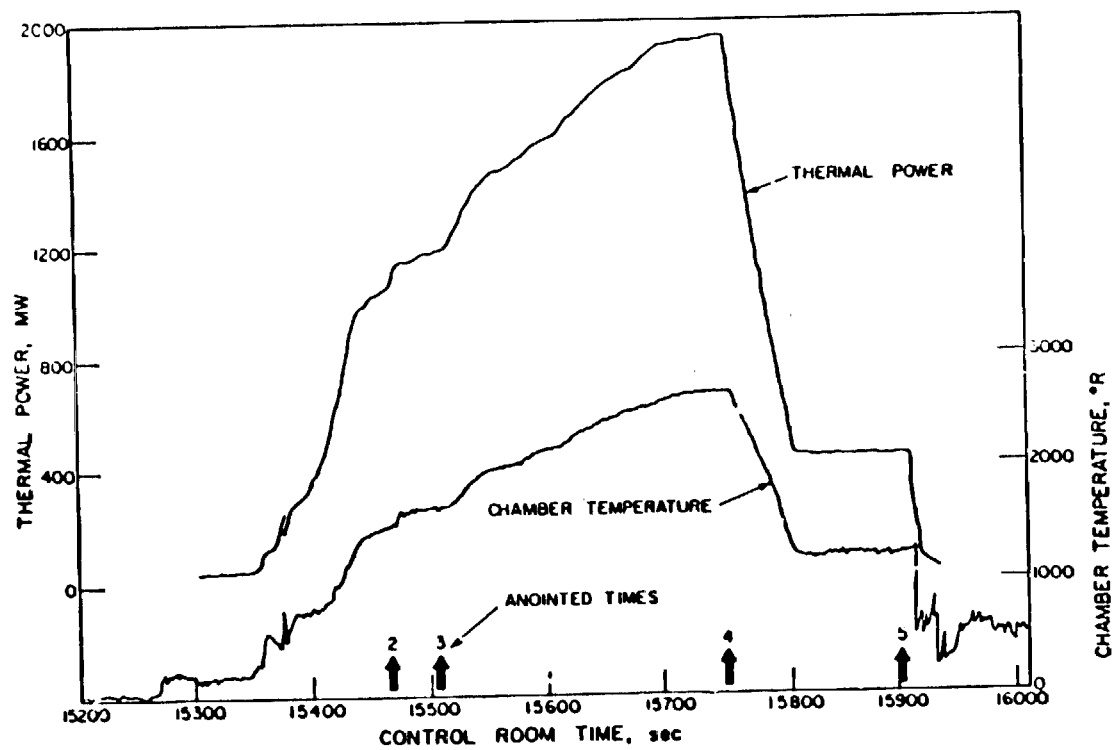


PHOEBUS 1B Propellant Pressure Core Exit

Figure D-33

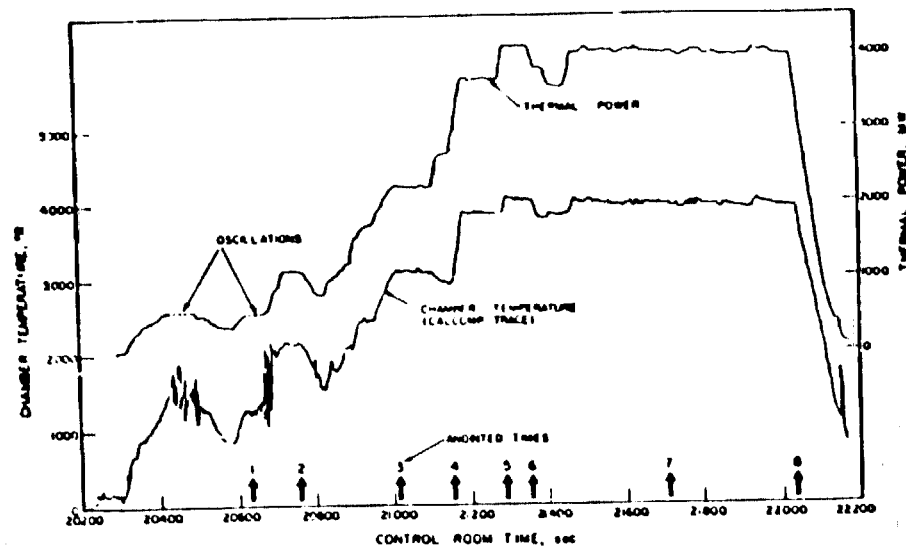
PHOEBUS 2 A

TEST DATA



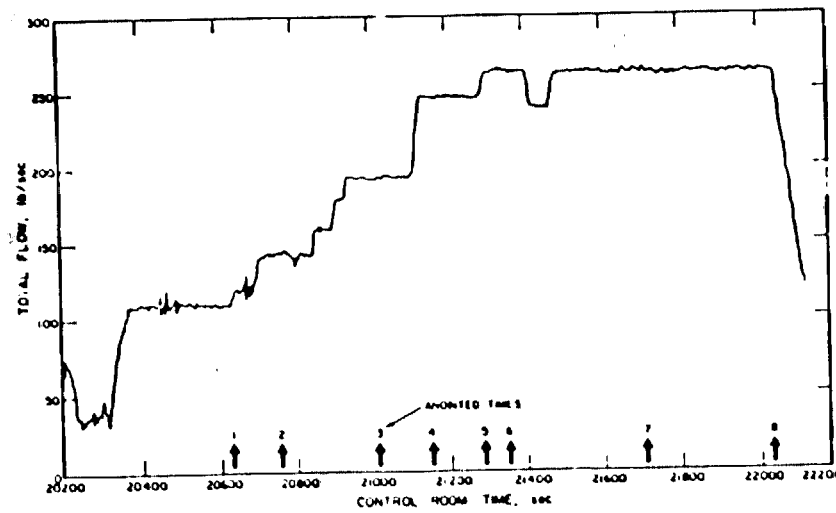
PHOEBUS 2A Computed Thermal Power and Chamber Temperature, EP-III.

Figure D-34



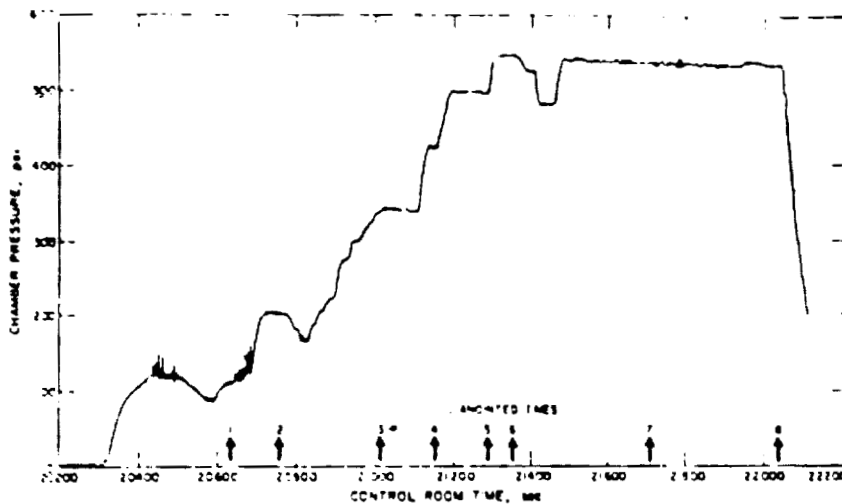
PHOEBUS 2A Computed Thermal Power and Chamber Temperature, EP-IV.

Figure D-35



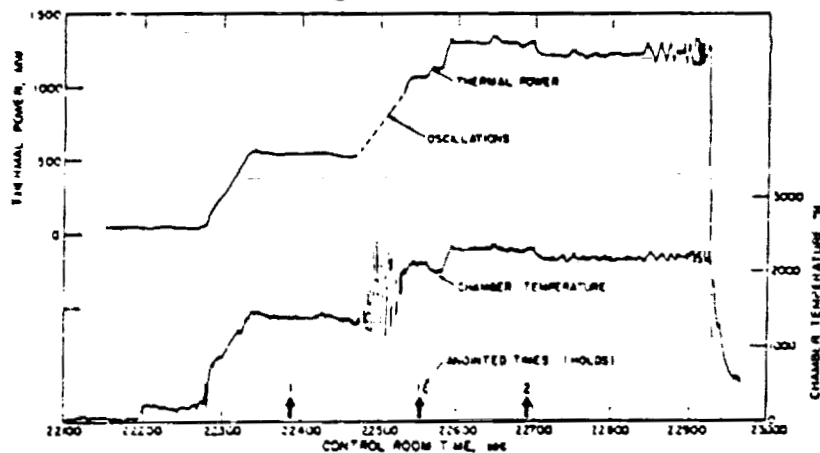
PHOEBUS 2A Computed Total Flow Rate, EP-IV.

Figure D-36



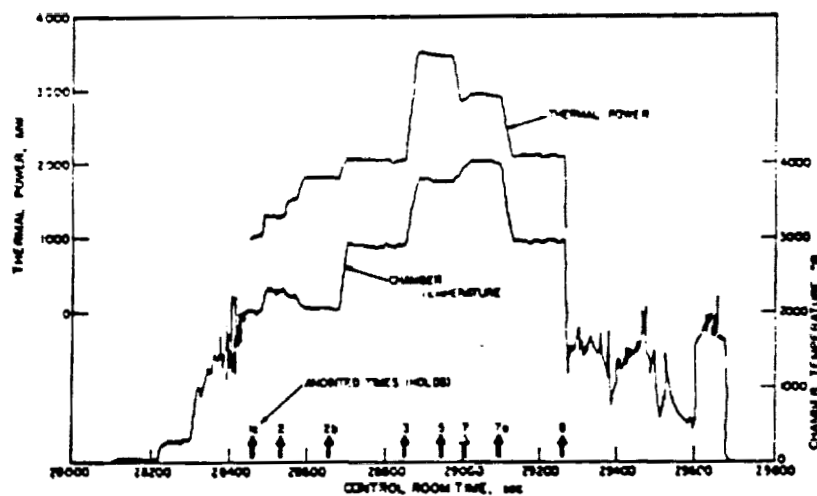
PHOEBUS 2A Chamber-Pressure Variations, EP-IV.

Figure D-37



PHOEBUS 2A Calculated Thermal Power and Chamber Temperature, EP-VA.

Figure D-38



PHOEBUS 2A Computed Thermal Power and Chamber Temperature, EP-VB.

Figure D-39

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PEWEE 1 -

TEST DATA

NOTE: Tabular Data Only. Time Traces Unavailable.

Experimental Plan II

Anointed Time	1	1A	2	3	4	5A	5	5	7	8	9	10	10A
Control-Room Time, sec	25180 -85	25215 -40	25288 -91	25322 -32	25394 -29	25505 -15	25599 -49	25509 -74	25636 -46	25715 -35	25805 -15	25915 -25	25940 -70
Total Flow, lb/sec	15.02	15.17	17.91	17.94	25.90	25.99	25.99	29.95	29.98	29.99	39.14	39.20	39.18
Moderator Flow, lb/sec	3.01	3.25	3.69	3.71	5.46	6.51	5.82	6.47	6.85	7.03	8.98	9.16	10.44
Chamber Pressure, psia	138.7	137.4	162.0	176.7	246.9	300.1	305.2	347.9	384.9	423.0	541.4	586.2	574.9
Chamber Temp., °R	1252	1206	1203	1419	1332	1931	1992	1951	2362	2822	2720	3150	3040
Fuel Exit Temp., °R ^a	1640	1600	1590	1880	1785	2705	2695	2645	3220	3865	3720	4300	4320
Drum Angle, °	73.7	71.1	68.7	75.0	65.4	78.9	81.3	77.9	64.9	91.3	84.0	85.1	86.7
Thermal Power, MW (DATUS calculated)	68.2	66.4	78.1	92.2	124.9	182.0	187.8	212.0	258.7	312.7	392.5	460.0	442.2

Experimental Plan III

Anointed Time	37	FP1A	FP1B	FP1C	FP1D	341	FP1A	FP1B	FP2C	FP2D	342	SD
Control-Room Time, sec ^b	23606	23700	24190	24300	24838	25140	25670	26400	26430	26660	26890	27300
Total Flow, lb/sec	30.37	41.48	41.48	41.53	41.52	27.90	41.55	41.56	41.55	41.56	27.87	10.46
Moderator Flow, lb/sec	7.14	9.69	11.10	10.27	10.27	6.03	10.29	10.27	10.26	10.24	5.99	1.92
Chamber Pressure, psia	386.1	633.5	621.3	630.7	630.5	218.6	630.1	631.0	631.3	630.2	217.0	83.0
Chamber Temp. (DATUS), °R	2318	3277	3160	3242	3242	908	3234	3241	3245	3233	897	931
Fuel Exit Temp. (DATUS), °R ^a	3190	4520	4530	4540	4540	1220	4535	4545	4555	4545	1200	1170
Drum Angle, °	84.7	90.9	89.2	91.5	95.0	48.0	99.5	102.5	107.0	109.9	69.9	82.9
Thermal Power, MW (DATUS)	257.0	503.5	488.4	503.2	503.0	92.1	501.9	503.3	503.9	502.0	90.0	35.5

^aWith measured hot-end seal ΔP flow values.

^bTen-sec intervals starting at the time shown.

PEWEE 1 Major Reactor Parameters

Table D-1

NUCLEAR FURNACE NF1

TEST DATA

	EP-II AT 12	EP-III ^(a) AT 6	EP-IV AT 10	EP-V AT 5
Thermal power, MW	44.1	44.0	43.5	44.5
Core (fuel cell) flow, lbm/sec	2.41	2.43	2.38	2.40
Periphery flow, lbm/sec	1.09	1.06	1.07	1.06
Total flow, lbm/sec	3.60	3.60	3.56	3.57
Inlet manifold - temp., °R ^(b)	508	545	484	491
temp., °R ^(c)	497	537	476	483
pressure, psia	724	715	715	713
Reflector - inlet temp., °R	502	541	482	486
inlet pressure, psia	717	709	709	707
ΔP, psi	19.0	24.3	18.6	18.7
Core - inlet temp., °R	622	635	606	612
inlet pressure, psia	695	680	687	684
outlet pressure, psia	468	447	469	460
ΔP, psi	228	233	218	224
Periphery bypass exit temp, °R	660	656	658	669

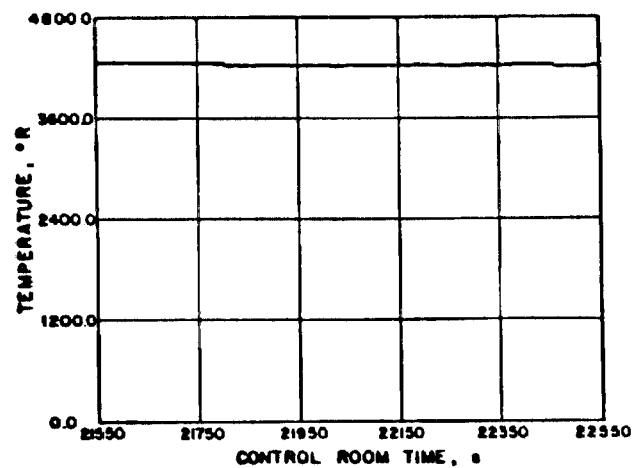
(a) Reflector not at equilibrium.

(b) Cart venturi resistance thermometer.

(c) Inlet manifold thermocouple.

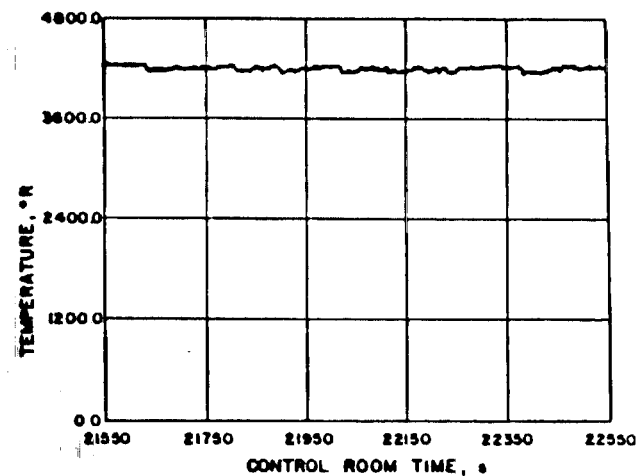
NF-1 HYDROGEN SYSTEM STATE POINTS AT END OF FULL POWER HOLD

Table D-2



NF1 Exit-gas temperature trace for composite element.

Figure D-40

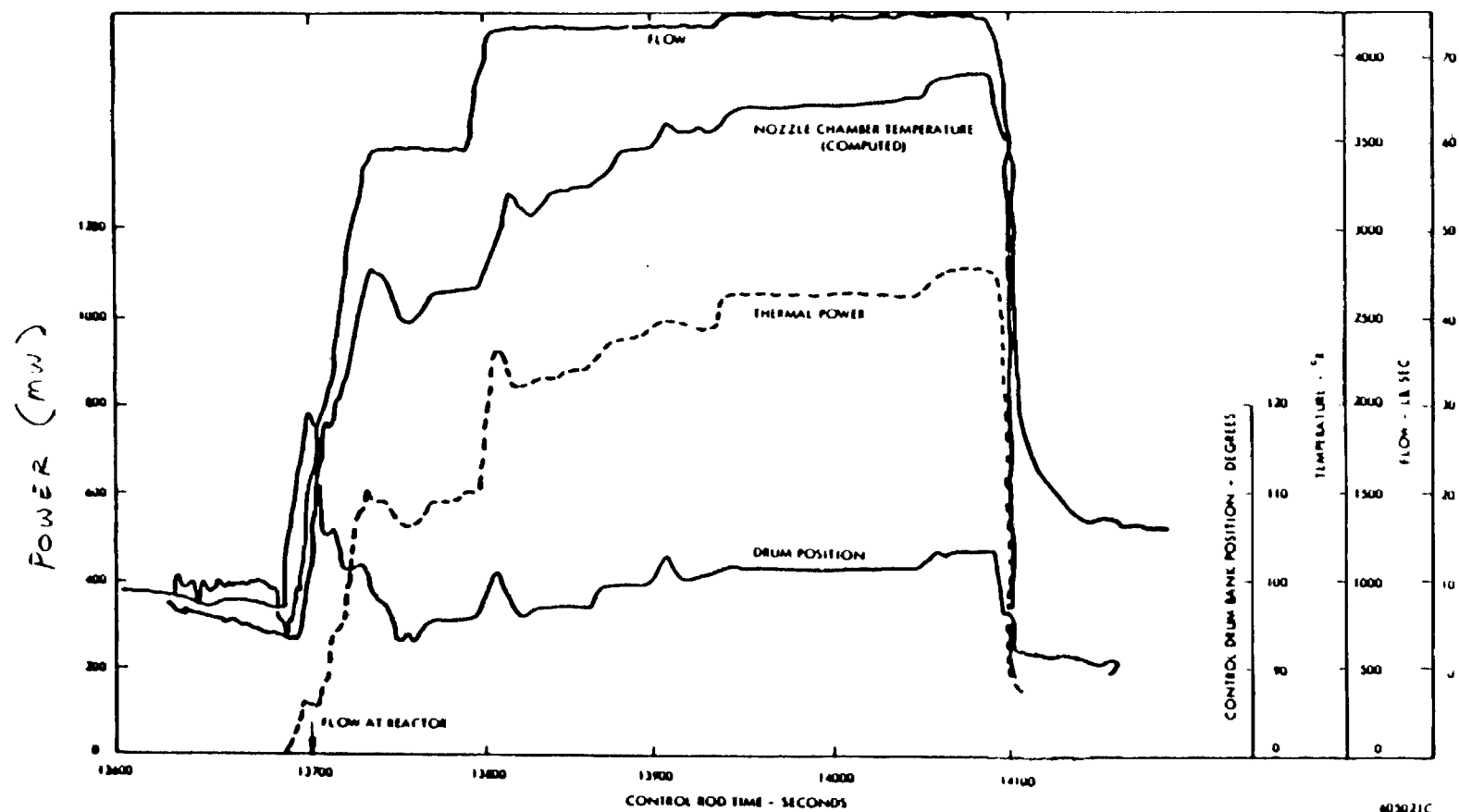


NF1 Exit-gas temperature trace from carbide element (Channel AT-186).

Figure D-41

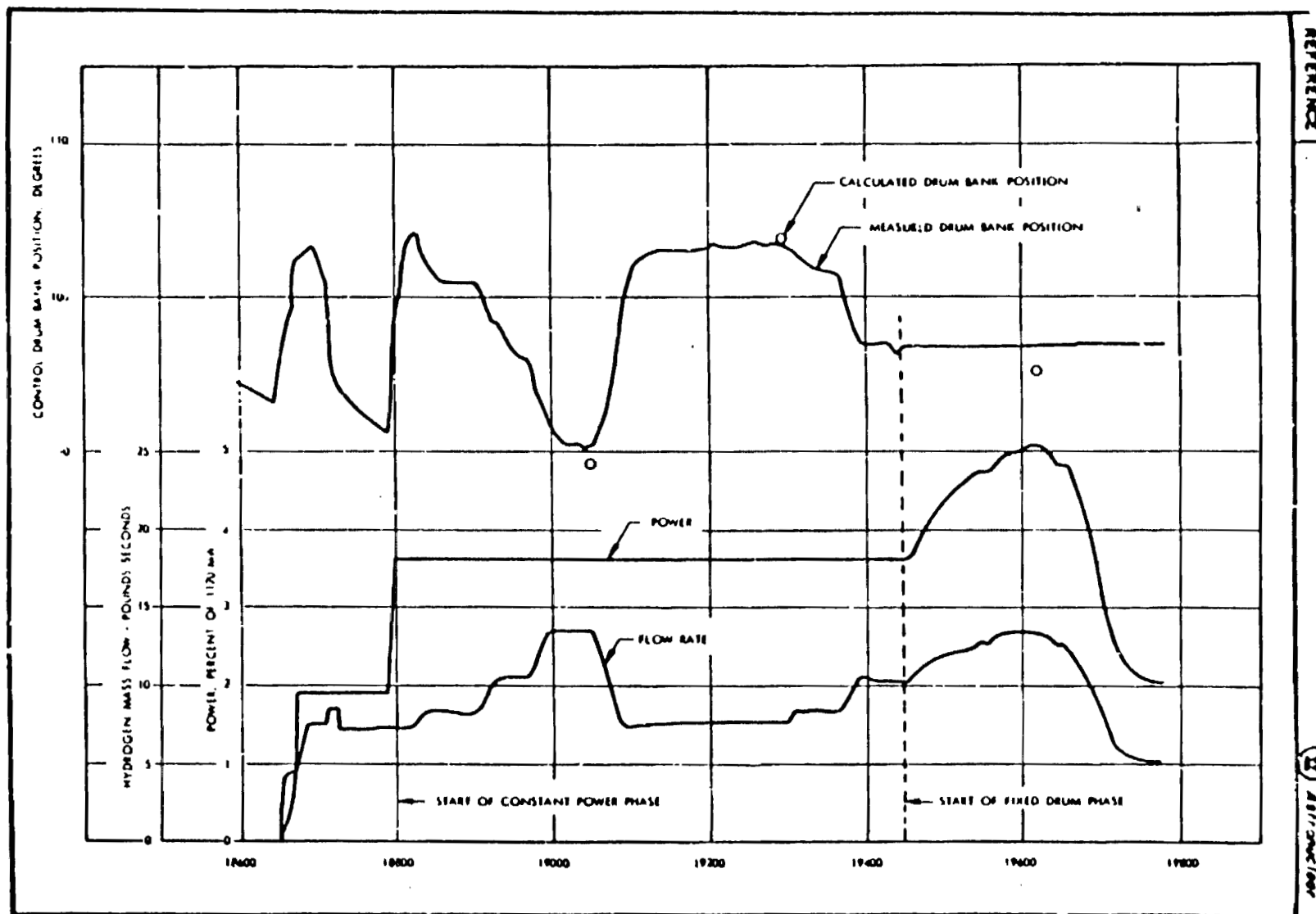
NRX A2

TEST DATA

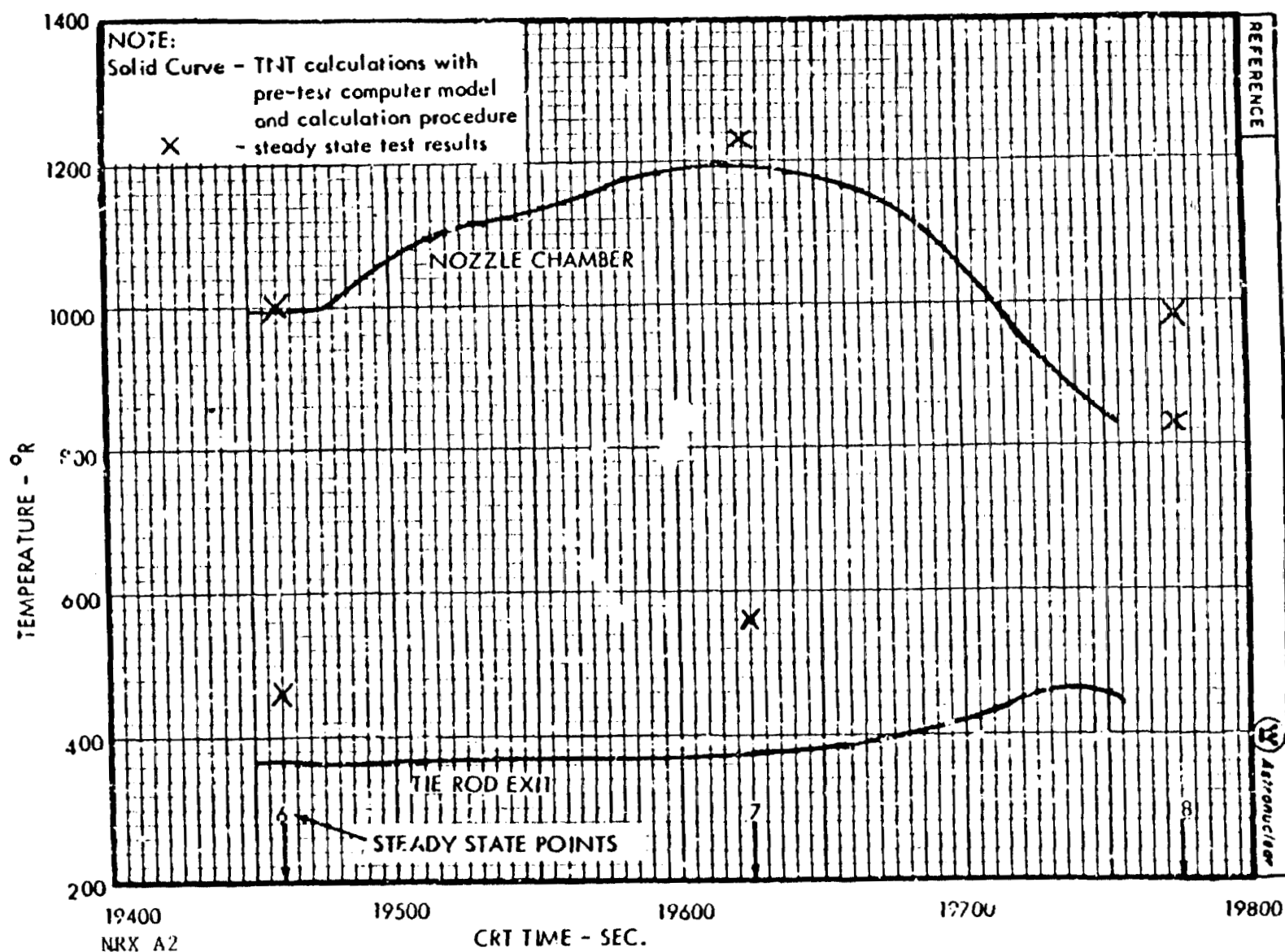


NRX-A2 EP-TV POWER, FLOW, CHAMBER TEMPERATURE
AND DRUM POSITION VS TIME

Figure D-42



CONTROL ROOM TIME - SECONDS
NRX-A2 EP-V LOW POWER MAPPING TEST



EP-V LOW POWER MAPPING TEST: FIXED CONTROL DRUMS
 NOZZLE CHAMBER AND TIE ROD TEMPERATURES

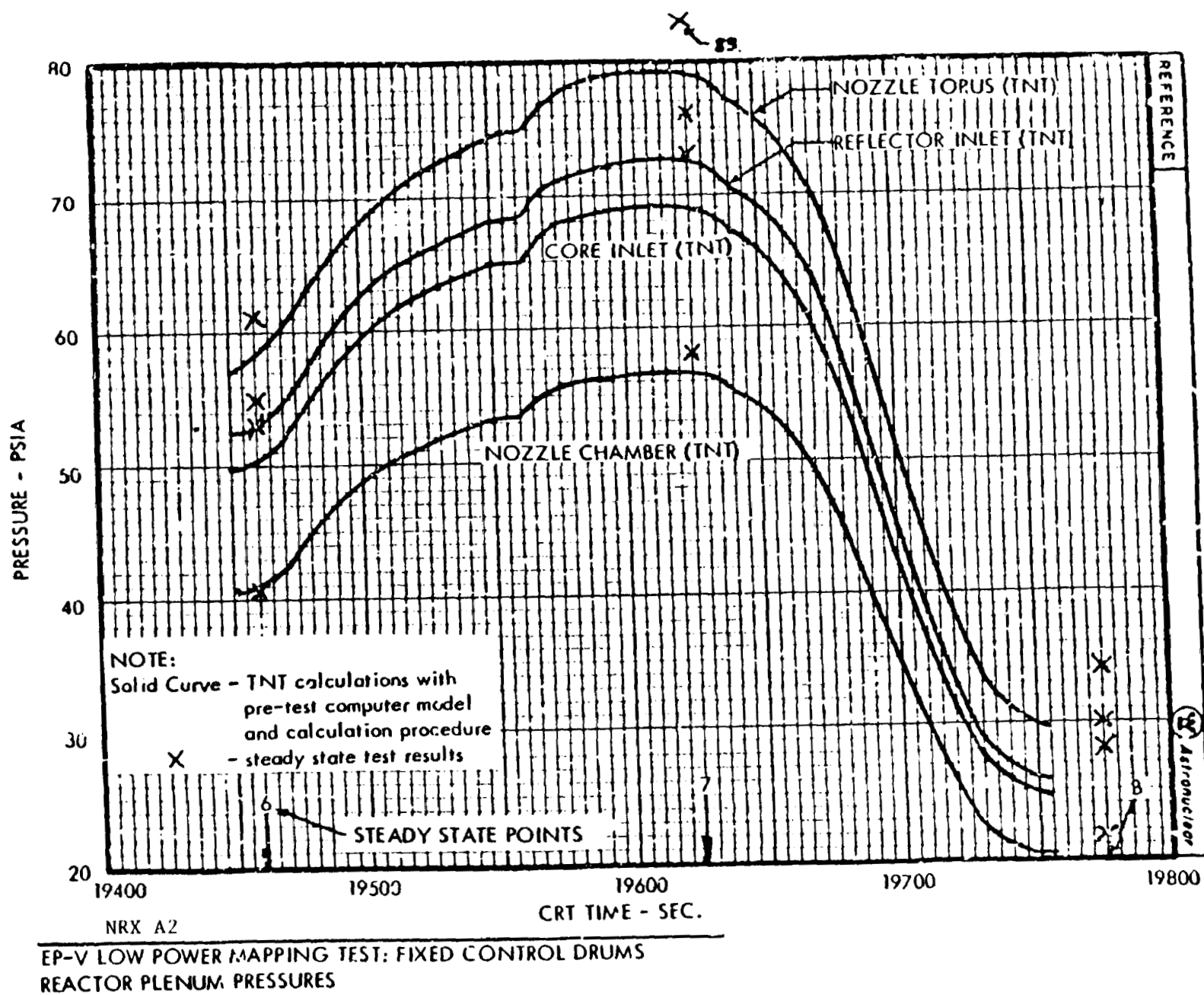
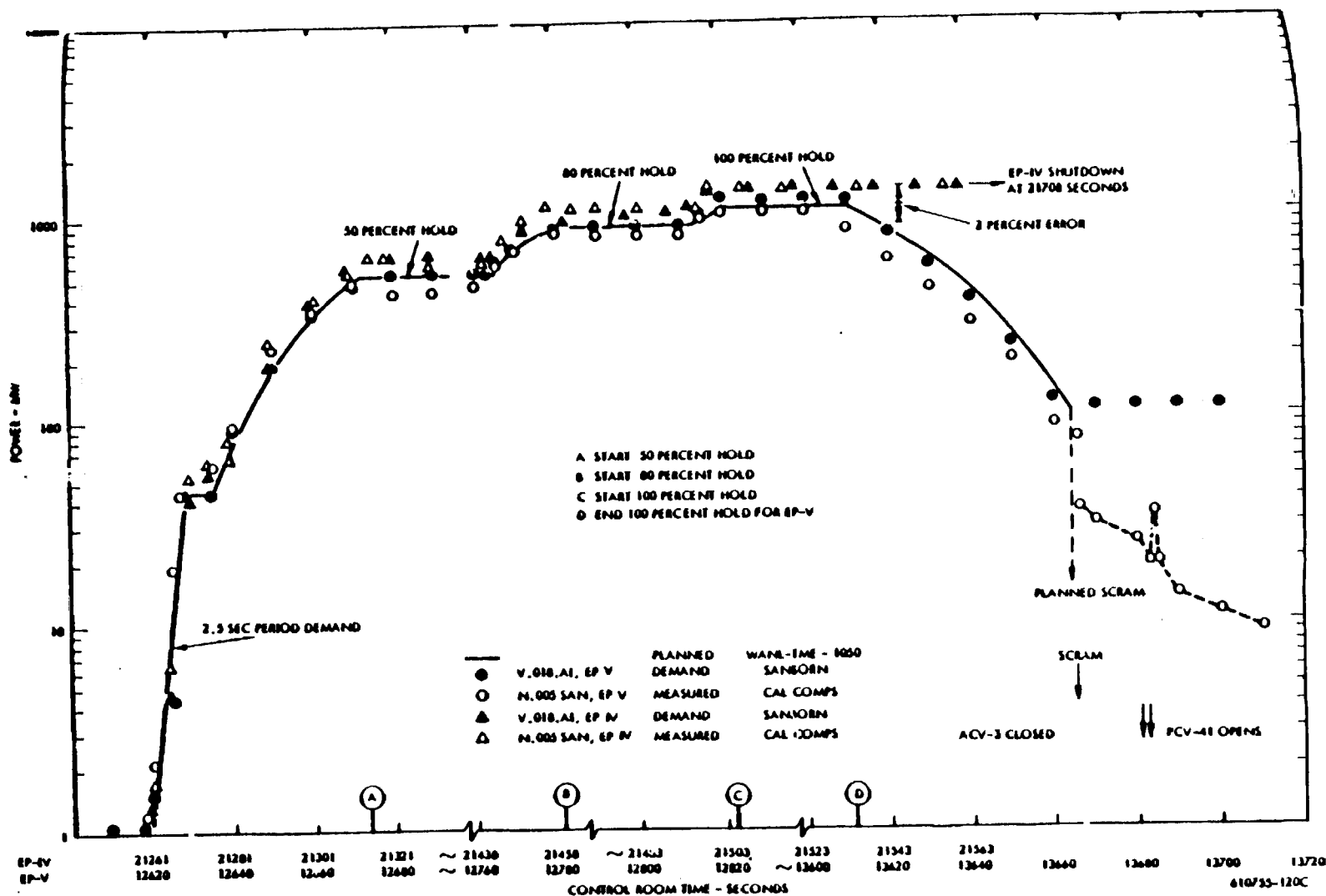


Figure D-45

NRX A3

TEST DATA

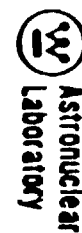
D-61



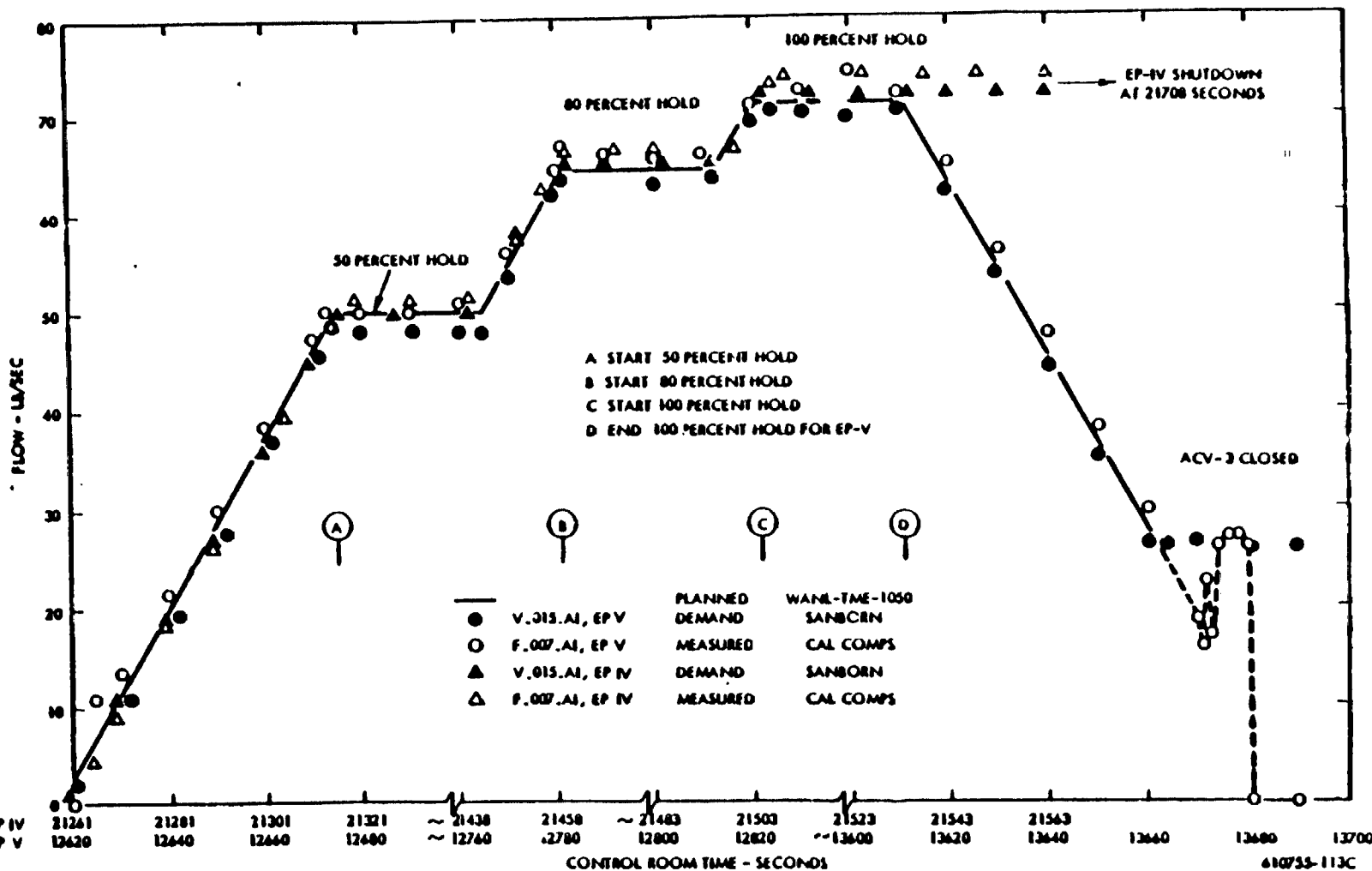
NRX-A3 Average Log Power Programmer

Figure D-46

WANL-TNR-202



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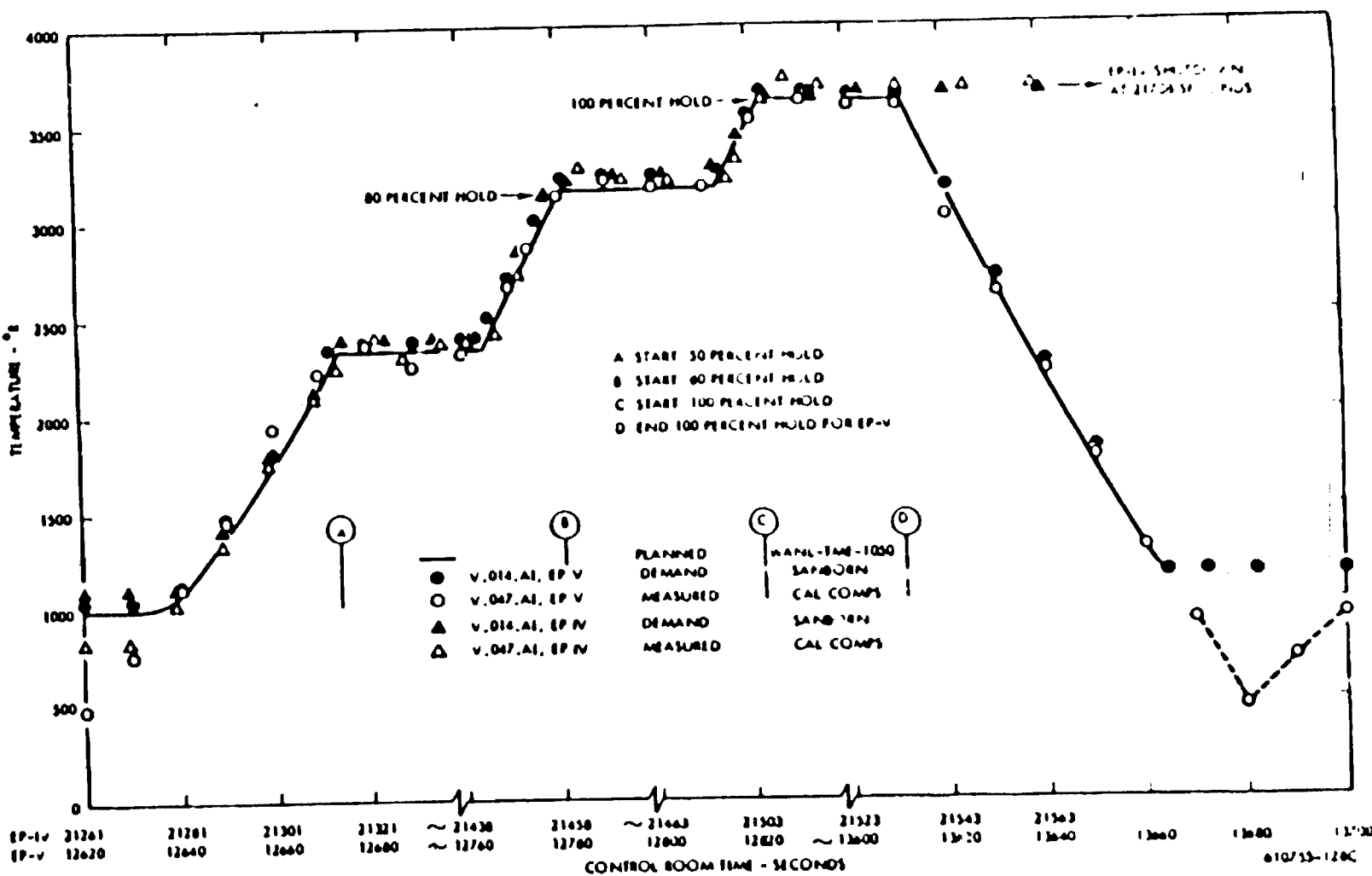


NRX-A3 Liquid Hydrogen Flow Programmer

Figure D-47

ASTRONUCLEAR
Laboratory
WANL-TNR-202

D-62
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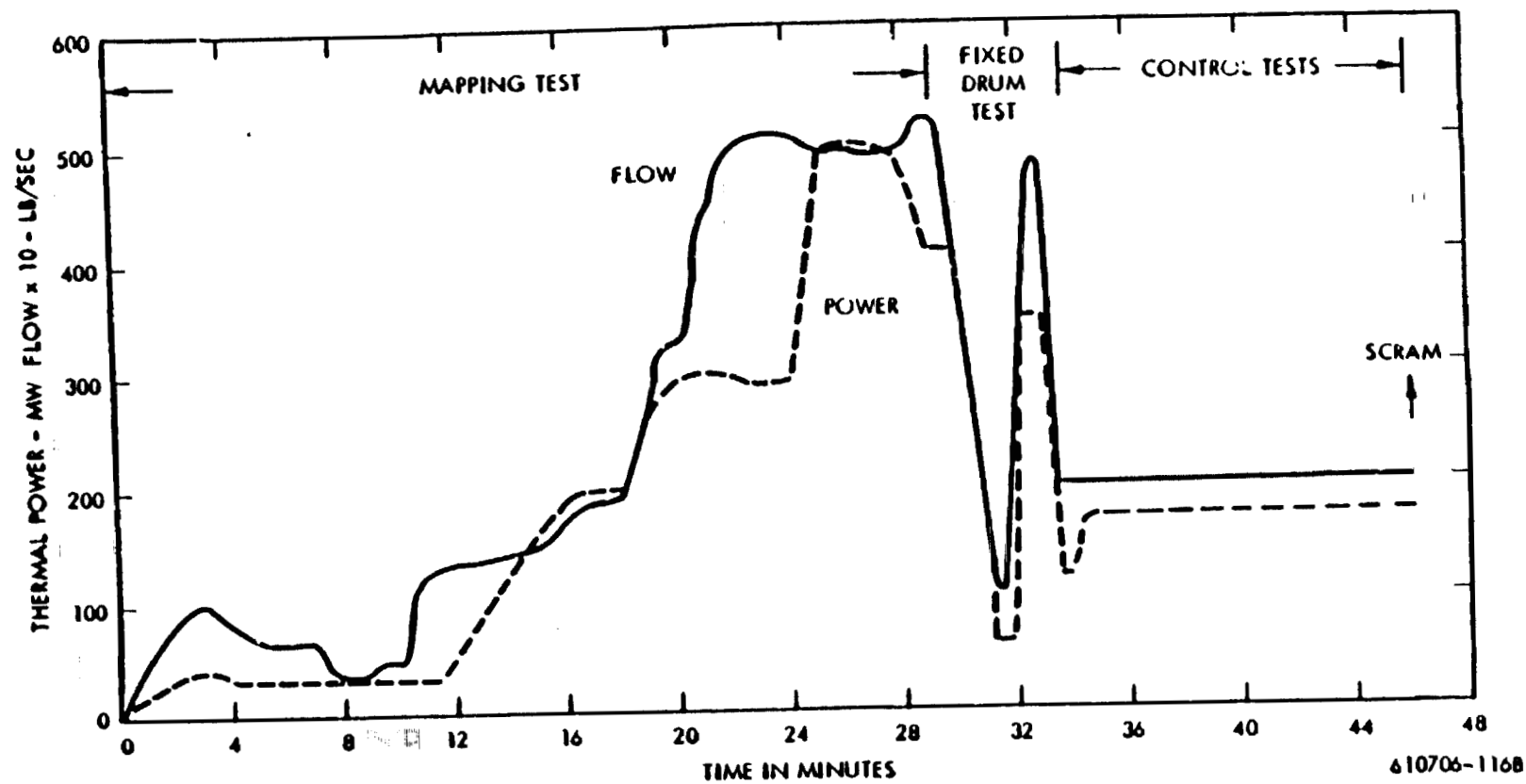
NRX-A3 Station 26 Temperature Programmer

Figure D-48

ASTRONUCLEAR
Laboratory
WANL-TNR-202

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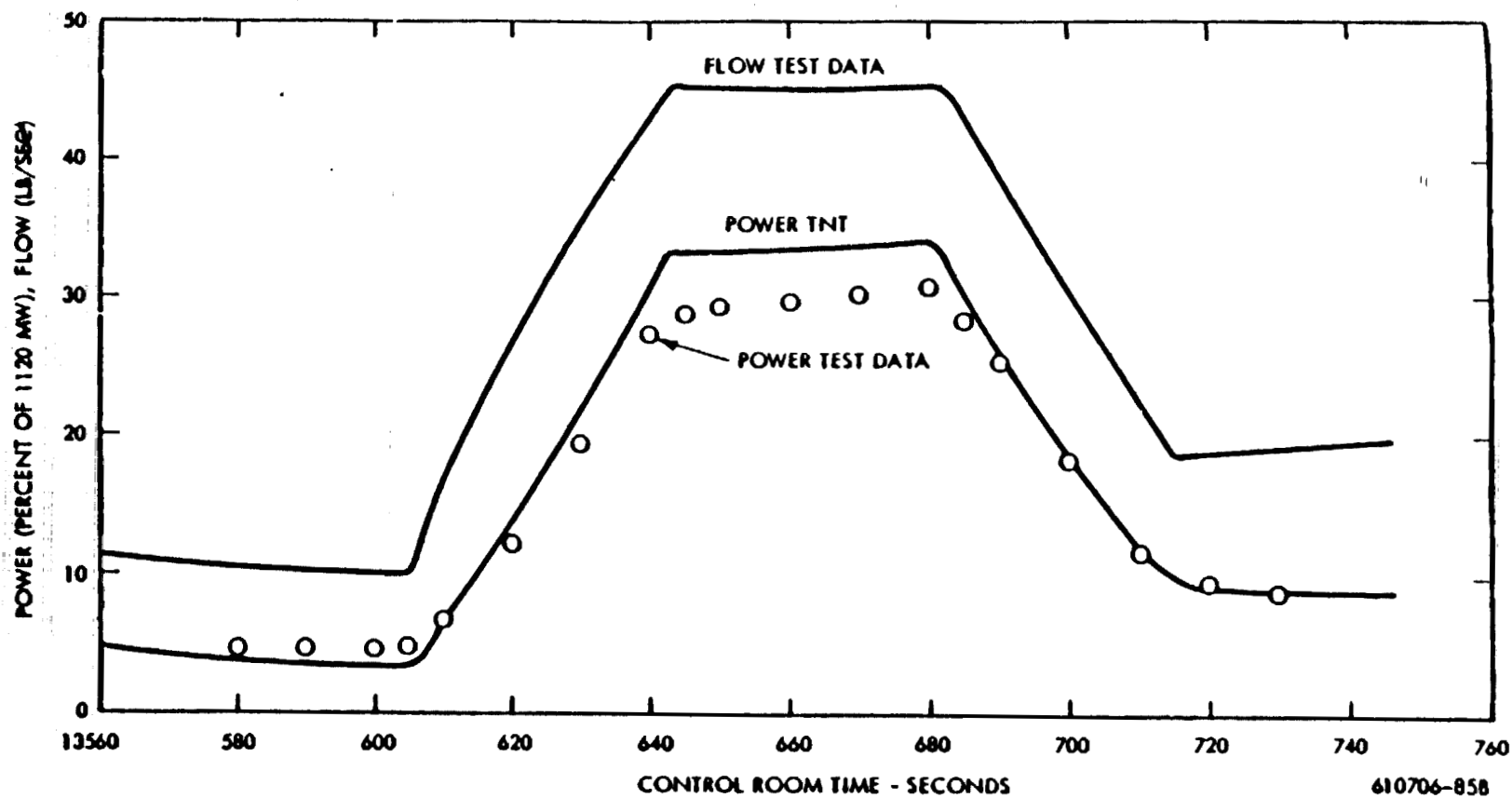
D-63



NRX-A3 Medium Power Mapping Test Power and Flow Profiles

Figure D-49

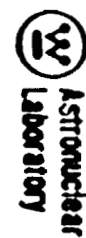
D-65



NRX-A3 EP-VI Fixed Control Drum Test Profile - Power and Flow

Figure D-50

MANL-TNR-202



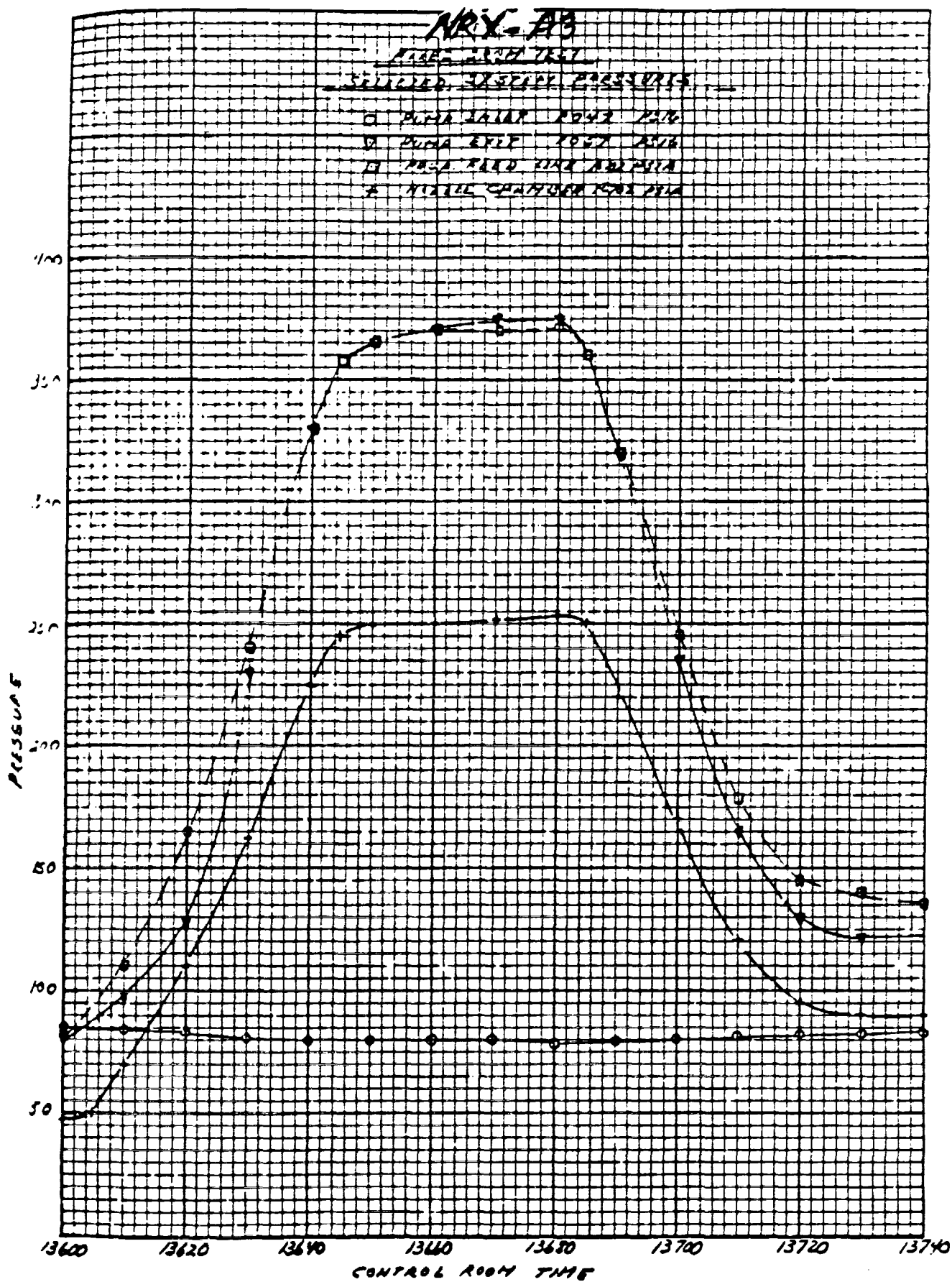
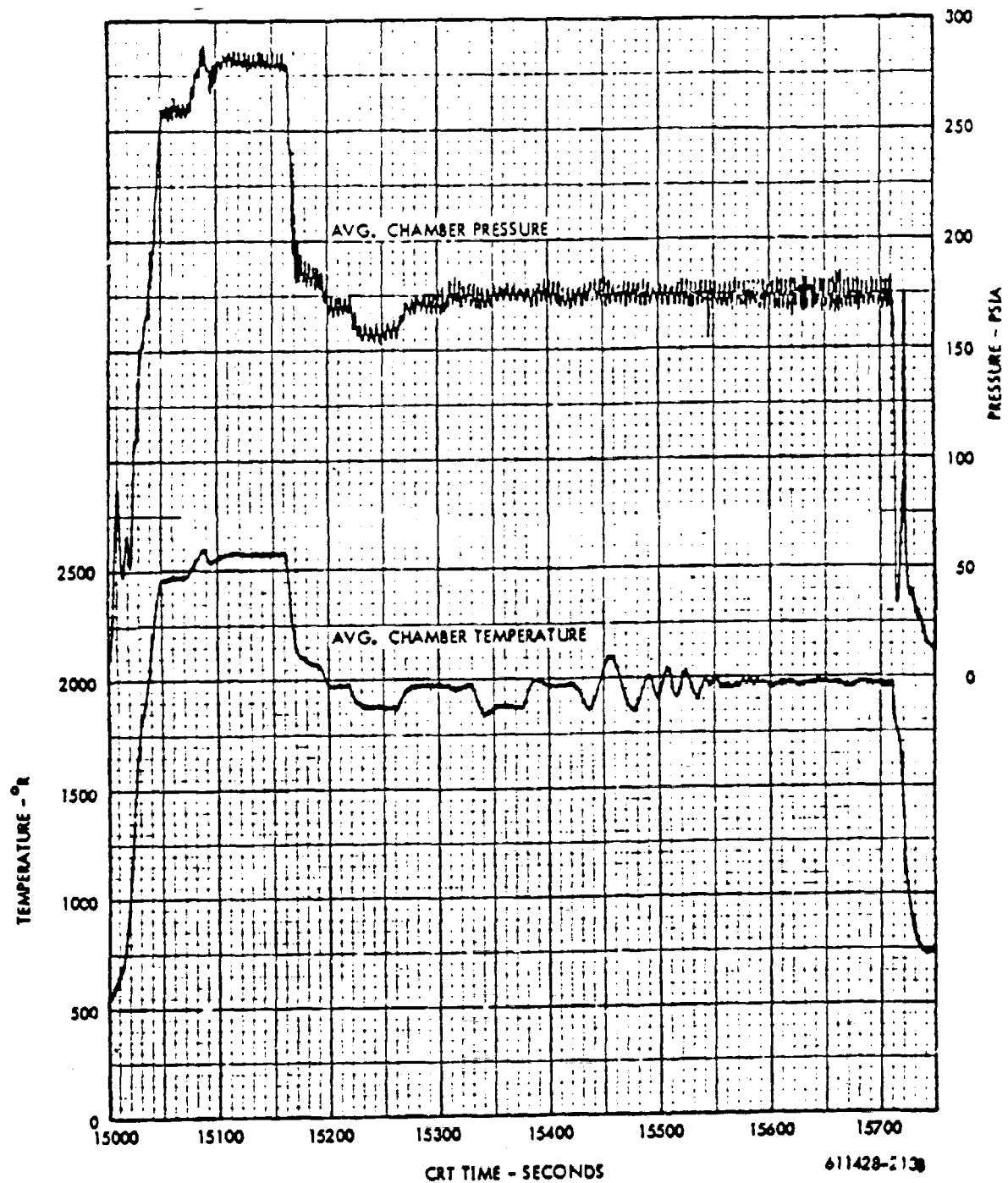


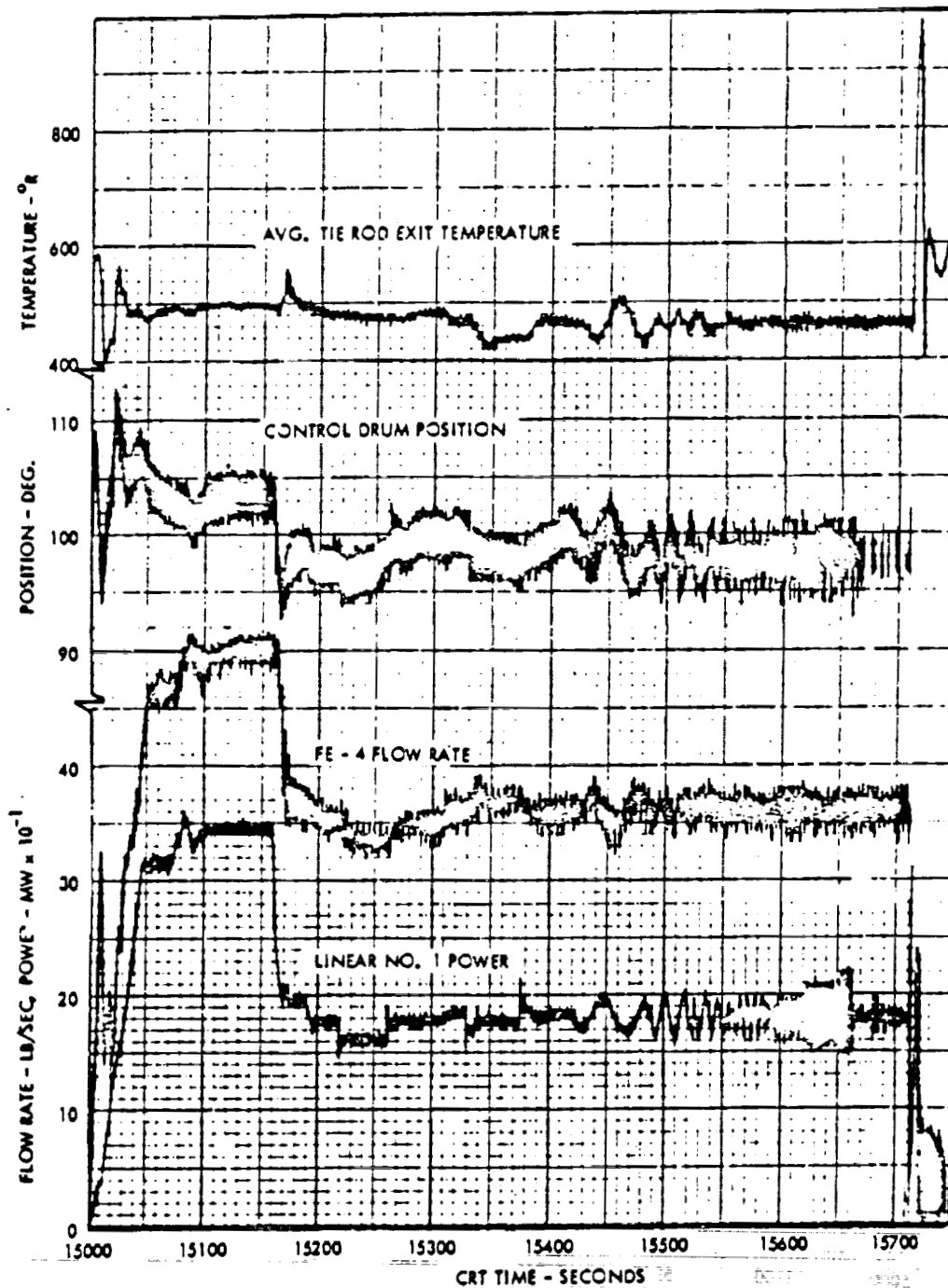
Figure D-51

NRX/EST
TEST DATA



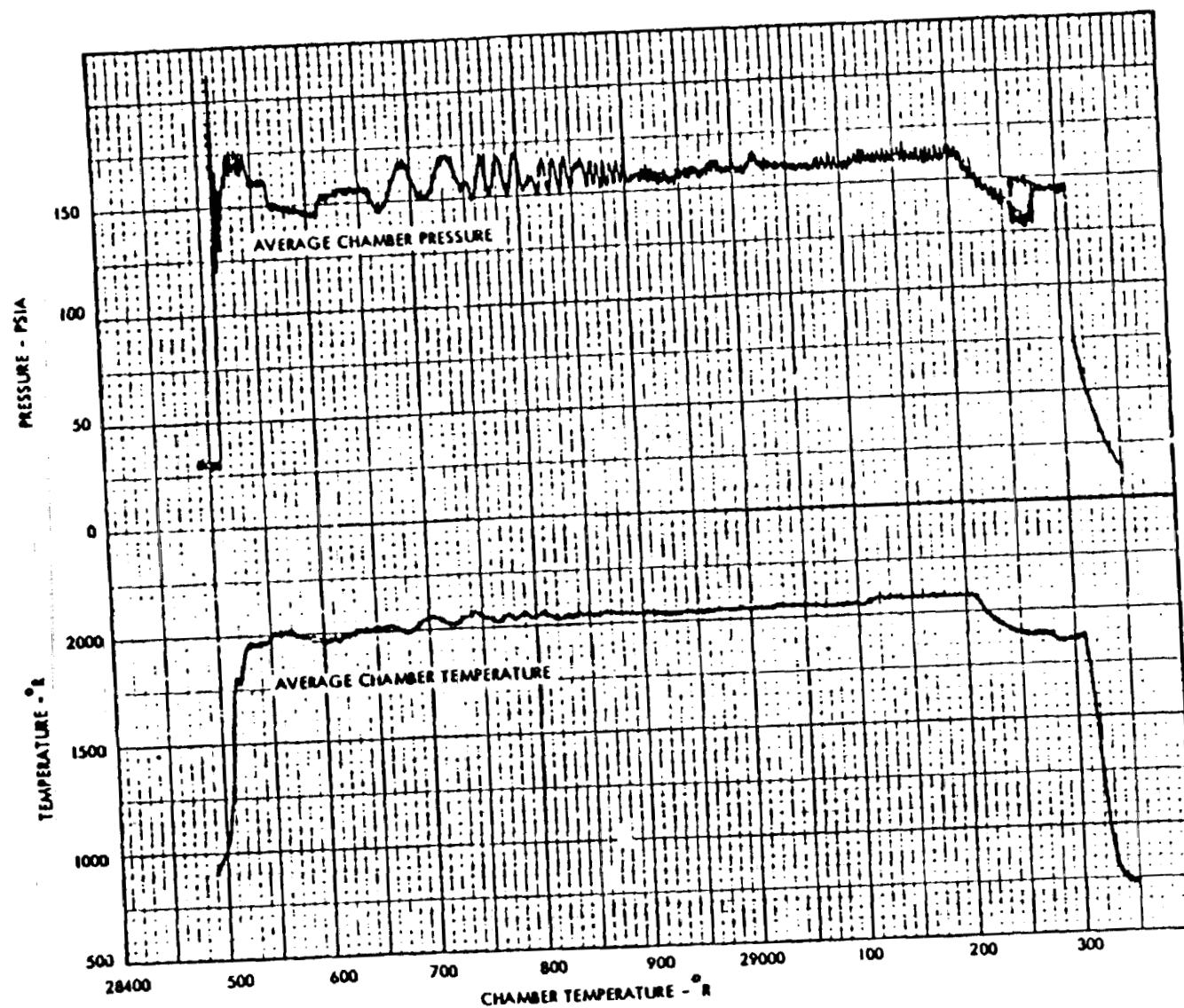
NRX/EST EP-IIB, Run 1 Profile (Sheet 1) - Complete Run

Figure D-52



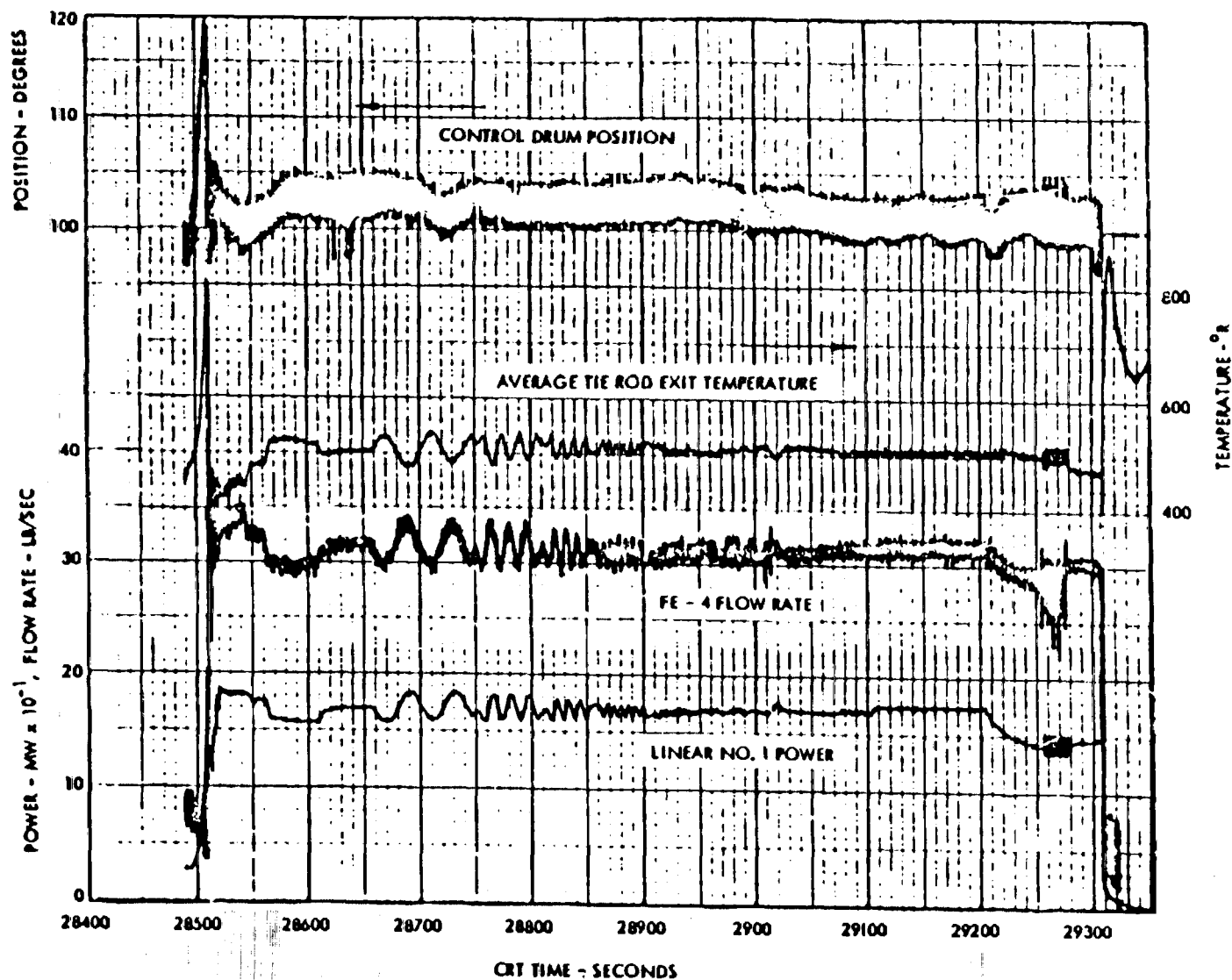
NRX/EST EP-IIIB, Run 1 Profile (Sheet 2) - Complete Run

Figure D-53



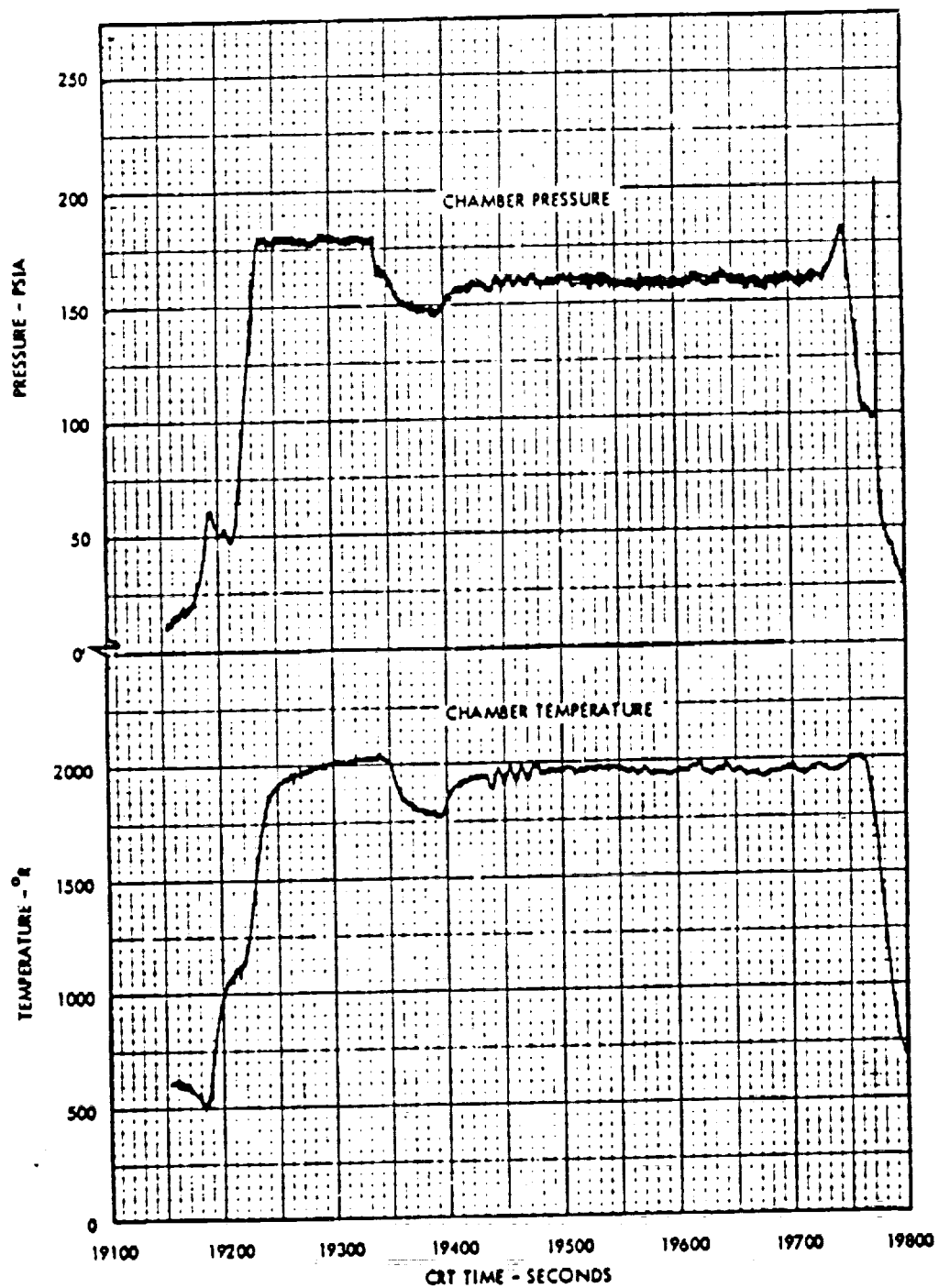
NRX/EST EP-IIB, Run 2 Profile (Sheet 1) - Complete Run

Figure D-54



NRX/EST EP-IIB, Run 2 Profile (Sheet 2) - Complet Run

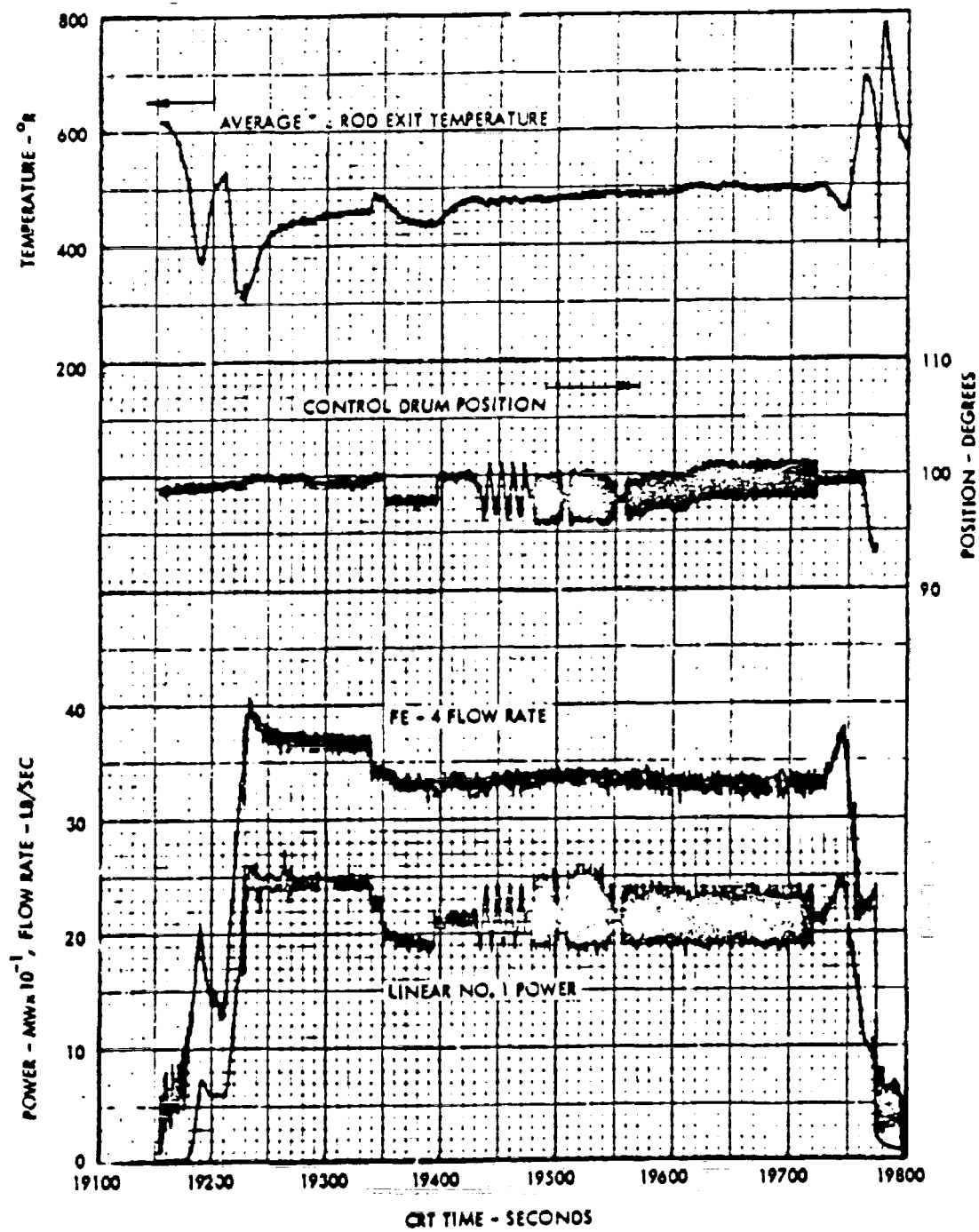
Figure D-55



NRX/EST. EP-IIC, Run 1 Profile (Sheet 1) - Complete Run

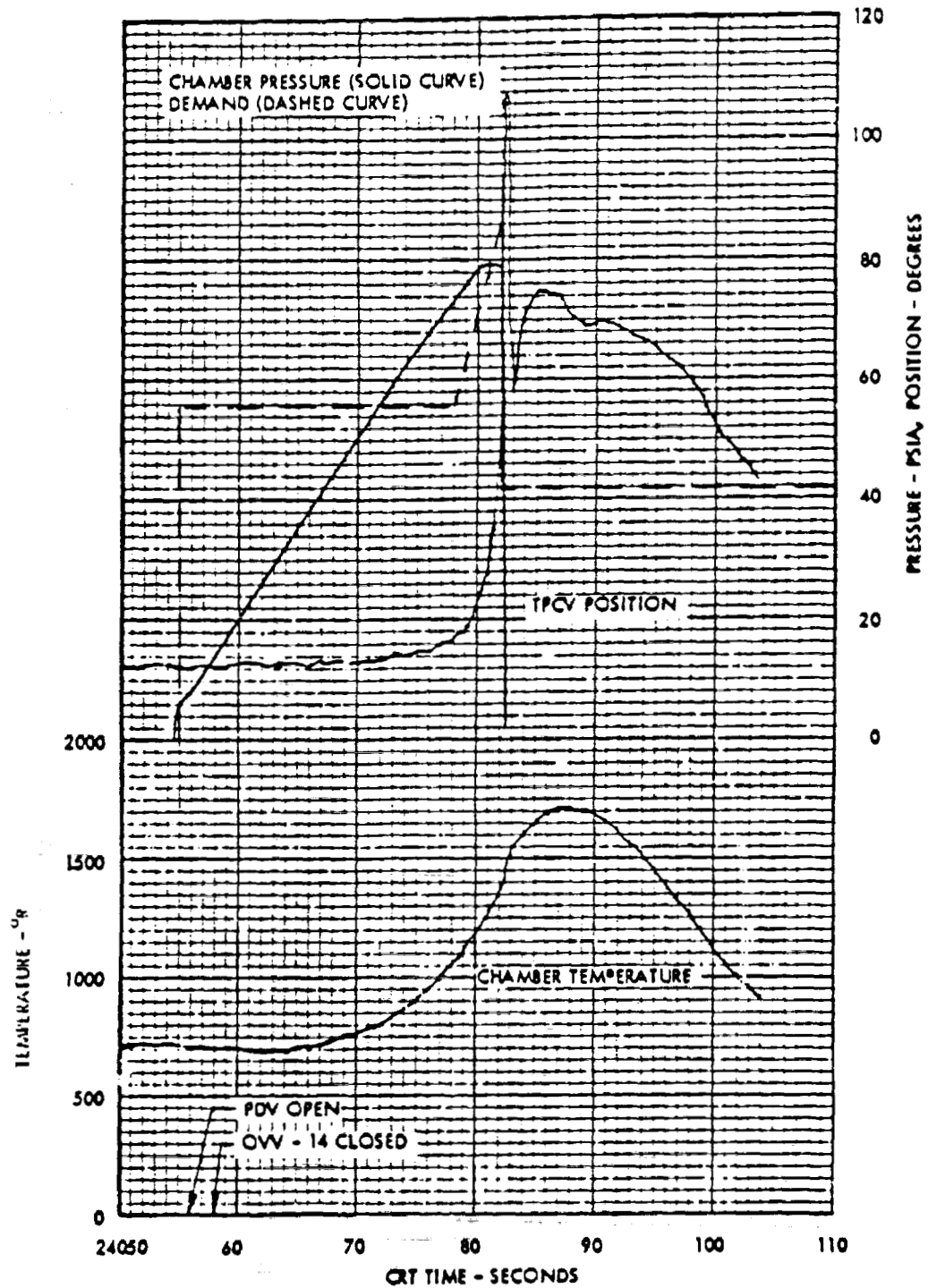
Figure D-56

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NRX/EST EP-IIC, Run 1 Profile (Sheet 2) - Complete Run

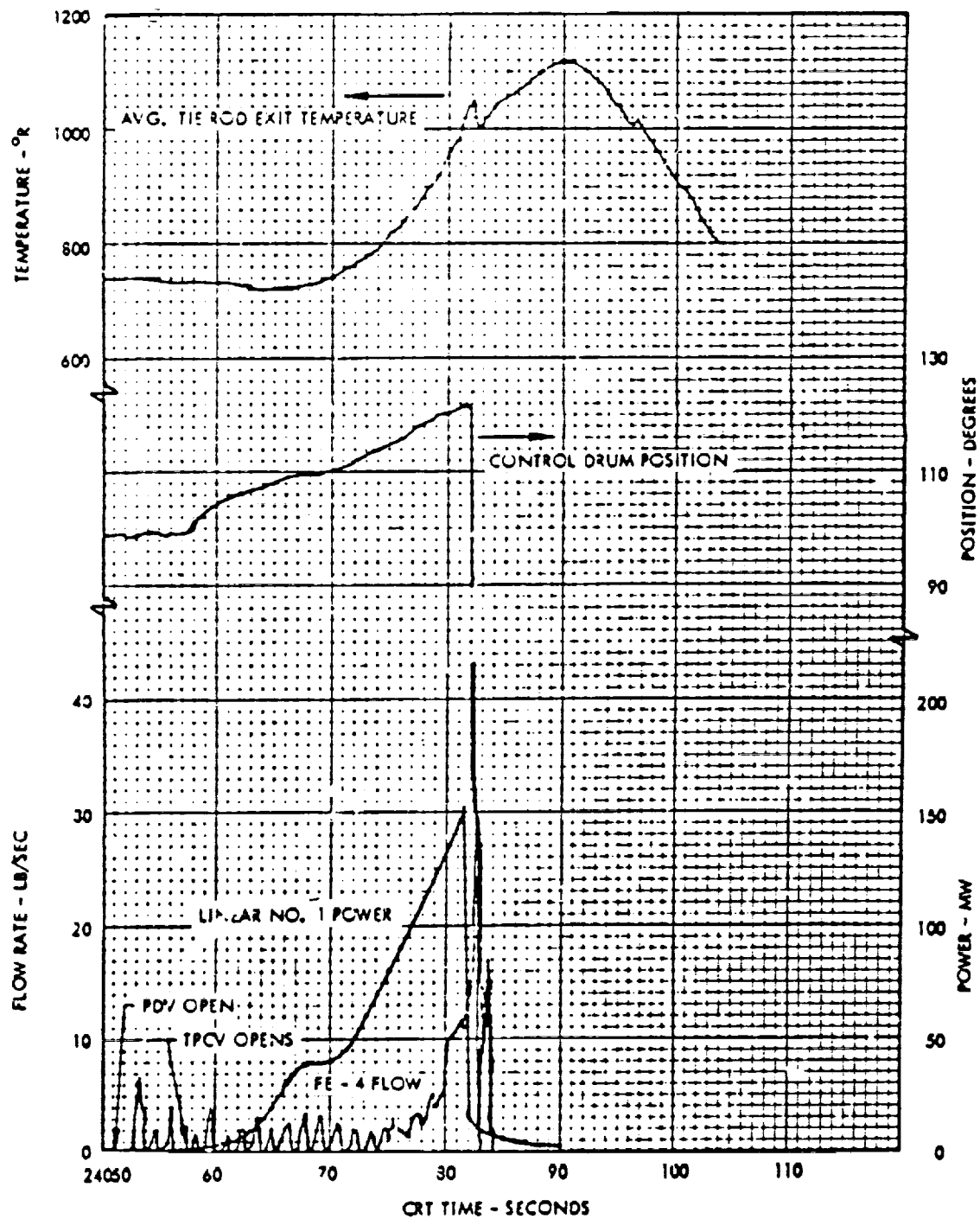
Figure D-57



NRX/EST EP-II C, Abort Run Profile (Sheet 1) - Low De'var Pressure Bootstrap

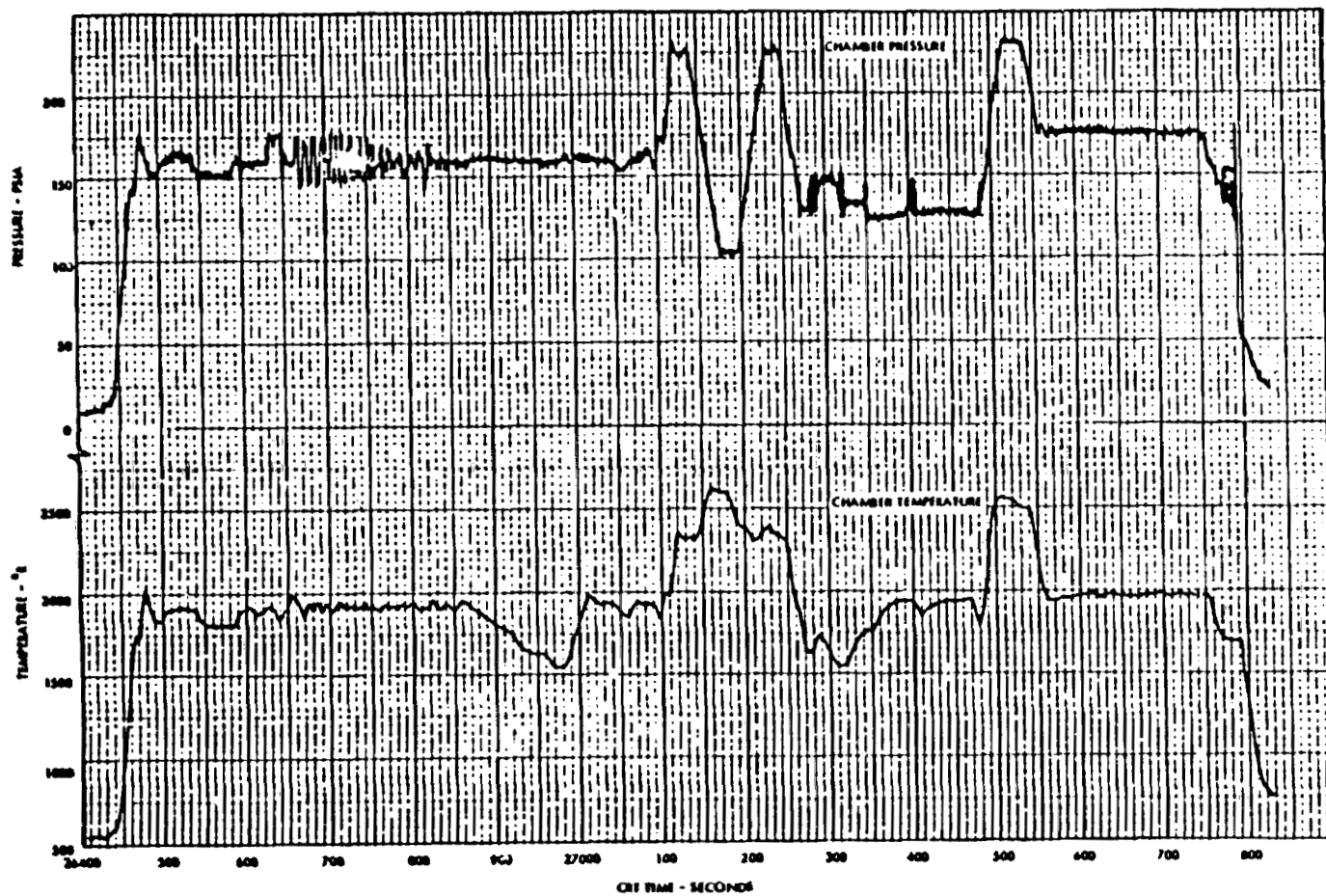
Figure D-58

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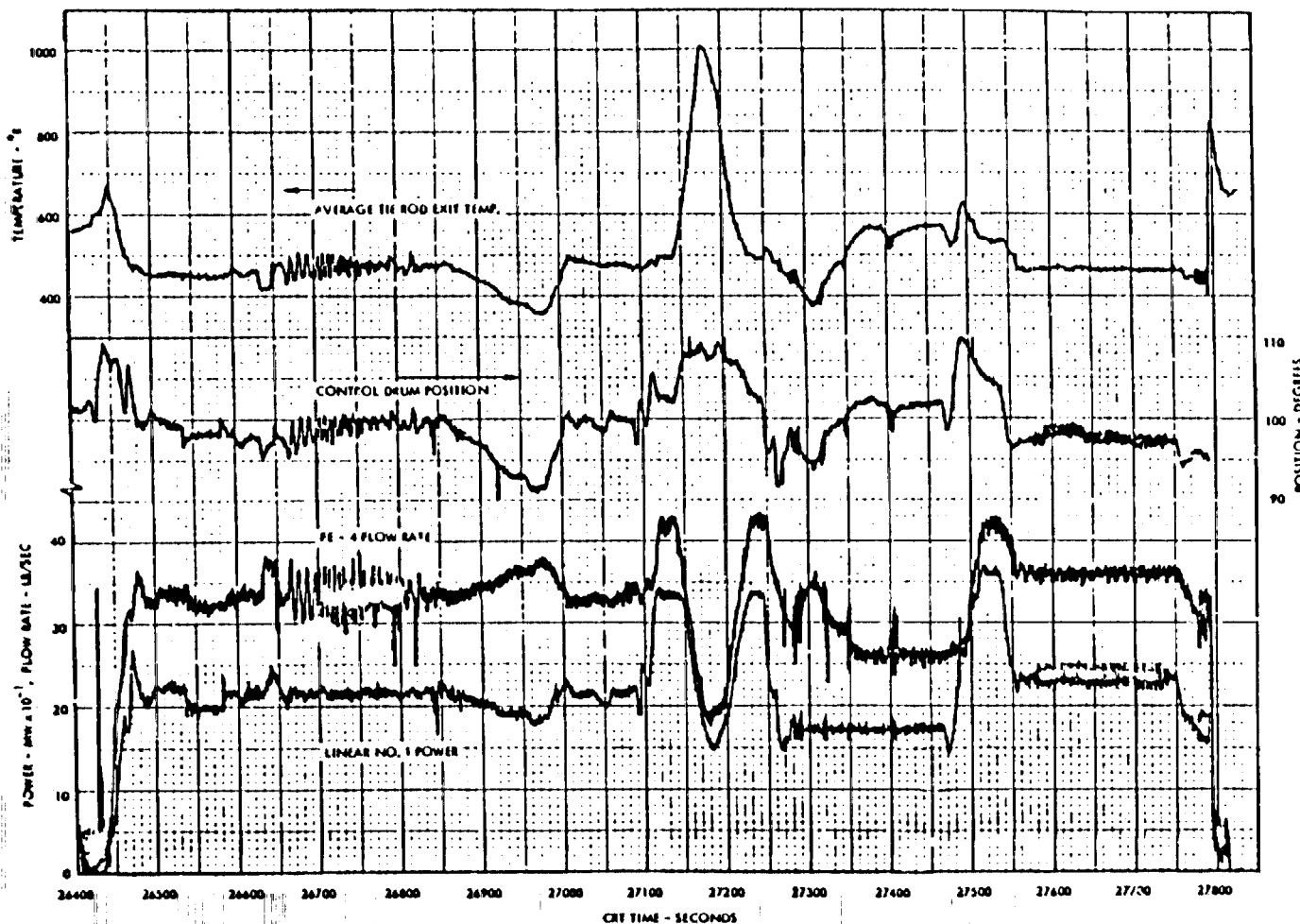
NRX/EST EP-IIIC, Abort Run Profile (Sheet 2) - Low Dewar Pressure Bootstrap

Figure D-59



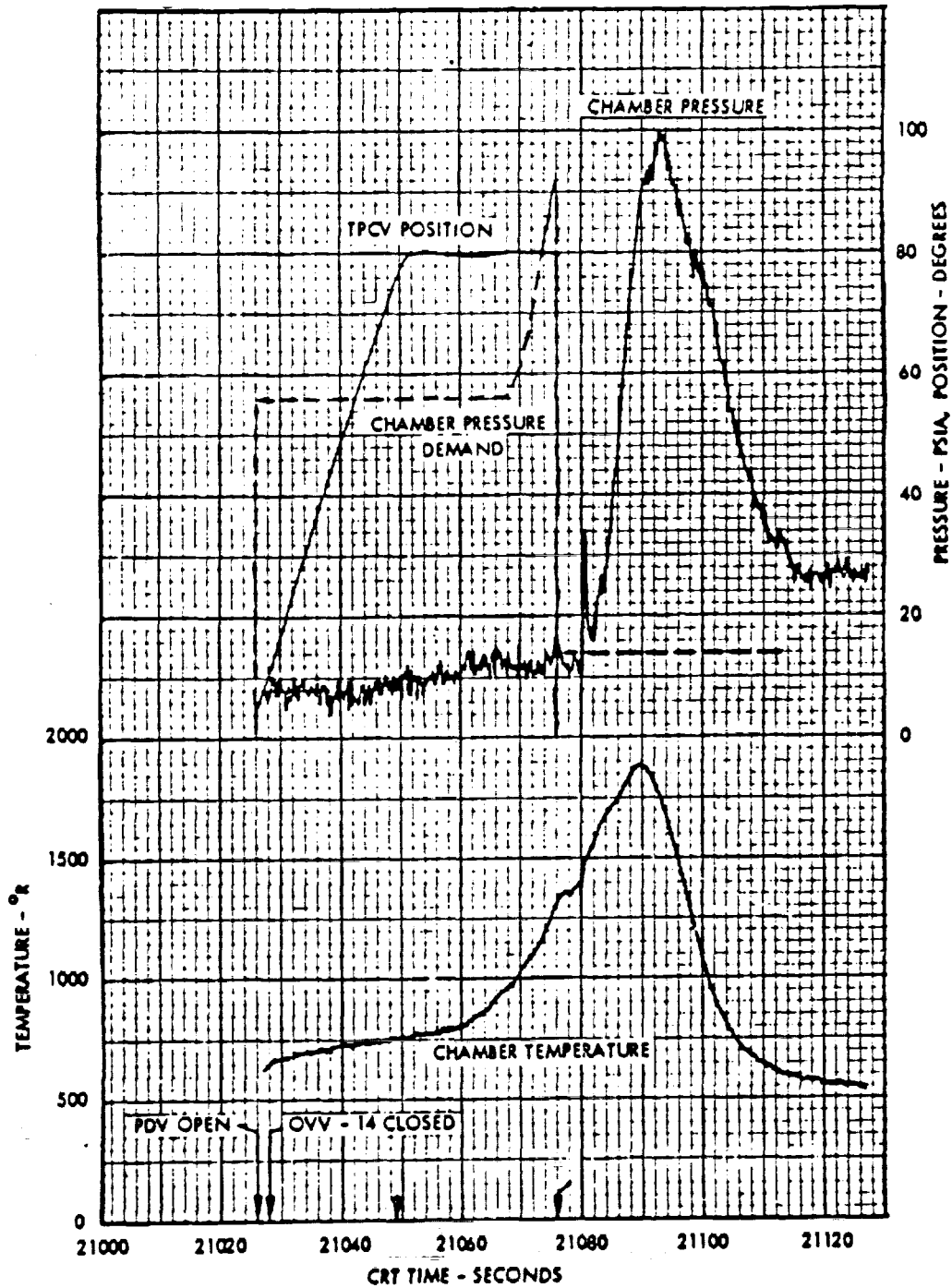
NRX/EST EP-IIC, Run 2 Profile (Sheet 1) - Complete Run

Figure D-60



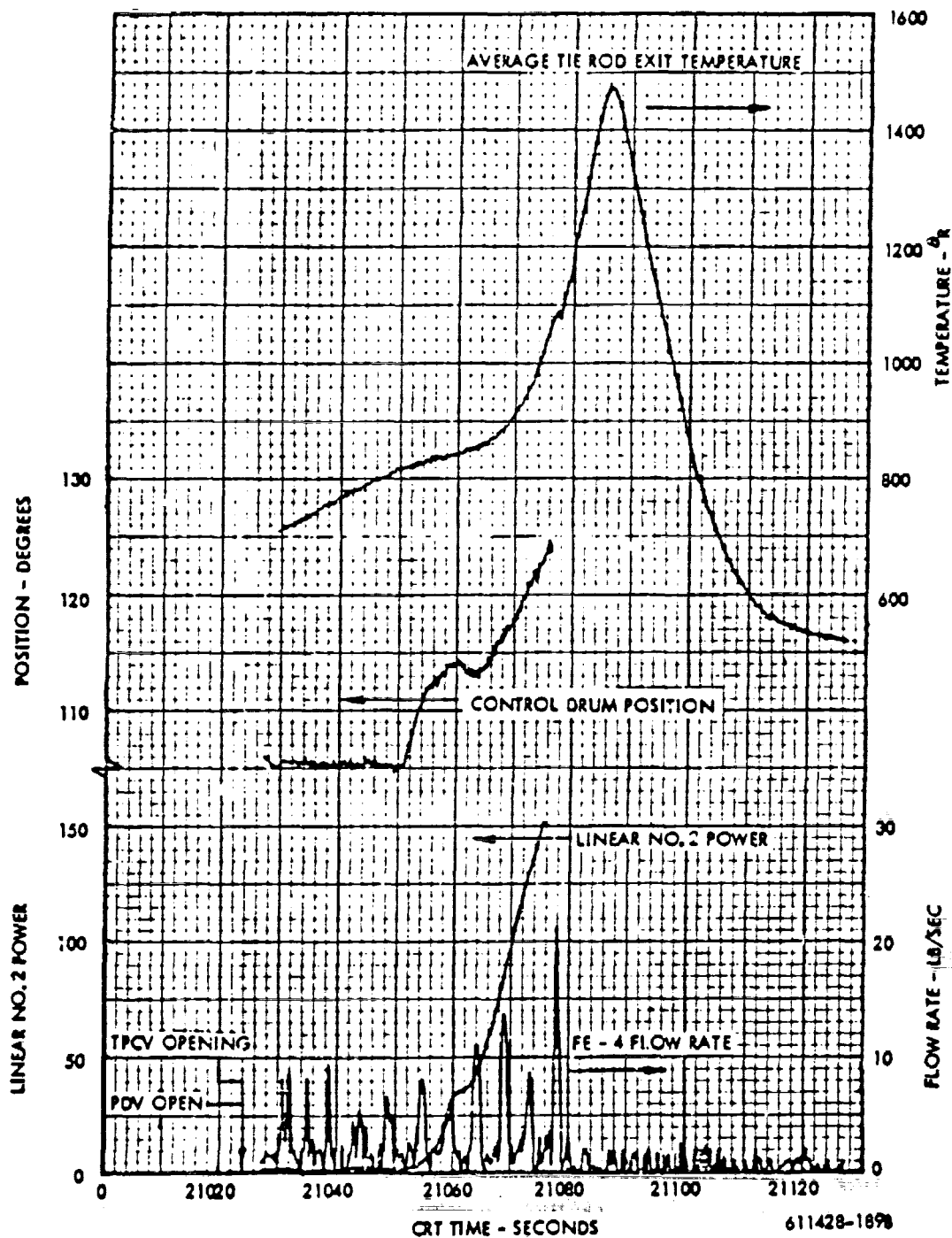
NRX/ESTEP-IIC, Run 2 Profile (Sheet 2) - Complete Run

Figure D-61



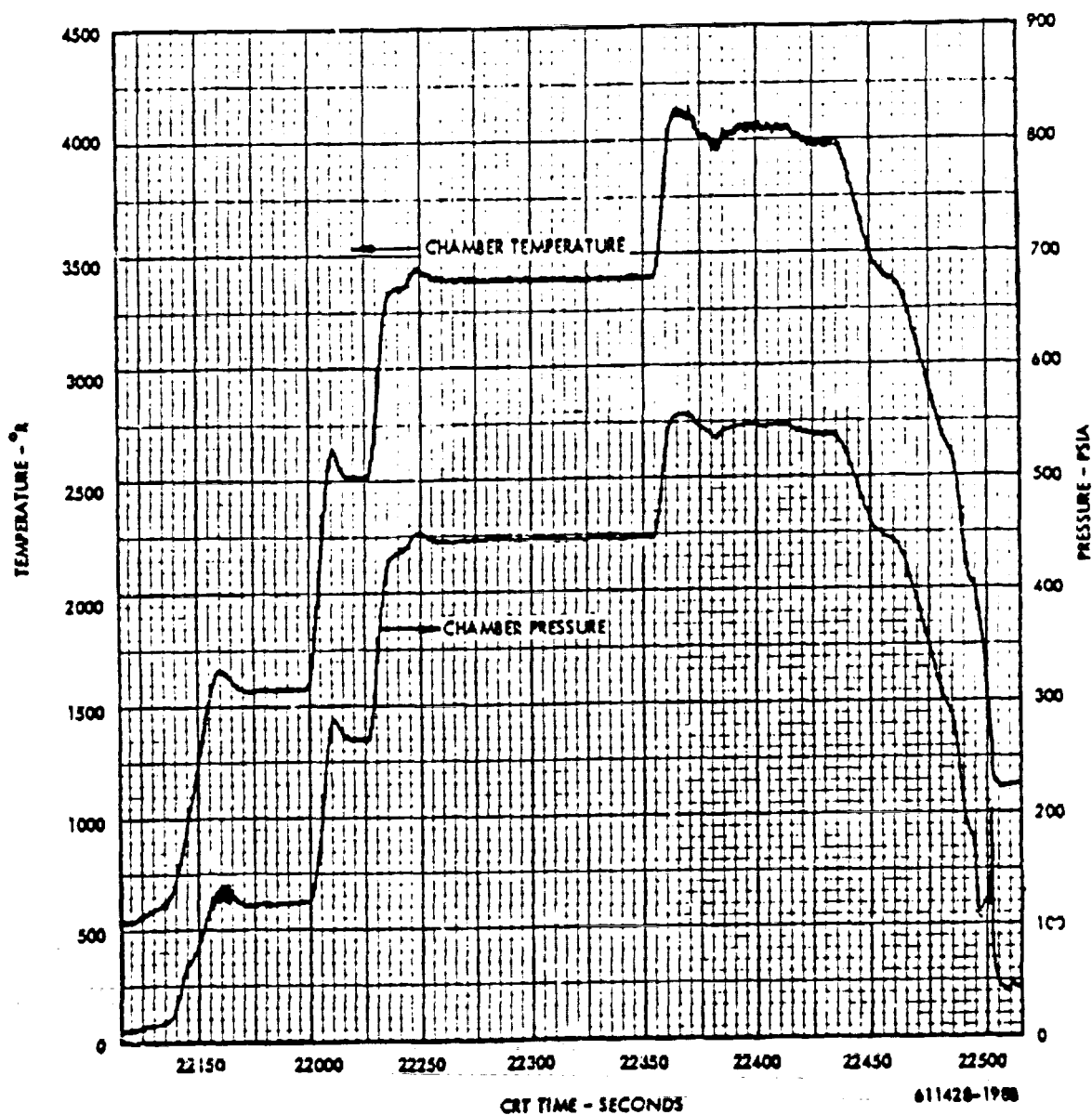
NRX/EST EP-III, Abort Run Profile (Sheet 1) - Low Dewar Pressure Bootstrap

Figure D-62



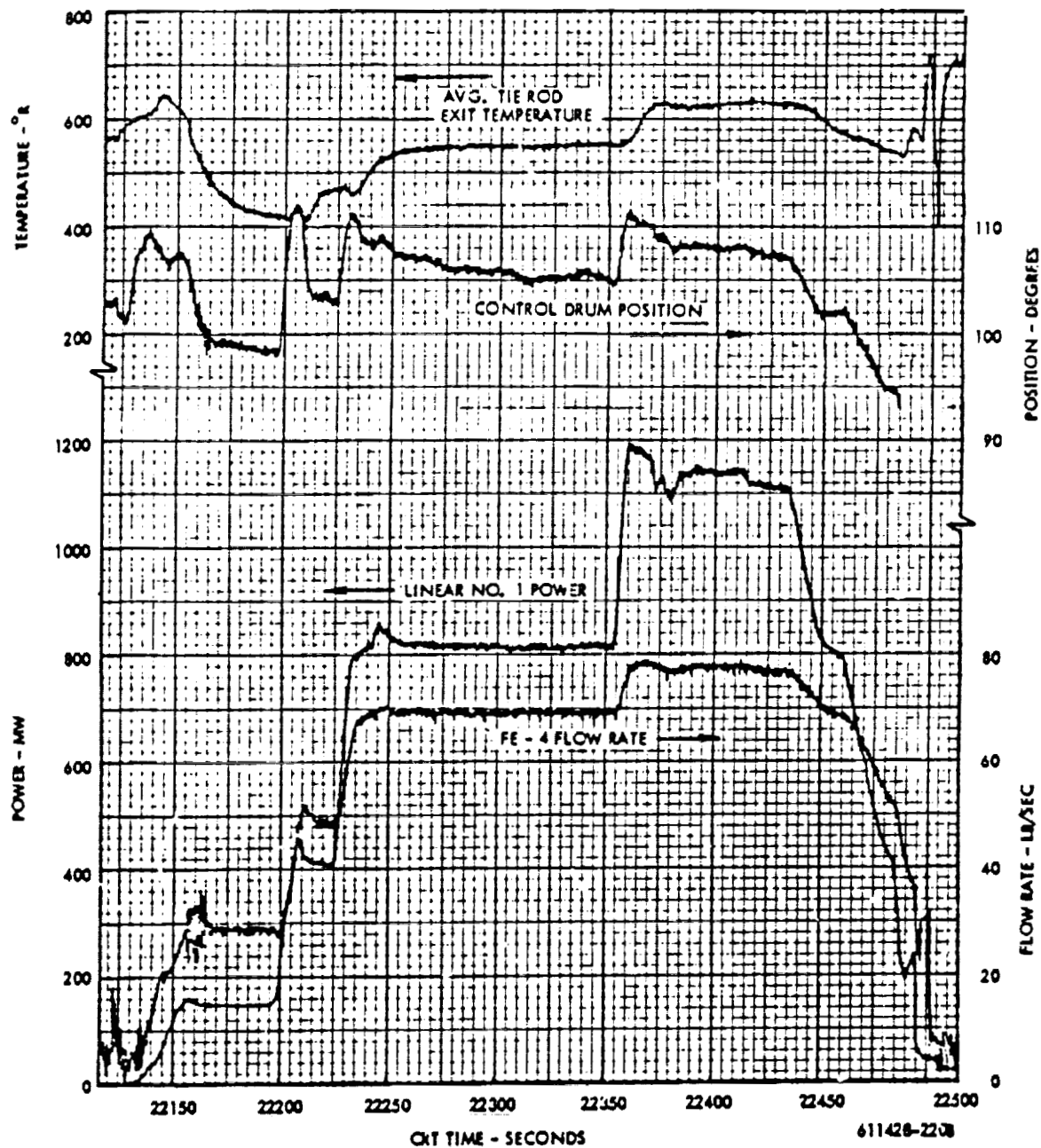
NRX/EST EP-III, Abort Run Profile (Sheet 2) - Low Dewar Pressure Bootstrap

Figure D-63



NRX/EST EP-III, Run 1 Profile (Sheet 1) - Full Power Test

Figure D-64



NRX/EST EP-III, Run 1 Profile (Sheet 2) - Full Power Test

Figure D-65

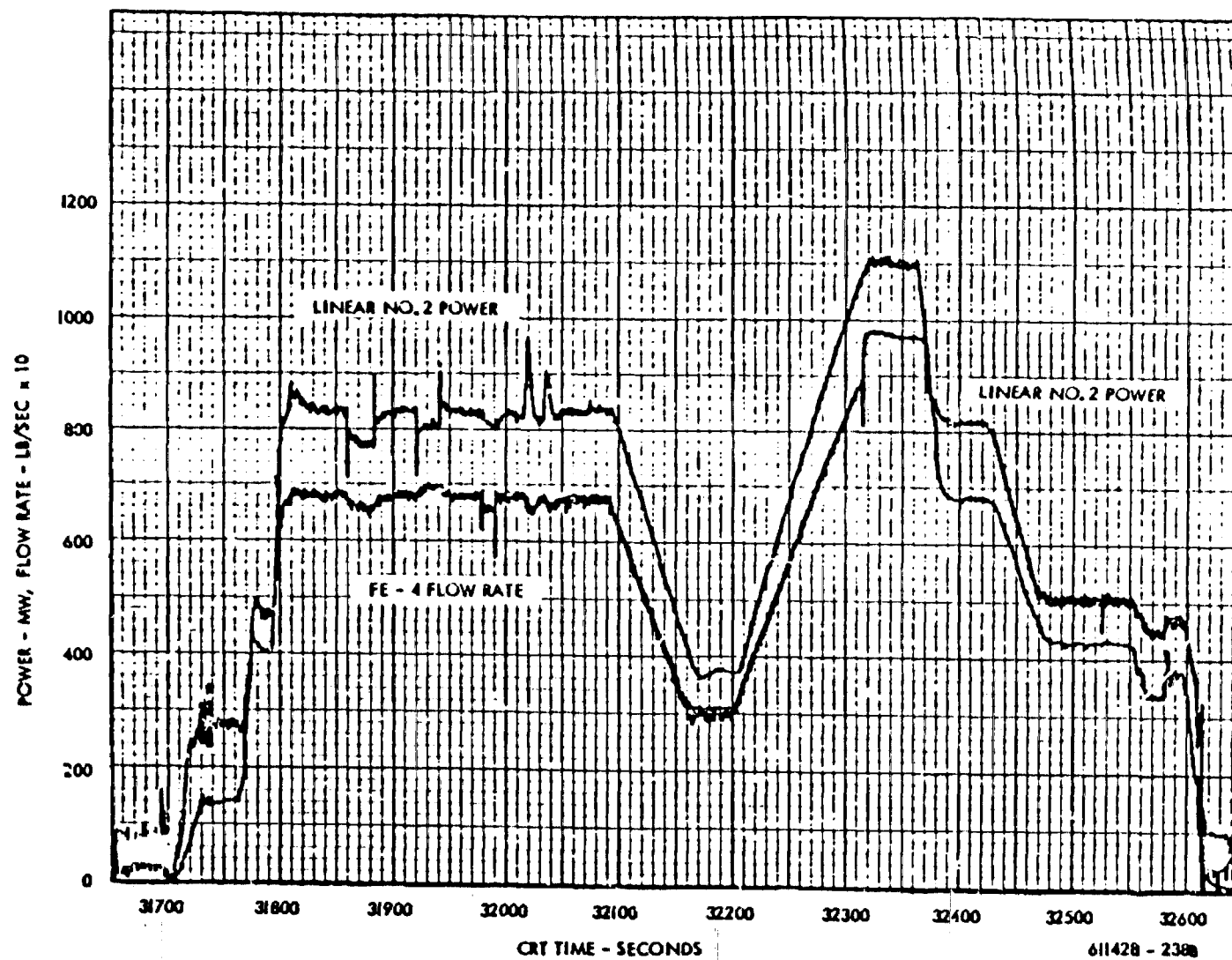
D-82

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NRX/EST EP-III, Run 2 Profile (Sheet 1) - 3400°R Mapping

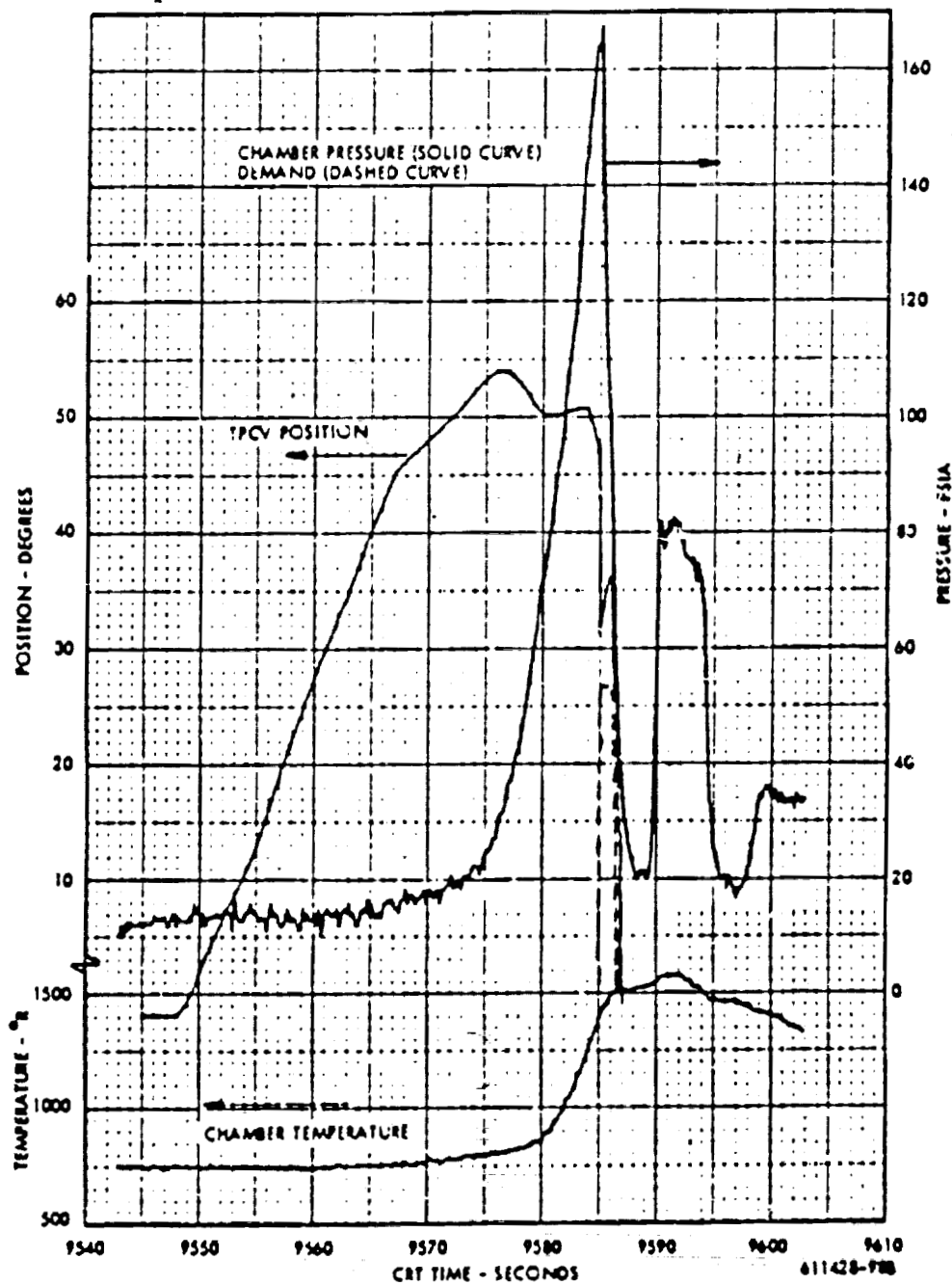
Figure D-66



ASTRONUCLEAR
LABORATORY

NRX/EST EP-III, Run 2 Profile (Sheet 2) - 3400°R Mapping

Figure D-67

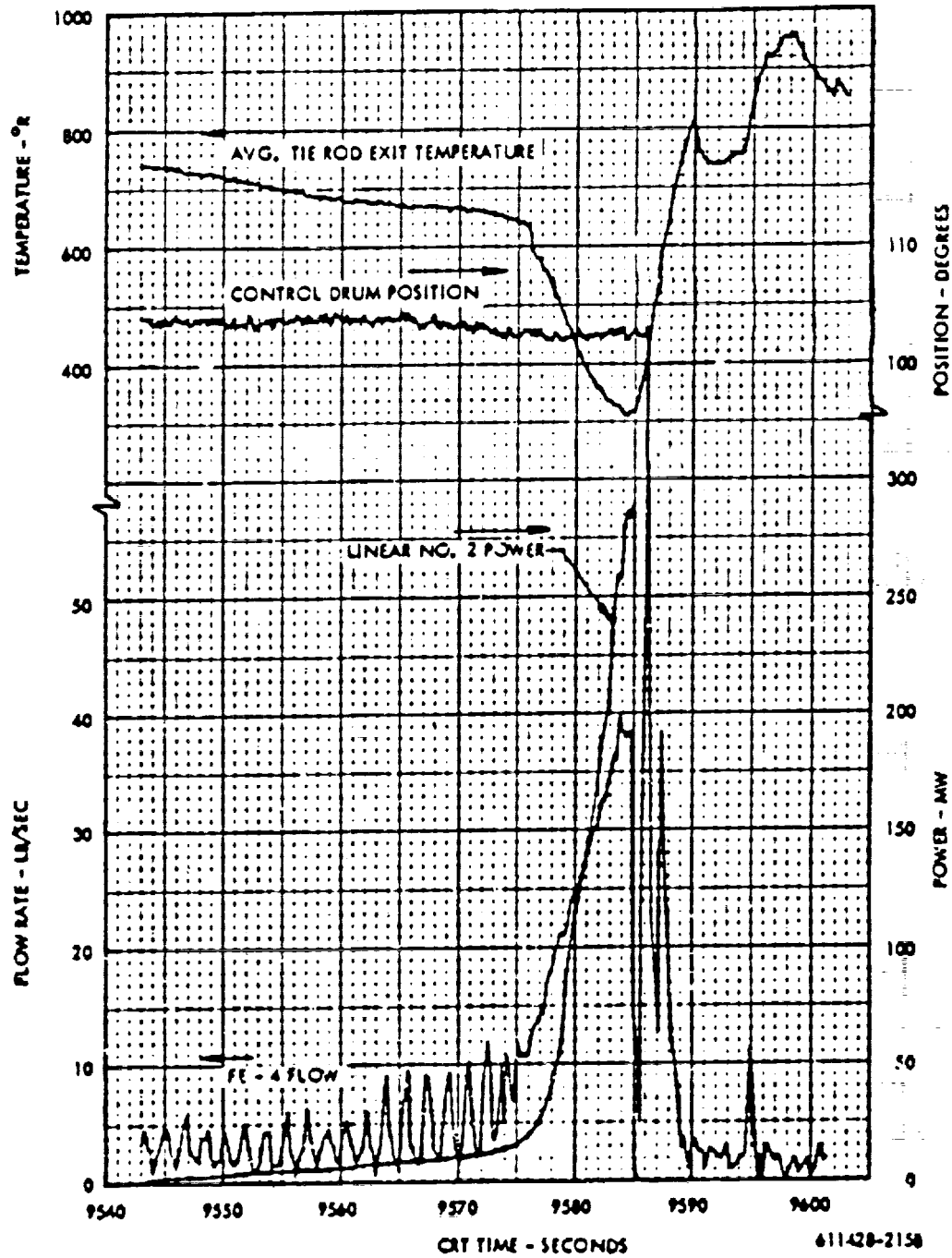


NRX/EST

EP-IV, Abort Run Profile (Sheet 2) - Fixed
Drum Low Dewar Pressure

Figure D-68

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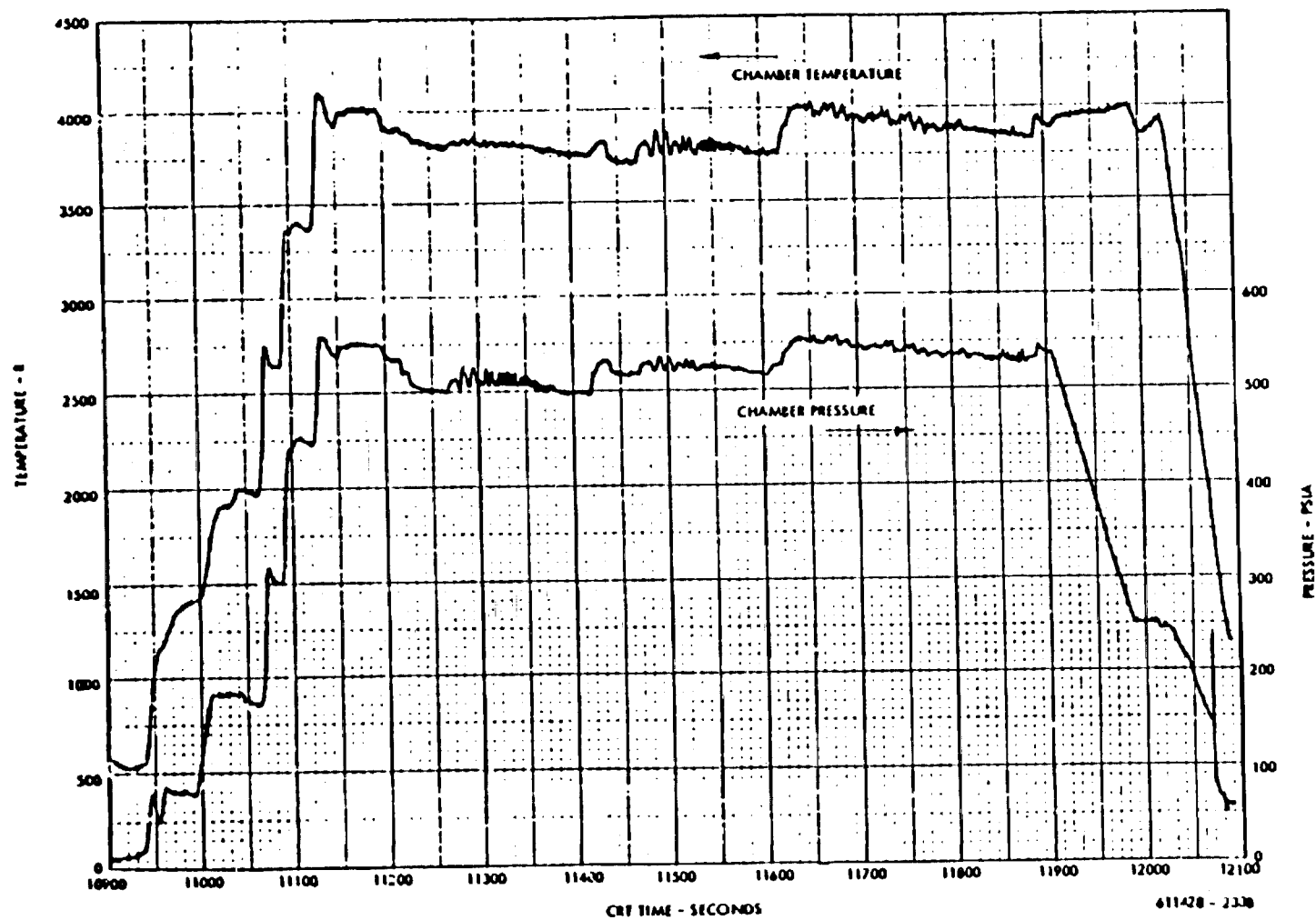


NRX/EST EP-IV, Abort Run Profile (Sheet 2) -
Fixed Drum Low Dewar Pressure

Figure D-69

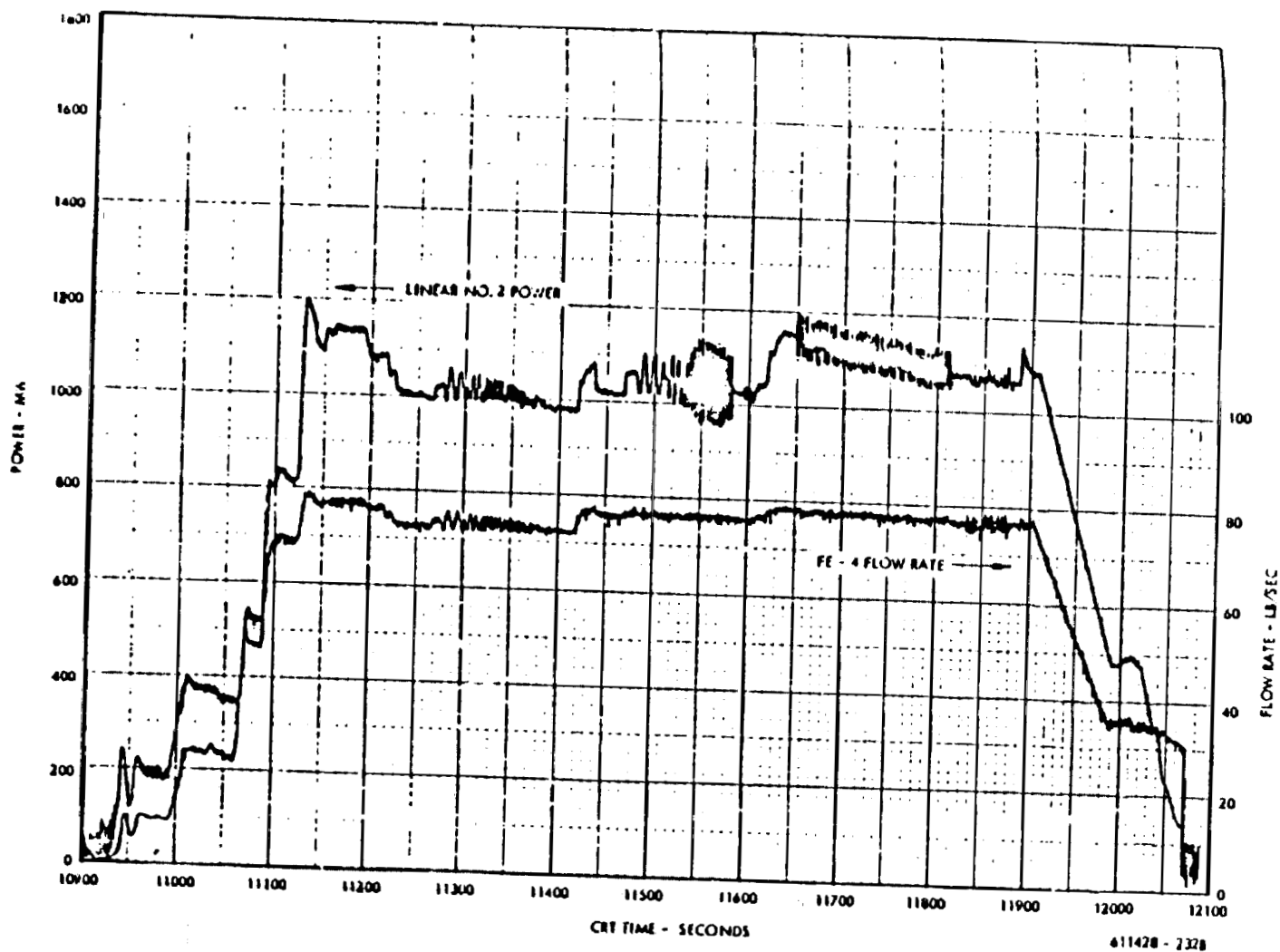
D-86

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NRX/EST EP-IV, Profile (Sheet 1) - Full Power Endurance Test

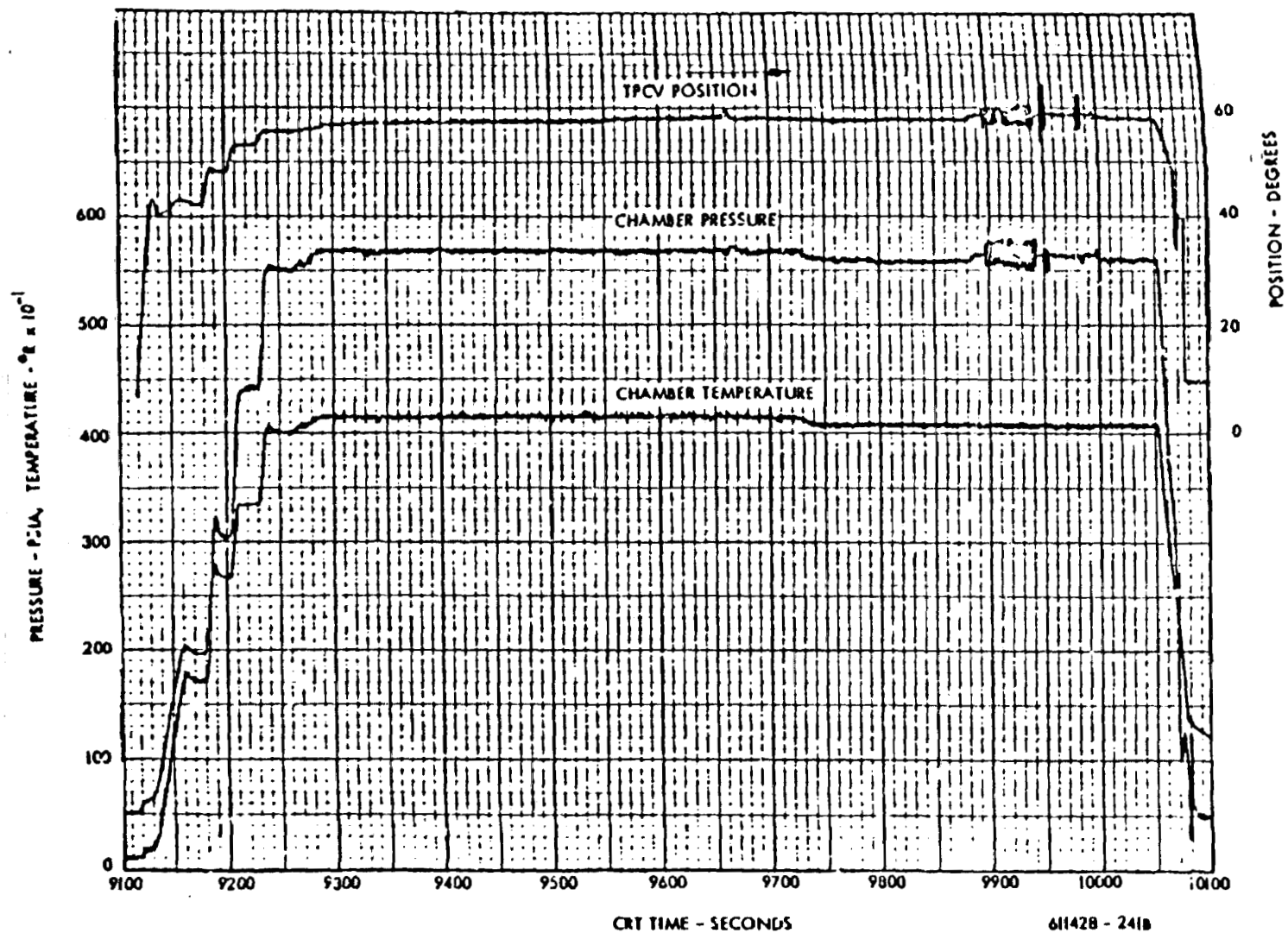
Figure D-70



NRX/EST

EP-IV, Profile (Sheet 2) - Full Power Endurance Test

Figure D-71



NRX/EST

EP-IVA, Profile (Sheet 1) - Full Power Endurance Test

Figure D-72

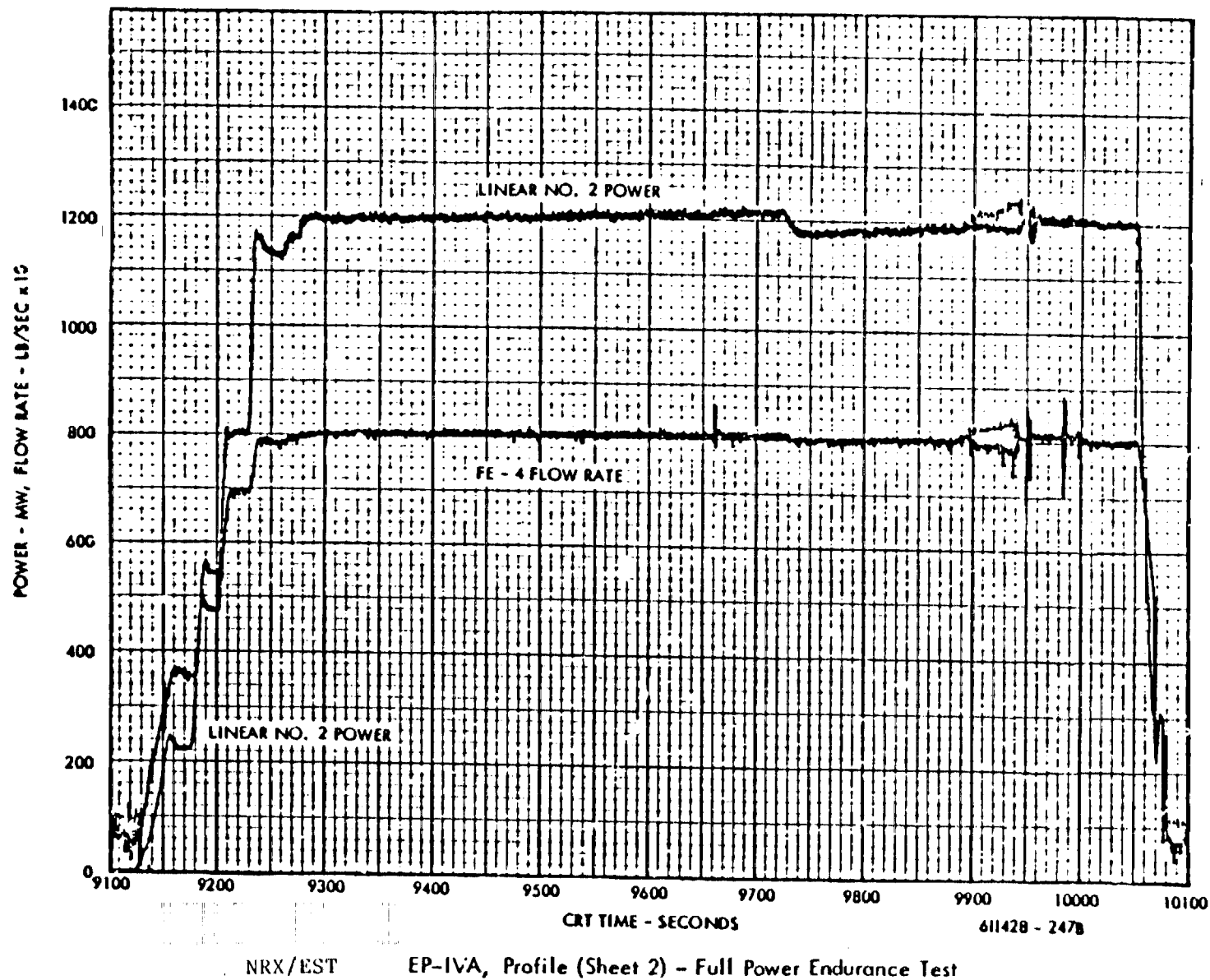
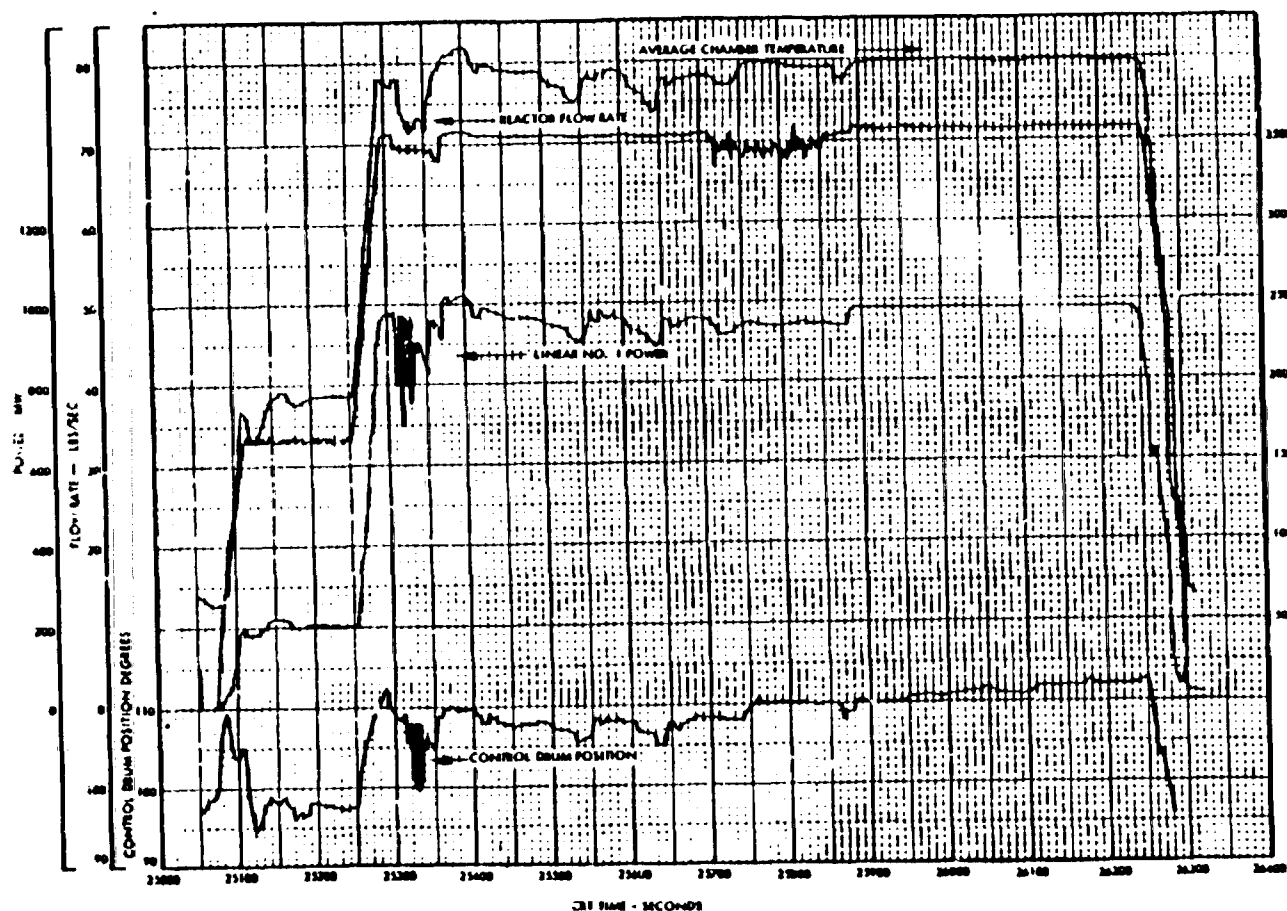


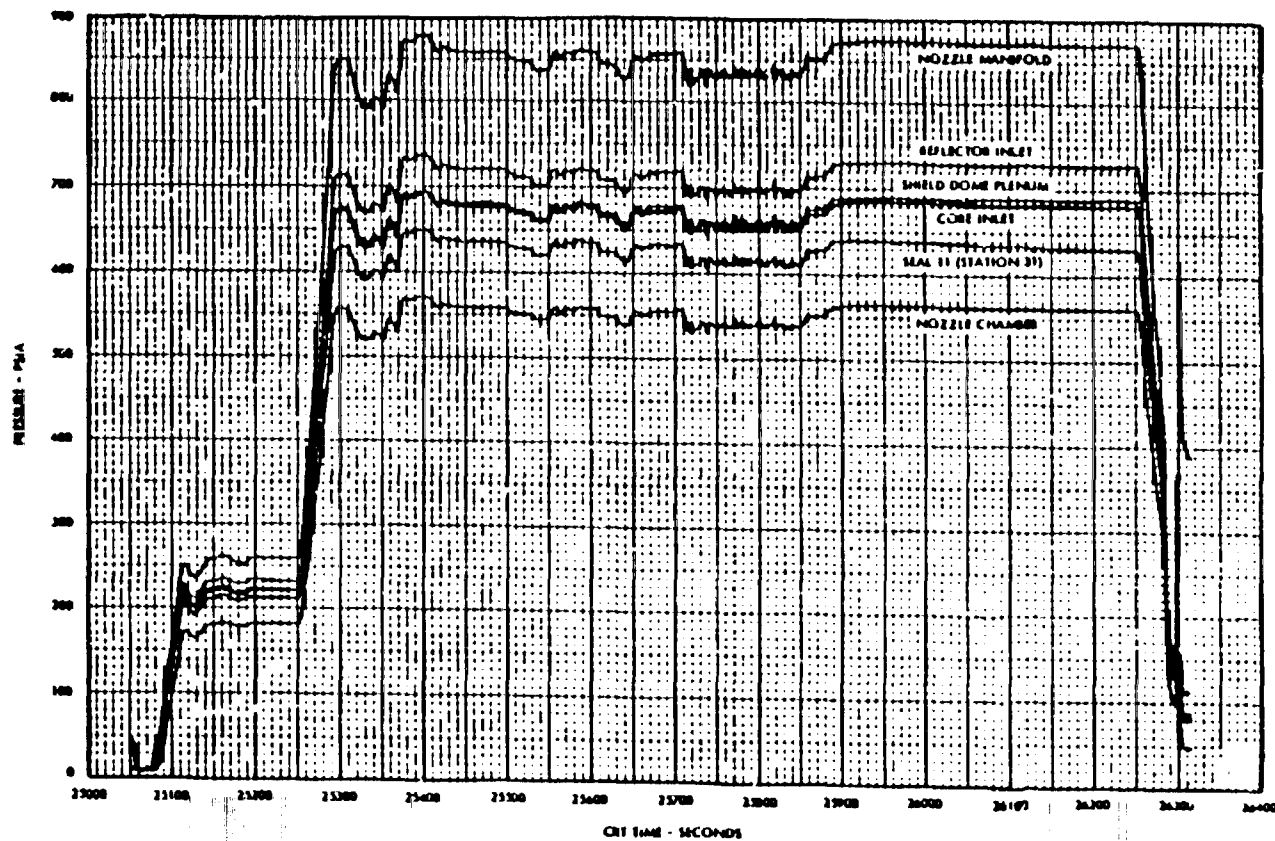
Figure D-73

NRX A5
TEST DATA



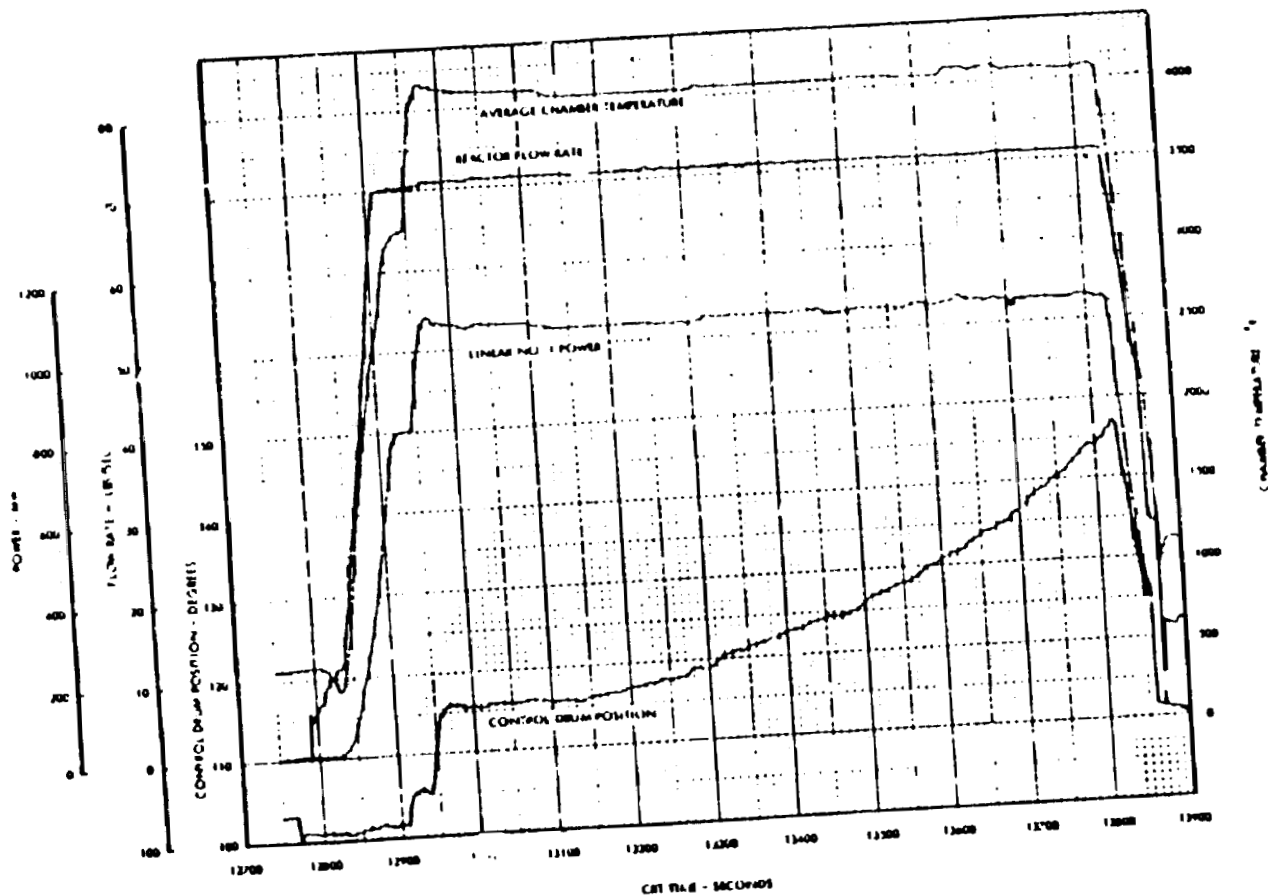
NRX-A5 EP-III Power Test Description: Chamber Temperature,
Flow Rate, Power and Control Drum Position

Figure D-74



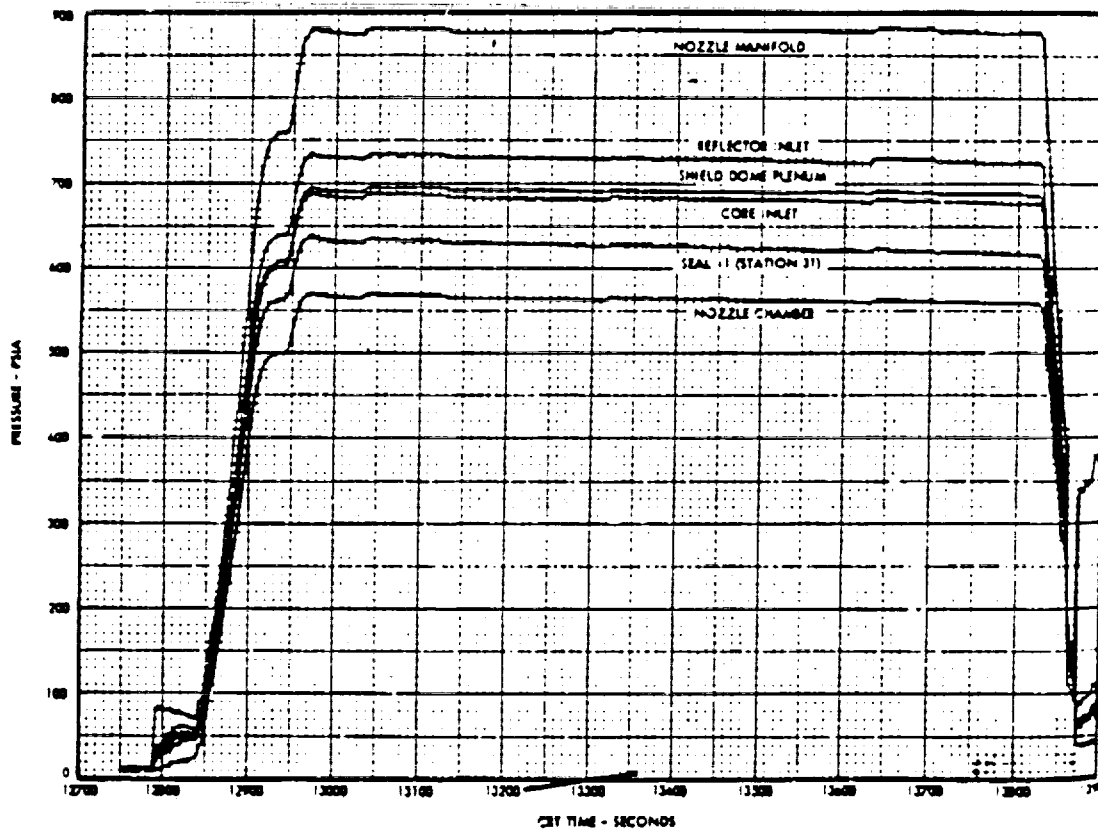
NRX-A5 EP-III Power Test Description:
Plenum Pressures

Figure D-75



NRX-A5 EP-IV Power Test Description:
Chamber Temperature, Flow Rate, Power and Control Drum Position

Figure D-76

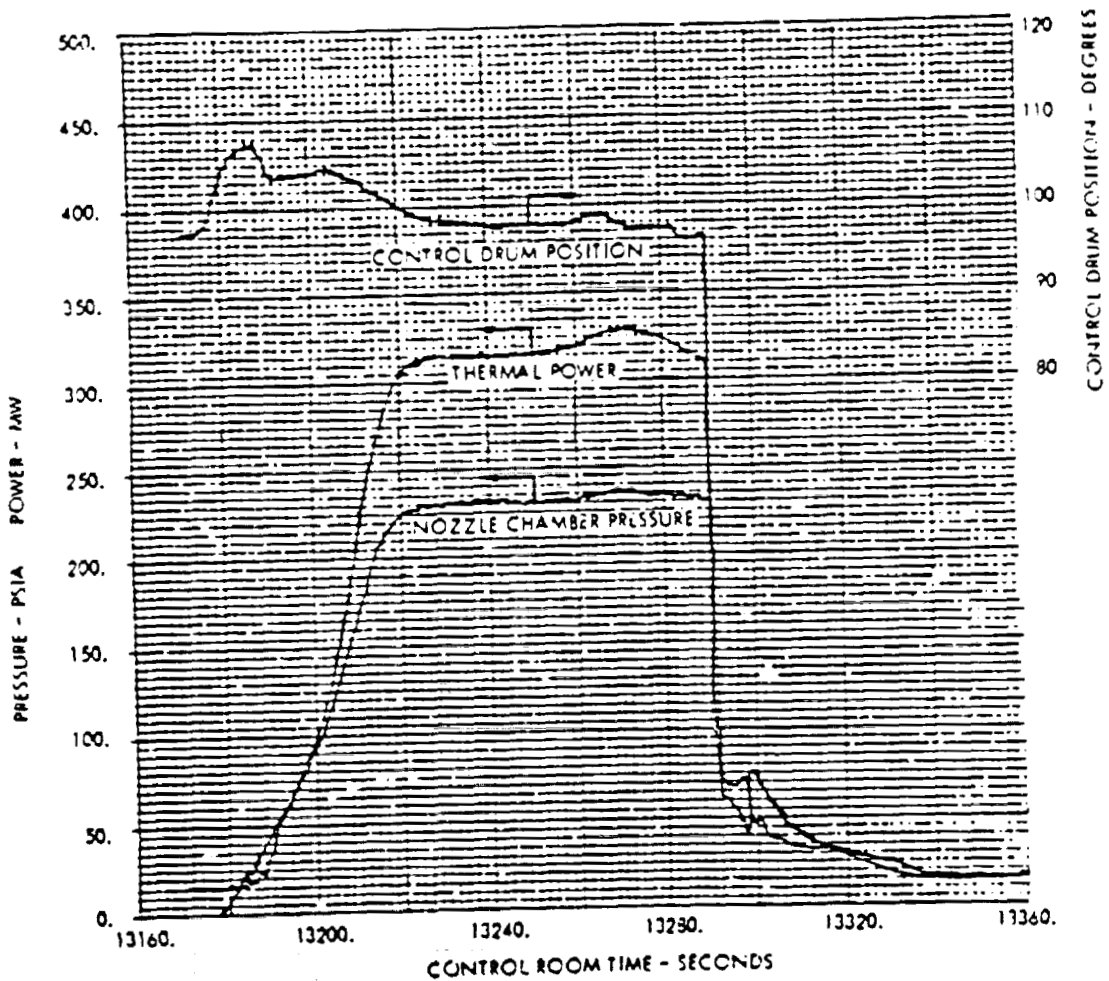


NRX-A5 EP-IV Power Test Description: Plenum Pressures

Figure D-77

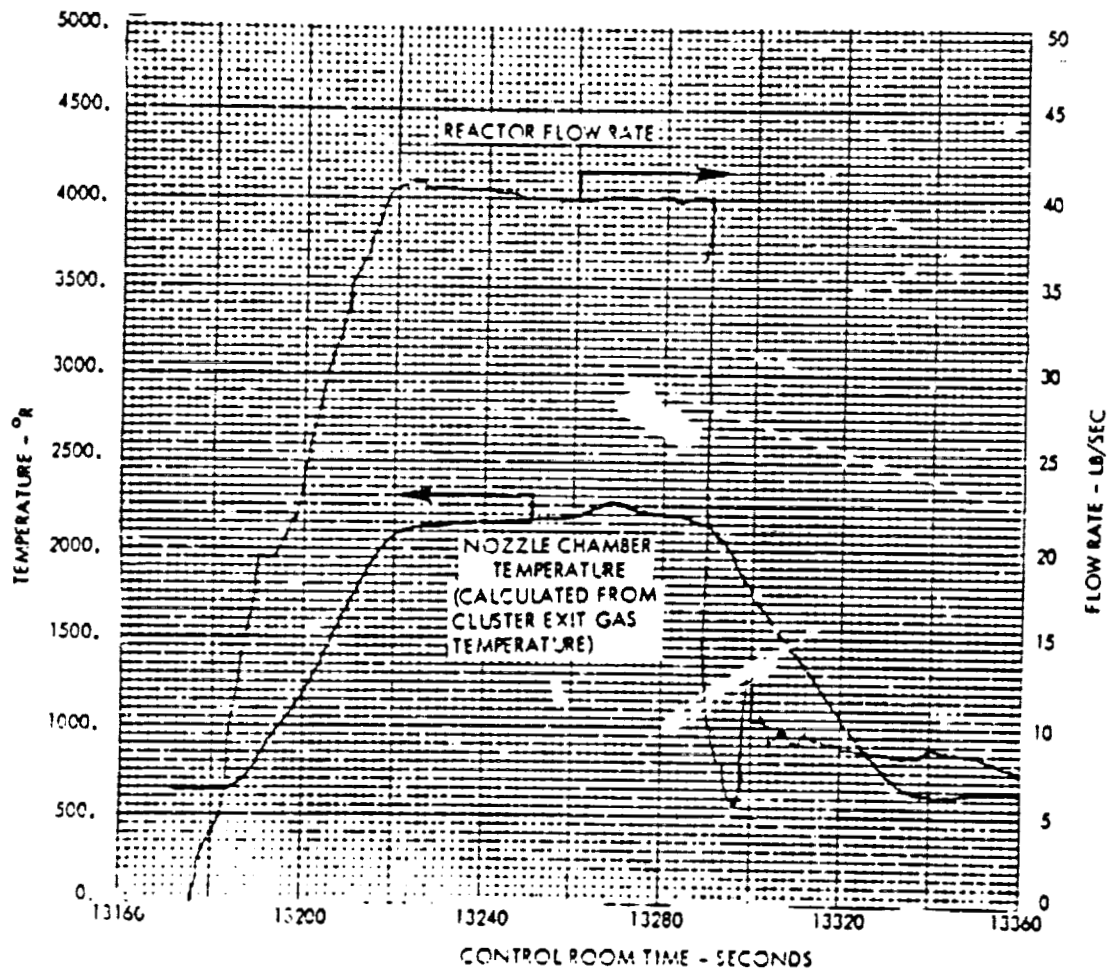
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NRX A6
TEST DATA



NRX A6 (CRD) EP-III - Thermal Power, Control Drum Position and Average Nozzle Chamber Pressure (U) *ABORT RUN*

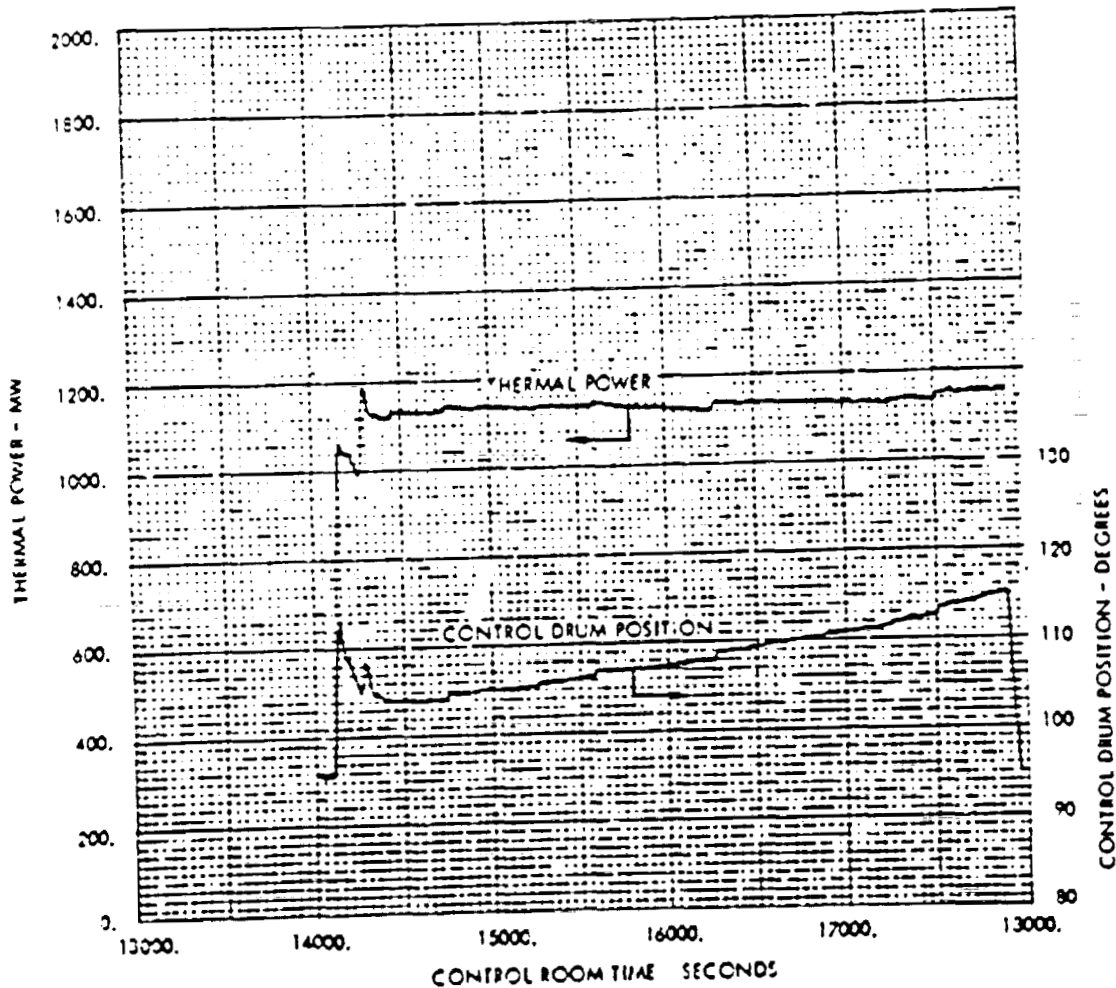
Figure D-78



NRX A6 (U) EP-III Chamber Temperature and Flow Rate
ABORT RUN

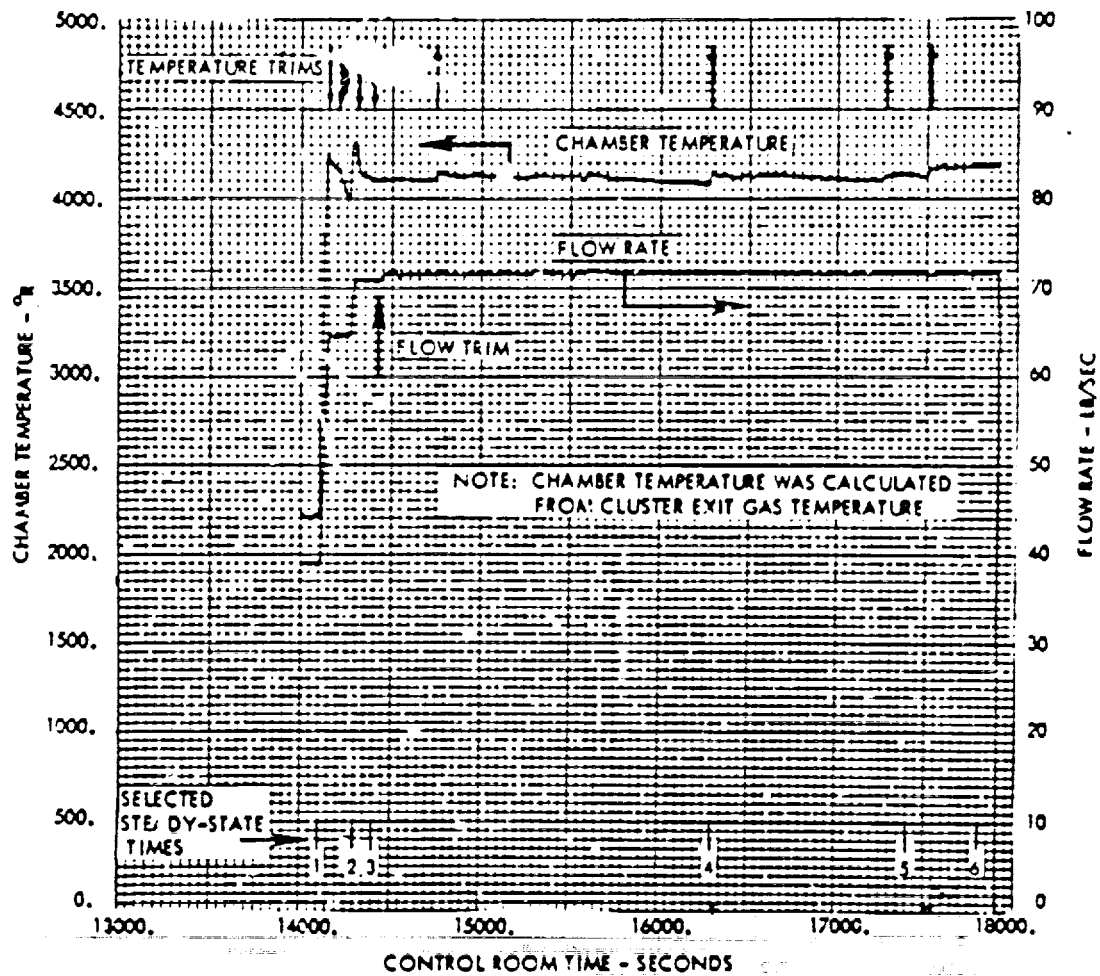
Figure D-79

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NRX A6 (CRD) EP-III A - Thermal Power and Control Drum Position (U)

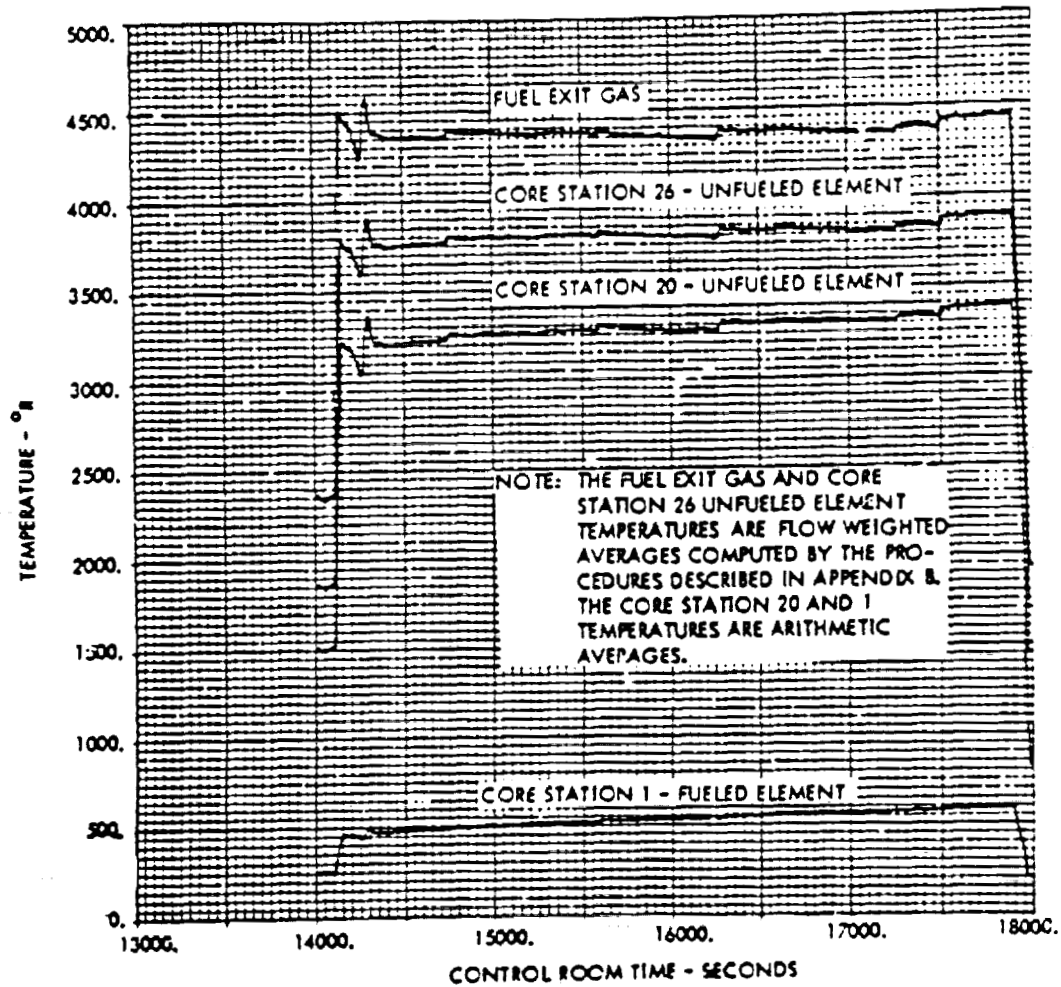
Figure D-80



NRX A6

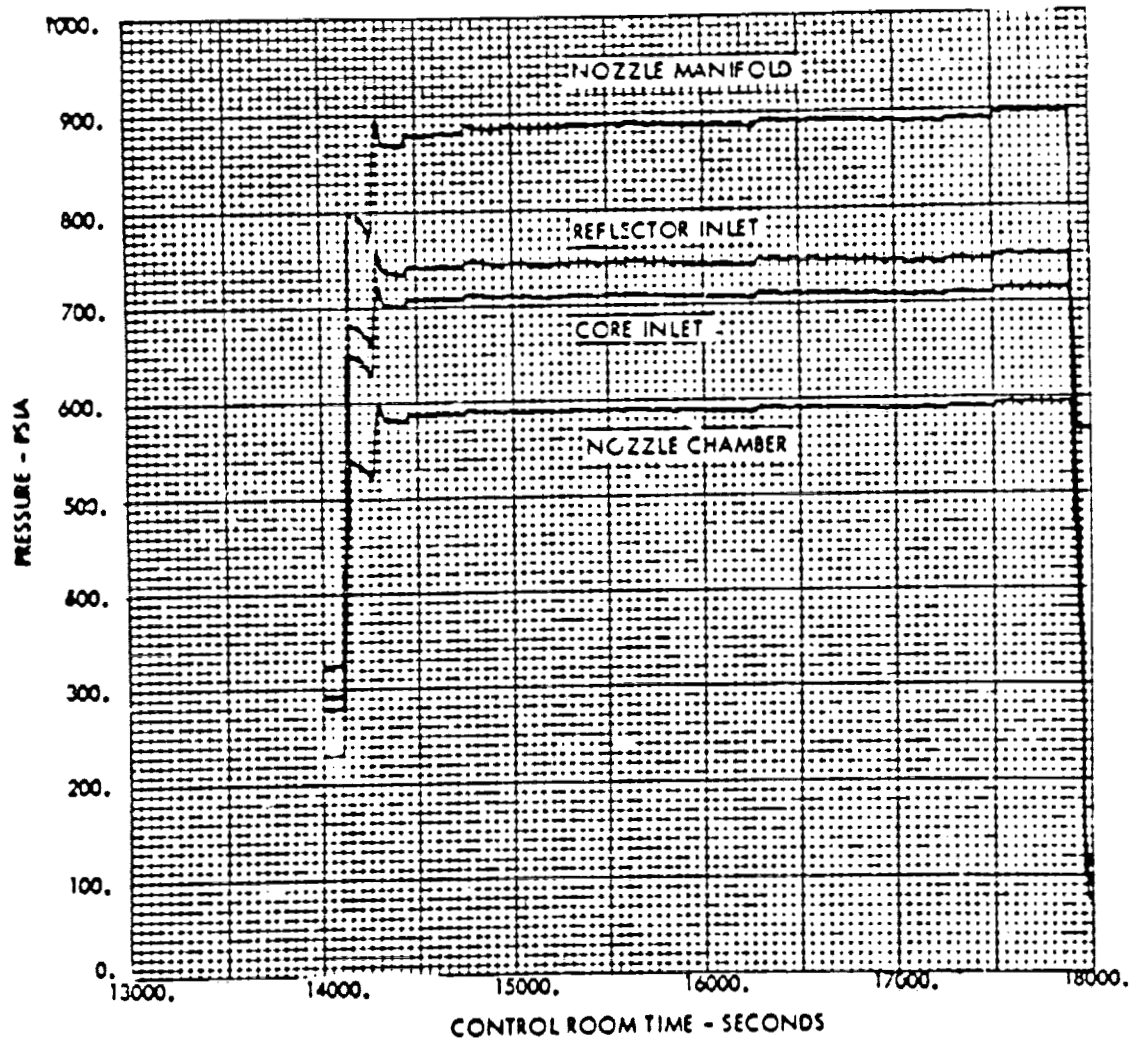
(U) EP-III A - Chamber Temperature and Flow Rate

Figure D-81



NRX A6 (CRD) EP-III A - Average Fuel Exit Gas and Core Station Temperatures (U)

Figure D-82



NRX A6 (U) EP-III A - Plenum Pressures

Figure D-83

XE PRIME

TEST DATA

NOTE: Tabular Data As Presented In Aerojet Report
RN-S-0510 Volume III Book 1

XE-PRIME

EP-6A STEADY-STATE HOLD POINT

$P_c = 51.3$ psia $T_c = 795^\circ\text{R}$ Range Time = 45075 to 45080

Turbine Power Control Valve Position	36.0°
Reactor Power	40.4 Mw
Turbopump Shaft Speed	4315 rpm
Ambient Pressure (Engine Test Compartment)	6.92 psia
Net Positive Suction Pressure	20.2 psi

<u>Station</u>	<u>Flow</u> (lb/sec)	<u>Pressure</u> (psia)	<u>Temperature</u> (Deg R)
Propellant Tank Outlet	16.0	36.1*	37.0
Pump Inlet	16.0	36.5	37.6
Pump Outlet	15.9	74.2	38.4
Nozzle Manifold Inlet	15.9	72.3	41.3
Reflector Inlet	15.9	64.7	49.3
Reflector Outlet	15.9	62.3	---
Shield I Outlet (Dome)	15.9	61.6	---
Core Inlet	---	61.4	---
Nozzle Chamber (T_c)	14.6	51.3	795
Diluent Bleed Inlet (Dome)	---	61.6	51.8
Diluent Bleed Outlet	---	56.2	---
Hot Bleed Port	---	51.3	795
Turbine Inlet	1.34	22.6	100
Turbine 2nd Stage Rotor Exit	1.44	---	---
Turbine Exhaust Nozzle	1.44	7.03	---

*Ullage Pressure at top of tank.

Table D-3

XE-PRIME

EP-6A STEADY-STATE HOLD POINTS

$P_c = 62.1 \text{ psia}$ $T_c = 1107^{\circ}\text{R}$

$P_c = 65.4 \text{ psia}$ $T_c = 1632^{\circ}\text{R}$

Range Time = 44700 to 44705

Range Time = 44565 to 44570

Turbine Power Control Valve Position	35.0	33.6 ⁰
Reactor Power	59.7	79.3 Mw
Turbopump Shaft Speed	5205	5678 rpm
Ambient Pressure (Engine Test Compartment)	6.53	6.43 psia
Net Positive Suction Pressure	20.4	20.5 psi

Station	Flow (lb/sec)	Pressure (psia)	Temperature (⁰ R)	Flow (lb/sec)	Pressure (psia)	Temperature (⁰ R)
Propellant Tank Outlet	16.7	36.2*	36.9	14.4	36.3*	36.9
Pump Inlet	16.7	36.6	37.4	14.4	36.7	37.4
Pump Outlet	16.6	90.1	38.4	14.2	99.9	38.5
Nozzle Manifold Inlet	16.6	88.1	41.6	14.2	98.4	43.1
Reflector Inlet	16.6	79.3	51.6	14.2	87.3	52.9
Reflector Outlet	16.6	77.1	---	14.2	84.1	---
Shield I Outlet (Dome)	16.6	76.3	---	14.2	82.8	73.1
Core Inlet	15.3	75.9	---	13.1	81.8	90.2
Nozzle Chamber (T _c)	15.0	62.1	1107	13.0	65.4	1691
Diluent Bleed Inlet (Dome)	1.29	76.3	---	1.16	82.8	73.1
Diluent Bleed Outlet	1.29	70.4	---	1.16	76.3	86.1
Hot Bleed Port	0.33	62.1	1131	0.02	65.4	1632
Turbine Inlet	1.57	27.26	110	1.18	26.0	162
Turbine 2nd Stage Rotor Exit	2.60	6.56	88	1.31	---	132.5
Turbine Exhaust Nozzle	2.60	6.54	83	1.31	6.42	---

*Ullage pressure at top of tank.

Table D-4

XE-PRIME

EP-4A STEADY-STATE HOLD POINT

$P_c = 93 \text{ psia}$ $T_c = 1386^\circ\text{R}$ Range Time = 61255 to 61260

	<u>Actual</u>	<u>Predicted</u>
Turbine Power Control Valve Position	36.8	40.3 ^o
Reactor Power	100	102 Mw
Turbopump Shaft Speed	7147	7096 rpm
Ambient Pressure (Engine Test Compartment)	5.77	6.28 psia
Net Positive Suction Pressure	16.6	19.9 psi

<u>Station</u>	<u>Flow (lb/sec)</u>		<u>Pressure (psia)</u>		<u>Temperature (Deg R)</u>	
	<u>Actual</u>	<u>Predicted</u>	<u>Actual</u>	<u>Predicted</u>	<u>Actual</u>	<u>Predicted</u>
Propellant Tank Outlet	22.7	22.4	35.5*	35.0*	37.2	36.4
Pump Inlet	22.7	22.4	35.6	34.6	38.1	36.5
Pump Outlet	22.3	22.2	135	132	39.6	37.9
Nozzle Manifold Inlet	22.3	22.2	131	127	---	38.0
Reflector Inlet	22.3	22.2	118	113	55	53
Reflector Outlet	22.3	22.2	---	110	---	55
Shield I Outlet (Dome)	22.3	22.2	114	110	55	55
Core Inlet	20.2	20.3	113	109	56	58
Nozzle Chamber (T_c)	20.1	20.1	93	90	1386	1400
Diluent Bleed Inlet (Dome)	1.95	1.95	114	110	55	55
Diluent Bleed Outlet	1.95	1.95	107	104	61	57
Hot Bleed Port	.16	.13	93	90	1386	1400
TPCV Inlet	2.11	2.08	88	90	136	156
Turbine Inlet	2.11	2.08	44	44	136	156
Turbine 2nd Stage Rotor Exit	2.54	2.33	---	6	---	136
Turbine Exhaust Nozzle	2.54	2.33	5.5	6	108	118

*Ullage pressure.

XE-PRIME

EP-8A STEADY-STATE HOLD POINT

$P_c = 119 \text{ psia}$ $T_c = 1693^\circ\text{R}$ Range Time = 59308 to 59313

	<u>Actual</u>	<u>Predicted</u>
Turbine Power Control Valve Position	37.5	41.3 ^o
Reactor Power	144	149 Mw
Turbopump Shaft Speed	8770	8760 rpm
Ambient Pressure (Engine Test Compartment)	4.49	5.53 psia
Net Positive Suction Pressure	10.2	20.0 psi

<u>Station</u>	<u>Flow (lb/sec)</u>		<u>Pressure (psia)</u>		<u>Temperature (Deg R)</u>	
	<u>Actual</u>	<u>Predicted</u>	<u>Actual</u>	<u>Predicted</u>	<u>Actual</u>	<u>Predicted</u>
Propellant Tank Outlet	26.3	26.7	35.1*	35.0*	40.0	36.8
Pump Inlet	26.3	26.7	35.5	35.6	40.3	36.9
Pump Outlet	26.2	26.4	182	184	42.4	39.3
Nozzle Manifold Inlet	26.2	26.4	176	176	44.7	39.4
Reflector Inlet	26.2	26.4	157	156	60.9	59
Reflector Outlet	26.2	26.4	151	151	---	82
Shield I Outlet (Dome)	26.2	26.4	150	150	64	83
Core Inlet	24.0	24.4	149	152	72	83
Nozzle Chamber (T_c)	23.8	24.2	119	120	1693	1700
Diluent Bleed Inlet (Dome)	2.09	2.04	150	150	64	83
Diluent Bleed Outlet	2.09	2.04	142	143	80	88
Hot Bleed Port	.23	.21	119	120	1693	1700
TPCV Inlet	2.32	2.24	117	118	210	248
Turbine Inlet	2.32	2.24	60.6	60	214	248
Turbine 2nd Stage Rotor Exit	2.48	2.54	6.7	8	145	216
Turbine Exhaust Nozzle	2.48	2.54	5.1	5.53	162	194

*Ullage pressure.

XE-PRIME

EP-5C STEADY-STATE HOLD POINT

$P_c = 151 \text{ psia}$ $T_c = 1957^\circ\text{R}$ Range Time = 39200 to 39205

	<u>Actual</u>	<u>Predicted</u>
Turbine Power Control Valve Position	40.5	43.3 ^o
Reactor Power	204	203 Mw
Turbopump Shaft Speed	10325	9871 rpm
Ambient Pressure (Engine Test Compartment)	243	4.29 psia
Net Positive Suction Pressure	16.4	21.5 psi

<u>Station</u>	<u>Flow (lb/sec)</u>		<u>Pressure (psia)</u>		<u>Temperature (Deg R)</u>	
	<u>Actual</u>	<u>Predicted</u>	<u>Actual</u>	<u>Predicted</u>	<u>Actual</u>	<u>Predicted</u>
Propellant Tank Outlet	32.7	30.9	35.4	35.0	37.8	36.8
Pump Inlet	32.7	30.9	36.0	37.0	38.3	36.8
Pump Outlet	32.5	30.5	244	226	41.4	40.1
Nozzle Manifold Inlet	31.5	30.1	238	217	---	40.2
Reflector Inlet	31.5	30.1	207	192	68	61
Reflector Outlet	31.5	30.1	199	186	---	85
Shield I Outlet (Dome)	31.5	30.1	199	185	94	86
Core Inlet	29.2	27.6	198	184	109	92
Nozzle Chamber (T_c)	28.8	27.3	151	150	1957	2000
Diluent Bleed Inlet (Dome)	2.31	2.46	199	185	94	86
Diluent Bleed Outlet	2.31	2.46	184	177	109	91
Hot Bleed Port	.4	.29	151	150	1957	2000
TPCV Inlet	2.71	2.75	150	144	---	297
Turbine Inlet	2.71	2.75	88.3	81	304	298
Turbine 2nd Stage Rotor Exit	294	3.08	9.7	11	---	214
Turbine Exhaust Nozzle	294	3.08	6.7	4.3	233	205

Table D-7

XE - PRIME

EP-5C STEADY-STATE HOLD POINT

$P_c = 192$ psia $T_c = 2388^{\circ}\text{R}$ Range Time = 38630 to 38635

	<u>Actual</u>	<u>Predicted</u>
Turbine Power Control Valve Position	40.5	43.8 ⁰
Reactor Power	284	283 Mw
Turbopump Shaft Speed	11541	11531 rpm
Ambient Pressure (Engine Test Compartment)	1.19	1.73 psia
Net Positive Suction Pressure	21.3	19.7 psi

<u>Station</u>	<u>Flow (lb/sec)</u>		<u>Pressure (psia)</u>		<u>Temperature (Deg R)</u>	
	<u>Actual</u>	<u>Predicted</u>	<u>Actual</u>	<u>Predicted</u>	<u>Actual</u>	<u>Predicted</u>
Propellant Tank Outlet	37.1	35.5	35.4*	35.0*	36.3	37.0
Pump Inlet	37.1	35.5	36.5	35.6	36.7	37.0
Pump Outlet	36.6	35.1	297	294	40.5	41.3
Nozzle Manifold Inlet	35.7	34.6	290	282	---	41.4
Reflector Inlet	35.7	34.6	254	246	76.2	71.0
Reflector Outlet	35.7	34.6	245	237	---	108
Shield I Outlet (Dome)	35.7	34.6	241	236	109	109
Core Inlet	32.9	31.8	240	234	124	117
Nozzle Chamber (T_c)	32.5	31.5	192	190	2388	2400
Diluent Bleed Inlet (Dome)	2.69	2.74	241	236	109	109
Diluent Bleed Outlet	2.69	2.74	226	224	118	117
Hot Bleed Port	.44	.37	192	190	2388	2400
TPCV Inlet	3.13	3.11	185	180	---	381
Turbine Inlet	3.13	3.11	109	103	356	381
Turbine 2nd Stage Rotor Exit	3.62	3.49	11.6	14	---	330
Turbine Exhaust Nozzle	3.62	3.49	7.9	4.67	259	298

*Ullage Pressure.

XE-PRIME

EP-5C STEADY-STATE HOLD POINT

$P_c = 261 \text{ psia}$ $T_c = 3128^\circ\text{R}$ Range Time = 38690 to 38695

	<u>Actual</u>	<u>Predicted</u>
Turbine Power Control Valve Position	42.4	46.0 ^o
Reactor Power	454	445 Mw
Turbopump Shaft Speed	14045	14071 rpm
Ambient Pressure (Engine Test Compartment)	1.13	.68 psia
Net Positive Suction Pressure)	20.5	19.2 psi

<u>Station</u>	<u>Flow (lb/sec)</u>		<u>Pressure (psia)</u>		<u>Temperature (Deg R)</u>	
	<u>Actual</u>	<u>Predicted</u>	<u>Actual</u>	<u>Predicted</u>	<u>Actual</u>	<u>Predicted</u>
Propellant Tank Outlet	43.6	42.5	35.4*	35.0*	36.2	37.0
Pump Inlet	43.6	42.5	35.7	35.2	36.7	37.0
Pump Outlet	43.0	42.0	423	418	42.4	43.4
Nozzle Manifold Inlet	42.1	41.4	412	402	---	43.5
Reflector Inlet	42.1	41.4	349	342	99	96
Reflector Outlet	42.1	41.4	334	328	---	152
Shield I Outlet (Dome)	42.1	41.4	330	326	149	154
Core Inlet	38.9	38.2	328	323	169	164
Nozzle Chamber (T_c)	38.3	37.7	261	260	3128	3099
Diluent Bleed Inlet (Dome)	3.09	3.16	330	326	149	154
Diluent Bleed Outlet	3.09	3.16	309	311	164	165
Hot Bleed Port	.61	.54	261	260	3128	3099
TPCV Inlet	3.70	3.70	251	241	---	582
Turbine Inlet	3.70	3.70	157	152	546	579
Turbine 2nd Stage Rotor Exit	4.29	4.15	16.5	20	---	501
Turbine Exhaust Nozzle	4.29	4.15	11.4	6.8	387	449

*Ullage pressure.

XE-PRIME

EP-8A STEADY-STATE HOLD POINT

$P_c = 298$ psia $T_c = 2418^{\circ}\text{R}$ Range Time = 57015 to 57025

	<u>Actual</u>	<u>Predicted</u>
Turbine Power Control Valve Position	51.1	55.7 ^o
Reactor Power	430	450 Mw
Turbopump Shaft Speed	15133	15423 rpm
Ambient Pressure (Engine Test Compartment)	.99	1.72 psia
Net Positive Suction Pressure	20.0	17.4 psi

<u>Station</u>	<u>Flow (lb/sec)</u>		<u>Pressure (psia)</u>		<u>Temperature (Deg R)</u>	
	<u>Actual</u>	<u>Predicted</u>	<u>Actual</u>	<u>Predicted</u>	<u>Actual</u>	<u>Predicted</u>
Propellant Tank Outlet	56.2	56.1	35.1*	35.0*	36.6	36.5
Pump Inlet	56.2	56.1	34.0	32.5	36.8	36.6
Pump Outlet	55.7	55.6	483	491	42.9	43.5
Nozzle Manifold Inlet	54.7	54.9	458	464	44.3	43.8
Reflector Inlet	54.7	54.9	393	388	85.8	80
Reflector Outlet	54.7	54.9	377	373	---	115
Shield I Outlet (Dome)	54.7	54.9	373	371	109	116
Core Inlet	50.4	50.7	368	368	130	124
Nozzle Chamber (T_c)	49.3	49.7	298	300	2418	2400
Diluent Bleed Inlet (Dome)	4.11	4.19	373	371	109	116
Diluent Bleed Outlet	4.11	4.19	351	355	119	125
Hot Bleed Port	1.11	1.02	298	300	2418	2400
TPCV Inlet	5.22	5.22	281	270	531	556
Turbine Inlet	5.22	5.22	216	209	531	559
Turbine 2nd Stage Rotor Exit	5.74	5.70	22.6	27	370	471
Turbine Exhaust Nozzle	5.74	5.70	15.4	9.1	390	432

*Ullage pressure.

Table D-10

XE-PRIME

EP-9A STEADY-STATE HOLD POINT

$P_c = 297.0$ psia $T_c = 4075^{\circ}\text{R}$ Range Time = 60550 to 60555

	<u>Actual</u>		<u>Predicted</u>			
Turbine Power Control Valve Position	43.8		46.5 ⁰			
Reactor Power	589		605 Mw			
Turbopump Shaft Speed	15696		15721 rpm			
Ambient Pressure (Engine Test Compartment)	1.29		1.72 psia			
Net Positive Suction Pressure	15.6		18.6 psi			

<u>Station</u>	<u>Flow (lb/sec)</u>		<u>Pressure (psia)</u>		<u>Temperature (Deg R)</u>	
	<u>Actual</u>	<u>Predicted</u>	<u>Actual</u>	<u>Predicted</u>	<u>Actual</u>	<u>Predicted</u>
Propellant Tank Outlet	42.7	42.4	35.4	35.0	38.4	36.5
Pump Inlet	42.7	42.4	34.8	33.6	38.7	36.6
Pump Outlet	42.2	41.9	511	510	46.0	45.3
Nozzle Manifold Inlet	41.2	41.2	497	494	48.2	45.4
Reflector Inlet	41.2	41.2	404	417	138	137
Reflector Outlet	41.2	41.2	384	398	---	241
Shield I Outlet (Dome)	41.2	41.2	378	395	204	244
Core Inlet	38.0	38.0	376	391	229	262
Nozzle Chamber (T_c)	37.5	37.5	297	300	4075	4089
Diluent Bleed Inlet (Dome)	3.03	3.24	378	395	204	244
Diluent Bleed Outlet	3.03	3.24	355	373	277	258
Hot Bleed Port	0.52	0.51	297	300	4111	4089
TPCV Inlet	3.55	3.76	286	283	783	790
Turbine Inlet	3.55	3.76	178	172	780	786
Turbine 2nd Stage Rotor Exit	4.10	4.25	14.6	24.0	---	682
Turbine Exhaust Nozzle	4.10	4.25	13.4	8.0	548	604

Table D-11

XE-PRIME

EP-5C STEADY-STATE HOLD POINT

$P_c = 337 \text{ psia}$ $T_c = 3644^\circ\text{R}$ Range Time = 38830 to 38835

	<u>Actual</u>	<u>Predicted</u>
Turbine Power Control Valve Position	45.2	48.7 ⁰
Reactor Power	644	632 Mw
Turbopump Shaft Speed	16462	16339 rpm
Ambient Pressure (Engine Test Compartment)	1.23	2.32 psia
Net Positive Suction Pressure	19.5	18.6 psi

<u>Station</u>	<u>Flow (lb/sec)</u>		<u>Pressure (psia)</u>		<u>Temperature (Deg R)</u>	
	<u>Actual</u>	<u>Predicted</u>	<u>Actual</u>	<u>Predicted</u>	<u>Actual</u>	<u>Predicted</u>
Propellant Tank Outlet	52.1	50.2	35.4	35.0	36.4	37.0
Pump Inlet	52.1	50.2	35.1	34.6	36.8	37.0
Pump Outlet	51.5	49.7	566	552	44.6	45.7
Nozzle Manifold Inlet	50.4	48.9	549	531	---	45.9
Reflector Inlet	50.4	48.9	453	442	119	120
Reflector Outlet	50.4	48.9	432	421	---	183
Shield I Outlet (Dome)	50.4	48.9	430	419	176	185
Core Inlet	46.6	45.2	426	414	198	196
Nozzle Chamber (T_c)	45.8	44.5	337	335	3644	3651
Diluent Bleed Inlet (Dome)	3.63	3.66	430	419	176	185
Diluent Bleed Outlet	3.63	3.66	401	400	195	199
Hot Bleed Port	.84	.74	337	335	3644	3649
TPCV Inlet	4.47	4.41	323	313	---	782
Turbine Inlet	4.47	4.41	219	209	750	778
Turbine 2nd Stage Rotor Exit	5.01	4.92	23.1	27.0	---	671
Turbine Exhaust Nozzle	5.01	4.92	15.9	9.2	535	602

Table D-12

XE-PRIME

EP-5C STEADY-STATE HOLD POINT

$P_c = 420$ psia $T_c = 3800^\circ\text{R}$ Range Time = 38858 to 38863

	<u>Actual</u>	<u>Predicted</u>
Turbine Power Control Valve Position	50.0	52.6 ⁰
Reactor Power	821	800 Mw
Turbopump Shaft Speed	18690	18477 rpm
Ambient Pressure (Engine Test Compartment)	1.36	2.78 psia
Net Positive Suction Pressure	18.2	17.7 psi

<u>Station</u>	<u>Flow (lb/sec)</u>		<u>Pressure (psia)</u>		<u>Temperature (Deg R)</u>	
	<u>Actual</u>	<u>Predicted</u>	<u>Actual</u>	<u>Predicted</u>	<u>Actual</u>	<u>Predicted</u>
Propellant Tank Outlet	63.3	60.5	35.4	35.0	36.5	37.0
Pump Inlet	63.3	60.5	33.8	33.7	36.8	37.0
Pump Outlet	62.7	59.9	713	695	46.4	47.5
Nozzle Manifold Inlet	61.4	59.1	686	664	---	47.7
Reflector Inlet	61.4	59.1	563	545	127	127
Reflector Outlet	61.4	59.1	537	520	---	193
Shield I Outlet (Dome)	61.4	59.1	535	516	178	195
Core Inlet	56.9	54.7	530	511	204	206
Nozzle Chamber (T_c)	55.7	53.6	420	413	3800	3800
Diluent Bleed Inlet (Dome)	4.37	4.41	535	516	178	195
Diluent Bleed Outlet	4.37	4.41	498	494	203	210
Hot Bleed Port	1.21	1.03	420	412	3800	3800
TPCV Inlet	5.58	5.45	398	381	---	902
Turbine Inlet	5.58	5.45	297	278	900	906
Turbine 2nd Stage Rotor Exit	6.21	6.02	31	62	---	768
Turbine Exhaust Nozzle	6.21	6.02	21.5	12.1	650	695

Table D-13

XE-PRIME

EP-5C STEADY-STATE HOLD POINT

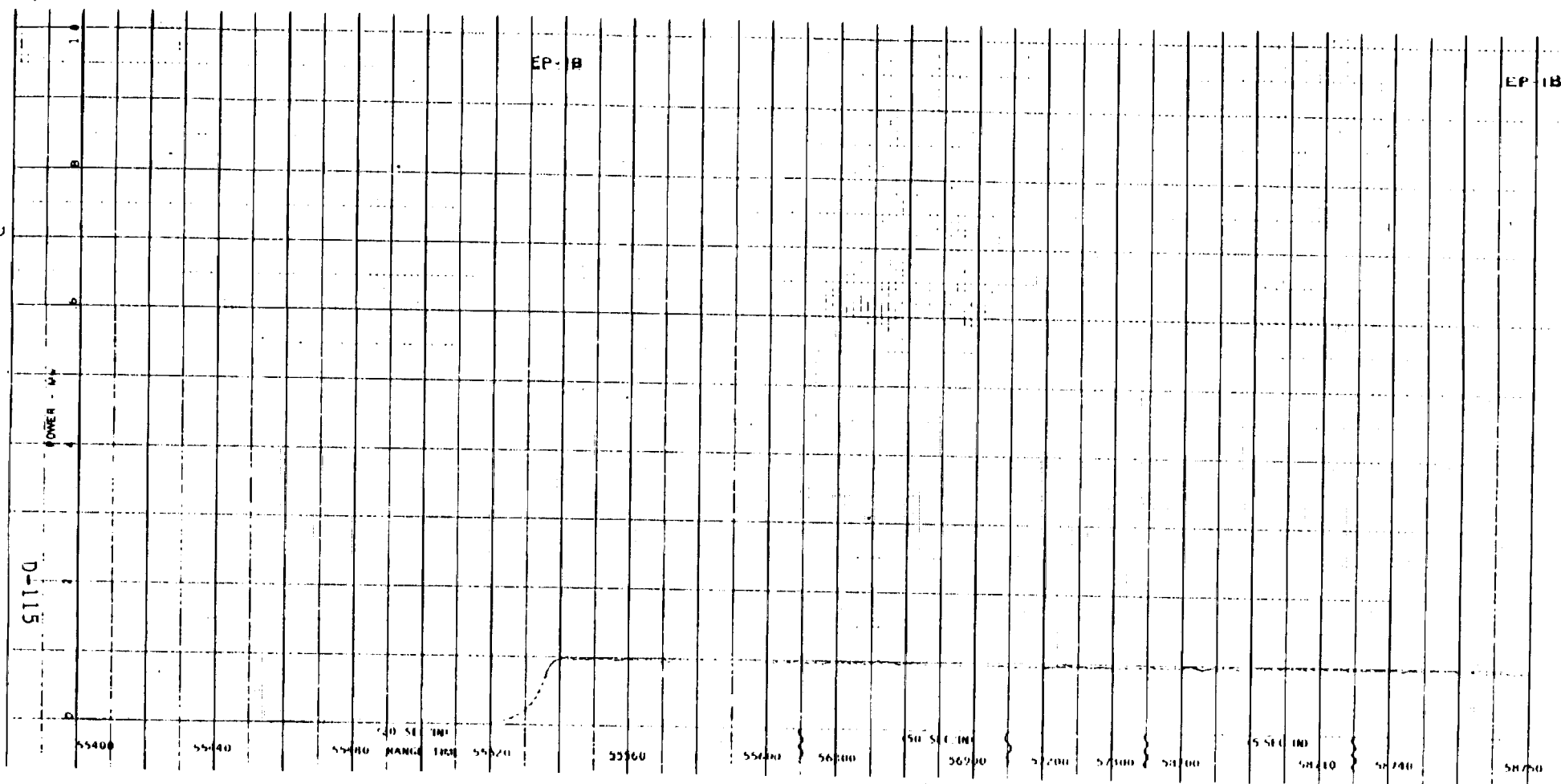
$P_c = 553 \text{ psia}$ $T_c = 4100^\circ\text{R}$ Range Time = 38935 to 38940

	<u>Actual</u>	<u>Predicted</u>
Turbine Power Control Valve Position	63.3	64.9 ⁰
Reactor Power	1137	1139 Mw
Turbopump Shaft Speed	21989	22271 rpm
Ambient Pressure (Engine Test Compartment)	1.64	3.66 psia
Net Positive Suction Pressure	16.4	15.5 psi

<u>Station</u>	<u>Flow (lb/sec)</u>		<u>Pressure (psia)</u>		<u>Temperature (Deg R)</u>	
	<u>Actual</u>	<u>Predicted</u>	<u>Actual</u>	<u>Predicted</u>	<u>Actual</u>	<u>Predicted</u>
Propellant Tank Outlet	79.7	79.0	35.4	35.0	36.6	37.0
Pump Inlet	79.7	79.0	31.7	32.2	36.9	37.0
Pump Outlet	79.0	78.4	970	977	49.9	51.4
Nozzle Manifold Inlet	77.9	77.4	923	927	---	51.9
Reflector Inlet	77.9	77.4	742	740	143	140
Reflector Outlet	77.9	77.4	706	704	---	209
Shield I Outlet (Dome)	77.9	77.4	703	699	196	212
Core Inlet	72.4	71.6	694	692	220	223
Nozzle Chamber (T_c)	70.5	70.0	553	560	4100	4090
Diluent Bleed Inlet (Dome)	5.42	5.73	712	699	196	212
Diluent Bleed Outlet	5.42	5.73	658	668	221	229
Hot Bleed Port	1.91	1.67	553	560	4100	4090
TPCV Inlet	7.33	7.40	513	509	---	1133
Turbine Inlet	7.33	7.40	435	424	1167	1140
Turbine 2nd Stage Rotor Exit	8.00	8.08	46.9	90	---	953
Turbine Exhaust Nozzle	8.00	8.08	32.2	18.2	870	874

*Ullage pressure.

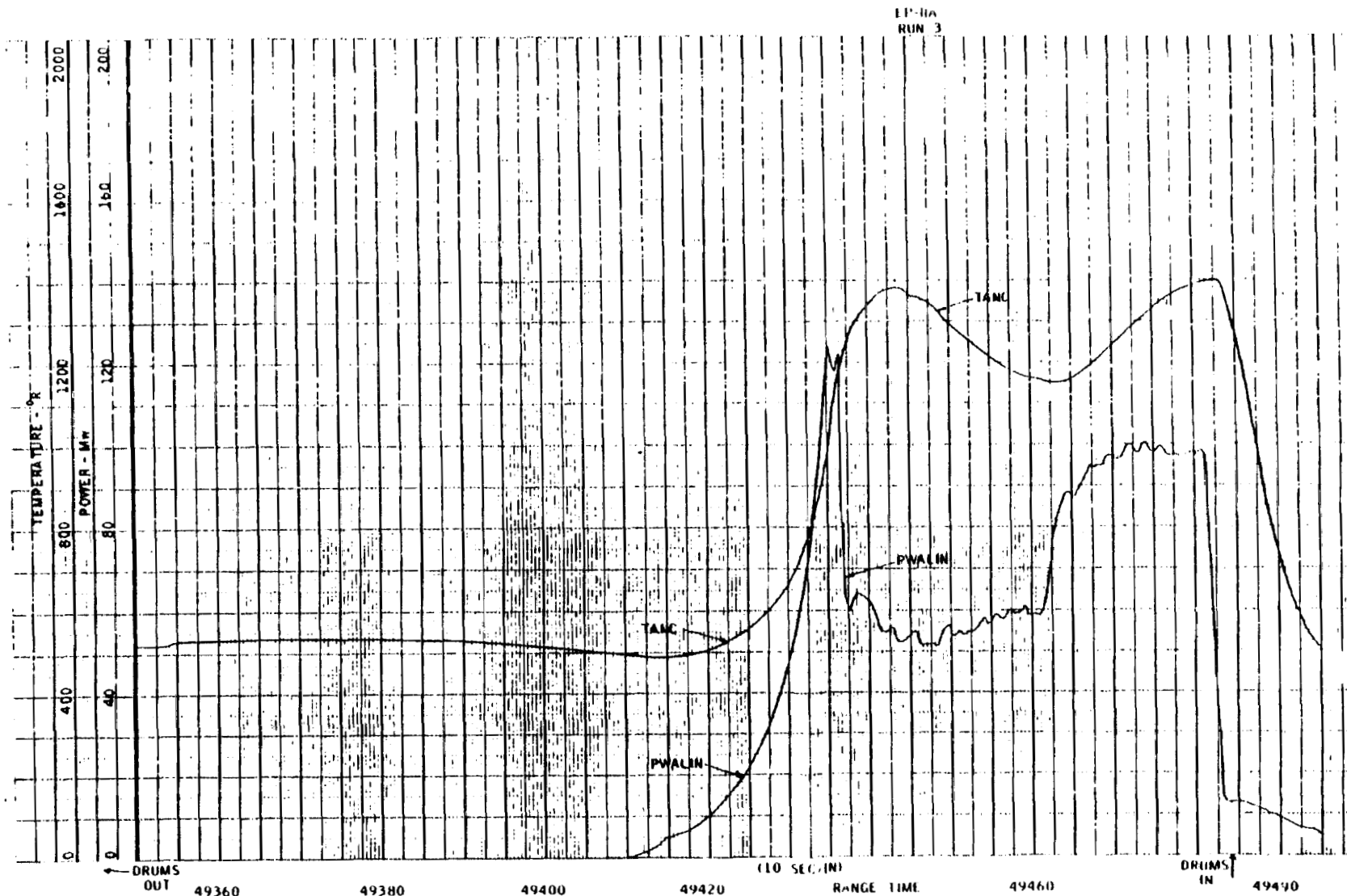
Table D-14



- EP-1B Chamber Temperature and Power vs. Time

Figure D-84

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EP-2A Chamber Temperature and Power Vs Time (Sheet 2 of 2)

Figure D-85

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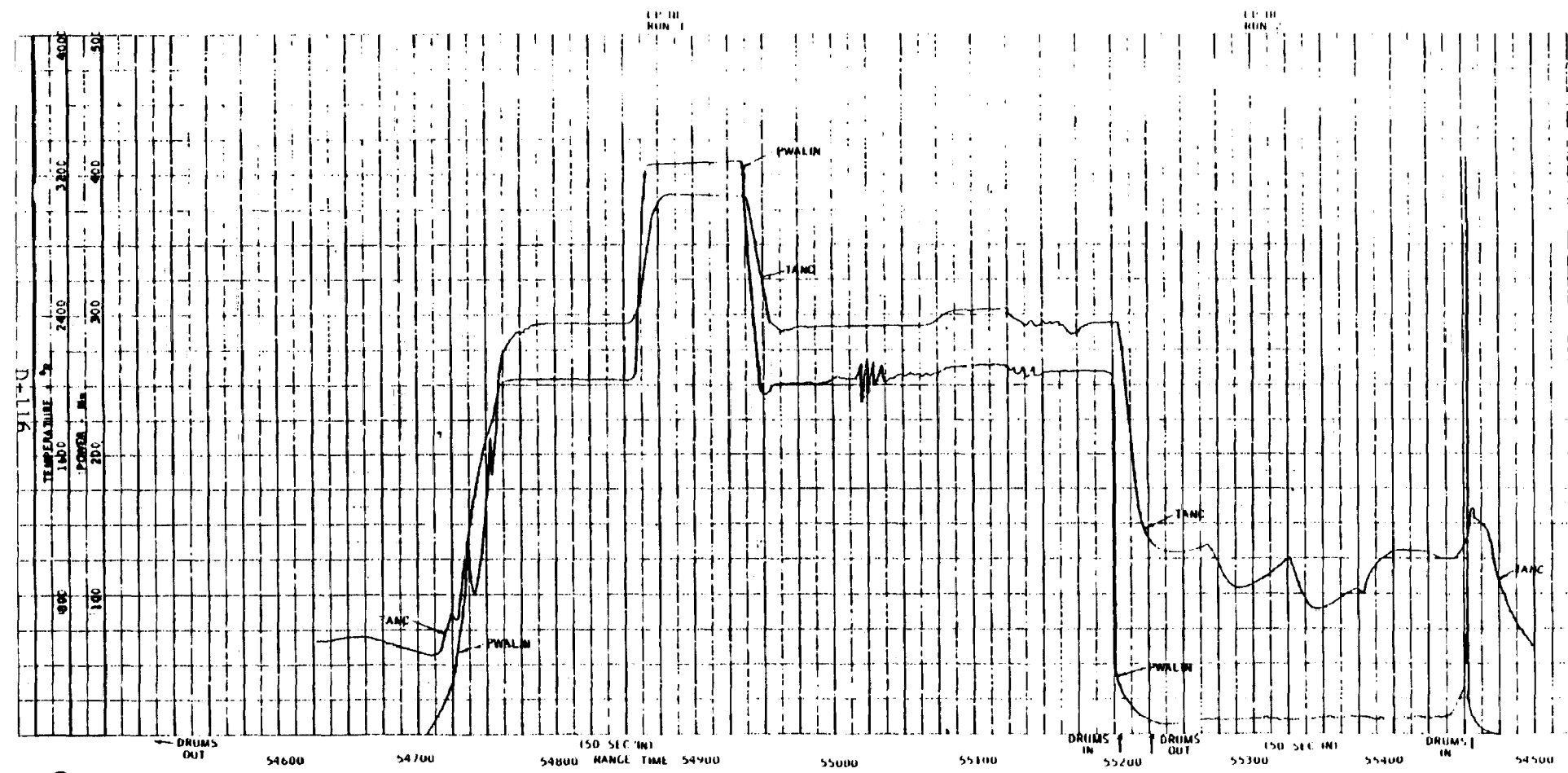
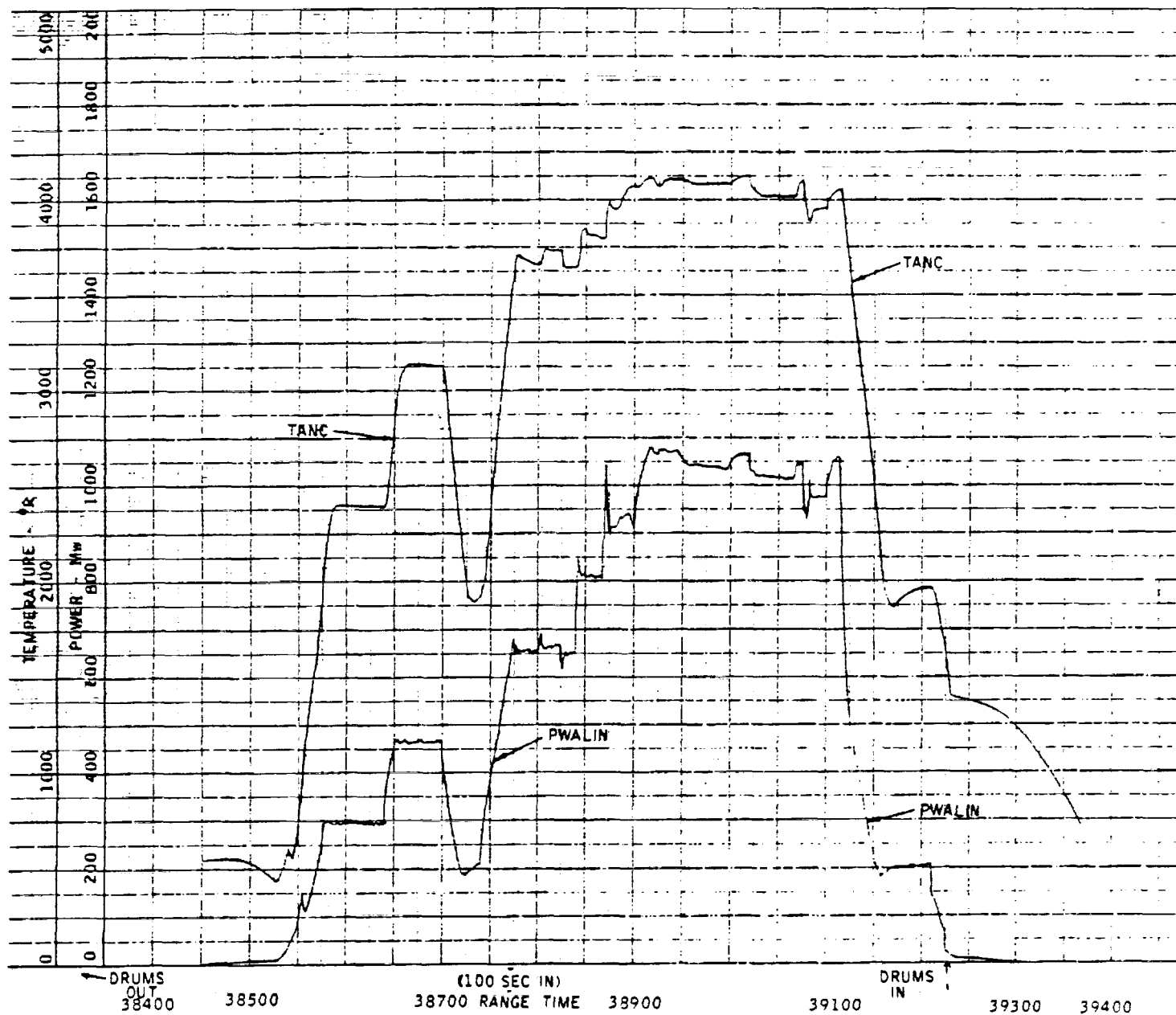


Figure D-86

EP-III Chamber Temperature and Power vs. Time

EP-5C

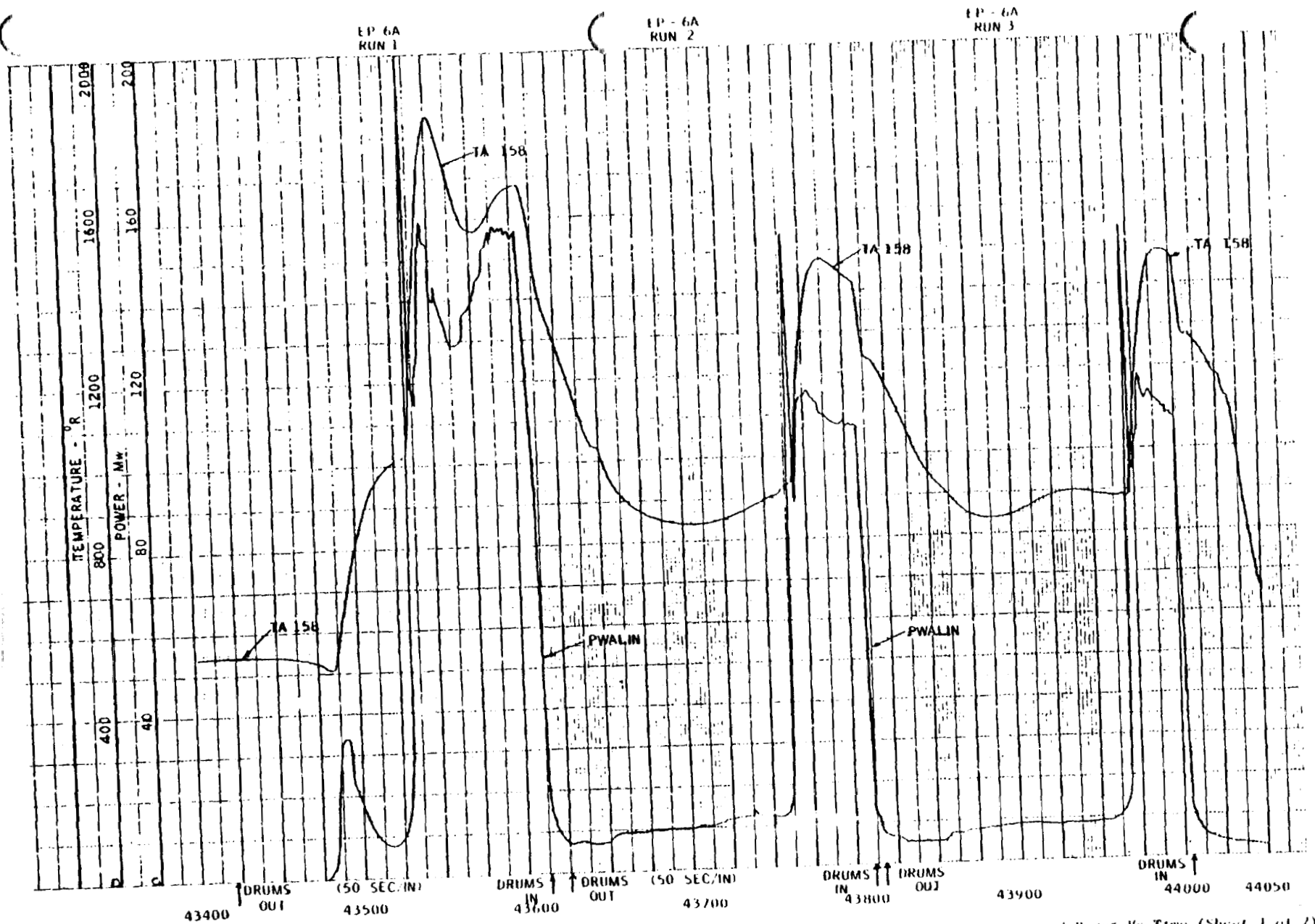


- EP-5C Chamber Temperature and Power Vs Time

Figure D-87

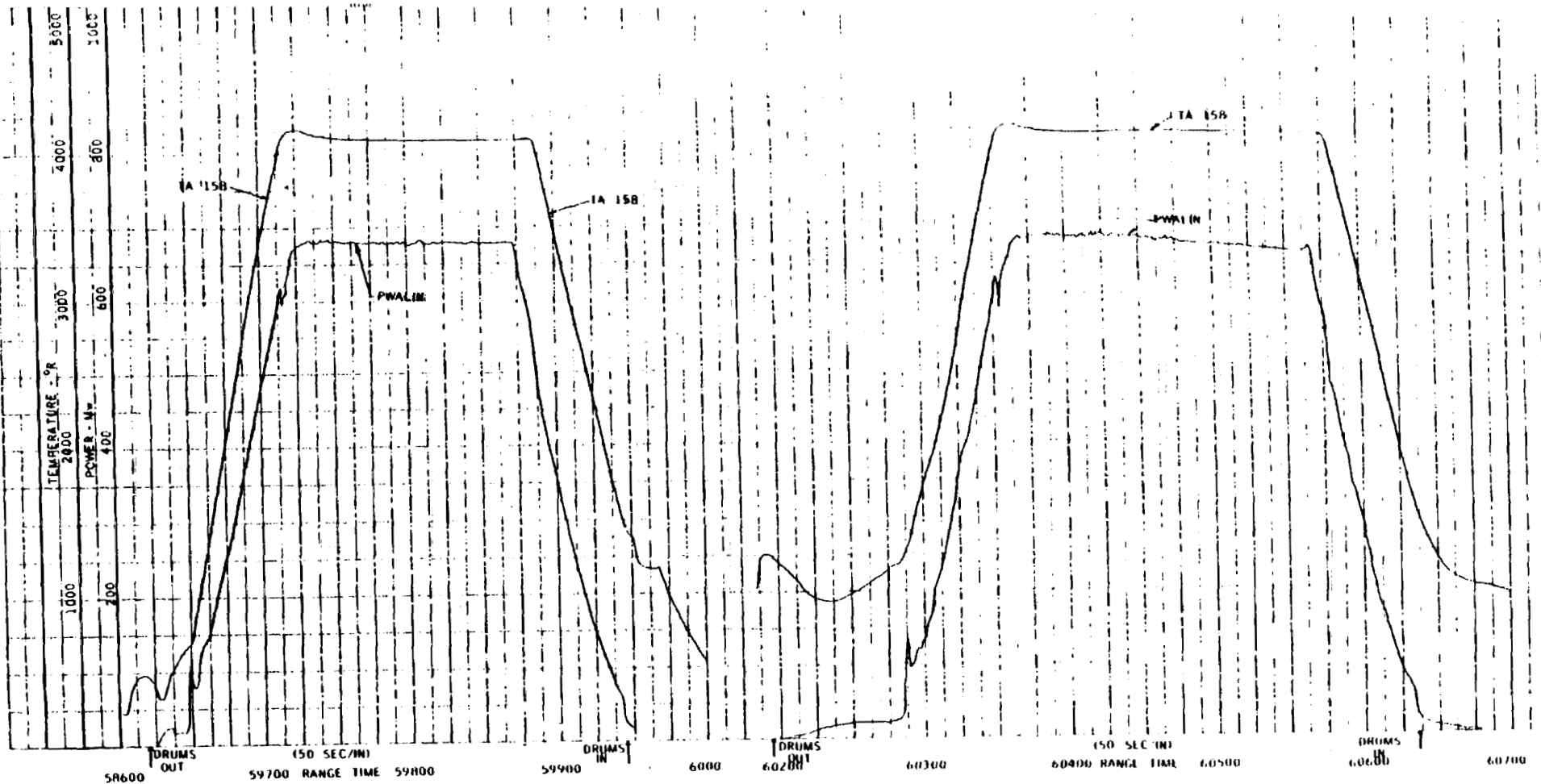
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EP-6A Chamber Temperature and Power Vs Time (Sheet 1 of 2)

Figure D-88



EP-9A Chamber Temperature and Power Vs Time

Figure D-91

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