UNCLASSIFIED

SECRET

UCRL-7249
Nuclear Reactors for
Ram-Jet Propulsion, C-90
M-3679 (30th Ed.)

This document contains 234 pages,
including pp. i-xii.
This is copy 147 of 186 Series A.

UNIVERSITY OF CALIFORNIA
Lawrence Radiation Laboratory
Livermore, California

Contract No. W-7405-eng-48

CLASSIFICATION CANCELLED
Date 11-21-73

Research & Technical Support Div

Exempt from CCRP Re-review Requirement
(per 7/22/82 Duff/Caudle memorandum)

N. Kinsler 11-30-11

THE TORY II-A REACTOR TESTS
FINAL REPORT
(Title: Unclassified)

James W. Hadley

May 3, 1963

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Atomic Energy Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

Noted
occu
use
inge
a the
em
that
ares
tract

RESTRICTED DATA

This document contains restricted data as defined in the Atomic Energy Act of 1954. Its transmittal or the disclosure of its contents in any manner to an unauthorized person is prohibited.

UNCLASSIFIED

SECRET

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

GF

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

.

.

.

.

.

.

~~SECRET~~

UCRL-7249

DISTRIBUTION

Series A

Copy No.

LRL Livermore,

Information Division	.	.	.	.	.	.	.	.	.	1 - 6
John S. Foster, Jr.	.	.	.	.	.	.	.	.	.	7
C. M. Van Atta	.	.	.	.	.	.	.	.	.	8
Roger E. Batzel	.	.	.	.	.	.	.	.	.	9 - 10
T. C. Merkle	.	.	.	.	.	.	.	.	.	11
Richard P. Connell	.	.	.	.	.	.	.	.	.	12
Albert J. Kirschbaum	.	.	.	.	.	.	.	.	.	13
Harry L. Reynolds	.	.	.	.	.	.	.	.	.	14 - 18
Henry C. McDonald, Jr.	.	.	.	.	.	.	.	.	.	19
Wallace Decker	.	.	.	.	.	.	.	.	.	20
W. Blake Myers	.	.	.	.	.	.	.	.	.	21 - 25
Robert W. Westbrook	.	.	.	.	.	.	.	.	.	26
Albert J. Rothman	.	.	.	.	.	.	.	.	.	27 - 28
James S. Kane	.	.	.	.	.	.	.	.	.	29
James H. Patterson	.	.	.	.	.	.	.	.	.	30
W. E. Humphrey	.	.	.	.	.	.	.	.	.	31
William G. Grayson, Jr.	.	.	.	.	.	.	.	.	.	32
Charles S. Barnett	.	.	.	.	.	.	.	.	.	33
Eugene Goldberg	.	.	.	.	.	.	.	.	.	34
James W. Hadley	.	.	.	.	.	.	.	.	.	35
Jerauld H. Moyer	.	.	.	.	.	.	.	.	.	36
Michael D. Mintz	.	.	.	.	.	.	.	.	.	37
Earl A. Sheridan	.	.	.	.	.	.	.	.	.	38
Wm. M. Wells, Jr.	.	.	.	.	.	.	.	.	.	39

DASA Livermore Liaison Office,

Norman G. Hunt	.	.	.	.	.	.	.	.	.	40
----------------	---	---	---	---	---	---	---	---	---	----

LRL Berkeley,

R. K. Wakerling	.	.	.	.	.	.	.	.	.	41
Hayden S. Gordon	.	.	.	.	.	.	.	.	.	42

~~SECRET~~

~~SECRET~~

UCRL-7249

DISTRIBUTION (Continued)

Series A

	<u>Copy No.</u>
LRL Mercury, Nevada,	
C. M. Bacigalupi	43
David P. Maniago	44
Paul E. Neal	45
Roy J. McConnell	46
Lockheed Missiles and Space Co.,	
Attn. J. G. Wenzel	47
Stanford Research Institute,	
Attn. Dorn Barclay	48
ACF - South Albuquerque Works	49
Aerojet General Corporation (NASA)	50
Aeronautical Systems Division	51 - 56
Air Force Special Weapons Center	57 - 58
Air University Library	59
Albuquerque Operations Office	60
Argonne National Laboratory	61
Army Combat Developments Command	62
Army Missile Command	63
ARO, Inc.	64
Atomic Energy Commission, Washington	65 - 68
Atomics International	69
Battelle Memorial Institute	70
Brookhaven National Laboratory	71
Bureau of Naval Weapons	72 - 75
Bureau of Naval Weapons (SPO)	76
Bureau of Ships	77
Canoga Park Area Office	78
Chicago Patent Group	79
Defense Atomic Support Agency, Sandia	80
Department of the Army	81
Director of Defense Research and Engineering (OAP)	82

~~SECRET~~

~~SECRET~~

UCRL-7249

DISTRIBUTION (Continued)

Series A

	<u>Copy No.</u>
duPont Company, Aiken	83
Edgerton, Germeshausen and Grier, Inc., Las Vegas . .	84
Edgerton, Germeshausen and Grier, Inc., Goleta . . .	85
General Electric Company, Cincinnati	86 - 91
General Electric Company, Richland	92 - 93
Institute for Defense Analysis	94
Jet Propulsion Laboratory	95
Johns Hopkins University (APL)	96
Ling Temco Vought, Inc.	97
Los Alamos Scientific Laboratory	98 - 99
Marquardt Corporation	100 - 103
NASA Langley Research Center	104
NASA Lewis Research Center	105 - 109
NASA Marshall Space Flight Center	110
NASA Scientific and Technical Information Facility . .	111 - 113
Naval Air Development Center	114
Naval Ordnance Test Station	115
Naval Postgraduate School	116
Naval Radiological Defense Laboratory	117
Nevada Operations Office	118 - 119
North American Aviation, Inc., Downey	120
Oak Ridge Operations Office	121
Office of the Assistant General Council for Patents (AEC) .	122
Office of the Chief of Naval Operations	123
Office of the Chief of Naval Operations (OP-03EG) . .	124 - 125
Phillips Petroleum Company (NRTS)	126
Pratt and Whitney Aircraft Division	127
RAND Corporation	128 - 129
San Francisco Operations Office	130
Sandia Corporation	131
School of Aerospace Medicine	132

~~SECRET~~

~~SECRET~~

DISTRIBUTION (Continued)

Series A

	<u>Copy No.</u>
Strategic Air Command	133
Union Carbide Nuclear Company (ORNL)	134 - 143
USAF Headquarters	144
USAF Headquarters (DCS/O)	145
Westinghouse Electric Corporation (NASA)	146
Division of Technical Information Extension	147 - 186

~~SECRET~~

CONTENTS

	<u>Page No.</u>
Abstract	ix
Preface and Acknowledgments	xi
1. Introduction	1
2. Preliminary Experiments	3
2.1 Completion of Reactor	3
2.2 First Critical Assembly	8
2.2.1 Experimental Results	17
2.2.2 Core Length Change	19
2.3 Second Critical Assembly	21
2.4 Blowdown Tests	26
2.5 Test Facility Alterations	27
2.6 Preliminary Nuclear Tests	31
2.6.1 Ambient Temperature Criticality	32
2.6.2 Elevated Temperature Measurements	32
2.6.3 Reactor Transfer Function	35
2.6.4 Dress Rehearsal	35
2.6.5 Reactor Power Servo Control Achieved	35
2.7 Final Preparations for High Power Tests	36
3. As-Built Reactor Characteristics	37
4. Summary of Intermediate- and High-Power Tests	44
4.1 Principal Purposes and Results	44
4.1.1 Intermediate Power Test	44
4.1.2 High Power Test 1	52
4.1.3 High Power Test 2	53
4.1.4 High Power Test 3	58
4.2 Analysis of Reactor Performance	61
4.2.1 Determination of Power and Airflow Rate	62
4.2.2 Axial Temperature Distribution	77
4.2.3 Reactor Inlet Pressure	82
4.2.4 Control Vane Angle	82
4.3 Reactor Effluent	85
5. Reactor Disassembly and Component Inspection	87
5.1 Disassembly and General Examination	87

CONTENTS (Contd.)

	<u>Page No.</u>
5.1.1 Disassembly Procedure	88
5.1.2 Condition of Core Components	95
5.2 Fission Product Inventory	100
5.2.1 Distribution of Gross Gamma Activity	104
5.2.2 Correlation of Local Power with Thermocouple Readings	109
5.2.3 Radiochemical Determination of Integrated Power	113
5.3 Fuel Element Cracks	118
5.3.1 Description of Crack Pattern	118
5.3.2 Expected Fuel Element Stress Distribution	143
5.4 Fuel Element Breaking Load Tests	146
5.5 Fuel Element Thermal Diffusivity	149
5.6 Fuel Element Adhesion	151
6. Test Procedure	152
6.1 Air Supply	152
6.2 Operation Plan	154
6.3 Test Conductor's Checklist	160
6.4 Operating Crew	165
6.5 Events of the Test	173
6.6 Operators' Sound Transcription	175
Appendices	
A. Appraisal of Airflow Parameter Measurements, by J. H. Moyer	193
B. Analysis of Fission Product Loss in Test Operation of Tory II-A, by C. F. Donaghy and E. Goldberg	200
C. Data System, by T. C. Varljen	206
References	222

~~SECRET~~

UCRL-7249

THE TORY II-A REACTOR TESTS
FINAL REPORT

James W. Hadley

Lawrence Radiation Laboratory, University of California
Livermore, California

May 3, 1963

ABSTRACT

Tory II-A was the first reactor in a series which will lead to development of a nuclear-powered ramjet engine in the Pluto program. The core embodied a design which is believed to be useable in an actual engine. All parts external to the core were intended only to form a suitable test bed, and had no relation to flyable hardware.

Test operations were conducted during the period December 1960 – October 1961. On May 14, 1961, following several months spent in low-power measurements and facility shakedown, the reactor was brought to 50 MW power ($1/3$ design) and 2580°F core temperature (300° above design) for a period of about one minute. All systems operated properly, and no indication of any damage was obtained.

After a period of facility alterations, to provide the capability of delivering test air at higher temperature and pressure, the reactor was run three times at power levels approaching or in excess of the design operating value of 160 MW. The second of these tests fully achieved design operating conditions, and the third substantially exceeded them in severity. Again no indications of failure were noted during operation.

Disassembly of the reactor has confirmed that it maintained its structural integrity. In general confirmation of previous experimental data, thermal stress cracks were found in many fuel element walls. Fewer cracks than expected, however, were found in the hottest part of the core, indicating that considerable stress relief through creep must have occurred. Even among the elements which cracked, no instance was found in which an air flow passage was blocked.

-ix-

~~SECRET~~

~~SECRET~~

UCRL-7249

Detailed behavior of the reactor with respect to heat deposition rates, temperatures, air pressures, radiation levels and other observed parameters generally confirmed calculated expectations.

It is concluded that the Tory II-A program has achieved full success, even surpassing its planned objectives in some instances.

~~SECRET~~

~~SECRET~~

UCRL-7249

PREFACE AND ACKNOWLEDGMENTS

This report relates the history of the Tory II-A reactor throughout the period following completion of its assembly. An account is given of preliminary test operations in Livermore and at the Nevada Test Site, of the series of tests at high power, and of the results of final disassembly and examination of the reactor.

It is assumed that the reader will be reasonably familiar with the reactor structure and characteristics, and no attempt is made to describe them in detail except for an account of a few important changes made just before and during the series of test operations. A general description of the reactor and its associated systems has been given in an earlier report (Ref. 1).

Some of the conclusions on reactor test conditions or post-test examination given here differ from those given in various informal reports which have been issued before. This disparity results from re-appraisal of the older data together with the addition of new data. Every effort has been made to consider all available information in arriving at the conclusions presented here, and they are intended to supersede any previously issued data which may seem to disagree with them.

Acknowledgment is gratefully made to the authors of the many internal memoranda from which material in this report has been taken. Sources of special value have been the reports of T. Varljen on test sequences and data; of J. Moyer on the interpretation of test data; of C. Barnett on circumstances of the Intermediate Power Test; of D. Maniago, B. Klem, and W. King on reactor disassembly, inspection, and fuel element testing; of C. Donaghy and E. Goldberg on reactor effluent material during tests; of J. Niday on fuel element fission product inventories, and of R. Womac on fuel element thermal diffusivity tests.

-xi-xii-

~~SECRET~~

1

2

3

4

5

6

SECRET

- 1 -

UCRL-7249

THE TORY II-A REACTOR TESTS FINAL REPORT

James W. Hadley

Lawrence Radiation Laboratory, University of California
Livermore, California

May 3, 1963

1. INTRODUCTION

It is intended that this report should accomplish three objectives: first, to describe final changes in the Tory II-A reactor and furnish a brief as-built description to supersede the design report¹ issued in 1959; second, to describe the test program in terms of reactor conditions and behavior; and finally to give an account of the preparation and procedures involved in carrying out the test program. An effort has been made to keep these different aspects separate in the report as much as possible.

A brief history of the Tory II-A program should be of aid to the reader who is not already familiar with it. Tory II-A* was conceived as a test device in 1957 when the Lawrence Radiation Laboratory came into the Pluto program for development of a nuclear ramjet engine.

It was decided that the objectives of this reactor should be limited to the test, under engine-like operating conditions, of a specific internal core design. This design would correspond in its interior elements to one which was regarded as suitable for an engine, but to minimize core cost, test air consumption, and nonessential complexities, the overall diameter would be

*Persons having intermittent acquaintance with this program may have noticed a possible source of confusion in the name of the reactor, which can be cleared up as follows: the reactor described here was originally called Tory II-A. Later, when building of two almost identical reactors of this type was scheduled, the first one, actually built and operated, became known as Tory IIA-1. The second was to be called Tory IIA-2. When the degree of success achieved with early reactor operations eliminated the need for a second reactor, Tory IIA-1 became again simply Tory II-A.

SECRET RESTRICTED DATA

kept as small as possible. Thus the Tory II-A core may be regarded as a full-length section from the center of an engine core whose full diameter would be perhaps one and one-half or more times as great as the II-A diameter.

In order to make the core neutronically critical, it was surrounded by a very thick graphite reflector. Reactor control elements were placed in this reflector rather than in the core, to make their design simple. Fuel loading was graded to produce a flat radial power distribution, but no effort was made to shape the axial power distribution. An account of the reactor design will be found in Ref. 1, supplemented by Chapter 2 of this report.

Design of the reactor was complete by early 1960, and assembly of the core took place during the summer and early fall of that year. Criticality was initially reached on October 7, 1960, in Livermore. Final assembly of the complete reactor took place shortly thereafter at the Nevada Test Site of the U. S. Atomic Energy Commission.

Test facility alterations, dry runs, and a series of zero- or low-power tests led up to the first reactor operation at power levels approaching the design value on May 14, 1961, when a power of 50 MW and a core temperature of 2580°F were attained.

Further facility alterations preceded the resumption of reactor testing in September, with three final test runs all achieving design power level and core temperatures as high as or, finally, much higher than design value.

A program of disassembly and inspection of the reactor took place from December 1961 to September 1962. Data from inspection of the fuel elements, where the bulk of the effort in this phase was concentrated, is presented in Chapter 5.

The bulk of the report is devoted to the four test runs made at high power levels, since these represent the final goal of the Tory II-A program and the source of all data by which the reactor may be evaluated in terms of degree of success in meeting its design objectives. Preliminary stages of operation, although essential steps in reaching eventual high power operation, were not of general interest in themselves and so have been described only in a very cursory manner.

2. PRELIMINARY EXPERIMENTS

2.1 Completion of Reactor

A general description of the reactor as of the latter part of 1960 is given in the report "Tory II-A: A Nuclear Ramjet Test Reactor" (Ref. 1). The most significant design development during the following year (occupied mainly by test vehicle and reactor assembly) was the final definition of the fuel elements and their assignment, by fuel content, to various positions in the core.

All of the fuel elements finally assembled in the core were made by the General Electric Aircraft Nuclear Propulsion Department to meet performance specifications furnished by LRL. These specifications were concerned with dimensions, density, freedom from flaws, strength, dimensional and fuel-retention stability at high temperature, and the like. The elements were extruded and sintered as hollow circular cylinders, then ground flat on six sides and cut off square on each end. Their material was a homogeneous mixture of beryllia and uranium oxide, with no impurities present in amounts greater than 100 parts per million by weight except for iron, sulphur, and calcium, present as a few hundred parts per million. Figure 2-1 is a drawing which forms part of the fuel element specifications.

Final detailed neutronic calculations, taking all available information on core parts dimensions, density, and composition into account, called for a reduction in total fuel loading below the value specified in Ref. 1. The original loading class definitions were retained, but six new classes were defined, 5% apart in fuel content as are the other loading classes. Table 2-1 lists the seventeen classes of fuel elements used, with the nominal weight percent of UO_2 in each class. The actual fuel loading in the manufactured elements varied over a smooth spectrum, rather than being concentrated at the nominal figure for each class. Separation into classes was made at loading values halfway between the nominal class values listed.

Fuel weight percent numbers are based on the assumption that all uranium is in the form of UO_2 .*

*The uranium is enriched to 93.2% in U^{235} .

Notes (Unless otherwise specified) Hexagonal

- 1 figure shown in view A-A & figure B are geometrically true.
- 2 At any axial station the wall thickness measured at the midpoints of the six flats shall not vary more than 0.005

View A-A tolerance band (scale none)
External profile of fuel tube at any axial station must lie wholly within limits of tolerance band shown.

GLL-638-1976

Fig. B volumetric limits (scale none)
Entire external surfaces of fuel tube to lie wholly within limits of right hexagonal figure with above cross section and length.

Fig. 2-1. Tory II-A fuel element: dimensional specifications.

Table 2-1. Fuel element loading classes

Class	Wt % UO_2	Class	Wt % UO_2
60	3.53	4	5.41
50	3.70	5	5.68
40	3.88	6	5.95
30	4.07	7	6.25
20	4.25	8	6.52
0	4.47	9	6.87
1	4.70	10	7.16
2	4.93	11	7.49
3	5.17		

The fuel loading averaged over the entire core fuel element mass, assigning the nominal loading value to each class, was 5.95 weight percent UO_2 .

Figure 2-2, and Fig. 2-3 in enlargement, show the assignment of loading classes to positions in the core. This map shows the result of a design decision made after publication of the original loading map in Ref. 1, namely that no effort would be made to flatten azimuthal power variations due to the control elements in the reflector, or due to local concentrations of unfueled beryllia in the core structural links. A great simplification in the map resulted, in that symmetrically placed modules now were given the same loading distribution no matter where in the core they occurred. The original map showed symmetry only from one core quadrant to another.

As in earlier stages of the design, the fuel concentration was not varied in the axial direction through the core. The loading class specified for a given station on the map applied to the full length of that column of fuel elements.

In other respects, the reactor configuration was nearly as described in Ref. 1. Four minor exceptions are noted:

Early plans called for a 1.5-inch thick tungsten gamma shield at the inlet end of the core, to protect the core support structure from radiation heating. It was later determined that this shielding was unnecessary, and the tungsten plates were replaced by 1/4-inch-thick stainless steel plates, which simply served to retain the fuel elements when the core was tilted with the inlet end down.

MUL-9560

Fig. 2-2. Assignment of fuel element loading classes to core location.

Fig. 2-3. Fuel loading assignments, enlarged from Fig. 2-2. The dark elements at the core periphery are unfueled, solid BeO elements.

The peripheral ring of ceramic structural links at the inlet end of the core and all of the last layer of structural links at the exit end were replaced by metal links, because of a feeling that the core structural integrity and flow passage alignment might be particularly sensitive to breakage of parts in these positions. Cracking of ceramic links in the core interior, it was felt, would have no serious adverse effect on core performance.

It was also believed that additional protection should be afforded against breakage of the ceramic structural links all around the cylindrical outer surface of the core, with possible resultant dislocation of the outer fuel elements. Consequently a perforated sheet-metal retainer was placed directly over the outer surfaces of each row of peripheral links, being retained in place by wires at each end and by the inward pressure of the surrounding wiper seals.

To reduce the possibility of flow passage misalignment at the exit reflector-baseplate interface, the hole diameter in the exit reflector tubes was increased to 0.230 inch (instead of 0.200 as in the rest of the core).

The reactor structure is shown in the drawings in Figs. 2-4, 2-5, and 2-6.

Figure 2-7 is a photograph of the core about halfway through assembly: many modules of fuel elements have already been placed on the assembly table and enclosed by ceramic structural links, and another module is about to be added.

The reactor core is shown in Fig. 2-8 as it was being lifted onto the test vehicle, and Figs. 2-9 and 2-10 are photographs of the complete test vehicle. Figure 2-11 furnishes identification of the various test vehicle and reactor components.

2.2 First Critical Assembly

By late September, 1960, the reactor was completely assembled except for the final tie tubes and base plates, the metal structural links mentioned above, and the exit reflector elements with oversize holes. The tie tubes used in assembly were almost identical to the rods which would be used for operation, but were not of acceptable mechanical quality. Aluminum base-plates were used in assembly, with dimensions identical to those of the final Mo-1/2% Ti plates. Two-thirds of the initially assembled exit reflector elements had 0.200-inch-diameter holes instead of the final 0.230-inch size.

MUL-11267-A

Fig. 2-4. Longitudinal section through reactor core and reflector.

Fig. 2-5. Section through reactor, transverse to axis.

Fig. 2-6. Detail of peripheral core module and adjacent reactor structure.

Fig 2-7 Core assembly. A bundle of fuel elements comprising one module is being added to the core. Other modules are already in place, retained by rows of structural links. Note the inlet and exit reflector regions at top and bottom

GLB-638-5114

Fig. 2-8. Core being lowered onto test vehicle. The remainder of the air duct is already in place, with the exit nozzle in the foreground of the picture.

57-27055

Fig. 2-9. Reactor and test vehicle complete.

Fig. 2-10. Reactor and test vehicle from inlet end.

Fig. 2-11. Diagram of test vehicle and reactor assembly.

Before test operation of the reactor, many columns of fuel elements were to be extracted for replacement by other columns with thermocouples installed in them. This operation was to be left until all other assembly operations had been performed on the core.

Thus, although the core was not yet really complete, it was close enough to its final configuration that neutronic differences would be very slight. Since delay in procurement of the baseplates and tie rods would hold up final assembly for many more weeks, it was decided that the initial criticality experiment should be performed in the intervening period. This experiment was to be performed at LRL in Livermore, where optimum facilities existed for working on the core if it should be found necessary to make major alterations to adjust reactivity.

2.2.1 Experimental Results

The reactor was assembled in a special fixture in a shielded containment building. The assembly fixture was arranged to provide the same physical arrangement of the reactor core and reflector as they would finally have in operation, but with greater flexibility and accessibility, and with the addition of extra control elements inserted in the central six tie tubes of the core.

Dry and Wet Results. The initial approach to criticality was made with core vessel and reflector cooling passages devoid of water. Criticality was reached on October 7, 1960, with all control vanes rotated to an angle of 90° from the full-shutdown position. The expected angle, for the reactor assembly as it existed at this time, was about 70° . This amount of deviation from predicted results was not considered surprising, or necessarily alarming.

Another approach to criticality was then made after filling the core vessel and reflector cooling passages with water. Instead of showing the predicted increase in k_{eff} of 0.023, the reactor showed a decrease of about 0.008 in k_{eff} at small vane angles and a much greater decrease at large vane angles, and in fact would not go critical at all.

Appraisal of Cause of Discrepancy. A review of calculational methods which followed this surprising result showed that the difficulty lay in an improper

geometrical description, in the input to the computer neutronics calculations, of the reflector and its cooling water. Calculations made with a more detailed description of the water-filled passages succeeded in showing a negative reactivity effect associated with the water, with a value reasonably close to that observed.

A really careful investigation of the physical causes for the lowering of reactivity by water, which one normally expects to have the opposite effect, has not been carried out. However, it is not difficult to see in what direction the explanation must lie: Even without water, neutrons in the reflector are quite effectively thermalized by the graphite of which it is composed. It is credible that any additional moderation due to the added hydrogen may be of little value, while its capture cross section produces a noticeable drain on the neutron economy.

The calculational review demonstrated also that the supposed initial near-agreement of the dry critical vane angle with predicted results was to a large extent invalid. In retracing the logical connections between one- and two-dimensional neutronic calculations and those describing r - z and r - θ views of the reactor, it was found that the most objectively acceptable grouping of results led to a predicted k_{eff} differing from that measured for the dry reactor by about 0.06 abs. The uncertainty in calculational results associated with recognized uncertainties in the exact physical description of the reactor is of comparable amplitude; so it is difficult to say how much of the discrepancy is directly ascribable to inadequacy of the geometrical approximations used to describe the reactor, how much to failure of the diffusion theory neutronics to approximate actual neutron behavior well enough, and how much to imperfect nuclear cross section values in use. All in all, however, it is felt that the design approach used was amply justified, and that a similar approach can be taken to the design of succeeding reactors. The basic characteristic of this approach is a strong reliance on a variety of critical experiments in very simple geometry to check calculational methods, and no requirement whatever for "mockup" critical experiments in a form closely approximating the final reactor.

Use of Heavy Water. Additional neutronic studies led within a few days to the conclusion that the substitution of heavy water for light water in the reflector cooling passages should be sufficient to bring reactivity up to desired levels.

This was confirmed on carrying out such a change, leaving light water in the core vessel and vernier control rod cooling passages.

Figure 2-12 shows effective multiplication factor k_{eff} (based on $\beta = 0.0069$) as a function of control vane angle, measured during these initial experiments with the cooling passages dry, or filled with H_2O or D_2O . As the figure illustrates, even with heavy water the maximum available excess k_{eff} , with control vanes at 180° , was only about 0.027 abs. This margin was expected to be reduced by about 0.003 abs. as a result of final installation of all core components (metal links, thermocouples, etc.). The predicted excess k_{eff} needed to achieve design point core temperature was, at this time, 0.023 abs. Thus it appeared that if no adjustments were made, it would be barely possible to reach design temperature with no margin left for error or unexpected loss of reactivity from other causes. For this reason a decision was made to use the provided capability of core length increase.

2.2.2 Core Length Change

Following completion of the critical experiments in Livermore, at the end of October, 1960, the reactor core was shipped to the Pluto facility at the Nevada Test Site of the Atomic Energy Commission. All further work on the core, as well as all subsequent reactor testing, took place at that location.

The reactor core had been designed and built with enough extra tie tube length to permit the addition of one layer of four-inch-long fuel elements, plus about one-and-one-half inches more if needed. Results of the critical experiments in Livermore had shown that even with substitution of heavy for light cooling water, more reactivity would be necessary to provide a good expectation of reaching design core temperature with sufficient margin for control.

Accordingly, it was decided that an additional four inches of fueled length would be provided, carrying with it an expected increase in k_{eff} at any control vane angle of 0.017 abs. Fuel elements were not available in each loading class in numbers sufficient to adhere strictly to the loading map of Fig. 2-2, but departures from this map were generally small, nowhere resulting in variations in total power delivered to individual air flow passages of more than about 2%. This discrepancy has been taken into account in

GLL-638-1977

Fig. 2-12. Effective multiplication factor determined from first criticality experiments, in October, 1960.

making the thermocouple correlations of Section 5.2.2. The new critical control vane angle, at room temperature with D_2O coolant and following addition of the new layer of fuel elements (as well as metal links and new exit reflector elements) was expected to be 64° .

Other neutronic changes due to the length change, as in power distribution, kinetics, etc., were expected to be of negligible importance. A small increase in air pressure drop through the core at design point would be experienced, but again this was of no real significance. A physical description and listing of operating parameters for the final core configuration is given in Section 3.

To carry out the length change, the core was placed on an assembly fixture at the Test Site, and stripped of the two layers of unfueled reflector elements at its inlet end. The additional layer of fuel elements was installed, and the reflector elements then were replaced. Following this operation, the core was inverted for exchange of 230-mil for the remaining 200-mil exit reflector elements and for installation of a layer of metal structural links (see Section 2.1); neither of these items had been fully available when the core was first assembled. The core was now complete except for final tie tubes and baseplates, which still had not been received, and for complete installation of core thermocouples, which would be the final step in core assembly.

Meanwhile, alterations had been carried out on the test vehicle and in the bunker at the test point to permit the circulation of heavy water through the reflector cooling system with a minimum of loss and contamination.

2.3 Second Critical Assembly

A program of qualifying the test facility for reactor operation had started in the summer of 1960. By the end of November, when the reactor alterations described above had been completed, most facility systems had been completed, checked out, and pronounced ready for use. Final readiness would not be assured, however, until air had been discharged through a reactor test vehicle (without core) at full design temperature and flow rate in simulation of an actual reactor test. It was planned that at least two such tests should be carried out with fully satisfactory operation of the facility before reactor testing would commence.

The reactor core at this time, although not fitted with its proper tie tubes and baseplates, had been increased in length to correct for the reactivity deficiency found during its initial critical assembly, and it was strongly desired to test the result of this alteration as soon as possible, rather than wait several weeks for availability of the missing parts. Consequently a decision was made to suspend facility checkouts and to run a new critical assembly test at once.

The reactor core was mounted in the test vehicle without any of its air duct except for the core vessel section in which the core was mounted. A different set of graphite reflectors was present at the test site from those used in Livermore, but there was every reason to believe that the two sets would be neutronically identical.

This configuration, with heavy water in the reflector, reached criticality on December 9, 1960, with a control vane angle of 65° , differing by only 1° from the expected value. A very close approach to criticality two days earlier had shown that the critical vane angle with the reflector devoid of heavy water would be approximately 71° .

By insertion of boron-loaded rods into the central six tie tubes, the core reactivity was lowered enough to permit turning the control vanes out to an angle of 132° before criticality was reached. By measuring neutron level as a function of vane angle in this range as well as in the small-angle range without the extra poison rods, it was possible to construct a curve proportional to k_{eff} as a function of vane angle. Measurement of a relatively short stable reactor period reached by moving the vanes a carefully measured amount past the critical angle then provided a normalization factor to yield, finally, the curve of effective multiplication factor as a function of vane angle given in Fig. 2-13. The effective delayed-neutron fraction β was taken as 0.0069 in generating this curve.

A large number of U^{235} foils were placed in the core tie tubes and fuel element flow passages for an activation run of about ten minutes' duration and 150 watts power. The activity levels of these foils, multiplied by local fuel concentration to give a measure of core local power density, are plotted as a function of core radius in Fig. 2-14, where each point represents an integration over core length for a single flow passage. Figure 2-15 shows axial variation of foil activity, measured near the core center; a calculated curve is shown for comparison.

Fig. 2-13. Effective multiplication factor measured following core modifications, in December 1960. The reflector was filled with heavy water, other cooling passages with normal water.

Fig. 2-14. Relative local power density, integrated over core length, as a function of radial position in core. December, 1960.

GLL-638-1980

Fig. 2-15. Axial foil activation profile from foil activation experiments of December, 1960, compared with calculated profile.

These data indicated that power profiles at high power and temperature levels would meet design intentions. The axial profile, which should not change appreciably with core temperature, was very close to calculated values inside the core region, and would be slightly closer if full correction were made for self-shielding in the 0.001-inch U^{235} foils used. The radial profile showed a dip at large values of radius which was expected to disappear as the control vane critical angle (and the effective reflector thickness) increased to compensate for high core temperature.

As will be shown in Section 5.2.1, these expectations were confirmed in operation of the reactor at high temperature.

2.4 Blowdown Tests

The low power critical assembly was one of the only two planned test procedures which could be carried out without producing too large a fission product inventory in the core to permit further direct personnel access for the long periods of time necessary for installation of final components. The other such procedure would involve discharge of ambient temperature air through the core at high flow rates, with the reactor subcritical, and of course would not contribute any increase whatever to the core radioactivity level. Indeed, it was desirable to carry out this operation before final core assembly in case it should show the need for modifications in core fittings, mounting arrangements, and the like.

Accordingly, the test vehicle was fitted with its complete air duct and stripped of special fixtures used during the critical assembly. Stored air was discharged through the core in a series of runs on December 17 and 18, being held for about thirty seconds duration each on flow rate plateaus of 40, 60, 75, 100, 225 and 330 pounds per second. Aside from the intention of a preliminary demonstration of core integrity under pressure loads, these tests had the following specific purposes:

The air pressure data system would be run under realistic conditions, and actual values of static and total pressure at various points in the air duct, aft and ahead of the reactor, would be compared with computed values to give an initial check on the validity of core air flow calculations.

In addition, strain and vibration instrumentation would record mechanical behavior of the core under load.

Results of the tests were satisfactory. No undesirable core behavior was detected, and strain and air pressure readings confirmed expected values.

Some small-amplitude vibrations were noted, apparently associated with acoustic resonances in the duct. It did not appear that sufficient energy was involved to endanger the core. This subject continued to be investigated during a number of later blowdown tests without the core in place, especially with regard to a resonance in the 100 cps frequency range which tended to be excited in the flow regime near establishment of choked flow at the nozzle throat. Although a high sound level was generated, very little vibration of the air duct itself took place, and the investigation was finally dropped without full enlightenment, but with good assurance that no threat to proper reactor operation was presented. Reference to the Operation Plan excerpts of Section 6.2 will show that some caution over this phenomenon (called "horning") was still in evidence at the time of the May, 1961 Intermediate Power Test.

2.5 Test Facility Alterations

After the blowdown tests were completed, the reactor core was returned to the test site Disassembly Building for installation of baseplates, tie tubes, and thermocouples. Test Vehicle 1 (a duplicate, vehicle No. 2, was also on-site) was installed at the test pad, complete in every way except that its core vessel was empty.

A program of simulated nuclear operations was now scheduled, involving actual discharge of air at flow rates and temperatures up to peak reactor design values. Besides verifying proper operation of all facility systems, this high-flow-rate operation was intended to sweep as much residual dirt and rust as possible out of the air ducts to reduce the hazard of erosion of the reactor core.

Clamp Failure

On January 11, 1961, during what was hoped to be a final system qualification test, at an air flow rate of 720 pounds per second and temperature of 1060°F, the clamp holding the exit nozzle to the air duct on the test vehicle broke, releasing the nozzle, which performed a free flight to a distance of 480 feet from the test point.

The broken item was a band clamped around the circumference of the nozzle joint, holding mating flanges together. Twelve such clamps were in use along the air duct. Careful examination of the clamps subsequent to this accident showed that the broken one, as well as all the others, were in an unsatisfactory metallurgical state, having very low ductility and impact strength.² In addition to the need for remedy of this condition, design changes which would greatly reduce the peak stress to which these parts would be exposed were found necessary. A complete set of new clamps was accordingly ordered, and a program of heat treatment, re-shaping, and qualification testing of the old ones started to provide parts useable for low-level operations pending completion of the new clamps, now designated V-band couplers.

Figure 2-16 shows the broken clamp.

Test Vehicle Accident

After appraisal of results of the nozzle loss, later in the same afternoon, it was decided that a rehearsal of radio-controlled disconnection and removal of the test vehicle from the test pad should be carried out as previously scheduled.

During this operation, the test vehicle was pulled by its electric locomotive along the railroad track from the test pad to a point just beyond the switch at the main track. Here, the electrically-controlled coupler between the locomotive and test vehicle spontaneously opened. Because of a grade in the track at this section, the test vehicle rolled away from the locomotive, traveled several hundred feet down the track (with a drop of perhaps six feet in elevation) and finally collided violently with the concrete face of the bunker at the test pad.

A superficial inspection of the resulting damage indicated that it was apparently confined to nonvital parts of the test vehicle and bunker. As detailed inspection later bore out, the bed of the vehicle itself was not distorted, nor were the graphite reflectors or their support carriages damaged. Nearly everything else on the vehicle was extensively damaged, however, requiring complete stripping and rebuilding of the superstructure of this vehicle to make it available for use. Visual inspection of the air duct on the test vehicle and inside the test bunker showed no apparent damage, although some support braces and nearby parts in the bunker were distorted by the thrust of nearly

GLB-638-5115

Fig. 2-16. Broken exit nozzle clamp.

half a million pounds, which the air duct is thought to have supported in doing most of the work of stopping the 80-ton test vehicle.

A detailed examination of the air duct was of course necessary before it could be used again. Fluorescent-penetrant and magnetic crack detection techniques were applied, x-ray photographs were made, and hydrostatic pressure tests applied. These tests revealed extensive cracking, as well as imperfect seams and slag inclusions in many parts, comprising almost all of the air duct downstream from the heater exit flange, including the spare sections on the second test vehicle, which of course was not involved in the accident. The only damage in the duct sections which could be ascribed to the vehicle accident was in a section near the point of impact, which broke circumferentially under hydrostatic test.

Some flaws were minor surface weld cracks which may have been present from the time of fabrication and would probably not have led to gross failures. Other cracks were large and appeared to have developed at some time after fabrication and initial inspection of the parts. Particularly in view of the nozzle clamp failure, and because of the very serious consequences which would follow rupture of the air duct during a high power test, it was decided that none of the parts would be used at high pressure until a repair program had brought their condition up to meet rigorous criteria for freedom from flaws of all kinds.

For the earliest planned reactor operation at significant power (the Intermediate Power Test of May 14, 1961) it was decided to use temporary sections of mild steel ducting where final replacement parts would not yet be available. Eventually, all test vehicle duct sections downstream from the inlet pipe were to be fully repaired by grinding and rewelding of flaws and cracks, and all sections from the air heater exit flange to and including the test vehicle inlet pipe were to be replaced by new parts fabricated from type 304 stainless steel rather than the original superalloys. This program was carried out as planned.

One other need for repairs had arisen, independently. When the reactor front support structure was examined after the blowdown tests described in Section 2.4, it was found that of the order of half of all of its welds had developed root cracks at the downstream face. Examination of the spare, identical support structure which had been kept in storage showed it to be in the same condition. The cracking, which had resulted from aging

rather than from applied loads, went too deep to be ground out and cast serious doubt on the reliability of these parts. Steps were taken at once leading to procurement of an auxiliary support structure, fabricated without relying on welds for primary load-carrying functions. This would have to be attached to the upstream side of one of the original support structures before reactor testing could proceed.

The early months of 1961 were spent in carrying out these repair programs and then in conducting a series of blowdown tests through a test vehicle without the reactor core, to qualify the facility for reactor testing.

Because of the temporary mild steel sections in the bunker air duct, the first nuclear operation would have to be confined to low values of supply air pressure and temperature. It was decided to use a temperature of 400°F, and to aim for a reactor power of 40 MW. At the design peak fuel element wall temperature of 2250°F, these conditions require an air flow rate of about 120 pounds per second. Facility qualification tests at this time, therefore, were limited to an air temperature of 400°F and a flow rate of 120 pounds per second.

These tests were completed successfully and the reactor, with core in place, was connected at the test pad on April 20, 1961.

2.6 Preliminary Nuclear Tests

During the next two weeks a series of reactor tests was conducted, with these objectives:

- 1) To re-determine the critical control vane angle at ambient temperature,
- 2) to determine the influence of a moderate core temperature increase on critical vane angle,
- 3) to measure changes in core power density distribution with an increase in core temperature,
- 4) to measure the low-power reactor transfer function,
- 5) to test reactor operation with closed-loop automatic servo control, and
- 6) to conduct a rehearsal of the first planned high-power operation, with reactor power the only computer-simulated parameter.

2.6.1 Ambient Temperature Criticality

Initial critical assemblies were made with the core at ambient temperature. With the reflector drained of cooling water, the critical control vane angle was 75° ; and with D_2O coolant it was found to be 67° . In the latter configuration the reactivity worth of each of the sliding control rods was measured at 25 cents.

With heavy water in the reflectors, a repetition of the foil activation measurements described in Section 2.3 was made. Measured values are plotted against core radius in Fig. 2-17, which should be compared with Fig. 2-14 showing corresponding data from the preceding test of December 1960. The apparent differences in power profile in the two sets of measurements have not been explained.

In addition to the foils used for these measurements, a small number of uranium foils in stainless steel envelopes were placed in core tie tubes. These latter foils could be exposed to elevated temperature, and thus would be used to obtain data showing the change in core power profile in going from ambient to elevated temperature.

2.6.2 Elevated Temperature Measurements

In a later series of tests, hot air discharged through the reactor core at about 25 pounds per second was used to raise its temperature first to $220^\circ F$, then to $440^\circ F$, and finally to $635^\circ F$. Figure 2-18 shows the critical vane angle measured at each temperature level. These results fell well within the range of expectations and gave a high degree of assurance that design core operating temperature could be reached with reactivity to spare.

The vanes were moved past the critical angle to put the reactor on a short period while at a temperature of $635^\circ F$. Reference to the inhour relationship, with the measured period and angle, showed an incremental reactivity value of $\$0.124$ per degree of vane angle, for all vanes turned in unison. The value measured at ambient temperature in the experiments of the preceding December was $\$0.125$ per degree, identical within experimental error.

Finally, the reactor was operated at 10 kilowatts power for about one minute, at a core temperature of $643^\circ F$, to produce activation foil data for comparison with the preceding ambient temperature results. Only a small

GLL-638-1981

Fig. 2-17. Relative local power density, integrated over core length, as a function of radial position in core. April, 1961.

GLL-638-1982

Fig. 2-18. Critical control vane angle measured at three values of uniform core temperature. April, 1961.

number of data points were measured with these stainless steel-enclosed foils, but a good indication was obtained that the core radial power distribution changed in form about as expected with increased temperature, and that it would be acceptably flat in high power operation.

2.6.3 Reactor Transfer Function

One test operation, at power levels up to 10 kilowatts, was devoted to measurements of the "zero-power" open-loop reactor transfer function, through use of one of the control rods as an oscillator rod. Results were in agreement with expectations, as has been discussed in Ref. 3. Satisfactory completion of this test was a necessary prelude to any attempt to close the automatic servo power control loop.

2.6.4 Dress Rehearsal

On May 3, 1961, a blowdown test was run, with stored air discharged through the core at temperatures up to 400°F and flow rates up to about 120 pounds per second. Air flow rate and temperature were programmed to duplicate the procedure planned for the Intermediate Power Test scheduled for May 12. Reactor power and temperature were simulated by an analog computer, and every effort was made to conduct this run in the same way as if these latter parameters were real rather than simulated.

The operation was carried out satisfactorily, without any special incidents.

2.6.5 Reactor Power Servo Control Achieved

Up to this time all critical reactor operation had been carried out by manual control of power. Each control element normally was positioned by a separate closed-loop servo system, and the position demand signals for these systems were supplied by the operator to raise or lower reactivity as he desired.

The first effort to achieve closed-loop automatic control of reactor power was made on May 3, following the blowdown test described above. Through operator error, the vernier control rod was not placed in its equilibrium position before closing the loop; and the resulting transient oscillation in rod position, on transfer to automatic control, caused tripping of a period scram interlock.

Following a day's examination of the control system recorder chart traces to make sure that this incident was well understood, another test was conducted in which automatic power control was achieved smoothly and without incident. A series of step changes in power demand were made, and were met by proper and stable response of the reactor and control system.³

The automatic log power and period control systems were tested in this procedure, but no effort was made to control the reactor on a linear power signal. As it developed, the automatic linear power control mode was never used throughout the life of the reactor, since the logarithmic power mode proved entirely satisfactory for all needs and was more convenient.

The remaining time before the Intermediate Power Test of May 14, 1961, was occupied by final preparation and checking of the facility. The conduct and results of that test are described in detail in other sections of this report.

2.7 Final Preparations for High Power Tests

Following the successful test of May 14, which was run with supply air at temperatures and pressures far below the design rated capacities of the facility, almost four months elapsed before final duct sections and V-band couplings were installed to permit design point operation of the reactor.

Meanwhile, one more facility difficulty had become apparent, but fortunately without causing further delay:

Studies of performance of the stored-energy heater during high temperature blowdowns showed that it contained an inadequate mass of steel to supply its rated quantity of heat to the test air, even when the initial temperature of the steel was brought up to its design figure of about 1350°F.

After a number of test operations, inspection of the oil-fired furnace which supplied hot air to charge the heater showed that its heat-exchange ducting had scaled quite severely, with complete penetration of the duct wall at a few points. To avoid complete loss of this furnace, it would have to be operated as much as 100° below its rated output air temperature for all succeeding operations, further lowering the amount of heat storable in the heater.

Because of these circumstances, it was not possible to achieve a test run of useful duration with the original design point air temperature of 1063°F.

By this time, the standard design point Mach number for the conceptual Pluto missile had become 2.8 instead of the original 3.0, to which the 1063°F ram temperature corresponds. It was felt entirely appropriate to redefine the Tory II-A design operating point to correspond to Mach 2.8 missile operation, with a corresponding change of inlet temperature to 946°F. The change was not significant with respect to operations of the reactor itself, but it permitted achievement of the originally contemplated test durations without rebuilding the furnace and heater.

When the test vehicle and facility had been made ready for use, a short series of blowdown runs was made, without the reactor core, to check the operation of all systems and to rehearse the planned high power tests. Figure 2-19 is a photograph made at this time, with cold air discharging from the test vehicle.

Re-installation of the complete reactor at the test point took place on September 15, 1961, and was followed by two preliminary test runs. In the first of these, proper operation of the (slightly altered) closed-loop power control system was verified at power levels up to 10 kilowatts. The second test involved discharge of 800°F air through the core, at low flow rates, to verify operation of the core thermocouples and associated data systems.

No other activities, except for final facility systems checkouts, took place before the final series of three full power tests on September 28, October 5, and October 6.

3. AS-BUILT REACTOR CHARACTERISTICS

The physical characteristics of the Tory II-A reactor core, as they existed during all elevated-power operation, are listed in Table 3-1. Corresponding values of operating parameters at the Mach 2.8 design point are given in Table 3-2, and Figs. 3-1, 3-2, and 3-3.

Definitions of symbols used in the figures follow:

T_w Flow channel wall surface temperature

T_i Internal wall temperature

T° Air total (stagnation) temperature

p Air static pressure

p° Air total (stagnation) pressure

g Power density distribution; $\int_0^1 g(\epsilon) d\epsilon = 1.$

GLB-638-5116

Fig. 2-19. Cold air blowdown test. Test vehicle is complete except for reactor core.

Table 3-1. Physical characteristics of reactor core

Core Length									
Inlet reflector	.	.	.	.	.	.	.	.	6.0 in.
Fueled region	.	.	.	.	.	.	.	.	48.5 in.
Exit reflector	.	.	.	.	.	.	.	.	<u>2.0 in.</u>
Total ceramic	.	.	.	.	.	.	.	.	56.5 in.
Equiv. circular radius	.	.	.	.	.	.	.	.	16.35 in.
Total frontal area	.	.	.	.	.	.	.	.	840.00 in. ²
Fractional frontal area displacements									
Fuel tube material	.	.	.	.	.	.	.	.	0.46
Propellant flow passages	.	.	.	.	.	.	.	.	0.32
Structural BeO	.	.	.	.	.	.	.	.	0.111
Tie rod material	.	.	.	.	.	.	.	.	0.025
Tie rod flow passages	.	.	.	.	.	.	.	.	0.027
Incidental void	.	.	.	.	.	.	.	.	<u>0.057</u>
									1.000
Number of fuel element flow passages*	.	.	.	.	.	.	.	.	8623
Fueled flow passage diameter	.	.	.	.	.	.	.	.	0.2005 in.
Total mass of uranium in core†	.	.	.	.	.	.	.	.	49.1 kg
Average weight proportion of UO ₂ in fuel element material	.	.	.	.	.	.	.	.	5.95%
Fuel element configuration	.	.	.	.	.	.	.	.	See Fig. 2-1

*Includes about 60 dummy tubes containing thermocouple leads.

†Enriched to 93.2% in U²³⁵.

Table 3-2. Design operating parameters

Equivalent missile flight condition		Mach 2.8 1000 ft. altitude 100°F ambient air
Inlet total temperature		946°F
Inlet total pressure*		316 psia
Total air flow rate		634 lb·sec ⁻¹
Air flow rate through fuel elements		495 lb·sec ⁻¹
Flow rate per fueled passage		25.9 g·sec ⁻¹
Max. fuel element wall temperature		2250°F
Fuel element exit air total temperature		1945°F
Mixed-mean exit air total temperature		1763°F
Exit plenum total pressure		180 psia
Reactor pressure drop		136 psia
Fuel element exit Mach number		0.62
Total power to air stream		150 MW
Total power delivered in fuel elements		145 MW
Power density		
averaged over total fuel element volume		278 watts·cm ⁻³
averaged over fuel element material volume		472 watts·cm ⁻³
peak value in fuel element material		616 watts·cm ⁻³
Power per fueled passage		16.8 kilowatts
Peak thermal stress in fuel elements		18,600 psi

*Corresponds to missile inlet pressure recovery of 0.80.

GILL-638-1983

Fig. 3-1. Fuel element design point operating conditions.

Fig. 3-2. Tie tube design point operating conditions.

Fig. 3-3. Shroud design point operating conditions.

M Mach number

σ Thermal stress, calculated on elastic model.

As explained in Chapter 2, the changes from original specifications can be ascribed to two causes: first, a need to gain increased reactivity over the original configuration, and second, a redefinition of the design point to correspond to missile operation at Mach 2.8 instead of Mach 3.0. Response to the former was of course necessary to provide the means of attaining criticality at high temperature. The latter move, while consistent with current Pluto missile design, was mainly occasioned by inability of the test facility air furnace and heater to meet their intended capacity.

No changes were made from the original design which were significant in terms of reactor performance or severity of reactor environment.

4. SUMMARY OF INTERMEDIATE- AND HIGH-POWER TESTS

4.1 Principal Purposes and Results

This section will give a general account of the four reactor tests at substantial power, presenting the purpose of each test and a description of its results in terms of the primary operating parameters such as power, temperature, and so forth. A more detailed analysis and interpretation of the results is given in Section 4.2, and a discussion of available information on effluent material in Section 4.3. Chapter 5 describes the condition of the core and its components as determined following complete disassembly.

Table 4-1 shows achieved values of the primary reactor operating parameters for each of the four main tests, together with design point values.

4.1.1 Intermediate Power Test

Early plans for the operation of Tory II-A called for a long series of tests leading up through progressively higher temperature and power levels to eventual operation at the design point. However, experience with the very considerable effort involved in accomplishing even a single test with a facility as complex as the Pluto NTS facility, realization of the relatively high degree of hazard to the reactor which is incurred with intermittent operation, and finally, greatly enhanced confidence in predictability of reactor behavior, led to abandonment of the original plans for numerous runs.

Table 4-1. Tory II-A Design and achieved operating parameters*

Test designation	Inlet air stagn. temp (°F)	Air-flow rate (pps)	Max power (MW)	Max av. fuel element temp (°F)	Av. exhaust air stagn. temp (°F)
Design op. pt.	946	634	150	2250	1763
IPT, May 14	393	114	46	2580	1840
HP-1, Sept. 28	394	445	144	2330	1560
HP-2, Oct. 5	925	650	166	2300	1810
HP-3, Oct. 6	402	432	162	2640	1745

*All parameters correspond to the "best value" operating points derived as shown in Section 4.2.1, except for the fuel element temperature values, which are averages of measured temperatures at the hottest station of the core, rounded off to the nearest 10°, and corrected to give wall surface temperature.

By the time the reactor test program started, in December 1960, it had been decided that all of the testing could be carried out in three steps: first, a series of preliminary runs at very low power, principally to test the neutronic behavior of the reactor and its control system; second, one or two tests at moderately high power, which would offer some possibility of discovering and eliminating potential causes of trouble without committing the reactor to the severe conditions of design point operation; and finally one or more tests at design point.

The preliminary tests detailed in Sections 2.2 through 2.6 gave good assurance that the reactor was ready for operation at high power. It was decided that one test should be run at a moderately high power level, a core temperature near the design value, and with inlet air temperature reduced as required by the temporary air duct sections in the test bunker (see Section 2.5).

This test, designated Intermediate Power Test (IPT), was intended to provide a first test of the reactor at power levels which would produce heat transfer conditions in the core similar to those expected at the design point. Specifically, heat transfer to the air stream would predominate over radiative and conductive heat transfer among the core components. A power of 40 MW was selected as being high enough to meet these conditions, while

still not producing thermal stress levels comparable with those expected at the 150 MW design point power. The average fuel element wall temperature at the hottest axial core station would be taken to the design point value of 2250°F. The corresponding airflow rate would be 120 lb/sec, at a reactor inlet temperature of 400°F.

Test air would be supplied through the smaller of the two available flow control and metering systems. A reliable measure of flowrate would thus be available at the choked flowmeter FM-2, but not at the reactor inlet diffuser, which would operate unchoked because of the reduced inlet air temperature.

The mode of operation, in brief, would be as follows:

- a. Bring reactor airflow rate up to 30 lb/sec, at ambient temperature.
- b. Raise reactor inlet air temperature to 400°F.
- c. Raise reactor power until peak core temperatures are 1900°F.
- d. Raise power and airflow rate simultaneously, keeping core temperature constant at 1900°F, until flow rate is 120 lb/sec.
- e. Raise power until peak core temperatures average about 2250°F, or until individual thermocouples reach 2500°F.
- f. Shut down approximately in reverse order.

It had been decided before the run that since some points in the core were likely to show temperatures well above the average over the hottest core region, individual thermocouple readings should be allowed to go as high as 2500°F in order to be sure of bringing the average up to 2250°.

Following two days' postponement after the scheduled date, because of adverse weather, the test run was started shortly before noon on May 14, 1961, with all components of the facility and reactor in satisfactory operating condition.

About two hours were spent in bringing the reactor up to the 100-kW power range, and then about one-half hour in going through the operating sequence listed above. This procedure was carried out approximately as planned (see Chapter 6 for a detailed account). Figure 4-1 shows a history of reactor power and airflow rate during the run, and the reading of a typical thermocouple in the hottest part of the core.

Fig. 4-1. Indicated power, temperature, and airflow rate during Intermediate Power Test. Power trace is from uncorrected ion chamber current.

The only striking departure from planned operation was in the core temperature, which exceeded the planned 2250°F value by about 300°. It had been recognized before the run that temperatures well beyond the design point might occur, and could occur without harm to the core. The peak local measured value of greater than 2600°F reached during the run was higher than had been planned, however. This was caused, in part, by a lack of clear information available to the reactor nuclear operator, and was eliminated before subsequent test operation. A discussion of the circumstances is given at the end of this section.

The highest reactor power that was held steady for any appreciable length of time during this test was 46 MW (actual peak was 48 MW). Duration of the period of near-peak power and core temperature was about 120 seconds, while the total period during which the power was at 10 MW or greater and the average peak core wall temperature at 1900°F or higher was about 800 seconds (13 minutes).

A short digression should be made here to comment on the concept of core temperature. At any moment during a run, different general regions of the core were at different temperatures. The design intention was that gross variations in core temperature should take place only in the axial direction. Azimuthal temperature variations at a given axial station were not expected to be large, and radial variations were kept small by the radial gradation in fuel loading density. Within the framework of this general picture, then, each axial station is characterized by a typical temperature. In detail, however, there were of course relatively minor variations in temperature at a given axial station caused by random and systematic variations in local fuel density, concentration of unfueled beryllia or structural metal, fuel element wall thickness, proximity to control elements, and so forth. Unless the exception is stated, core temperatures referred to in this report are values characteristic of a given axial station averaged over the radial and azimuthal variations. Finally, it must be pointed out that a temperature gradient, supporting outward heat flow, existed in each fuel element normal to the tube axis. Temperature differences up to 80°F or so existed between the wall surface adjacent to the air stream, and the hexagonal outer surface of an element. Temperature at the wall exposed to the air stream is the value of by far the greatest importance, since this is the location

of peak tensile stress in the elements as well as the site of corrosive attack on the element material. Consequently, unless the contrary is stated, the term core temperature should be taken to mean temperature at the air-cooled surface. Thus "average peak core temperature" means the value of flow passage wall temperature, at the hottest axial region of the core, averaged over radial-azimuthal variations.

The early 1900°F-10 MW plateau was held for a long period of time (ten minutes) during the Intermediate Power Test in order to review all data system readings, as a last check for possible trouble spots before going to higher power. Most attention was given to readings of thermocouples in the fuel elements and core structural components, where heat was being generated internally for the first time. No cause for alarm was found, although depletion of the stored air supply made it necessary to go on to the main part of the run before all readings could be studied as thoroughly as had been wished.

The high-power portion of the run was then carried out, with the nuclear operator adjusting power to produce desired core temperature indications from a selected group of core thermocouples. The pronounced variations in power level during this period, as seen in Fig. 4-1, were the result of several factors, notably the slow rate of temperature response at this power level and the difficulty of interpreting the somewhat conflicting temperature information available. The power level at the brief flat top of the last peak was 46 MW.

No evidence of damage to the reactor was noted during this test. No unexplained fluctuations in air pressure, core temperatures, or control element position were observed. Effluent air radioactivity measurements are somewhat lower than expected, although the difference was within experimental uncertainties. Photographs were taken looking through the core against a bright background after the run, and showed no evidence of blocking of air flow passages.

The methods used in determining reactor power and temperature, together with a more extensive selection of measured or derived data for this test, are presented in Section 4.2.1 and Appendix A. Chapter 6 gives a detailed account of the preparations for and progress of the test, and a description of the data system will be found in Appendix C.

Circumstances of Unintentional High Core Temperature. In order to clarify the circumstances resulting in the deviation from planned core temperature in the Intermediate Power Test, a brief discussion will be given here of the manner in which temperature information was displayed at the time of this run.

The Nuclear Operator's console was equipped with ten temperature-indicating meters of a vertical in-line type thought to be adapted to easy comparison at a glance. It had been thought that a good fraction of the core thermocouples might be lost during the first run, so this type of display had been chosen to allow the operator to ignore those readings which were out of step with the majority.

A hot-wire probe had been inserted into the core air flow passages, with the reactor at the test pad, to ensure that output channels in the control room actually corresponded to couples in the proper core locations. Only limited success was achieved with this probe, owing to difficulty in inserting it in some holes and in delivering enough heat to get a clear indication. The locations of 22 fuel element thermocouples were clearly established, however.

The selection of thermocouples for the Nuclear Operator's meters was based upon these rules:

- a) eight were to read fuel element temperatures, two to read exit air temperature;
- b) fuel element thermocouples were to be at an axial station (fraction of core length measured from the front) $X/L = 0.7$ or 0.9 ; and
- c) only probe-located fuel element couples were to be used.

Five separate transmission systems were used. Six core thermocouple signals were transmitted via an ASCOP multiplex telemetering system, and two core thermocouples and the two exit air thermocouples via four single-channel transmission systems.

At another station in the control room, the Core Data Observer could watch temperature readings from all core and air thermocouples (over 100 in number) and announce any situation in need of correction.

Data on the ASCOP system, as well as control room voice communication, was recorded on magnetic tape and could be played back after the run, duplicating operators' meter readings, other temperature readings, and

operators' conversation during the run. Doing so led within a short time to a thorough understanding of what happened during the test, which may be summarized as follows:

During the preliminary 1900°F plateau period, the Nuclear Operator and Core Data Observer correlated their temperature readings and decided which of the operator's meters appeared to be most reliable. Three were definitely designated unreliable at this time.

After taking the power to near-peak value, the Nuclear Operator found additional divergence among his meter readings, and had to give up use of the one first thought most reliable when it became noisy. He finally centered his attention on one meter (meter "D" reading thermocouple #147) and leveled it at 2250°F. As it turned out, although meter "D" gave one of the highest readings at the 1900° plateau, it had the lowest reading of the meters still in action at peak power.

This could have happened with any of the thermocouples, but the reason it did happen with this particular couple became clear when it was discovered that, owing to an error in interpreting records, the couple was not actually in a fuel element but rather in an unfueled component whose temperature could be expected to diverge from that of the fuel elements as power and air flow were increased. The difference between this meter reading and that of the others was not great enough to attract the operator's attention easily. The only one reading above 2500°F happened to be farthest from the operator, and the two nearest him were reading only 2270° and 2340°.

The Core Data Observer, after looking at many core and air temperature indications, did realize that the core temperature was over the agreed-upon limit, but by the time he was sure of this he knew that the time for shutdown was almost at hand anyway, and so he decided to avoid confusion by keeping quiet.

At the time of the three later tests at full power, these sources of error had been eliminated. Avoidance of the unfueled part was of course called for. The couples used for later runs were selected on the basis of satisfactory performance during the Intermediate Power Test. Finally, the Nuclear Operator was given a large paper chart recorder display of the average reading of six core thermocouples. This system proved entirely satisfactory (see Appendix C-4.2).

4.1.2 High Power Test 1

Following final qualification of the test facility for operation at reactor design point conditions, one more test run at conditions short of the design point was planned. This was to take the reactor approximately to design power and design core temperature, with reduced inlet air temperature. Fuel element conditions (temperature and thermal stress) as severe as those at design point would thus be achieved, while mechanical loads and structural component temperatures would be below design point values.

A power level of 150 MW was selected, with a peak fuel element wall temperature of 2250°F and an inlet air temperature of 400°F. The sequence of operations would be essentially the same as had been used in the IPT, except that the 1900° plateau preceding full power operation was eliminated. As before, the air flow rate and reactor power were separately controlled, through servo systems, by their respective operators. The Airflow Operator followed a preplanned program, and the Nuclear Operator, with much the faster-responding system, adjusted power to produce desired temperature values.

Again, because of the low reactor inlet air temperature, the inlet diffuser would operate unchoked and so would not provide a useful measurement of airflow rate. Air was provided through the small flow valve (AV-4) and flowmeter (FM-2). The tank farm pressure necessary to drive the required 448 lb/sec flow rate through this line was quite high, about 2000 psi, so that only half of the stored air from an initial pressure of 3800 psi was available for the test. This forced a relatively brief period of operation, putting a premium on getting up to full power and temperature as rapidly as possible to maximize the time available there. The advantage of having a choked flowmeter to provide a feedback signal to the airflow servo-control system was felt to outweigh this disadvantage, however.

Two changes at the Nuclear Operator's console were of special interest for this run. The core temperature averaging system discussed in Appendix C-4.2 had been installed and was expected to eliminate the difficulty in properly gauging core temperature which had been experienced during the IPT run of May. In addition, the earlier system of setting automatic power demand with a potentiometer had been changed to one in which the operator could select a rate by pushbutton, then change power demand at that rate by

pressing a switch. Both of these changes were intended to bring smoother and more accurate control of power and temperature.

The test took place on schedule, on September 28, 1961, without any special incident. All systems worked well and all test objectives were achieved. Again, no indication of damage to the reactor or of any degradation in performance was observed.

Figure 4-2 shows the history of reactor power, peak core temperature, and airflow rate during run HP-1. As shown, reactor power and peak core temperature were essentially constant over a period of about 50 seconds at their highest values, with the temperature differing only slightly from its planned value.

The highest steady reactor power during this test was 144 MW, with the core temperature simultaneously at its highest value of 2330°F. Additional measured and derived data from this test will be found in Section 4.2 and Appendix A.

The data taken during this run indicated that no difficulty should be anticipated in taking the reactor to design point conditions. The only specific doubt on this point that remained after the initial run in May (IPT) had concerned possible overheating of the wiper seals, and the new run (HP-1) showed that with increased power and airflow the wiper temperature dropped to expected and proper values, so that even with the inlet air temperature increased to the design point value, no excessive temperatures should be expected in these components.

4.1.3 High Power Test 2

No changes to the facility or reactor systems were required after run HP-1, and so the next test was scheduled four days later, Monday, October 2. The objective of this test was to achieve design point conditions, or slightly higher, in every respect. The power level would be 150 MW, peak core temperature 2250°F, airflow rate 634 lb/sec, and inlet air stagnation temperature 946°F. The program followed would otherwise be identical to that of HP-1, with the possible time at high power again somewhat limited, not now by stored air capacity but by the heater capacity. The high power plateau was to be ended just before the inlet air temperature began to drop due to exhaustion of stored heat.

GLL-638-1987

Fig. 4-2. Indicated power, temperature, and airflow rate during High Power Test 1. Power trace is from uncorrected ion chamber current.

The high airflow rate required for this run made it necessary to use the larger of the two control valve-flowmeter systems (AV-3 and FM-1). Because of the elevated reactor inlet air temperature, the inlet diffuser would operate with choked flow for the first (and only) time in any of the reactor tests, and would provide a second measurement of airflow rate. A change in operating procedure was made, in that airflow rate over the peak portion of this run was varied according to a preset program by an automatic device in the airflow servo-control system. The feedback signal for airflow automatic control was taken from a total pressure transducer at the choked inlet diffuser.

All systems were ready for operation on the scheduled date, but adverse weather conditions forced postponement of the test from that day and each of the two following days. On October 5 a steady southwest wind held and approval was received for operation. The test was completed smoothly, and with all objectives achieved, the core power and temperature both having been held slightly above design point values for a period of 40 seconds.

Figure 4-3 shows traces of power, peak core temperature, and airflow rate during HP-2. Exceptionally smooth control of temperature and power was achieved in this run, partly because of increased operator experience and partly because the physical provisions for airflow and power control were optimum for the run, in that the improved automatic power demand panel and the temperature averaging network were in use, and that airflow control from a choked orifice near the reactor gave a very steady flow rate.

The values of power and core temperature on the plateau were 166 MW and 2300°F, with 650 lb/sec airflow rate. Additional data in more detail can be found in Section 4.2 and Appendix A.

Figure 4-4 is a reproduction of the core temperature averaging network trace for HP-2. It illustrates the ease with which core temperature could be followed by the nuclear operator even though some thermocouples chosen for his use failed during the run. The sharp vertical motions of the trace at 1080 sec, 1110 sec, and 1205 seconds mark points where individual couples failed (producing a shift in the reading) and were then switched out of the network (returning the reading to about the previous level). The Core Data Observer was responsible for monitoring the single outputs of the couples and for switching out any erratic ones.

GLL-638-1988

Fig. 4-3. Indicated power, temperature, and airflow rate during High Power Test 2. Power trace is from uncorrected ion chamber current.

Fig. 4-4. Output of temperature-averaging network during High Power Test 2.

With completion of this test, proper operation of the reactor at design point had been demonstrated. No evidence had been seen during any of the three runs at elevated power of damage to the reactor; although, of course, any relatively minor internal core damage such as fuel element cracking, incipient coating failures, breakdown of non-vital parts and the like was not likely to be observable.

4.1.4 High Power Test 3

Although the smooth operation of the reactor up to this time was gratifying, some question remained as to whether the optimum yield of information had been secured. There seemed to be a reasonable probability, for instance, that the limiting operating conditions of the core had not been approached closely enough to produce much permanent change in its components.

It was decided, then, that one more test should be run, with the aim of providing information on what degree of severity of operating conditions beyond the design point could be tolerated. Reactor power would be taken to about the design point value, with core temperature raised to at least 2500°F, the design value for the successor reactor, Tory II-C. Inlet air temperature would be 400°F, permitting the greatest possible duration of run (required airflow rate decreases as reactor inlet air temperature decreases, and the air heater limitations are also avoided).

The operating program would be similar to that of the preceding high power runs, with control of airflow by operator action throughout the program rather than by automatic programmer during the highest power portion.

Because of the high airflow rate and long run duration desire, it was necessary to supply air through the large control valve and flowmeter (AV-3 and FM-1). FM-1 would not choke except for brief transient periods and could not supply a directly flow-proportional pressure signal for feedback to the flow servo-control system. At an air temperature of 400°F, the reactor inlet diffuser would not choke either, and so also would not provide a flow-rate feedback signal. Consequently the automatic flow control system could not be used, and control would be accomplished by operator regulation of valve position.

With only a relatively small charge required in the stored-energy heater, the time necessary to prepare for operation was fairly short. Good weather conditions continued, and the test was conducted on the following day, Friday, October 6, 1961.

High Power Test 3 was entirely successful and was completed according to plan. The peak average core temperature achieved was about 2640°F, at a power of 162 MW and an airflow rate of 432 lb/sec. Individual thermocouple readings indicated that some local regions of the core were at temperatures well above 2700°.

Figure 4-5 shows traces of the main operating parameters throughout run HP-3. Manual control of airflow rate resulted in considerable unsteadiness in comparison with automatic control. The Nuclear Operator's efforts to prevent core temperature wobbles resulting from the airflow rate variations account for the irregular power trace. Duration of the period of peak power and temperature was 95 seconds, while the total time during which the average peak core temperature exceeded 2250°F was about 150 seconds. Detailed data will be found in Section 4.2 and Appendix A.

Throughout this final test, as in all of the preceding ones, the reactor operated smoothly and gave no evidence of damage or irregular operating conditions. Again, it should be realized that no very sensitive indicators of minor damage were available. One could not expect to see fluctuations in the airflow rate, for instance, before really gross structural failure of the core had started. Local hot-spot damage might chance to occur where no thermocouple was situated to record it. The two indicators that were regarded as most encouraging in giving no sign of damage were control vane angle, which varied only as could be explained by kinetics requirements and changing core average temperature, and the relatively low level of radioactive material observed in the exhaust air stream by downwind detector stations. It was estimated that loss of as much as 1% of the reactor fuel or 0.1% of the BeO in the core during a run would have been observable in an increase in control vane angle, and the loss of as much as 1% of the fission product inventory rather than the predicted 0.1% would have been clearly distinguishable in the effluent activity. Section 4.3 describes the effluent characteristics more fully.

Fig. 4-5. Indicated power, temperature, and airflow rate during High Power Test 3. Power trace is from uncorrected ion chamber current.

As will be discussed in detail in Chapter 5, observations made following complete disassembly of the core confirm that damage was minor. No case was observed in which integrity of a fuel element air flow passage was lost, although extensive fuel element cracking due to thermal stress and stress-corrosion effects did occur. One baseplate coating failure occurred, leading to formation of a 1/8-inch-diameter pit in the base metal. Many unfueled ceramic links cracked, but it is not clear to what extent this occurred in operation, and to what extent in the disassembly operations. A number of metal links at the exit end of the core partially melted, indicating that temperatures in some of the module-periphery triangular flow passages were higher in relation to the fuel elements than expected. These links had been intended to provide extra assurance that exit reflector tubes would not shift away from their matching baseplate holes and cut off coolant flow. Fortunately, it appears that they were not needed, and that their loss caused no additional damage.

The basic objectives considered in planning the series of test operations were, first, to verify proper operation of the reactor at its design point conditions; and second, to expose it to sufficiently more severe conditions to provide information on what safety factor was available in beyond-design operation and to identify the probable first sources of failure, without actually destroying the reactor and making post-test examination too difficult to interpret. It is felt that these objectives were met in all respects.

4.2 Analysis of Reactor Performance

The present section will describe the detailed reactor performance, as recorded during the four elevated power tests or derived from test data.

Before undertaking to analyze and interpret the recorded data, one must make some basic assumptions and choose a frame of reference. In design analysis, it is usual to adopt the point of view that the basic operating parameters (power, air flow rate, inlet air temperature, etc.) are independent variables and may be specified at whatever values one wishes, and that one may then inquire as to the values of a number of parameters describing detailed effects, such as local pressure, fuel element temperature, etc., the latter being regarded completely as dependent variables.

The situation is different in handling the data from an actual reactor test, where the available information consists of a large number of measurements of single parameters (mostly temperature or pressure values) taken at various locations. With each measurement is associated an estimated experimental uncertainty. The process parameters usually assumed as basic, especially total power and airflow rate, are not known accurately, and in fact are usually not even the subject of really direct measurement. By making use of aerodynamic and heat transfer theory, one can derive these process parameters from any one of many possible sets of measured parameters (qualified by experimental uncertainties), but at least one logical pitfall must be avoided, which is this: an otherwise attractive means of deducing the total power and airflow operating conditions may involve assuming that the reactor internal heat transfer and airflow processes operate exactly according to design intentions. It must be remembered, after all, that the reactor is the object of testing, and that its detailed functioning cannot be assumed known until actually measured. For example, it would not be legitimate to use measurements of peak core temperature or reactor inlet air pressure to determine the reactor power and total flow rate, since assumptions on heat transfer and pressure drop within the reactor must be made to carry out such a determination.

It must also be noted that substantial experimental uncertainties are associated with most of the measurements made, and so necessarily also with any derived quantities, such as reactor power. The available data have been treated in such a way as to reduce these uncertainties as much as possible.

4.2.1 Determination of Power and Airflow Rate

Nuclear Measurements. Two classes of measurement are available from which reactor power levels may be inferred.

In one class are measurements based directly on the rate of nuclear reactions in the core. Ionization chamber measurements of neutron leakage fluxes during the test runs, and radiochemical determination of residual fission product concentrations after the test series fall into this class. Unfortunately, both of these measurements have strong disadvantages.

Ion chamber current is of course only, at best, a relative measure of neutron flux, which must be calibrated in some way against an absolute measurement. An effort was made to do this during the preliminary reactor tests, but unfortunately the only calibration* made with the final detector configuration was invalidated through accidental sample contamination. Even without this accident, a sizeable experimental error would have been introduced in going from a low power calibration to a high power test run, through lack of ideal linearity of the detectors as well as the necessity of relating several overlapping stages of detection, at increasing power levels. Further, the detectors do not measure actual core power, but rather the flux of neutrons leaking from the outside of the reactor and traversing various paths to the detectors. Two particularly strong sources of error are involved in the latter issue: The logarithmic-amplified ion chambers P_1 , P_2 , and P_3 , which were used for automatic reactor control, as well as the linear chamber C_1 whose signal was displayed on a chart recorder for the nuclear operator, were located in a position where the neutron flux reaching them was strongly influenced by changes of air density in the main air duct to the reactor. The other linear ion chambers - L_1 , L_2 , and L_3 - were located where most of the flux reaching them was leakage from the reflector, and so was sensitive to the control vane position and hence to the average core temperature. Some such difficulty would have to be accepted no matter where the detectors were placed.

Determination of various fission product inventories after reactor disassembly can at best only give a measure of the total number of fissions which have occurred in the core during each run or group of runs occurring on or near a given date, with no distinction between the contributions of several runs occurring only days apart, such as HP-1, HP-2, and HP-3. Moreover, to infer the value of reactor power at a particular instant in a particular run from this data, one must make use of the ion chamber data to go from integrated power to instantaneous power.

Radiochemical power determination, discussed in Section 5.2.3, will be treated as confirmatory evidence bearing on the power levels attained, rather than as primary data useable in determining these values.

* The foil activation run of April 24, 1961: see Section 2.6.1.

Calorimetric Measurements. Two general kinds of calorimetric power determinations are possible. One would involve the effects of charging and discharging the heat capacity of the core material, and so would proceed from a study of rates of temperature change of core components in transient power or airflow conditions. Because of the great uncertainty in knowing just what the transient conditions were at any moment, this method has not been used.

The other type of calorimetric measurement, which is the source of the power and flow rate determinations accepted as valid here, involves measurements of the thermodynamic conditions of the reactor air stream at various points in its duct, upstream and downstream from the reactor, with all operating conditions as near steady as possible.

The specific primary parameters which were measured for this purpose are the following:

- $\dot{w}$ Airflow rate at bunker flowmeter FM-1 or FM-2 (temperature and pressure are the quantities actually measured, but they are always used in the form of the single variable, air mass flow rate).
- T_0° Air total temperature upstream from the reactor (unchanging from heater through inlet diffuser to reactor inlet face).
- p_0° Air total pressure at the inlet diffuser throat and in the duct just upstream from it.
- T_2° Air total temperature downstream from the reactor.
- p_2° Air total pressure in the reactor exit plenum and nozzle.

These variables, together with reactor power, can be related to one another by two equations, which describe the processes of heat addition to an air stream, and mass flow rate through a choked nozzle, in terms of the well-known thermodynamic properties of air. No assumptions are needed as to other interactions between the reactor and the air stream. These equations are:

$$P = \dot{w} \int_{T_0^\circ}^{T_2^\circ} c_p dT \quad (4.1)$$

$$\dot{w} = f(T_i^\circ) C_N A_N \frac{p_i^\circ}{(T_i^\circ)^{1/2}} \quad (4.2)$$

where f is a very slowly varying function of T_1° ,
 P reactor power delivered to the air stream,
 $\dot{w}$ air mass flow rate,
 c_p specific heat of air at constant pressure,
 C_N nozzle (or diffuser) discharge coefficient, and
 A_N nozzle (or diffuser) area.
 i is 0 at the diffuser or 2 at the exhaust nozzle.

Equation (4.1) expresses conservation of energy in the reactor-air stream system. Most standard texts on compressible fluid flow show the derivation of the well-known choked nozzle condition stated by Eq. (4.2), as well as the form of the function f . See for instance Ref. 4.

If known exactly, values of any three of the five primary parameters listed above, taken with Eqs. (4.1) and (4.2), would be sufficient to determine exactly the values of the others as well as the value of reactor power. Knowledge of all five measured parameters would provide several alternate derivations of power and flow rate, which ideally would coincide. In reality, these parameters are not known accurately, but only to within some experimental error. Consequently, determinations of power and flow rate made from different combinations of the parameters do not agree. Ideally, all five parameters could be used simultaneously, according to proper statistical principles, to derive a single most probable set of values of power and flow rate, with final standard deviations derived from the estimated standard deviations of the individual measurements. To obtain such solutions analytically is a very tedious process and was not thought to be worth undertaking here. Some early estimates of power and flow rate were obtained by making only partial use of the measurements available, but this involves making arbitrary subjective choices among the data and so is rather unsatisfying.

The process of data reduction finally chosen is a graphical one which makes systematic use of all of the data available, and has the additional advantage of making easily apparent the relationship of each measurement to the final result. No effort has been made to prove the rigorous general validity of this method, but it is intuitively satisfying and is believed to give essentially an optimum result. For special cases it can be shown analytically to give a solution having minimum variance corresponding to expected uncertainties in the measured parameters.

Method of Determination of Power and Airflow Rate. The process may be described as follows: The six variables P , $\dot{w}$, T_0° , p_0° , T_2° , p_2° are related by three equations: Eq. (4.1) and Eq. (4.2) applied to both diffuser and exhaust nozzle. Thus there are three independent variables. Consider the three-dimensional space with coordinates representing P , $\dot{w}$, and T_0° . Identification of a single point in this space serves to identify all six of the variables of interest. Assignment of a specific value to any one variable will define a surface in this space. Thus to any given single measurement there corresponds a surface in this three-space, indefinite in position in proportion to the estimated standard deviation of the measurement to which it corresponds. Imagine the set of these surfaces corresponding to all of the experimental values determined during a given test run. If the measurements are perfectly accurate, the surfaces will all intersect in some common point. For a set of real measurements, the surfaces will fail to intersect at a common point, but there will be an identifiable region in the three-space in which they all come closest to intersecting. The center of this region, located with appropriate weighting, will represent the most probable set of simultaneous values of the variables. If valid standard deviations have been assigned to the measurements, it will be found that the corresponding surfaces tend to miss intersecting the "best-value" point by distances comparable to those representing the standard deviation of each.

The difficulties of handling three-dimensional graphs prevent carrying this process out literally as described, but a solution can be obtained quite satisfactorily by looking at a plane slice through the three-space, namely a surface of constant T_0° corresponding to the measured value of that variable. The other experimental value-surfaces intersect this plane as lines. Neglecting for the time being the effect of uncertainty in the actual value of T_0° , note that the position of each such line depends only on the result of a single measurement and is statistically completely independent of the uncertainties in position of all of the other lines. Figure 4-6 shows an example of such a plane, with curves corresponding to measured values of $\dot{w}$ (flow-meter), p_0° , T_2° , and p_2° . Uncertainties in the position of each curve corresponding to standard deviations of the associated measured values are indicated by crosswise marks. The position of the curves is of course only uncertain with respect to normal translation, not slope (to the first order).

GLL-638-1990

Fig. 4-6. Relationship between power and airflow rate implied by each of the measured variables $\dot{p}_0$, $\dot{p}_2$, $\dot{W}_{FM-1}$, and T_2° . Estimated standard deviations are indicated.

Intuition now suggests that the point representing the true values must lie somewhere in the middle of the region where the lines tend to cluster together.

The diagram is repeated in Fig. 4-7, now with a tentative best-value point identified by a dot. Its distance from the line belonging to the variable "x" has been shown as the segment ℓ_x , and the estimated uncertainty in the position of line "x" is indicated as $\pm \delta_x$. The criterion adopted for selection of the best-value point is that the quantity Δ be minimized, where

$$\Delta = \sum_i \left[\frac{\ell_{x_i}}{\delta_{x_i}} \right]^2. \quad (4.3)$$

An estimated standard deviation must be assigned to the position of the best-value point. Since this position has been determined by making use of all the available data, it presumably is more accurately known than would be the point most accurately determinable from any single set of three measurements, such as the points at any intersections of two lines. A simple and conservative way of assigning the final standard deviations has been chosen here, namely to make them equal to those associated with the best-determined line intersection; that is, the intersection of the two lines having the smallest individual standard deviations δ_x , δ_y . Figure 4-8 repeats the previous diagram again, with an ellipse of error drawn in over the intersection of lines p_0° and T_2° , (which bear the smallest values of δ).

Finally, it must be remembered that the position of most of the lines on the diagram is influenced by the original choice of a value of T_0° , and the standard deviation in T_0° must be combined with the resultant of the others. The dotted lines parallel to the other lines in Fig. 4-8 show the displacement of each line resulting from an increase in T_0° by the amount of its estimated standard deviation (the $\dot{w}_{FM-1}$ line is not affected). The corresponding displacement of the best-value point is shown by the small hollow circle at the end of an arrow. The horizontal and vertical components of the arrow represent standard deviation of the best-value point due to T_0° alone, and are to be combined with the (statistically independent) components of the previously derived standard deviation as the square root of the sums of squares, in the conventional way.

The final derived best values of P and $\dot{w}$, with standard deviation, are indicated near the corresponding axes in Fig. 4-8.

GLL-638-1991

Fig. 4-7. Identification of a trial point representing tentative best values of power and airflow rate, on the diagram of Fig. 4-6.

GLL-638-1992

Fig. 4-8. Determination of standard deviations for the derived best values of power and airflow rate.

This process, as outlined above, has been used to find most probable values of power and flow rate for each of the four elevated power reactor tests, as will be shown below.

It may be noticed that the measured value of T_0° was simply accepted as valid. In principle, the data reduction process should proceed one step further, to find the value of T_0° which would give the overall minimum value of Δ , including the square of the deviation of T_0° itself. This would add greatly to the labor of data reduction, and moreover, would not really add significantly to the validity of the result. In the first place, the estimated standard deviation of the measured T_0° is quite small, and would tend strongly to pull the final "best-point" toward the measured value anyway. Second, in only one of the runs, HP-2, would this extra step make any difference whatever, as a careful examination of the diagrams pertinent to each run will bear out. The relationship of the $\dot{w}_{\text{flowmeter}}$, p_2° , and T_2° lines is not affected by changing T_0° , so that the part of Δ due to $\dot{w}$, p_2° , and T_2° remains unchanged by varying T_0° , and of course the total value of Δ is minimized by making $\ell_{T_0^\circ} = 0$. If a p_0° line is present, as is the case only in HP-2, then a slight improvement could probably be made by choosing a different T_0° value. This has not been considered worth doing here.

Numerical Results. The selection of basic data to be used in determining power and airflow rate has been made by J. H. Moyer, as described in Appendix A. Table 4-2 presents the selected values.

Figures 4-9 through 4-12 show the diagrams used in these determinations as described in the preceding paragraphs. The spread in position of each line corresponding to the estimated standard deviation of the measurement to which it refers is indicated by a shaded band. One would expect, if the estimated standard deviations were valid, that something like half of the shaded bands in all of the diagrams would cover the best-value point. This is in fact found to occur in eight of the thirteen cases, which seems to indicate satisfactory consistency. If anything, the estimated standard deviations may have been slightly too large.

The explanation for the presence of the fourth line (p_0°) in only Fig. 4-11 lies in the fact that the inlet diffuser was choked only in run HP-2, and so only in that run was the measured value of p_0° useable with Eq. (4.2) to contribute to the determination of power and airflow rate.

GLL-638-1993

Fig 4-9. Determination of best-value power and airflow rate for the Intermediate Power Test.

Fig. 4-10. Determination of best-value power and airflow rate for High Power Test 1.

Fig. 4-11. Determination of best-value power and airflow rate for High Power Test 2.

GLL-638-1995

GLL-638-1996

Fig. 4-12. Determination of best-value power and airflow rate for High Power Test 3.

Table 4-2. Primary airflow parameters

Parameter	Value	Unit
Intermediate Power Test		
$\dot{w}_{FM2}$	117.3 ± 3.5	lb/sec
T_0°	393 ± 10	$^\circ F$
T_2°	1840 ± 34	$^\circ F$
p_1°	61	psia
p_2°	29.2 ± 1.7	psia
High Power Test 1		
$\dot{w}_{FM2}$	443.4 ± 5.8	lb/sec
T_0°	394 ± 15	$^\circ F$
T_2°	1558 ± 30	$^\circ F$
p_1°	206	psia
p_2°	119.8 ± 0.8	psia
High Power Test 2		
$\dot{w}_{FM1}$	661.4 ± 11.0	lb/sec
T_0°	925 ± 18	$^\circ F$
T_2°	1820 ± 27	$^\circ F$
p_1°	337	psia
p_2°	185.4 ± 1.5	psia
p_0°	391 ± 2	psia
High Power Test 3		
$\dot{w}_{FM1}$	416.2 ± 9.0	lb/sec
T_0°	402 ± 10	$^\circ F$
T_2°	1726 ± 32	$^\circ F$
p_1°	216	psia
p_2°	124.3 ± 1.6	psia
Fixed Throughout		
Nozzle discharge coefficient 0.988		

The derived best values of power P delivered to the air stream, and airflow rate $\dot{w}$ were summarized in Table 4-1, together with the assumed values of T_0° .

The set of values for each test run correspond to average values over a 5-second period chosen when the reactor was at or near to peak power for the run. The starting times for these intervals, on the base time scales shown in Figs. 4-1, 4-2, 4-3, and 4-5, are respectively, 1179 sec, 1271 sec, 1180 sec, and 1080 sec.

4.2.2 Axial Temperature Distribution

For the selected best values of power and flow rate corresponding to each test, axial temperature distributions through the reactor have been calculated for the fuel element material and the air stream, using the measured axial power distribution. Figures 4-13 through 4-16 show these results; thermocouple readings, taken at four axial stations, are also indicated. Both peak fuel element internal temperature T_i and fuel element wall temperature T_w are shown. The thermocouple junctions are buried at locations which ideally would be at some intermediate temperature between these. In indicating thermocouple readings on these figures, no temperature readings in fuel elements next to structural links were used. The readings of such thermocouples were lower, typically by 200°F or so, as may be seen in the detailed listings of Section 5.2.2. The agreement between measured and expected temperature values is very satisfactory.*

*This conclusion has been somewhat shaken by new heat transfer calculations, which strongly indicate that the unfueled dummy column of elements next to each thermocouple-bearing column should have depressed the thermocouple readings by perhaps 200°F below the temperature of unperturbed fuel elements at similar locations. This matter is still being analyzed at the time of writing.

Fig. 4-13. Axial temperature distribution of air and of fuel elements in the Intermediate Power Test.

GLL-638-1998

Fig. 4-14. Axial temperature distribution of air and of fuel elements in High Power Test 1.

Fig. 4-15. Axial temperature distribution of air and of fuel elements in High Power Test 2.

Fig. 4-16. Axial temperature distribution of air and of fuel elements in High Power Test 3.

4.2.3 Reactor Inlet Pressure

Measured reactor inlet plenum pressures p_1° may be compared with expected values as a test of the design calculations. The two static and three total pressure measurements which were made at this station during each test are listed in Appendix A, all converted to total pressure. The mean and standard deviation values of the measurements are tabulated below, together with the value for each test computed from power, airflow rate, and inlet air temperature. Agreement is excellent.

Reactor Inlet Total Pressure p_1°		
Test	Computed value (psia)	Measured value (psia)
IPT	60	60.7 ± 5.7
HP-1	207	205.9 ± 4.5
HP-2	326	337.0 ± 12.0
HP-3	209	216.2 ± 6.0

4.2.4 Control Vane Angle

The actual dependence of critical vane angle on core temperature may be compared with design calculations through records of the control vane angle during the tests. Since the reactor servo-control system functioned to keep the vernier rod close to its null position except for brief transient motions, the average angle of the ganged vanes provides a measure of the changing reactivity contributions from various causes during the run. The major effect is that of core temperature. Reactivity changes to bring about variations in power may be seen as relatively minor perturbations on the vane position history.

Figure 4-17 shows average control vane angle plotted against core temperature at the hottest axial station of the core, for each of the four tests. Recall that of these tests, three were run with inlet air temperature at about 400°F, and one, HP-2, with inlet air temperature of 925°F. The

Fig. 4-17. Average control vane angle versus core temperature (thermo-couple No. 156) during the four tests at elevated power.

core internal temperature distribution was somewhat different in the latter run from what it was in the other three, with a higher average core temperature corresponding to a given peak core temperature.

From Fig. 4-17, correcting for estimated perturbations due to kinetics, it is estimated that the steady-state critical vane angle at 60°F would have been about 69.5 degrees, and at 2250°F would have been 94.5 degrees, assuming 400°F inlet air temperature (in interpreting these numbers one must remember that the lower one refers to a uniform 60° temperature throughout the core, while the higher refers to a peak temperature of 2250°F, an average core temperature of about 1970°F, and a front support structure-neutronically important - temperature of 400°F to 500°F).

It is difficult to cite an objective pre-run prediction of high-temperature critical vane angle, because a variety of predictions were actually made, based in various degrees on experience gained in the preliminary reactor tests and on personal correction factors. However, a careful review of pre-run documents made at the time of this writing by E. Goldberg of "R" Division, and the author, leads to the conclusion that the proper predicted value of critical vane angle at 2250°F peak core temperature, corresponding to the other conditions outlined above, and depending solely on calculated results independent of other Tory II-A test data except for the cold critical vane angle, would have been 91.5 degrees, in good agreement with the observed 94.5 degrees.

By way of comparison, the most firmly stated pre-run prediction of this angle was 95 degrees, assuming two vernier rods half-inserted and a cold critical angle of 65.5 degrees. For one half-inserted rod and the measured cold angle of 69.5 degrees, this would become 97 degrees, also in good agreement with the actual results.

The reproducibility of vane angle from test to test, at a given core temperature (thus excluding HP-2), as shown here, was highly encouraging evidence during the period of test operations that losses of material from the core could not be taking place to any substantial degree. In the 80-90° region of vane angle, one degree of vane angle was worth about 0.001 in k_{eff} ,

thus about the same amount, say roughly one-tenth of one percent in BeO density change; or about seven times as much, or two-thirds of one percent in change of fuel content.

The apparent excellent agreement between the three tests with 400° inlet air temperature in going from 1000° to 2000° core temperature might be fortuitous to some extent, if any hidden causes of reactivity change took place. Note, for instance, that the core-reflector alignment may have been shifted slightly (by perhaps 1/8 inch) after the IPT, when the main air duct was replaced. Increased air density in the core would tend to increase the vane angle at high temperature for the HP runs slightly over that of the IPT, but could not account for as much as one degree. Most of the discrepancy at large angles and high temperature among the various runs is believed due to kinetics, the reactor being supercritical by substantial amounts at times.

On the whole, it is felt that these curves imply a loss of fuel amounting to less than one percent, over all runs.

4.3 Reactor Effluent

An analysis of the available data concerning fission product release from the reactor, together with a discussion of the expected release mechanism and quantity, is given in Appendix B. The results are summarized below.

Fission products generated during high power operation of the Tory II-A reactor can escape by erosion and corrosion of the core material, by diffusion out of the material, or by direct and immediate recoil from the fission events. The yield from the first two processes was expected to be negligible in comparison to the last, which according to theory should deliver about one-tenth of one percent of all generated fission products to the effluent air stream.

A variety of programs were undertaken to determine the actual amount of fission products released during each test. These were carried on by several separate agencies:

LRL itself operated an array of gamma dose rate detectors on an arc at about 1.3 miles from the reactor, with data telemetered to a central recorder. It also pumped a continuous-flow air sample from the reactor exhaust into a radiation detector system in the test bunker.

An Air Force Special Weapons Center-Convair team set up an extensive array of fallout trays, high volume air samplers, radiation rate detectors, and other detection devices over a large area generally downwind from the reactor.

Reynolds Electrical and Engineering Company, under contract to the AEC for Test Site radiological safety services, carried out a monitoring program involving both stationary detectors and teams of personnel with portable detectors roving over a large part of the Nevada Test Site. They have reported their detailed findings separately.⁵

The U. S. Public Health Service, with responsibility for determination of radiation hazards at off-site locations, collected a variety of data at points both within and outside of the Nevada Test Site boundary, as described in their separate report.⁶ One of their activities was effluent cloud tracking by portable radiation monitors on a small aircraft.

The data which is most directly applicable to the problem of estimating total fission product escape fraction is the gamma radiation rate recorded as a function of time at the various stations of the 1.3-mile LRL detector net. Comparisons of radiation levels at the two or more stations nearest to the point of transit of the effluent cloud, together with other data available on the height of the cloud, permits a rough calculation of the total number of radioactive fission products in the cloud, with the result accurate to within a factor of ten. This process was carried out for the Intermediate Power Test of May, 1961, and for the tests HP-1 and HP-2 of September 28 and October 5, 1961. Unfortunately, a breakdown of the detector system prevented useable data being obtained for the final test, HP-3.

Agreement between the expected and observed cloud activities was closer than the estimated uncertainty in the data for each of the three tests, and the average values over all three tests taken together agree to within about 30 percent. This degree of agreement is probably fortuitous, but little doubt can remain that the expected amount of fission product release was confirmed to better than an order of magnitude. The accepted value for the fraction of fission products escaping, then, remains 0.001.

Other data which were collected during and after the tests, while not useable for quantitative checks of fission product release, do confirm that release of radioactive material was slight. The Reynolds Electric survey teams observed a maximum gamma radiation intensity on the ground of 11 mr/hr during passage of any effluent cloud (at ten miles from the reactor following HP-3). Film badges with a 30-mr limit of sensitivity placed throughout the possible fallout area showed no detectable integrated dose. No alpha activity above background levels was detected, either on open fallout trays or in the high volume air sampler filters. Beta activity was observed throughout the general area of expected fallout, at levels ranging over several orders of magnitude. The greatest concentration was found on a fallout tray at 500 feet from the reactor, with an activity of $1\mu\text{Ci}/\text{m}^2$ two days after a test. Detailed results may be found in Ref. 5.

The Public Health Service survey, which covered the area generally lying between ten and one hundred miles distance from the reactor, found only very low levels of activity, indeed barely detectable outside the Nevada Test Site boundary. Detailed findings are available in Ref. 6.

High volume air samples collected by Reynolds Electric were analyzed for beryllium, but no detectable difference from normal background levels was found.

Radioactivity levels in the LRL test bunker continuous air sampler were too low to be distinguishable from background radiation scattering into the bunker from the reactor.

Results of the Air Force study are not yet available (time of writing 7-1-63).

5. REACTOR DISASSEMBLY AND COMPONENT INSPECTION

5. 1 Disassembly and General Examination

The reactor disassembly program was directed toward two principal ends: first, exploration for damage or dislocation in the core structural framework or its components; and second, determination of the condition of the fuel elements.

The following section will describe the process of breaking the reactor core down into its separate pieces, with cursory mention of special conditions observed, and will then review the condition of all of the most important components.

Succeeding sections will be devoted to a general discussion of fuel element condition and to presenting the results of special tests applied to the elements.

5.1.1 Disassembly Procedure

After reactor testing had been completed, with run HP-3, the reactor on its vehicle was moved to the disassembly building. Remotely operated tools were used to sever the wires, water hoses, and other connections between the core vessel and test vehicle. Clamps at each end of the core vessel were released, and this section, with the core inside it, was lifted away from the test vehicle and attached to a large tilting table (welding positioner) where it remained during all subsequent handling. Figure 5-1 shows the core package at this time, with a man working at the core inlet end. Occasional operations required direct personnel access, typically for times less than one minute or so. The man in the picture is working in a gamma field of about 8 r/hr.

Before any further work was done on the core, a set of photographs was taken looking into the exit face, with a brightly lit white wall providing a background. In each photograph, light from the wall shines through a group of fuel element flow passages except where they are blocked by components of the front support structure at the other end of the core, or by obstructions within the flow passages themselves. About half of the core flow passages are illuminated in these photographs, and in no case does any obstruction appear which does not correspond to known positions of dummy tubes carrying thermocouple leads, or of the front support structure. Figure 5-2 is a view of the upstream end of the core taken just prior to disassembly. The auxiliary front support structure (Sec. 2.5) is in the foreground, with the original front support structure visible behind it, and some of the module plates at the actual core surface behind that. Figure 5-3 is an example of a photograph showing light transmission through the core. Here, the illuminated tubes are mostly in modules K_a , L_a , and L_b (L_a is above the central module, K_a , and L_b is to the left of these two). The continuous bands of dark passages are recognizable as members of the support structures seen in Fig. 5-2. In addition, five single dark passages are visible, marking the positions of thermocouple leads in columns L_a -124, and K_a -60, -61, -109, and -110.

Next, the core vessel and other duct sections, and the front support structures, were removed from the core, so that it was encased only in its shroud. The exit end, terminated by the baseplates, was turned upward. Disassembly plans called next for removal of the tie rods and replacement with dummy rods adapted to the disassembly handling tools.

Fig. 5-1. The reactor core after final removal from the test vehicle.

GLE-638-5117

Fig. 5-2. Upstream end of the core just prior to disassembly, showing the auxiliary front support structure in the foreground.

GLB-638-5118

Fig. 5-3. View through the core just prior to disassembly. Some passages are obscured by the front support structures at the far end, or by thermocouple leads.

At this point, the first real difficulty of the disassembly process appeared, when it was found that the tie rods strongly resisted removal. Nearly all required the application of great force before coming out, and a few, resisting all efforts at pulling, were finally driven out of the way in the opposite direction. Baseplates were removed when sufficient surrounding tie rods had been taken out to free them. This process unfortunately involved placing large uneven loads on local regions of the core, both statically and dynamically, and is believed to have been the cause of breaking many ceramic structural links of the core.

The source of the difficulty lay in platinum shims at the baseplate-tie rod junctions, which had bonded to the aluminum oxide coat on both the baseplate and tie rod surfaces. Figure 5-4 illustrates the relationship of the tie rods and baseplates and shows some effects of the violent means required to move the tie rods. The baseplate has been removed from module N_a in the foreground, exposing exit reflector elements damaged by a tool used in prying off the baseplate. The tie rod to the left could not be pulled up, and so was finally pushed down, leaving torn fragments of its platinum shim behind. The next tie rod to the right has been partially pushed down, and again shows the damaged remains of a shim. The extreme right-hand tie rod has been pulled out to a considerable distance while being rotated, as the spiral score marks on it show.

At this time it was also discovered that some of the metal structural links, which replaced ceramic links at the exit end, had melted at the center of their crosspiece, depositing slag on the sides of the baseplates downstream. No other damage resulted from this melting.

Following removal of all baseplates, the extraction of fuel elements proceeded according to original plan: barbed drill rods were passed through many flow passages in a selected module until they hooked behind a previously-installed perforated plate at the other end of the module. The rods were then pulled upward, drawing the entire module with them into a can. This load of fuel elements was pushed out into a horizontal examination tray, permitting the separate fuel elements to be inspected layer by layer, and identified for storage and eventual detailed examination.

Module M_h was the first module extracted, and Fig. 5-5 shows the void left in the core by its removal. Close attention to the ceramic links forming the walls of this cavity will show that a number of them are cracked, as was

Fig. 5-4. Core exit end, showing effect of the forceful means required to remove baseplate N_a .

Fig 5-5. Void left in core by extraction of module M_h .

characteristic of the rest of the links in the core. As mentioned above, these may have been broken either during core handling prior to operation (least likely), during test operation, or during the efforts to remove tie rods and baseplates. The core, and the individual links, appear greatly foreshortened in Fig. 5-5, which was taken through a mirror and telescope.

Dismantling of the core continued module by module in this way, but with increasing difficulty as the core structure grew progressively looser. An increasing tendency appeared for the elements of a module or parts of the extraction tool to snag adjacent core parts during extraction, and even minor damage or dislocation suffered during the extraction of one module could make the later removal of its neighbors very hard to accomplish. Spills of fuel elements from a module being extracted or from an adjacent module became frequent. Finally, after 26 modules had been removed from the core (out of a total of 37), it was decided that no others offered a reasonable chance of successful extraction. Fuel element samples were desired from a variety of locations in the remaining modules, and these were extracted in single columns. The remaining fuel elements in the core were dumped out with no further attempt made to preserve identification.

5.1.2 Condition of Core Components

As discussed before, the general condition of the core was found to be excellent. No damage was serious enough to impair core performance in any way at the time testing was stopped, and no instance was found of conditions which appeared to be approaching the threshold of rapidly propagating failure. Some damage had commenced in various components, however, as will be described. Only those components will be discussed in which observable changes occurred.

Baseplates. Baseplate N_d was found to have a pit about 1/8-inch in diameter and of comparable depth on one edge of its upstream face, as shown in Fig. 5-6. Apparently, a flaw in the baseplate coating either existed prior to the test runs or was made during them, permitting oxygen to attack the underlying metal. No other coating or metal damage was found except for numerous chips known to have resulted from the disassembly process (see also Fig. 5-4).

Fig. 5-6. Baseplate N_d , showing small pit and accumulation of slag from melted structural link.

Some of the N-ring (core periphery) baseplates were coated with a heavy black slag on their side surfaces, as is the case with the side adjoining the pit shown in Figure 5-6; the surfaces so coated lay just downstream from metal structural links which had melted (see below). Chemical analysis showed that principal constituents of the slag are Ni, Cr, and Co, with Mo, Al, and Ti also present. These correspond to the composition of the melted structural links.

The pit in baseplate N_d was adjoined by a particularly heavy slag deposit. One may speculate as to whether this was involved in the cause of the pit.

Patterns of varying shades of gray and brown were visible on the baseplates before the test operations. These generally persisted through the tests, with some changes in color and intensity. A yellow coloring appeared on large areas of the baseplates after the tests, as well as a gray-to-black sooty coat which appeared principally on areas exposed directly to the air stream emerging from the core. Efforts to analyze the composition of regions colored in this way have not yet given conclusive results. The black-streaked area, it is felt, may have resulted from powdered oxide scale originating in the furnace tubes of the air supply. This material was known to be present in all test air passed through the heater in the bunker, and is also believed to be the source of deposits on the fuel elements (see below).

The platinum shims bonded to the baseplate and tie rod coatings have already been discussed in Section 5.1.1.

Metal Structural Links. All of the ceramic structural links between tie rods in the last layer at the downstream end of the core had been replaced (see Section 2.1) by parts made of a high-temperature metal alloy (Udimet-500) because of concern over the possibly serious effects of broken links at this position. Heat transfer and airflow analysis prior to operation indicated that the temperature of these parts would be about 2000°F at reactor design point operation, with no harmful effects expected since only negligible stress would be present.

During core disassembly it was found that some dozen of these metal links had melted through at the centers of their connecting webs. All of the melted links connected tie rods in the outermost ring with rods in the next ring inward. Subsequent laboratory furnace tests showed that the links would melt at a temperature of 2470°F.

The first and last of the reactor test runs are known to have taken the core to temperatures about 300°F higher than those corresponding to the design point, thus to about 2300°F predicted for the metal links on the basis of a flat radial power distribution. The upward hook in actual radial power distribution at the core edge appears to account for the melting in that region.

Fortunately, the supposed extra structural safety afforded by the metal links appears not to have been needed, since no other observable failure can be ascribed to their loss.

Ceramic Structural Links. A large number of ceramic structural links ("dogbones") were found to be broken after disassembly. While no careful attempt was made to count them, a very rough estimate of one-half of all parts seems reasonable for the fraction broken. The breaks were especially likely to take the form of a clean crack at or near the junction of web and loop.

Unfortunately, there is no way to be sure when this breakage took place. The links were known from laboratory experiments to be especially sensitive to axial loads applied at single tie rod stations, and these are just the sort of loads which were applied, often violently, during efforts to pull tie rods free of the core during disassembly.

The most that can be said with respect to possible link breakage during operation is that, if it did occur, it produced no traceable secondary effects. Link breakage was felt, before the test program, to be very likely to occur, but was also felt to be more acceptable than most other imaginable kinds of damage. These links were not load-bearing parts, but rather served as fuel element spacers and tie rod thermal insulators.

Fuel Elements. Figure 5-7 shows fuel element module M_h as it lay in the examining tray immediately following its extraction from the core. The white elements at the left-hand side make up the exit reflector, and individual fuel elements can be distinguished in the dark gray part of the bundle, with slight separations between their ends where they have spread apart during discharge onto the tray.

Several of the exit reflector elements are seen to be crushed and broken (from forces exerted during module extraction from the core). Not all of the fuel element gaps occur between the ends of elements: the two gaps

Fig. 5-7. Fuel element module M_h in examination tray following its extraction from the core.

nearest the reflector, as well as the large gap at right center can be identified as transverse breaks in fuel elements. The white spots on the sides of the elements are the residue of aluminum paint code marks which identified the fuel content of each piece. Steel rods have been threaded through the outermost fuel element columns to hold them in position.

Figure 5-8 shows module N_c after removal of several layers of fuel elements. The region seen is about one-quarter of the core length in from the front end, with air flow from left to right. The elements toward the right show a crazed appearance, which was the most noticeable change in the condition of the fuel elements as they were first examined.

Another immediately noticeable change in fuel element appearance was a dark surface glaze which appeared on some elements in the hottest part of the core.

Pairs of fuel elements here and there in the downstream region of the core were found cemented together, apparently by a dark substance which was visible between the elements, sometimes filling the triangular gaps where the corners of three elements joined. This material was left sticking in little ridges and peaks on the corners and faces of a number of elements.

Small pits were found in the corners of a few fuel elements; this is seen in element M_h -124D, Fig. 5-9.

In confirmation of earlier evidence, all fuel elements were found to be lying in their original position, without dislocation of parts such as to interrupt the air passages, except for seemingly obvious cases of disassembly damage.

To go beyond the superficial impression recorded above, it will be necessary to describe the results of individual tests performed on the fuel elements to determine their condition. The following sections will discuss various detailed findings.

5.2 Fission Product Inventory

A convenient indication of the relative power developed in the various individual fuel elements of the core is given by the gross radioactivity level of each element following the series of reactor operations. The residual activity level represents, of course, the sum of all the incremental contributions made throughout the reactor operating life, adjusted according to the varying decay time. Strictly, then, the fuel element activity is proportional

Fig. 5-8. Layer of fuel elements in module N_c , showing stress-corrosion crazing. The direction of air-flow is from left to right.

Fig. 5-9 Pit in corner of fuel element M_h -124D

to the time integral of reactor power, weighted in favor of the most recent test runs; there is no way of analyzing the measured activity to derive exact power profiles at any moment. However, nearly all (say, around 90%) of the time-integrated power was developed at core temperatures near the design point, and hence with a spatial power distribution characteristic of the design point. The local variations in spatial power distribution amounted to only 10% in going from ambient temperature to design point (compare Figs. 2-17 and 5-11) so the contribution of all of the sub-design temperature integrated power can have contributed no more than $10\% \times 10\%$, or 1%-sized variations to the total integrated power profiles. Furthermore, only the radial power profile is concerned, since the axial profile is essentially constant with temperature change.

In summary, then, the residual activity levels of the fuel elements at a given time may be taken as proportional (within about one percent) to the power developed in the respective elements with a core temperature in the range 2250-2550°F.

A great number of fuel elements were selected from the core for determination of gamma activity by means of a scintillation counter, biased to accept gamma energies above 0.36 MeV. The counting of these elements took place over a period of many months, but the effects of radioactive decay and of possible varying counter sensitivity were eliminated by reference to a set of five standard elements, always included in any set of elements to be counted.

All selected elements were counted for gross activity by being placed at a fixed distance (about one meter) from the counter. In addition, the lengthwise distribution of activity in many of the farthest-upstream fuel elements was determined by moving them across a 1/8-inch slit between two shielding blocks facing the counter, with 1/8-inch increments of motion.

In some fuel element columns, each element was counted, although in the majority of cases, only two elements from a given column were counted. Where possible, the fuel elements of thermocouple-junction bearing columns were counted, but many such columns could not be recovered after disassembly because of the difficulty of remote operations on the core.

5.2.1 Distribution of Gross Gamma Activity

Figure 5-10 shows axial profiles of fuel element activity in the core. The solid line is representative of the central region of the core, including all but the outermost (N) ring of modules, while the dotted line fits data from fuel elements in the last inch or so next to the core-side reflector interface. The two curves have been normalized to coincide at their peak values. Although the available data indicates that the transition between these two forms takes place almost entirely within a few inches of the core periphery, it is not possible at present to show in greater detail how it takes place.

The activity distribution of the farthest-upstream fuel elements, determined by counting through a slit, was combined with the gross relative fuel element activity data to produce the detailed shape of the dip in the curves just downstream from the inlet reflector, at $X/L \approx 0.15$.

The curves of Fig. 5-10 have been corrected to show the distribution of power density as if all elements contained identical fuel loading. In fact, very many of the farthest-upstream elements were loaded differently from the rest of the elements in their column (see Section 2.2.2, Core Length Change), so that few fuel element columns actually had a distribution of power density exactly as shown in Fig. 5-10. To obtain the power distribution for any actual column, one must make a stepwise break in the profile at the interface between the first two fuel elements (at $X/L = 0.181$), raising or lowering the profile upstream from that point by the same percentage amount that the first element differed in loading from the others in its column. This correction would have a very small effect (less than experimental error) on any of the data except for the correlation of observed with expected thermocouple readings described in Section 5.2.2. There, in the case of the thermocouple junctions embedded at $X/L = 0.27$, individual loading variations in the first element would clearly be important, and were used in constructing Table 5-1.

Note that the areas under the two curves are very nearly the same in spite of the difference in shape, so that relative peak values of power density in two or more fuel element columns may be taken also as representative of relative values of power integrated over their length.

Fig. 5-10. Axial profiles of fuel element radioactivity, corresponding to core power distribution.

Relative values of peak power density (and thus length-integrated power) of 163 individual fuel element columns are shown, distributed over core radius, in Fig. 5-11. Two smooth curves have been drawn in, with ordinates differing everywhere by 10%. The shape of these curves was chosen so that as many as possible of the data points would be included between them, and has no other basis.

In laying out the core fuel loading map, zones of nominally constant fuel loading were placed so that a spread of 5% in power density would be expected within a single zone, in consideration of the spatial distribution of neutron flux in the core. The tolerance band for actual fuel content in each fuel element loading class was also set at 5%; so that an overall spread of 10% in the power delivered to individual fuel element columns was expected, even without the effect of possible additional causes of minor local variation for which no specific provision was made, such as local concentrations of metal or unfueled material, flux depression due to control elements, and so forth.

Of the 163 data points shown, a total of 144, or slightly over 88%, fall within the area between the two lines, showing good confirmation of the expected results.

A curve midway between the two curves of the preceding figure, thus showing radial core power distribution with local variations smoothed out, is presented in Fig. 5-12. It was of course intended, in designing the reactor, that this curve should be flat; the sharp upswing at the outer edge is ascribed to the effect of replacing normal by heavy water in the reflector. The moderate nonuniformity of this power distribution must have been reflected in the broad radial temperature profile of the core, but the degree of departure from flatness is not great enough to be visible over the random variation in individual thermocouple data available (see Sec. 5.2.2 below). One might have expected that thermocouples in the last layer of fuel elements at the core periphery would show temperatures a few hundred degrees above the core average as a result of the 20% rise in power there, but in fact they do not, apparently due to their having transferred more heat than expected to the triangular air flow passages against the outer structural links.

Reference to Fig. 2-17 will show that a similar upswing in power density at the core periphery was present during low-temperature operation, although

GLL-638-2003

Fig. 5-11. Length-integrated core power versus radius. Individual flow passages are indicated, with lines drawn 10% apart so as to include most of the data points. Note that the vertical scale has a suppressed zero.

Fig. 5-12. Integrated power versus core radius, with local variations smoothed out.

the entire curve dips somewhat more strongly from the center going outward, due to the decreased importance of the reflector at low core temperatures.

The question may arise as to whether at least part of the gross fuel element gamma activity observed could be due to neutron activation products other than fission products. The following argument is offered to deny this: many fuel elements were counted at different times up to six months apart. No difference in the relative activities of elements from different parts of the core was observed with the passage of time, so they were all decaying at the same rate. If the activity were from a mixture of fission products and extraneous neutron-activated materials, the proportions of the mixture would vary from one part of the core to another. This would produce different decay rates at different locations in the core, in contradiction to the observed data.

One exception is possible, namely an activity produced by a substance present in the fuel elements in the same local concentration as the fuel itself, and having an energy-dependent neutron capture cross section very similar in form to the U^{235} fission cross section. Since such an activity would be everywhere proportional to the actual power, however, its presence would not affect validity of the data in any case. Furthermore, chemical analysis of the new fuel elements, as part of the acceptance testing procedure, virtually eliminated the possibility of such an extraneous material being present.

5.2.2 Correlation of Local Power with Thermocouple Readings

Examination of the actual core thermocouple readings obtained during the four reactor tests at elevated power shows that a considerable scatter existed among values obtained at a given core station. Part of this was due to inherent limitations of the thermocouples and their telemetering system, with an expected accuracy of about $\pm 30^\circ\text{F}$. In addition, real variations in temperature at the various thermocouple locations could be expected to result from local variations in neutron flux intensity, from variations in individual element fuel loading, from heat conduction into adjacent unfueled regions or from variations in airflow rate through individual passages.

There were two axial stations in the core at which substantial numbers of thermocouples could be compared: Readings are available from eight

couples installed in the upstream end of the third fuel element from the core inlet ($X/L = 0.27$); and from nineteen couples in the upstream end of the ninth element ($X/L = 0.68$), of which thirteen were in fuel element columns adjacent to structural links and the other six were separated from the links by at least one layer of fuel elements. These latter six couples, in "normal" unperturbed fuel elements, tended to run about 200°F hotter than the couples in elements next to structural links. All of the couples at $X/L = 0.27$ were in interior elements.

The readings from thermocouples within a group, for any run, show a considerable scatter. In order to eliminate as much as possible of the random component of this scatter, values of relative temperature representative of all four runs have been derived in this way:

From each temperature value, the reactor inlet air temperature has been subtracted, to give a value of temperature rise ΔT which ideally would be proportional to the local power divided by (approximately) the local air-flow rate. For each run, a standard value of $\overline{\Delta T}_{\text{abs}}$ was then obtained by averaging together the values of ΔT for that run, using only the couples which survived all four runs. Individual relative values of ΔT were then obtained, dividing the actual ΔT values by the $\overline{\Delta T}_{\text{abs}}$ for their run. These are listed for the three groups of couples in Tables 5-1, 5-2, 5-3, as "Relative ΔT ." Values of $\overline{\Delta T}_{\text{abs}}$ for each run are shown at the foot of each table.

For each couple, the values of this quantity for as many runs as the couple survived were averaged together, giving $\overline{\Delta T}_{\text{rel}}$, also listed in the tables, which indicates quantitatively by how much each couple read high or low with respect to others of its group.

Values of the relative power density in the fuel element columns bearing the couples were determined from the gamma activity data described in the preceding section. Where the actual gamma activity of a particular column of elements had not been determined, a value taken from other elements with the same fuel loading, at the same radius, was used. This latter process involved making corrections for the cases of fuel elements which belonged to loading classes different from the remainder of the column. The corrected relative local power values are listed in the tables as " P_{rel} ."

Table 5-1. Comparison of thermocouple readings with relative integrated power determined from activation measurements

Interior fuel element thermocouples, axial station X/L = 0.27								
Couple No.	Radius (in.)	Relative ΔT				$\overline{\Delta T}_{rel}$	P_{rel}	$\overline{\Delta T}_{rel}/P_{rel}$
		IPT	HP-1	HP-2	HP-3			
97	1.3	1.01	0.99	0.99	0.96	0.99	1.07	0.92
98	5.5	0.99	1.02	1.00	0.98	1.00	0.97	1.03
99	10.1	0.93	couple failed -----			0.93	0.98	0.95
100	13.8	1.04	1.01	1.02	1.03	1.02	0.99	1.03
132	14.6	0.99	1.01	1.00	1.00	1.00	0.98	1.02
133	4.9	0.95	1.00	0.98	1.01	0.98	1.03	0.95
134	10.2	0.88	0.94	0.96	----	0.93	0.98	0.95
136	13.8	1.03	0.97	1.00	1.01	1.00	0.97	1.03
$\overline{\Delta T}_{abs}$ (°F)		1087	1131	760	1304			
R. M. S. Deviation---						0.031		0.044

Table 5-2. Comparison of thermocouple readings with relative integrated power determined from activation measurements

Interior fuel element thermocouples, axial station X/L = 0.68								
Couple No.	Radius (in.)	Relative ΔT				$\overline{\Delta T}_{rel}$	P_{rel}	$\overline{\Delta T}_{rel}/P_{rel}$
		IPT	HP-1	HP-2	HP-3			
110	5.5	0.96	1.01	failed	----	0.99	0.95	1.04
115	10.1	0.95	1.02	0.98	----	0.98	1.02	0.96
119	14.0	0.97	0.99	0.98	1.00	0.99	0.98	1.01
140	8.9	0.99	----	----	----	0.99	0.99	1.00
156	14.8	1.03	1.01	1.02	1.00	1.02	1.02	1.00
159	10.4	0.98	----	----	----	0.98	0.98	1.00
$\overline{\Delta T}_{abs}$ (°F)		2202	1975	1418	2281			
R. M. S. Deviation -----						0.013		0.023

Table 5-3. Comparison of thermocouple readings with relative integrated power determined from activation measurements

Thermocouples in fuel elements adjacent to structural links, axial station $X/L = 0.68$								
Couple No.	Radius (in.)	Relative ΔT				$\overline{\Delta T}_{rel}$	P_{rel}	$\overline{\Delta T}_{rel}/P_{rel}$
		IPT	HP-1	HP-2	HP-3			
105	10.1	0.93	0.92	0.95	0.97	0.94	0.96	0.98
106	8.0	1.03	failed	-----		1.03	1.01	1.02
112	2.8	1.03	1.04	1.06	0.99	1.03	1.05	0.98
114	8.0	1.00	1.02	0.98	1.00	1.00	0.98	1.02
118	13.2	1.04	1.05	1.04	1.05	1.05	0.98	1.07
125	12.8	1.02	0.99	1.01	1.03	1.01	1.03	0.98
139	7.1	1.01	0.97	0.98	1.01	0.99	1.01	0.98
142	11.2	1.06	1.05	1.07	1.08	1.07	0.98	1.09
148	2.8	1.03	1.05	1.05	1.04	1.04	1.05	0.99
150	8.1	0.99	0.99	0.98	0.98	0.99	0.99	0.99
153	12.8	0.92	0.96	0.87	0.92	0.92	1.00	0.92
154	13.3	1.05	1.07	1.07	1.02	1.05	1.03	1.02
158	10.3	0.92	0.91	0.92	0.93	0.92	0.94	0.98
$\overline{\Delta T}_{abs}$		1965	1810	1350	2017			
R. M. S. Deviation ---						0.053		0.044

If all of the variations among thermocouple readings are due to variations in local power, and none to random error, heat loss to adjacent components, or variation in airflow through the flow passages, one would expect the values of $\overline{\Delta T}_{rel}$ and P_{rel} for each couple to correspond. The degree to which they fail to correspond, then, is a measure of the importance of these other effects. To put this on a quantitative basis, the ratio $\overline{\Delta T}_{rel}/P_{rel}$ has been computed for each couple and entered in the extreme right-hand column of each table. The root-mean-square deviation from the mean of each group of $\overline{\Delta T}_{rel}$ and $\overline{\Delta T}_{rel}/P_{rel}$ values is entered below the respective columns in the tables. This index of the degree of scatter shows no improvement after the correction for local power variations has been made.

If one looks closely at the columns of figures for the axial station $X/L = 0.68$, in Tables 5-2 and 5-3, where a given percent deviation in power has the advantage of producing a larger absolute temperature difference than is the case near the inlet end, a correlation between many of the temperature and power numbers does seem to appear, perhaps being obscured in the comparison of R. M. S. values by a few large deviations which do not correlate. The differences concerned here amount only to a few tens of degrees in temperature, however, and certainly do not warrant further analysis.

The values listed in the final columns of Tables 5-1, 5-2, and 5-3 should provide an indication of whether any large systematic variation in local air-flow rate exists across the face of the core; they are shown distributed according to radial position in Fig. 5-13 a, b, and c. It does not appear that any such variation exists in an amount of more than one percent or so.

5.2.3 Radiochemical Determination of Integrated Power

Determinations of the time-integrated reactor power have been made through radiochemical analysis of a number of separate fuel elements, all of which also were counted for gross gamma activity as described in Section 5.2.1. By making use of the gamma activity data, it is a straightforward matter to relate the total number of fission events in any single fuel element to the overall total for the core.

Before the radiochemical analysis was made, values of the time-integrated power for each of the reactor test runs were derived from calorimetric data. It will be of interest to compare the results here, and to make a final comment on the estimated values of power achieved during each run.

By making use of the power values deduced for a specified moment in each run in Section 4.2.1, together with traces of reactor external neutron flux as a function of time, a value of time-integrated power was derived for each of the four major runs, as shown in Table 5-4. To convert these values to total numbers of fission events, the energy delivered to the air stream per fission event must be known. This conversion factor is estimated as follows:

Fig. 5-13. Local fuel element temperature/power ratio: (a) Axial station $X/L = 0.27$; module-interior fuel elements. (b) Axial station $X/L = 0.68$; module-interior fuel elements. (c) Axial station $X/L = 0.68$; fuel elements adjacent to structural links.

Energy per fission:

Fission fragments	165 MeV
Prompt gammas	7
Neutron kinetic energy	5
Gammas from neutron capture in core	2
Fission fragment decay beta and gamma	<u>6</u>
	185 ± 4 MeV

Allowance is made for escape of about ten percent of all gamma energy from the core, as well as for similar loss of roughly twenty percent of the kinetic energy carried by neutrons. The energy released by fission products in equilibrium at a steady power level would be 13 MeV per fission event, but

Table 5-4. Integrated power and fission rate values* (from calorimetric power data and ion chamber curves)

Test date	Integrated power (MW sec)	Integrated fissions	Sub-total fissions
IPT 5/14/61	1.80×10^4	6.07×10^{20}	6.07×10^{20} IPT
HP-1 9/28/61	1.61×10^4	5.42×10^{20}	
HP-2 10/5/61	2.09×10^4	7.05×10^{20}	
HP-3 10/6/61	3.13×10^4	10.54×10^{20}	23.01×10^{20} All HP
TOTALS	8.63×10^4 MW sec or 1.00 MW day		29.08×10^{20} Fissions

* The experimental uncertainties which must be assigned to these values are rather high. Individual experimental errors in calorimetric power values range from 2.2 to 4.8 percent (see Figs. 4-9 through 4-12). The absolute error in integrating the power versus time traces is estimated at nearly 10 percent. Finally, an uncertainty of perhaps 2 percent is associated with the value of heat delivered per fission. The resultant estimated uncertainties for this table are: for ratios of values within the table, 5 percent; for absolute values, 10 percent.

in the four Tory II-A test runs, the power had been at or near a steady high level for times ranging from about 45 to 160 seconds, permitting only partial buildup of the fission product inventory. The corresponding rates of fission product energy contribution, derived from Ref. 7, range from 44 to 57 percent of the equilibrium value; 6 MeV was taken as nearly correct for all runs.

Values of the integrated number of fission events for each run are listed as the second column of Table 5-4, together with the total number of fission events for all of the last three runs taken together.

Unfortunately, no really direct experimental data on relative fission rate as a function of time was available. The traces of external neutron flux measured by various detectors do not correspond exactly in power level changes; and although every effort was made to choose the most reliable data in producing the values of Table 5-4, it is believed that substantial uncertainties remain due to this cause as well as others (see the summary at the bottom of the table). In addition, it must be pointed out that the values of "Integrated Power" in the table need qualification: they are based on the assumption of a constant ratio of heat release to fission rate, which is not strictly correct, owing to the changing fission product inventory. This qualification of course does not apply to the integrated fission values: the curves which were time-integrated represented, strictly, fission rate rather than power.

Table 5-5 lists, for a number of individual fuel elements, the number of fissions which occurred in each as determined from their relative gamma activity and the calorimetrically measured integrated total power. Next to these are presented the corresponding quantity determined by radiochemical analysis of the elements, using the fission products Cs^{137} and Ce^{144} . By treating the data as if all fissions produced during the HP tests had occurred at a single weighted average time (namely at noon on October 5, 1961), the known concentrations of these two isotopes are sufficient to determine the numbers of fission due to the IPT and the lumped HP runs separately.

Inventories of the fission products Zr^{95} and Y^{91} were also determined for a few of the fuel elements listed, and gave results in excellent agreement with the Cs-Ce results, both with respect to total numbers of fissions and to the ratio of Intermediate Power Test fissions to total fissions. The latter

value determined from Cs and Ce alone was 0.22, and from all isotopes about 0.24, with uncertainties of something greater than 10%. The value derived from the calorimetric data of Table 5-4 is 0.21 ± 0.01 .

Table 5-5. Comparison of calorimetrically and radiochemically determined fissions per fuel element

Element	Fissions		Rad Ch. /Calor.
	Calorimetric	Rad Chem	
K _a 19-H	3.70×10^{16}	4.10×10^{16}	1.11
M _h 62-G	3.35	3.81	1.14
M _h 124-G	3.50	3.84	1.10
M _h 124-L	2.44	2.74	1.12
M _c 80-E	2.95	3.25	1.10
M _c 80-G	3.34	3.60	1.08
M _l 107-G	3.47	3.80	1.10

The absolute numbers of fissions derived from calorimetric data have an estimated standard deviation of 10%, corresponding to that of the data presented in Table 5-4. Relative to one another, however, the values in the first column have a standard deviation of 1%, estimated from the reproducibility of gamma-counting data.

The standard deviation of the radiochemical data is about 2% with respect to absolute values, and somewhat less with respect to relative values.

The last column of Table 5-5 lists the ratios of radiochemically-determined power to the calorimetric values. These ratios agree quite well, and have an average value of 1.11, indicating that the actual integrated reactor power was higher than the value derived from the neutron detector curves. This degree of correspondence is entirely satisfactory, considering the estimated uncertainty of 10% in the derived value. The best value of time-integrated reactor power over all operation, then, is taken to be

$$\int P \, dt = 1.11 \text{ MW day.}$$

Since most of the uncertainty in the calorimetrically derived value of integrated power lies in the process of integration rather than in the instantaneous power values for the various runs, this correction cannot be applied to those power values. It does, however, tend to confirm that the power values were at least as high as supposed.

5.3 Fuel Element Cracks

Although the fuel elements in Tory II-A maintained their form and position throughout the test operations, they suffered cracking of various sorts, as was expected in view of cracking experienced in laboratory tests at similar temperature and stress levels. Many elements in the part of the core where the highest thermal stress existed did crack; however, many others which were expected to crack did not, as will be seen below. A limited region of the core also showed a special stress-corrosion type of cracking which was characteristic of the non-stabilized fuel element material used in this reactor. Section 5.3.1 will present a descriptive account of the fuel element condition, and a discussion of calculated thermal stress levels and expected cracking is in Section 5.3.2.

5.3.1 Description of Crack Pattern

Many features of the fuel element behavior with respect to cracking were evident at once when the core modules were laid out in an examination tray after extraction.

Figures 5-7 and 5-8 show examples of transverse breakage and crazing in two of the first modules removed.

Figures 5-14 through 5-19 show views along the entire length of one layer of elements in module N_C, one of the core edge modules. Reading from top to bottom, the identities of the columns of elements are as follows: at the top, an unidentified column, probably from the previous layer removed; next, 39-54-70-87-105-123-141-159-177-195-213-229-242-252-257-258-259. The last two columns belong to the next two underlying layers. Lying against the near edge of the tray are a number of unfueled solid BeO filler bars which occupied spaces at the outer edge of the module. Lettering on the back of the tray identifies the approximate position of the individual elements, starting with O and N, the 2-inch and 4-inch unfueled elements at the core inlet end, then running from M to B through the fuel elements proper, and ending with A, the unfueled 2-inch element at the core exit.

The fuel content of the elements shown ranges from about 7.5 percent at the rear down to 4.9 percent at the near side, with the exception of column 252, which is unfueled (originally to carry a thermocouple lead, which was not installed).

Figure 5-14 shows elements at the inlet end, which retain the same appearance as when new. The irregular stagger pattern of the element positions was caused by shifting during discharge into the examination tray. The greater part of the "M" fuel elements is shown.

Figure 5-15 shows part of the L's at the left, all of the K's, and part of the J's at the right. The K elements start to show a strongly crazed appearance, which differs in nature from one tube to another. In general the amount of crazing increases with increasing fuel loading, although individual variations among elements are very great. Not much crazing appears in the three columns at the bottom of the photograph, which were at the core edge and more strongly cooled than the others.

Figure 5-16, showing J, I, and H elements, illustrates how the crazing disappears again in fuel elements farther downstream, persisting the farthest in the elements of higher fuel loading and in elements next to over-cooled non-fueled components, at the near edge.

Figure 5-17 includes elements H, G, and F. Little crazing remains except in the relatively cool elements at the core edge, where it is seen in Figs. 5-18 and 5-19 as far downstream as element C. Several pieces of exit reflector tubes, broken in disassembly, are seen lying on the fuel elements in Fig. 5-19.

Figure 5-20 shows F and G fuel elements of module M_h , with a variety of crazing patterns. The surfaces exposed are those which faced structural links at the module periphery. The elements at the upper left corner are in columns 26 and 14, with 5.4 weight percent fuel loading. The next two columns below them, 15 and 16, have 5.2% fuel loading. The next is column 7, with 4.9% loading, followed by columns 1 and 2 at the bottom, both with 4.7% loading. Records show that all elements of each common fuel loading shown here were members of common batches received from the factory. Each batch appearing in the figure shows its own strongly characteristic appearance except for the two elements of column 7. The elements at the bottom are not really crazed, but rather show a few simple thermal stress cracks.

Examination of many layers of fuel elements exposed in the examination tray, as well as with cracks revealed by a fluorescent-penetrant treatment, shows that the types of crack fall into three general groups:

GLB-638-5122

Fig. 5-14. Fuel elements in module N_c . White elements at the left belong to the (unfueled) inlet reflector. White spots are aluminum paint used to identify fuel loading classes.

GLB-638-5123

Fig. 5-15. Fuel elements near upstream end of core, in module N_C. The elements to the right show stress-corrosion crazing characteristic of the unstabilized fuel element material used in Tory II-A.

Fig. 5-16. Fuel elements slightly upstream from core center, in module N_c .

GLB-638-5125

Fig. 5-17. Fuel elements in module N_C , about midway along the core length.

Fig. 5-18. Fuel elements in module N_c , downstream from core center.

Fig 5-19 Fuel elements at exit end of core, in module N_C . The loose pieces of white beryllia are from exit reflector elements broken in disassembly.

Fig. 5-20 F and G fuel elements in module M_h , showing several different typical crack patterns.

1) Many fuel elements which were located in the region of highest power density in the core show cracks which are all oriented parallel to the direction of air flow. These mostly do not run the full length of an element, although some do. This type of crack is believed to represent simply the effect of thermal stress levels in excess of breaking stress. The amount of this cracking varies with fuel loading, being at a minimum in elements with the lowest concentration of fuel, in the central module (K_a). Figures 5-21, 5-22, and 5-23 show fluorescent-penetrant treated fuel elements making up the entire column 120 in module L_d (fuel loading 4.0%). Recall that element M is at the inlet end of the core, L next downstream, and so forth to B at the exit end. The cracks here exemplify the type under discussion.

2) A second type of cracking, identified as a stress-corrosion effect, is found in elements which spent a substantial amount of time at high stress levels while at a temperature in the range 1800-2200°F. This is the crazing effect which was shown in a well-defined region of the core in Figs. 5-14 through 5-19. It results from high stress, high oxygen pressure, and temperatures in the cited range, all of which must be present together. This effect has been studied in the laboratory, being produced under well-controlled conditions with very similar appearance to that of the reactor-exposed fuel elements.⁸ It is specifically a property of the fuel element material used in the II-A reactor; the material developed for the Tory II-C reactor is not subject to it.

Figures 5-24, 5-25, and 5-26 are photographs, again using fluorescent-penetrant crack detection, of the fuel elements making up column 71 of module L_e (fuel loading 4.3%). Some crazing exists, but of a very mild degree. A moderate degree of crazing is shown in Figs. 5-27, 5-28, and 5-29, in the elements of column 58, module M_m (loading 6.2%); and severe crazing in column 108, module N_c (loading 7.5%), in Figs. 5-30 through 5-33. The exit reflector element A, and the inlet reflector elements N and O are included in the latter set. Notice in Fig. 5-33 that element B and the downstream part of C, adjacent to it, show an indication of general surface porosity. A similar appearance was found on other exit-end elements of high fuel loading.

Figures 5-34 and 5-35 are photographs of elements from module M_h , column 125 (fuel loading 5.4%), which were cut into halves, lengthwise, and

Fig. 5-21 Fluorescent-penetrant treated fuel elements from column L_D -120, showing "pure" thermal stress cracks

Fig. 5-22. Fluorescent-penetrant treated fuel elements from column L_D -120, showing "pure" thermal stress cracks.

Fig. 5-23. Fluorescent-penetrant treated fuel elements from column LD-120, showing absence of cracking in the hot, downstream region of the core.

Fig. 5-24. Fluorescent-penetrant treated fuel elements from column L_e-71, showing mild crazing.

Fig. 5-25. Fluorescent-penetrant treated fuel elements from column L_e -71, showing mild crazing.

Fig 5-26. Fluorescent-penetrant treated fuel elements from column L_e -71, showing absence of cracking in hot downstream region of core

Fig. 5-27. Fluorescent-penetrant treated fuel elements from column M_m -58, showing crazing of intermediate severity.

Fig. 5-28. Fluorescent-penetrant treated fuel elements from column M_m -58, showing crazing of intermediate severity

Fig. 5-29. Fluorescent-penetrant treated fuel elements from column M_m -58, showing absence of cracking in downstream part of core even for near-peak fuel loading (6.2% by weight).

Fig. 5-30. Fluorescent-penetrant treated fuel and reflector elements from column N_c-108, at inlet end of core.

Fig. 5-31. Fluorescent-penetrant treated fuel elements from column N_C-108, showing severe crazing.

Fig 5-32. Fluorescent-penetrant treated fuel elements from column N_C-108, showing severe stress-corrosion crazing, grading into simple stress cracks farther downstream.

Fig. 5-33. Fluorescent-penetrant treated exit end fuel elements, and exit reflector element, from column N_C-108. Notice the surface porosity, increasing toward the exit end, indicated by retention of large amounts of the penetrant material. These elements have the highest fuel loading (7.5 weight %) used in the core.

Fig. 5-34. Fluorescent-penetrant treated fuel elements H and I from column M_h -125. The elements have been cut lengthwise into halves, with the outer surfaces shown here.

Fig. 5-35. Fluorescent-penetrant treated fuel elements H and I from column M_h-125. The elements have been cut lengthwise into halves, with the cut surfaces and flow passage surfaces shown here.

treated with fluorescent-penetrant to display cracks. The inner and outer surfaces of each piece are shown, so that the manner in which the cracks go through the element walls is revealed.

3) Fuel elements next to unfueled parts suffered generally severe crazing and transverse cracking over a large part of the core length. Depression of the temperature of such elements, together with presence of a transverse temperature gradient, seems to account for these effects; the crazing is believed to represent the same process discussed in the preceding paragraphs. The series of photographs in Figs. 5-14 through 5-19 showed this phenomenon next to a column of dummy elements and next to the outer structural links of module N_c .

The absence of cracks in large regions of the core where calculated (elastic) thermal stress exceeded the measured breaking stress of the fuel elements is a particularly interesting result. Cracks and crazing were distributed in the following way in elements which did not adjoin unfueled structural parts:

Elements at the hot, exit end of the core were in good condition, with B, C, and D showing cracks in only a few scattered instances. The M elements at the inlet end were completely free of damage. In the lowest fuel loading classes, elements E, F, and L were also free of cracks or crazing; G generally showed one or two simple cracks; and H, I, J, and half of K showed mild crazing. In the elements of highest fuel loading, the crazing became severe in H, I, J, and K, and extended into elements G and the downstream end of L with less severity; while E and F showed relatively simple cracking. Elements in the intermediate fuel loading classes ranged between these two extremes, with severity of damage generally increasing with fuel loading.

An atlas of fuel element photographs is in preparation, to illustrate these findings, and will appear as a UCRL report.⁹

5.3.2 Expected Fuel Element Stress Distribution

The crazing which appeared throughout the central part of the core length corresponded to stress-corrosion behavior found in laboratory tests of individual fuel elements. The intensity and pattern of crazing is evidently

strongly correlated with fuel loading concentration as well as with other unknown factors in the fuel element makeup; photographs in the preceding section illustrate the wide variety of crazing appearance.

Thermal stress cracking in other regions of the core, however, continues to be of great interest. Elastic-theory analysis of the stress distribution to be expected in Tory reactor-type fuel elements, as well as laboratory experiments on breakage induced by electric heating of BeO specimens and an extrapolation of the results to predictions of crack distribution in the II-A reactor have been reported elsewhere.^{10, 11, 12} The results will be presented here without their theoretical or experimental background, which can be found in the references.

The theoretical thermal stress in a fuel element is a function of the power delivered to air flowing through the element, of the size and shape of the element, and of the material elastic modulus E , the coefficient of thermal expansion α , the thermal conductivity k , and Poisson's ratio ν . In a long hollow circular cylinder (a good approximation for the Tory II-A elements), the peak tensile stress σ can be written as

$$\sigma = \frac{E\alpha Q b^2 f(\Gamma)}{k(1-\nu)}$$

where Q is the constant volumetric power density generated throughout the element, b is the outer radius of the cylinder, and Γ the ratio of coolant flow area to total cross-sectional area. This can be rewritten

$$\frac{\sigma k}{E\alpha} (1 - \nu) = Q b^2 f(\Gamma)$$

where the entire left-hand term is a convenient thermal stress parameter, independent of material properties. Determined uniquely by the power density and the fuel element geometry, it is a direct measure of the effectiveness of the element as a source of power to the coolant air stream.

If the thermal stress parameter $\frac{\sigma k}{E\alpha} (1 - \nu)$ is determined from local power density at each point along the length of a reactor fuel element column, and then plotted against the local temperature at corresponding points, a curve results, with each point along its length corresponding to an axial station in the reactor. The three solid curves in Fig. 5-36 were generated in this way,

GLL-638-2006

Fig. 5-36. Thermal stress parameter $\sigma^k/E\alpha$ versus fuel element temperature for runs HP-1, HP-2, and HP-3. The general region of stress-corrosion crazing is indicated, as well as the boundaries between regions in which elastic theory predictions called for no cracks, for occasional cracks, or for all elements to be cracked.

and represent the three reactor tests HP-1, HP-2, and HP-3. Points corresponding to the centers of fuel elements are identified by letter. The quantity $\frac{\sigma_k}{E_a}$ is plotted, with the constant factor $(1 - \nu)$ omitted.

The lower dashed line represents the upper boundary of a region in which no cracking was expected, and the upper line a boundary above which all the tubes were expected to crack.* These were taken from Ref. 11. Now as pointed out in the previous section, cracking actually did not occur farther downstream than the inlet end of element G, except for elements in the highest fuel loading classes. In the low and medium loading classes – say up to at least 5% fuel – elements B, C, D, E, F, and most of G were free of cracks although most of them extended far into the temperature-stress region where elastic theory predicted that they would break. The fact that they did not is ascribed to the effect of creep relief of thermal stresses.

The two wiggly lines at the top of the figure roughly define the temperature region within which stress-corrosion crazing took place (in the elements of lower fuel loading).

Note again that these results apply only to the unstabilized Tory II-A fuel element material. The stabilized material developed for the Tory II-C fuel elements is not subject to stress-corrosion crazing.

5.4 Fuel Element Breaking Load Tests

A number of fuel elements, in positions selected at random from designated regions of the core, were subjected to breaking loads in a three-point beam bending apparatus. The regions chosen were the seven core modules nearest the center, where fuel loading ranged from 3.5 to 4.3 weight percent. The load was applied at the rate of ten pounds per minute, with two supports three inches apart opposed by a counteracting force midway between them. A total of 68 columns of 12 fuel elements each were tested, minus a few elements lost in disassembly.

Figure 5-37 shows the data obtained in these tests. Individual element data from four axial stations – B, E, J, and M – is displayed in histogram form. Each small solid square indicates the breaking load for an element of 3.5% fuel loading, and each open square an element in the fuel loading range

*These limits of 0% and 100% cracking assume that all tubes have identical breaking strength, which is far from true. See Section 5.4.

GLL-638-2007

Fig. 5-37. Three-point breaking loads measured on fuel elements removed from the reactor. Average values are given for each axial fuel element station, as well as the distribution of single measurements at four representative stations. The distribution of breaking loads for unused elements is included for comparison.

3.7 to 4.3%. For comparison, a histogram at the side shows the distribution in breaking load of new fuel elements. The latter had a higher fuel content, ~7%, but little new-element data is available in the lower fuel ranges for production elements. That which does exist indicates no difference in average strength or distribution.

The average breaking load for each axial station is indicated by a circle, and these are connected by straight line segments to show a quasi-continuous curve of breaking load versus axial position. The standard deviation of the average values is estimated as about one pound.

The effect of crazing is quite evident in the dip in the curve around elements I and J, where this behavior was most severe. However, the gradual decline in strength from the exit end (B) toward the middle of the core indicates that a loss of strength took place due to other effects than visible crazing. Figures 5-24, 5-25, and 5-26 show the representative appearance of a column of fuel elements, L_e-71, in the region from which the samples were chosen for breaking. As shown there, no cracking is visible in elements B through G, nor in L and M. Notice in Fig. 5-37 that element H is not much more depressed in strength than G, although H typically shows much more visible damage in this fuel loading range.

It is interesting to compare the appearance of the fuel elements of column L_e-71 with their actual breaking load values, measured after the photographs were taken:

B 83.8 lb	F 40.3	J 64.2
C 87.9	G 66.6	K 80.9
D 71.6	H 55.9	L 87.2
E 76.0	I 58.6	M 87.2

A significant point is now apparent, that even such a network of cracks as that in element L_e-71J may not greatly reduce the strength of the part.

The applied load, with the tube geometry, may be used to compute modulus-of-rupture values from this data. Such values have not been presented because the concept of a material modulus is not believed to be appropriate to the effect studied here. However, by way of orientation, the average undamaged breaking load of 80 pounds corresponds to a modulus of rupture of 23,000 psi.

5.5 Fuel Element Thermal Diffusivity

Another indication of changes undergone by the fuel elements during reactor operation is provided by measurements of their thermal diffusivity. A source of heat at about 300°F temperature is applied to one end of an element, at ambient temperature, and the time required for the temperature to rise at the other end (over an interval of a given percentage of total temperature rise) is measured. This time interval is inversely proportional to the effective thermal diffusivity of the fuel element material.

Elements containing various fuel loadings have been tested. The data, which has been averaged in four different loading ranges, is presented in Fig. 5-38 as relative thermal diffusivity. The points for the range of lowest loading represent five whole columns of fuel elements, the two middle ranges two columns each plus a few extra, and the highest range only one column. The latter was included principally to show the effect of crazing.

The range of values, on the same scale, characteristic of non-reactor-exposed fuel elements is indicated in the figure. Whether or not thermal diffusivity depends in a regular way on fuel concentration in the non-exposed fuel elements cannot be determined from the small amount of data presently available, but preliminary indications are that it does not. The regular arrangement of data points in order of fuel concentration in Fig. 5-38 is believed to be meaningful, however.

The effect of stress-corrosion crazing is clearly visible in the large dips at elements I-J-K. Crazing was quite mild in the 3.5-4.5% fuel range, being at worst like that of Figs. 5-24 and 5-25. In the 4.5-4.7% loaded tubes it was less mild, comparing to that shown in Figs. 5-27 and 5-28; and in the 6.1-6.4% tubes it was quite severe, similar to that illustrated in Figs. 5-31 and 5-32. Damage in the highest loading class was as severe as any in the core; the J element shown as the lowest point on the graph fell apart in handling. Clearly the data points in this region have no meaning in terms of thermal diffusivity as a material property; heat transmission through a network of gross cracks is measured.

Special crazing effects aside, it is seen that the thermal diffusivity of these fuel elements remains almost unchanged at stations C, D, E, and F, and then is progressively more depressed going toward the upstream direction. Element B is slightly depressed below the unirradiated value.

GLL-638-2008

Fig. 5-38. Fuel element relative thermal diffusivity following reactor operation.

The depression of thermal diffusivity shown here is interpreted as a radiation damage effect, whose severity is controlled not only by the integrated radiation exposure but also by the temperature of the fuel element at the time of exposure and afterward. Temperatures at the peak values attained during the Tory II-A tests were high enough to anneal out virtually all damage, as in elements C through F.

General consistency is shown here with results obtained using Tory II-C fuel elements, reported in Ref. 13; although the integrated radiation doses given to the two types of element still vary too widely to permit a really quantitative comparison.

It must be emphasized that the measurements reported here were made at or near room temperature. Theoretical considerations indicate, and preliminary experimental data from this laboratory confirm, that the effect of radiation damage on thermal conductivity measured at elevated temperature is relatively slight compared with the effect seen at room temperature. Thus any increase in thermal stress caused by radiation damage in a reactor core operating at high temperature is expected to be of little importance. This apparent masking of thermal diffusivity change while at high temperature is independent of the permanent annealing effect of high temperature which is mentioned above.

5.6 Fuel Element Adhesion

As mentioned in Section 5.1.2, pairs of fuel elements here and there throughout the core were found to be stuck together. In some cases they could easily be separated, but in others, a piece of one element would break off and remain attached to its neighbor when the two were pulled apart. A deposit of foreign material seemed to be associated with this sticking, sometimes almost filling the small space where the rounded-off corners of three elements came together.

Fuel elements pressed together in a laboratory furnace and subjected to reactor-like temperatures proved unwilling to stick together. However, it is known that finely powdered furnace-tube scale was discharged with air from the test facility, and when some of this material was sprinkled between fuel elements in the laboratory furnace, they did stick together. Chemical analysis of the extraneous material found on adhered tubes from the reactor

showed large amounts of iron, chromium, and nickel, supporting the conclusion that the sticking agent was oxidized furnace tube material which caught in the interstices of the reactor core.

A black surface glaze observed on some fuel elements may also have been associated with this metal oxide deposition, although conclusive chemical analysis has not been available.

6. TEST PROCEDURE

An account will be given here of how an actual reactor test run was carried out. The example chosen is the Intermediate Power Test of May 14, 1961, which was fairly typical of all of the four principal tests with respect to preparation and execution. It has special interest also in being the first high-level reactor test operation of the program.

A short description of the test air supply is given in Section 6.1 below to provide a background for the discussion of the reactor run. Appendix C, on the data system, can be consulted for an account of the information available to the operators.

6.1 Air Supply

The air supply, shown schematically in Fig. 6-1, functioned to provide air to the reactor core at flow rates up to about 1000 pounds per second over a period of 100 seconds, or at flow rates up to 25 pounds per second for an indefinite period. The temperature of this air could be regulated between ambient and somewhat above 1000°F.

The two blowers labeled 4 and 5 in the figure provided continuous low-flow-rate air, regulated by valves AV-7 and AV-8, which admitted air to the reactor or dumped it into the atmosphere. Air for high flow rates was stored in the tank farm, which could be closed off at AV-1 and AV-2. The flow rate was controlled by AV-3, and measured at the venturi flowmeter FM-1; or alternately, for low flow rates, was controlled by the smaller valve AV-4 and measured at the smaller meter FM-2. Hot air from the furnace was routed, before a test operation, through the stored energy heater by means of valves AV-6A and AV-6B, raising the temperature of a bank of steel rods in the heater. During a test, air for the reactor was routed through or around the heater by means of valves AV-5A and AV-5B, affording regulation of air

Fig. 6-1. Schematic diagram of test air supply system.

temperature at the reactor inlet. AB -3A and AB -3B are pressure relief valves which would open automatically to protect the air system from accidental overpressure. Fig. 6-2 shows the two flow-meters FM-1 and FM-2.

The air control valves were located in a concrete bunker, at one wall of which the reactor was connected for testing. The control console was located in a building about two miles distant from the test bunker. Figure 6-3 shows the general layout of the test facility.

6.2 Operation Plan

A written plan was provided for each reactor test, listing objectives, special conditions, and so forth. Excerpts from the Stage 4 Operation Plan, covering the Intermediate Power Test, are given below:

The Intermediate Power Test is the last major event which occurs before the Full Power Test. The overall objective is to demonstrate that all reactor and facility components perform satisfactorily at a significant but somewhat less than full power level. Prior to this test no nuclear heat of consequence will have been generated.

Some specific objectives are:

1. Attain fuel element design wall temperature.
2. Attain design exit air mixed-mean total temperature.
3. Attain a nuclear power to 40 MW.
4. Observe the behavior of core thermocouples when subjected to high internal heat generation rate.
5. Gain operator experience under conditions which very nearly approach those in a full power run.

General Circumstances

1. For this test the bunker air piping will have two temporary mild steel sections installed as replacements for the gimbal section and AV-6C spool. These temporary sections, which will limit the inlet air temperature to values below 600°F, will have been tested in three blowdowns at 400°F.
2. Because of the low inlet air temperature, maximum flow rate will be 113 pounds per sec (pps) instead of 200 pps which corresponds to the design inlet temperature of 1060°F.

Fig. 6-2. Airflow meters FM-1 (below) and FM-2 (above) in the test bunker.

MUL-8140

Fig. 6-3. Contour map showing layout of test facility.

3. Total integrated flow for this run is considerably less than Tank Farm capacity, therefore maximum Tank Farm pressure will be 3000 psig instead of the design value of 3600 psig. This will allow somewhat smoother control since the lower pressure means a lower air flow control valve gain.
4. The small air flow control valve (AV-4) will be used. This valve is quite adequate to provide the maximum flow required and allows smoother control of air flow than does the large valve (AV-3).
5. Recent blowdown experience both with and without the core installed in the ducting has shown a "horning" phenomenon which occurs over a band of nozzle pressure ratios. This is manifested as a loud low frequency noise superimposed on the usual white noise. To minimize horning during the run the flow and reactor power will be programmed in a special way, as follows:

Air mass flow rate will be increased to 30 pps and maintained. Then the reactor power will be increased until exit gas temperature 200° to 300° below design point value is obtained. After this temperature is stabilized, both air flow rate and nuclear power will be increased together by separate operators, in such a way as to maintain constant core temperature, until design point flow rate is reached. Power will then be increased until design point temperature is reached. The same steps will then be followed in reverse order to terminate the test run.

6. Peak fuel element wall temperature will be brought up to the design point value, 2250°F.

Execute Pre-Run Checklists:

Forty-eight hours prior to the scheduled runtime the test area will be put under a limited access condition. Access will be limited to those persons who are to participate in the execution of pre-run checklists. These checklists are detailed procedures for checking out all the systems which will be used in the run.

Simultaneous with establishment of the limited access condition all electronic rack doors will be closed and locked at both the Control Point and Test Bunker. It is probable that during the checkout period a condition will arise which will require opening of an electronic rack. The work to be done will require written approval and record as specified in Section ____.

As the pre-run checklists are completed they will be signed off by the engineer in charge of the given system and presented to the Pre-Operation Manager for inspection. When the lists are completed to his satisfaction, he will deliver them to the Test Group Director with notification of readiness. They will be inspected, together with all change records, by the Test Group Director and Test Conductor before test operation proceeds.

Final Inspection of the Forward Area

After completion of the pre-run checklists the TGD, Test Conductor, and others designated by the TGD will make a final inspection of the forward area. The purpose of this inspection is to look for such things as:

- a) crucial transducers left disconnected after calibration;
- b) interference problems between pieces of hardware, especially on the Test Vehicle;
- c) loose items improperly left on Test Vehicle or Test Pad.

This inspection group will include members of the final entry party whose job is to remove the boral blanket.*

Removal of the Boral Blanket and Evacuation of Final Entry Party

At completion of the final inspection all members of the inspection team except the final entry party will evacuate the test area. The final entry party, consisting of two men, will remove the boral blanket, remove the local neutron counter from the Test Vehicle, pull out the control vane locking pins, open the Test Pad building doors, and remove the nozzle cover.

*A boron-loaded jacket placed over the core for criticality safety during periods of shutdown.

The conditions which will exist when the boral blanket is removed are:

- a) Control system hydraulic power supply "off"
- b) reflectors at maximum separation.

Blowdown Through the Core

After evacuation of the final entry party, the main phase of the test begins. An outline of the steps follows:

- a) Obtain confirmation from the External Operation Control Officer that all personnel are clear of the forward area.
- b) Bring reactor critical, increase power until compensated ion chamber current is high enough to allow transfer to log power automatic control.
- c) Start blowers, but keep blower valve AV-7 closed.
- d) Open AV-4 to give approximately 30 pps.
- e) Increase inlet air temperature to 400°F.
- f) Increase nuclear power to obtain core temperature of 1800-1900°F.
- g) Increase flow rate to 113 pps, with proportional increase of power.
- h) Increase power to raise core temperature to design point, hold for at most 60 seconds, then reverse sequence through this and two previous steps.
- i) Decrease reactor power until compensated ion chamber current is 10^{-8} amp, switch to manual control, run all vanes to 0°.
- j) Decrease inlet temperature to minimum value.
- k) Close AV-4 and open blower valve AV-7 to allow blower air to pass through the core for aftercooling. Maintain test condition until aftercooling requirement for air discharge is finished.

Radiation Survey

Before reopening access to test area, Test Group Director will arrange dispatch of Health and Safety survey teams to determine radiation levels and set up necessary exclusion areas.

Transfer of Test Vehicle to Disassembly Building

At some time within a few days after test completion, the Test Vehicle will be moved to the Disassembly Building. If

radiation levels allow, this operation will be done manually. If not, the transfer will be made under remote control with observers stationed at various points along the track. Locomotive and Test Vehicle will be cabled together if access is possible within allowable radiation dose limits. The NTS 401 Standard Operation Procedure for Railroad Operations gives specific procedures which will be followed:

6.3 Test Conductor's Checklist

Progress of the test run was closely guided by a checklist which the Test Conductor followed: The final checklist, as used, is reproduced below.

1. Test Personnel:

Test Conductor -	C. Barnett
Internal Operation Control Officer -	E. Laine
External Operation Control Officer -	J. McShea
Nuclear Operator -	R. Finke
Nuclear Engineer -	R. Staker
Air Flow Operator -	K. Shyne
Air Temperature Operator -	C. Dykes
Coolant Engineer -	C. Meier
Data Coordinator -	T. Varljen
Offner Data Scanner -	R. McConnell
Console Data Scanner -	J. Moyer
Console Data Scanner -	J. Heieck
Core Data Observers -	P. Uthe, M. Mintz
ASCOP Data Scanners -	L. Newton, J. Penuelas
Instrumentation Engineer -	J. Piccolo
Communications -	R. Holloway

2. Communication Requirements:

- 2.1 Simplex Radio - External Operation Control, Security.
- 2.2 Channel 11 Headset - Control Room Operators.
- 2.3 Channel 3 Headset - Internal Operation Control, External Operation Control.
- 2.4 Intercom - External Operation Control, Control Room.

3. Initial Conditions:

- 3.1 Tank Farm Pressure - 3600 psig.
- 3.2 Heat Mass Temperature - 710°F.
- 3.3 Test Vehicle No. 2 plugged into Test Bunker.
- 3.4 Power Supply - diesel generators.
- 3.5 AB-3A, -3B set at 400 psig.

4. Immediate Pre-run Checklist:

- 4.1 Verify that P_{TO} and P_{FM_2} rate and level clamps place AV-4 in hold.

- 4.2 Set clamps as follows:

P_{TO} level <u>70</u> psig	P_{FM_2} level <u>240</u> psig
P_{TO} rate <u>8</u> psi/sec	P_{FM_2} rate <u>±25</u> psi/sec

- 4.3 Exercise AV-5A, AV-5B, AV-4 in all modes of control.

- 4.4 Verify that the following air valves are in the specified condition.

AV-1 <u>C</u>	AV-6A <u>C</u>
AV-2 <u>C</u>	AV-6B <u>C</u>
AV-3 <u>C</u>	AV-7 <u>C</u>
AV-4 <u>C</u>	AV-8 <u>O</u>
AV-5A <u>C</u>	AV-9 <u>C</u>
AV-5B <u>O</u>	AV-10 <u>C</u>
	AV-14 <u>C</u>

- 4.5 Verify that all nuclear recorders are running and inking.

- 4.6 Verify that nozzle cover is removed and is left on crane hook.

- 4.7 Verify that Test Pad doors are open.

- 4.8 Verify that core ball check drain valve is open.

- 4.9 Remove telephones from hooks.

5. Detailed Steps:

- 5.1 I. O. C. Are all personnel clear of the Test Area?

- 5.2 All operators and data observers acknowledge readiness.

- 5.3 C. D. S. Sound Test Pad siren.

- 5.4 C. D. S. Turn on area warning lights.

- 5.5 I. O. C. Is EG&G ready?

- 5.6 A. T. O. Arm air supply.
- 5.7 A. T. O. Start blowers 4 and 5.
- 5.8 C. E. Start light water pumps.
- 5.9 C. E. Is water flow satisfactory?
- 5.10 C. E. Start heavy water pumps.
- 5.11 N. O. Verify that reactor is in the scram mode.
- 5.12 N. O. Verify that the vane demand is at 0°, rod key switches in servo position, vane key switches in override position.
- 5.13 N. O. Verify that all scrams and resets except L_1 , L_2 , & L_3 are unbypassed.
- 5.14 N. E. Start control system hydraulic pump.
- 5.15 N. O. Clear scram.
- 5.16 N. O. Place vane key switches in servo position one at a time.
- 5.17 N. O. Center vernier rod.
- 5.18 N. O. Close reflectors.
- 5.19 N. O. Begin approach to criticality using standard startup procedure,* level out power when P_2 current reaches 10^{-8} amp.
- 5.20 N. O. Correlate P_1 , P_2 , P_3 with C_1 and establish level set points for 5.39.
- 5.21 N. O. Unbypass fast resets.
- 5.22 N. O. Set period demand at 20 sec, set log power level scrams a factor of 10 above present power level. Set reset level a factor of 2 above present level.
- 5.23 N. O. Center vernier rod.
- 5.24 N. O. Verify that log power control is selected.
- 5.25 N. O. Verify that P_2 level and P_3 period selected for control.
- 5.26 N. O. Arm automatic control system.
- 5.27 N. O. Zero log power error signal.

*Standard startup procedure is listed after this checklist.

- 5.28 N. O. Transfer to log power automatic control.
- 5.29 N. O. Pulse demand to verify automatic operation.
- 5.30 A. F. O. Verify that Air Valve 4 is closed, in servo mode.
- 5.31 A. F. O. Open Air Valve 2.
- 5.32 C. D. S. Start process air system recorders.
- 5.33 I. O. C. Alert EG&G to standby for a timing pulse.
- 5.34 O. D. S. Actuate timing pulser.
- 5.35 I. O. C. Start camera No. 1.
- 5.36 A. F. O. Increase P_{FM_2} to 37 psig and maintain.
- 5.37 A. D. S. Start Ampex recorder.
- 5.38 A. T. O. Increase T_{TO} to 400 °F at rate of 100°F/min-ute and maintain.
- 5.39 N. O. Set scram and reset points as follows for log level and period (to be determined from 5.20).*
<u>Reset</u>	<u>Scram</u>
P_1 _____ amps	P_1 _____ amps
P_2 _____	P_2 _____
P_3 _____	P_3 _____
Period <u>2</u> sec	Period <u>1</u> sec
- 5.40 I. O. C. Start camera No. 2.
- 5.41 N. O. Increase nuclear power until core temperature of 1900 degrees is reached.
- 5.42 D. C. Read all indicating chambers. Switch C_1 to 10^{-5} amp.
- 5.43 N. O. Read off your core and exit gas temperatures.
- 5.44 C. D. O. Are you in agreement with these temperatures?
- 5.45 O. D. S. Start strain gauge recorders.
- 5.46 I. O. C. Start camera No. 3.
- 5.47 A. D. S. Start reading Bourdon gauges.
- 5.48 A. F. O. Increase P_{FM_2} linearly to 200 psig in 60 sec and hold.
- 5.49 I. O. C. Start camera No. 4.

*Reset and scram values were filled in during the run.

- 5.50 N. O. Increase power until core temperature reaches 2250 degrees. Hold for 45 sec. Decrease temperature to 1900 degrees.
- 5.51 A. F. O. Decrease P_{FM_2} to 37 psig.
- 5.52 O. D. S. Stop strain guage recorders.
- 5.53 N. O. Decrease P_2 current to 10^{-8} amp, and shut down.
- 5.54 A. T. O. Decrease T_{TO} at a rate of $100^\circ\text{F}/\text{minute}$ until Air Valve 5A is closed.
- 5.55 A. F. O. Close Air Valve 2.
- 5.56 A. T. O. Open Air Valve 7 and close Air Valve 8.
- 5.57 A. F. O. Close Air Valve 4.
- 5.58 D. C. Collect your charts.
- 5.59 A. D. S. Are reflector temperatures low enough to secure heavy water pump?
- 5.60 C. E. Secure heavy water pump.
- 5.61 A. T. O. Open Air Valve 8 and close Air Valve 7.
- 5.62 A. T. O. Stop blowers 4 and 5.
- 5.63 Remove all control keys.
- 5.64 I. O. C. Notify E. O. C. of Tank Farm pressure.
- 5.65 I. O. C. Record perimeter radiation monitoring and RAMS system readings.
- 5.66 Announce test termination.

The standard startup procedure referred to in step 5.19 is shown below. The Nuclear Operator completed each numbered step of this procedure, then requested permission from the Test Conductor to proceed to the next.

Startup Procedure

1. Bypass permissive chain
2. Raise S_2 count rate to 250 c/sec
3. Raise S_2 count rate to 1000 c/sec
4. Retract S-counters to lower count rate one decade.
5. Raise S_2 count rate to 1000 c/sec (watch I_2)
 - Switch to Intermediate
 - Retract S-counters into bunker
 - Bypass S_1 recorder

6. Raise I_1 count rate to 1000 c/sec (watch C_1 and P_4)
Unbypass P_4
Bypass all I recorders and meters
Bypass power range period meters
7. Raise C_1 current to 3×10^{-10} amp ($P_4 = 3 \times 10^{-7}$ amp,
 $P_{123} = 2 \times 10^{-10}$ amp)
Unbypass power range meters
Bypass C_1 and P_4 , and bypass startup chain
8. Raise P_{123} current to 10^{-8} amp

6.4 Operating Crew

The operating crew has been listed in Section 6.3 above. Figure 6-4 shows a diagram of the control area, with the operators' locations noted.

Figures 6-5 through 6-9 show sections of the operating and data console within view of the principal operators.

The following pages list the functions of each crew member, together with data displayed for his use or information:

Test Conductor

Directs the test operation, by means of verbal instructions to crew members. Receives verbal reports from them on progress of run or data of importance.

Internal Operation Control

Keeps External Operation Control informed of progress of run, and relays messages between him and Test Conductor if they become necessary.

External Operation Control

Remains outside of control room, handling site activities not directly related to actual conduct of reactor test. Relays any necessary messages to or from Test Conductor via Internal Operation Control.

Nuclear Operator

Adjusts reactor controls to make reactor critical and regulate power level. Through power, regulates the core temperature. Observes indicators

Fig. 6-4. Plan of control area, showing location of the operators.

Fig. 6-5. Control console, cooling water and air temperature control sections.

GLB-638-5143

AIR FLOW

GLB-638-5144

Fig. 6-6. Control console, air flow control section.

Fig. 6-7. Control console, nuclear startup instrumentation.

Fig. 6-8. Control console, nuclear control section (left).

Fig. 6-9. Control console, nuclear control section (right).

showing reactor control element position, power level and period, core temperature, and exit air temperature.

Nuclear Engineer

Observes (internal) operation of nuclear control system, keeping Nuclear Operator advised of any apparent trouble. Watches display of control element positions, power level, period, servo system error signals, hydraulic fluid pressures and temperatures, etc.

Air Flow Operator

Operates valves AV-1, AV-2, AV-3, and AV-4 to regulate air flow through reactor. Observes pressure at diffuser on the test vehicle (P_{TO}) and flow indication from flowmeter in the test bunker, as well as valve position indicators.

Air Temperature Operator

Operates valves AV-5A and AV-5B to adjust temperature of air flow to reactor. Operates blowers 4 and 5, and valves AV-7, AV-8, AV-9 and AV-10 to supply low pressure air to reactor for aftercooling or sustained operation at low power. Observes air temperature entering reactor (T_{TO}), valve positions, and blower motor current.

Coolant Engineer

Operates pumps supplying cooling water to reactor air duct and control rods, and heavy water to reflector. Observes temperatures and pressures of water.

Data Coordinator

Keeps master log book, marks and collects recorder charts. Records miscellaneous information during test. Assists Nuclear Operator in taking neutron counts during approach to criticality.

Offner and Console Data Scanners

Observe data presented on strip chart recorders during test, and keep recorders running properly. Offner Data Scanner supplies common timing pulse to all recorders at several instants during tests.

Core Data Observers

Observe core, exit air, and structural component temperatures appearing on ASCOP display during test. Interpret data and furnish recommendations for operation, and power estimates, to Test Conductor.

ASCOP Data Scanners

Observe miscellaneous data displayed on ASCOP system during test. Observe and record air pressures from television display of direct-reading Bourdon gauges at test bunker (P_{T2} and P_{T4}).

Instrumentation and Communications Engineers

Observe miscellaneous data and stand by to repair equipment breakdown.

Overall responsibility for the test facility and reactor was held by the Test Group Director, who stationed himself in the Control Room during tests, but took no active part in carrying out the test procedure.

6.5 Events of the Test

The Intermediate Power Test was scheduled for 11:00 AM, Friday, May 12; and several days before that date were set aside for final preparations. The last previous reactor test, which was a check of closed-loop automatic operation of the nuclear control system, was held on Friday, May 5. The following Monday and Tuesday were occupied with installation of new lifting fixtures on the reactor core vessel and completion of installation of a few transducers on the test vehicle. Wednesday and Thursday were reserved entirely for final calibration and checking, with no more system alterations or additions permitted.

On Wednesday and Thursday and during Friday morning, all components and systems of the test facility and reactor which would be involved in the test were carefully checked for proper operation, according to standard check lists used in preparing for all test runs. The checklisting showed all systems to be in complete readiness for the test run.

On the day scheduled for the test, access to Site 401 was limited to one road (from Mercury via Jackass Flat), and a special guard posted there prevented entrance of any but test personnel and invited guests. On Site 401,

during the period of the test, personnel were all located in three buildings, to provide safety in case of a wind reversal with possible transport of high fission product concentrations toward the control area. Guests and some site personnel viewed display meters and television monitors in the Disassembly Building, to follow progress of the test. Safety and emergency personnel were gathered at the Check Station, another group of guests viewed a display console in the Control Building basement, and all personnel concerned with test operation remained on the main floor of the Control Building. In the Control Building, Test Organization personnel from LRL, AEC, U. S. Weather Bureau and other organizations remained in the office rooms and in a large room adjacent to the control room, where the External Operation Control officer provided communication from the control room. Occupancy of the control room itself was limited to the Operating Crew, the Test Group Director, and the Criticality Monitor (an independent observer to give backup appraisal of possible operation hazards).

On Friday morning, after all system checks had been completed except for the final Nuclear Operator's scram checks, which involved last minute verification of a number of modes of scram actuation, a party of about six persons made an inspection of the test bunker, pad, vehicle and reactor to search for loose objects, obstructions to moving parts, temporary clipleads or jumpers, and other improper conditions.

After completion of this inspection and of final scram checks, the facility was armed for testing: key switches were closed at the bunker and test vehicle to provide power to the main air shutoff valve (AV2) and to the reflector carriage drive system. The nozzle cover was removed, and the boral blanket (usually left on the core for criticality safety) was lifted off, and the last party of test personnel (two persons) left the test area.

The probability of having acceptable weather had seemed very low during the morning and early afternoon, but some chance did exist, and so the two reflector halves were started closing on the core at 2:00 PM, in order to hold them within 6 inches of a fully closed position and avoid delay if weather clearance should be given. At 2:15, it became apparent that the chance of good weather was very slight, and the reflector closure was stopped, both halves being brought back out to a fully open position. At 3:45 PM the test was definitely cancelled for that day, and re-scheduled for 11:00 AM the following day. The test facility was then disarmed and secured.

On Saturday morning, May 13, some improvement in the weather was seen and it was expected that test operation might be possible. No deterioration in readiness condition of the facility was to be expected from the overnight delay, but to ensure reliability the nuclear control system checklisting was repeated. The facility was again made ready for operation, and at 11:30 AM the reflectors were closed to within 6 inches of their inner limits and then held at that position. The wind direction oscillated in and out of the acceptable zone throughout the day, and when, at about 4:30 PM, it became evident that no reasonable probability of sustained acceptable conditions existed, the reflectors were withdrawn and the facility secured, with the test scheduled for 11:00 AM the following day.

On Sunday, May 14, weather conditions were excellent from the beginning of the day. Again, the nuclear control system pre-run checklisting was repeated, and a final inspection was made of the test bunker and reactor. Clearance to proceed was obtained from the AEC Test Manager at 11:20 AM.

Figures 6-10 through 6-14 reproduce the reactor log book record of the test operation, showing that the succeeding time up to 1:45 (1345) was mainly occupied with bringing the reactor power up to a high enough level to permit transfer to automatic control. A supplementary — and much more detailed — record of the remainder of the run is furnished by the operators' sound record presented in Section 6.6.

Following completion of the run and shutdown of the reactor and facility, entry parties were permitted to return to the test area. The first party went in about an hour after reactor shutdown, made a preliminary survey of radiation levels along the entry road and within the test bunker, and then shut down equipment in the bunker. Succeeding entry parties recovered radiation detectors, effluent collector trays, motion picture films, and so forth. Shutdown of the reactor was completed by placement of the nozzle cover on the exit nozzle, accomplished remotely with the test pad crane, through use of television monitors.

6.6 Operators' Sound Transcription

The following pages present an excerpt from a tape recording of sound on the operating crew's headset system. It should be recalled that general progress of the run was guided by the Test Conductor's Procedure Checklist

155

STAGE IV RUN 1Saturday May 13, 1961 **SECRET R D**

0700 System has been left alone overnight. Immediate pre-run checklists in progress.

1028 Checkout of reflector limit switches begun.

1112 Boral Blanket off.

1129 Starting reflectors in to 6" E: 6.5 W 7.7

Waiting for weather clearance.

1220 All operators on posts - talk through begun.

1638 Wind continues to come from the wrong direction. Began withdrawing reflectors. Decision was made not to run today.

1700 Boral Blanket replaced, nozzle cover on. Reactor secured for the night.

Sunday, May 14, 1961

0700 Immediate pre-run checklists ~~are~~ being made.

1040 Checkout of reflector limit switches begun.

1045 ~~REFLECTORS~~ Boral blanket removed.

1106 Started to close reflectors. Nuclear Operator completed check-listing controls.

Background Count reflectors full back.

S_1	S_2	I_1	I_2
358 c/m	516 c/m	15 c/m	9 c/m

1123 Reflectors to 6" E: 6.4 W: 7.4
Clearance has been given to run.

1127 All operators ready.

1128 Air supply armed - Blowers #4 & 5 turned on.

Note: PAS @ 3500

1136 Rod #3 selected as vernier. Scram cleared. Saffties out, vernier centered. Began closing reflectors.

SECRET R D

GIL-636-1489

Fig. 6-10. Control room log book, pages for May 14, 1961.

156 SUNDAY May 14, 1961 STAGE III RUN 1

1140 Reflectors at inner limits. Count taken: reflectors in, vernier at mid-position.

S_1	S_2	I_1	I_2
918 c/m	1120 c/m	14 c/m	6 c/m

~~1140~~ 1145 Began to run vanes out to approach criticality1148 Vanes at 20° 1151 " " 51° 1153 " " 60° **SECRET R D**Brought vanes to about 66°

1158 V_1 66.64	V_3 67.39	V_5 65.93	V_7 67.00	} 66.51°
V_2 66.06	V_4 65.61	V_6 66.67	V_8 66.79	

S_1	S_2	I_1	I_2
12,349 c/m	13,753 c/m	16 c/m	16 c/m

1205 Count rate on S_2 recorder raised to 1000 c/sec

V_1 68.44	V_3 67.33	V_5 67.78	V_7 68.86	} 68.4°
V_2 67.90	V_4 67.42	V_6 68.56	V_8 68.87	

S_1	S_2	I_1	I_2
50,414 c/m	56,759 c/m	60 c/m	42 c/m

Critical Vane angle estimated to be 69.1°

1209 S counters retracted to give a factor of 10 reduction in count rate. Increasing power to raise S's decade.

1215 Reactor Critical.

1220 Leveling count at 1000 cps on S_2

1224 Switch to intermediate range recorders.

1226 S counters retracted into bunker

1227 S recorders ~~after~~ bypassed.1228 Began increasing I_1 count rate to 1000 c/sec using vernier rod @ 10" to give ~~period~~ positive period.

1230 C. off pin beginning to read.

SECRET R D

GILL-636-1490

Fig. 6-11. Control room log book, pages for May 14, 1961.

157

Sunday May 14, 1961 STAGE IV RVN 1

1242 Count rate level

SECRET R D C_1 3.5×10^{-12} amps I , 900 c/s
 P_4 3.8×10^{-9} amps

1243 P_4 unbypassed
 All I meters and recorders bypassed.
 P_1, P_2, P_3 period scrams bypassed.

1247 Going for 5×10^{-10} amps on C_1 , from 3×10^{-12} amps
 P_4 ~~indicates~~ indicates 100 sec period.

1256 Levelled for readings.

C_1 1.2×10^{-10} P_3 1×10^{-10}
 P_1 1.1×10^{-10} P_4 1.6×10^{-7}
 P_2 1.1×10^{-10}

1300 Levelled for readings.

C_1 5.5×10^{-10} P_3 4.7×10^{-10}
 P_1 5.6×10^{-10} P_4 6.6×10^{-7}
 P_2 5.9×10^{-10}

1305 Unbypassed P_1, P_2, P_3, γ meters (scrams)

1318 Bypassed C_1 recorder P_4 level & rate.
 Startup chain bypassed.

1320 C_1 4.5×10^{-9} P_3 3.9×10^{-9}
 P_1 4.8×10^{-9} P_4 6.0×10^{-6}
 P_2 5.2×10^{-9}

1321 Going up to 10^{-8} amps on the C.I.C.'s

SECRET R D

GILL-636-1491

Fig. 6-12. Control room log book, pages for May 14, 1961.

158

Sunday May 14, 1961

STAGE IV RUN 1

- 1330 Reached 10^{-8} amps on C.I.C.'s. Indicated periods appear very noisy. Too noisy to attempt to transfer to automatic control at this point.
Looked at period and level signals from each channel.
SECRET R D ~~Ch~~ Chose to control on P1 period - P2 level.
- 1336 Unbypassed fast reset on Log IV.
1339. Transferred to log power automatic control. Smooth transfer, power held constant.
- 1340 C, 2×10^{-8} P_3 1.95×10^{-8}
P, 2.3×10^{-8}
 P_2 2.6×10^{-8}
- 1341 Demand pulsed to verify automatic operation. Period control verified. Power level held in automatic control @ 2×10^{-8} on C.
- 1344 AV2 open.
- 1344 Timing pulse actuated
- 1345 Increased PFM2 to 37 psig and held.
Began ~~increasing~~ increasing T70 to 400°F at 100°F/min.
Set scram and reset levels
- 1349 Reached 400°F on T70.
Began increasing nuclear power until core temperature reached 1900°F
- 1400 Checked core and exit gas temperatures - correlated well.
- 1401 Increased PFM2 linearly to 200 psig (slightly more) and held.
- 1402 Reached 200 psig on PFM2 began bringing up power until core temperature at about 2250°.
- 1404 At 2250, holding fairly steady. Timed for 45 sec. Then power reduced to get flat 1900°F.
- 1406 PFM2 decreased to 37 psig.

SECRET R D

At top of plateau:

C, 3.8×10^{-6}	P_2 4×10^{-6}	L_1 121%
P, 4×10^{-6}	P_3 2×10^{-6}	L_2 ∞
		L_3 15%

GLL-636-1492

Fig. 6-13. Control room log book, pages for May 14, 1961.

Sunday May 14, 1961 STAGE IV RUN 1 159

Note:

At 1349	C_1	7.5×10^{-7}	$L_1 = 12\%$
	P_1	9.0×10^{-7}	$L_2 = 45\%$
SECRET R D	P_2	9×10^{-7}	$L_3 = 2\%$
	P_3	5×10^{-7}	

At 1358	C_1	$.8 \times 10^{-6}$	$L_1 = 16\%$
	P_1	10^{-6}	$L_2 = 52\%$
	P_2	10^{-6}	$L_3 = 2\%$
	P_3	5.5×10^{-7}	

1408 Decreased reactor power to about 10^{-7} amps. Because of delay X-n's, reactor was difficult to control

1408 Nuclear operator scrambled reactor.

1409 Began moving reflectors out.
~~140~~ decreased to ambient

1412 D₂O flow ~~was~~ secured.

1414 AV2 closed.
 AV4 closed.

1419 West reflector ~~indicator~~ indicator jammed at 5". Stopped moving out East reflector only.

1420 Bunker face camera (zoomar lens) appears fogged. Other cameras O.K. Can't monitor reflector opening. Held reflectors until better picture is available.

1427 Closed blowers - core temperature seems to hold O.K.

1513 Both reflectors to outer limits (West ~~meter~~ meter still jammed)
 Appear to be full out.

SECRET R D

GIL-636-1493

Fig. 6-14. Control room log book, pages for May 14, 1961.

(Section 6.3). The Test Conductor followed this list, reading each item in turn, and response was given by the operator to whom the instruction was addressed. Many incidental exchanges of comments occurred between the checklisted items, and have also been included. The crew's names are listed by function in Section 6.3, the Test Conductor's Checklist.

The time in seconds, corresponding to the time scale of Fig. 4-1, is shown next to most of the instructions, responses, and comments below. Where no time is indicated, it should be taken for granted that the comment occurred almost at once after the preceding one, or even partly overlapping it.

A set of notes, referenced by number next to the recorded comments, is given following the sound transcription to clarify items which might otherwise be difficult to interpret.

Operators' Sound Transcription

<u>Time</u>	<u>Speaker</u>	<u>Notes</u>
54	Barnett	(1) Very good. Air temperature operator, increase T_{TO} to 400 F at a rate of 100 F per minute and maintain.
60	Dykes	Increasing T_{TO} to 400 F.
	Staker	(2) You'll get some interaction here, Rex (Finke).
72	Barnett	Keep an eye on your levels again - you'll start warming, the core'll change -
75	Finke	The vane angle is coming down -
	Barnett	Well, it'll change temperature.
	Finke	Right, core is cooling.
80	Uthe	Confirm that, the core is cooling.
152	Barnett	Nuclear Operator -
	Finke	Yes?
	Barnett	Would you set your scram and reset set points as follows: I will read off the reset first.
161	Finke	Reset first.
173	Barnett	P_1 is 1.15×10^{-5} (since we only have one set point we'll choose the -
	Finke	Highest.
	Barnett	- highest one, which is 1.3×10^{-5}).

<u>Time</u>	<u>Speaker</u>	<u>Notes</u>
181	Finke	All right, reset going up - 1.3×10^{-5} .
	Barnett	Your scrams, P_1 2.3×10^{-5} -
191	Finke	Say again?
	Barnett	P_1 , 2.3×10^{-5} ;
	Finke	P_1 , 2.3×10^{-5} .
200	Barnett	P_2 , 2.6×10^{-5} .
204	Finke	P_2 , 2.6×10^{-5} .
	Barnett	P_3 , 1.96×10^{-5} ; 1.9×10^{-5} .
	Finke	P_3 , 1.9×10^{-5} .
249	Dykes	Air Temperature Operator, stable at 400°F.
	Barnett	Very good.
258	Staker	(3) It's a little noisy here; I hope we don't stay long . . . (inaudible)
261	Barnett	Console Data Scanner, we can do without the timer on P_{TO} .
266	Shyne	(4) P_{AS} is 3500.
	Barnett	P_{AS} 3500.
272	Barnett	All right, everyone get ready.
277	Barnett	Internal Operation Control - Start camera No. 2.
	Laine	Start camera No. 2.
284	Barnett	Nuclear Operator, increase nuclear power until core temperature of 1900° is reached.
289	Finke	Nuclear power coming up, on a 20 second period.
309	Finke	I'd like a mark from somebody when the temperature starts up.
	Staker	Temperature is increasing very slightly, Rex.
337	Finke	We're coming up on L_2 and L_1 ; nothing on L_3 yet.
355	Uthe	We're getting good power now, Rex.
357	Finke	We're at 10^{-6} .
361	Finke	We're at 55 percent on L_2 .
365	Finke	Is the temperature starting up?
	Barnett	You can watch your meters, Rex; you have temperature indication here.
	Finke	Right,

<u>Time</u>	<u>Speaker</u>	<u>Notes</u>
385	Barnett	You also have the recorder.
	Finke	Right; am now controlling off of L_1 , and the temperature.
414	Finke	(5) Mike Uthe, the temperatures of 18 and 17 are 700 and 1000 at this particular moment. Which one do I control from? Give me the meter here.
	Uthe	One minute Rex, I read you.
	Finke	It looks like the meter is stuck on No. 18.
423	Uthe	No, it's a bad couple, Rex.
	Finke	Bad couple.
	Uthe	Let's take No. 17.
	Finke	All right.
487	Finke	Number 17's the highest of my thermocouples.
	Uthe	17 is here, too, Rex.
491	Finke	I read 1800 degrees; I'll start to level off.
500	Finke	Leveling off in power.
522	Finke	I am level on thermocouple No. 17 at 1900 degrees; rising slightly, possibly.
529	Barnett	Is your power level, reasonably?
535	Staker	You are level on the recorder -
536	Barnett	Data collector, read all indicating chambers.
	Staker	P_1 , 9×10^{-7} ; -
548	Barnett	Just write them down, and hand them to him, please.
557	Barnett	Is your temperature holding steady, Nuclear Operator? Seems to be.
	Finke	Temperature's holding steady with a variation of about 10 degrees, at about a reading on thermocouple No. 17 of 1920.
569	Uthe	Charlie, just a minute, or rather Rex; are you reading to me now?
574	Finke	I can't hear you.
	Uthe	I missed that call; did you read to me?
	Finke	I'm just verifying that the temperature's level at 1920, on thermocouple No. 17.

<u>Time</u>	<u>Speaker</u>	<u>Notes</u>
582	Barnett	Nuclear Operator, will you now read off all your core and exit gas temperatures so ASCOP can verify.
587	Uthe	Request one-minute hold, please.
	Barnett	All right, delay that.
591	Finke	We're dropping slightly in temperature.
596	Finke	We're at 1890.
	Barnett	Tank farm?
604	Shyne	P _{AS} is 3150.
	Barnett	Very good.
617	Uthe	I am now clear for temperature readings.
620	Finke	(6) All right. Thermocouple No. 27, is 1550; thermocouple No. 20, 1640; No. 18 is stuck at 700; 17 is at 1880; D is at 1890; C is at 1930; GB, gas temperature is at 1090; GA, 1000 even; thermocouple 21 is at 1840; and 79 is at 1780 - and the level was fairly constant during the reading; did not change by 10 degrees.
674	Finke	We're at 1890, 1880 on thermocouple 17.
688	Barnett	We're waiting for a response from you, ASCOP.
	Uthe	Yes, I'm clear, I'm checking.
696	Staker	You're rising slightly, Rex.
	Finke	Okay.
731	Uthe	We will discard channel 18.
	Finke	Right.
746	Shyne	3000 on P _{AS} .
	Barnett	3000 on P _{AS} .
754	Varljen	Request permission to put C ₁ on 10 ⁻⁵ scale.
	Barnett	Permission granted.
766	Staker	You are level, Rex.
	Finke	Right.
775	Finke	You can continue to tell me these things; I'll be listening.
813	Staker	You are still level, Rex.
832	Shyne	P _{AS} is 2900.

<u>Time</u>	<u>Speaker</u>	<u>Notes</u>
848	Staker	Still level, Rex.
870	Varljen	(7) Chuck, I now have a correlation on all meter readings at this level.
	Barnett	Very good.
879	Finke	Are we going to recalculate the scram levels here?
	Barnett	Ah, yes, if we - We're running a little short of time; the tank farm is being depleted.
885	Uthe	Charlie?
	Barnett	Yes.
	Uthe	Your new control points will be channels 20 and 27.
894	Barnett	Read that, Nuclear Operator?
	Finke	(8) You want me to raise the lowest temperatures to the T_2 of 2250?
	Barnett	Right.
910	Uthe	The preferred one is No. 20.
	Finke	The preferred one is No. 20, all right, it reads 1700 degrees at this moment.
	Uthe	Rex, can you give me a retake on G_A , and G_B ?
916	Finke	G_A is at this moment 1075 and G_B is at 1160.
930	Barnett	ASCOP, we're going to have to go on up in power.
	Uthe	(9) Okay. We have a discontinuity here in gas temperatures, from ours as compared to yours. I would suggest that you disregard - uh, I recommend at this time going ahead with the program as scheduled. Estimated power right now is about 9 megawatts.
950	Barnett	Offner Data Scanner, start strain gauge recorders.
956	Barnett	Internal Operation Control, start camera No. 3.
	McConnell	Recorders all on.
962	Shyne	2750 P_{AS} .
	Barnett	ASCOP-Start reading Bourdon gauges.
968	Barnett	Air Flow Operator, increase P_{FM_2} to 200 psig in 60 seconds and hold.
	Shyne	Understood.
974	Finke	Give me a mark when you start, please.

<u>Time</u>	<u>Speaker</u>	<u>Notes</u>
976	Shyne	Rising now.
	Staker	Temperature's steady, Rex.
	Finke	It's dropping.
	Staker	It's dropping slightly.
992	Staker	Dropping slightly, Rex.
	Staker	Still dropping slightly.
	Staker	You're leveling off now, Rex; coming up slightly.
1004	Barnett	Start camera No. 4.
	Staker	(10) Reactor temperature rising slowly, etc., etc.
1028	Finke	Going up intentionally.
1039	Shyne	P _{FM2} at 200.
	Barnett	Very good.
	Barnett	Nuclear Operator, increase power until core temperature reaches 2250°F.
1046	Finke	Increasing power.
1052	Barnett	Stand by to lose some of your thermocouples; it may happen here.
1060	Finke	I'm controlling on thermocouple 20.
1070	Finke	We seem to be having an oscillation in the thermocouple reading.
1074	Uthe	You're clear here, Rex, you're doing fine.
1080	Finke	I'm going to switch control to thermocouple D because it's reading the same as 20 but is more stable.
1090	Finke	At 2200 degrees.
	Barnett	Give me a mark when you're steady at 2250.
1094	Finke	Starting to bring power back down to level.
1106	Staker	Dropping slightly, Rex.
1115	Finke	Dropped the power a little too much.
	Barnett	Are you steady?
1118	Finke	Not steady, I'm dropping.
1124	Finke	Now I'm coming back up.
	Barnett	We're running a little low on air, so let's -
1129	Finke	All right, the temperature is at 2250 at this moment -

<u>Time</u>	<u>Speaker</u>	<u>Notes</u>
1134	Barnett	All right, we're timing you now.
	Finke	- and it is level.
	Barnett	We're timing you now.
1148	Finke	Temperature is level.
1163	Finke	We've dropped about 20 degrees.
1168	Barnett	Have about 8 seconds - 5 seconds - 3 seconds - decrease temperature to 1900 degrees.
	Finke	Temperature coming down.
1203	Finke	We are 100 degrees down.
1210	Finke	200 degrees down.
1220	Finke	I'm leveling at 1900 degrees.
	Barnett	Very good.
	Uthe (11)	Request 100 more degrees down, 1800.
	Barnett	Decrease to 1800 degrees.
	Finke	18? You want another hundred lower.
	Barnett	One hundred degrees lower.
	Finke	All right, I'll let it drift downward.
1238	Finke	I am now at 1800 degrees.
	Barnett	All right; Air Flow Operator, decrease P_{FM_2} to 37 psig.
1246	Shyne	Understood.
	Barnett	Nuclear Operator, hold your temperature as best you can.
	Finke	All right.
1249	Shyne	Coming down.
1284	Finke	We just lost thermocouple 27, looks like.
	Uthe	You're doing real fine, Rex, just real fine.
1304	Shyne	P_{FM_2} at 37.
	Barnett	Very good. Offner Data Scanner, stop strain gauge recorders.
1310	Barnett	Nuclear Operator, decrease P_2 current to 10^{-8} amps and shut down.
	Finke	Power level coming down.
1334	Finke	We went up about 200 degrees on that, following the air flow.

<u>Time</u>	<u>Speaker</u>	<u>Notes</u>
1340	Shyne	P_{AS} is 1700.
	Barnett	P_{AS} is 1700.
1346	Staker	Hydraulic pressure is fluctuating slightly.
1350	Barnett	(12) Those vanes are pretty busy.
1362	Finke	(13) Rod is oscillating more than I'd like.
1374	Finke	I think I am out of phase with it.
1383	Finke	Scram?
1385	Finke	Scram.
	Barnett	Reactor is scrammed; start your airplane.
1392	Finke	Yes. I was trying to keep the temperature down.
1398	Barnett	Air Temperature Operator, decrease T_{TO} at a rate of 100°F per minute until Air Valve 5A is closed.
	Dykes	Decreasing T_{TO} .
1403	Uthe	(14) Ah, correction, please — recommend if you can, holding a few minutes more; it's coming down very slowly, Charlie.
1410	Barnett	Delay that, Air Temperature.
	Dykes	Delayed.
	Uthe	(15) Yeah, that's the shroud problem.
1422	Barnett	Hold inlet air temperature.
	Dykes	Holding inlet air.
1458	Barnett	Nuclear Operator, you were interested in moving the reflectors back.
	Finke	Right.
1464	Barnett	You have permission.
1472	Finke	Reflectors are coming out.
1494	Staker	Tom, do you want these gamma readings? We're being held up in power —
	(Unknown)	sshhh.
	Uthe	Recommend, uh —
1507	Barnett	Let's knock off the unnecessary chatter, we're not through yet.
	Uthe	Recommend decreasing T_{TO} at this time.

<u>Time</u>	<u>Speaker</u>	<u>Notes</u>
	Barnett	Air Temperature Operator, decrease T_{TO} at a rate of 100 °F per minute until Air Valve 5A is closed.
	Dykes	Decreasing T_{TO} .
1554	Barnett	Having any trouble holding flow, Air Flow?
	Shyne	Negative.
	Barnett	Very good.
1607	Meier	Chuck Barnett?
	Barnett	Go ahead.
1610	Meier	Could I get an indication of what the reflector temperatures are at this moment?
	Barnett	ASCOP, would you get us information on reflector temperatures?
1620	Uthe	Understood.
1621	Dykes	AV5A closed, Chuck.
	Barnett	Very good.
	Mintz	Reflector temperatures are 100.
1626	Meier	Approximately 100 degrees?
	Mintz	Correct.
	Meier	Chuck?
	Barnett	Yes.
1634	Meier	Is there any reason to continue the D_2O flow?
		We're down at a hundred degrees now.
	Barnett	Very good. You have permission to secure the D_2O flow.
1642	Meier	Securing D_2O pumps.
	Hadley	Better watch reflector temperatures; you're still -
1652	Barnett	Stand by, Coolant Engineer on D_2O flow - watch the temperatures in case they start to increase, we'll start it again.
1660	Meier	Understood.
1670	Uthe	(16) Recommend transferring to blowers when ready.
	Barnett	Understood, ASCOP.

<u>Time</u>	<u>Speaker</u>	<u>Notes</u>
1681	Barnett	Air Flow Operator, close Air Valve 2 — hold one on this, hold one. Air Temperature Operator, as soon as you can bring in the blowers on the line you will go ahead and do so. Is that understood?
1692	Dykes	That's correct, Chuck.
	Barnett	All right, very good.
1694	Barnett	Air Flow Operator, close Air Valve 2.
	Shyne	Understood.
1733	Shyne	AV2 is closed.
1735	Dykes	AV7 coming open.
	Barnett	Very good.
	Barnett (aside)	Give us the bunker face camera?
1763	Dykes	AV7 open.
	Barnett	AV7 open, right.
1770	Dykes	Request close AV8?
	Barnett	Close AV8.
1801	Barnett	Air Flow Operator, you've closed Air Valve 4, is that correct?
	Shyne	Yes, I have.
	Barnett	Very good.
1866	Barnett	Having any trouble there, Air Temperature?
	Dykes	Having trouble regulating the blowers, Chuck, no real trouble.
	Barnett	But we have flow through the core now, do we not?
	Dykes	Definitely.
	Barnett	Very good.
1881	Barnett	ASCOP, give us another reading on reflector temperatures, please.
1893	Uthe	90 degrees Fahrenheit, Charlie
	Barnett	Very good.
	Uthe	Little bit higher.
1911	Uthe	Do we have time for a couple of comments here, Chuck?
	Barnett	Yes.

<u>Time</u>	<u>Speaker</u>	<u>Notes</u>
	Uthe	It looks like our power was about 44 megawatts, and our temperature was about 2350 degrees, about 2500 max; 2550.
1924	Barnett	2350 average, 2500 max?
	Uthe	Yeah, 2550 max.
	Barnett	Very good, very, very good.

Notes for Transcription

- (1) The transcript starts with Barnett reading item 5.38 from the checklist, just after the tape recorder was turned on.
- (2) The next six items discuss the expected change of control vane angle (automatic) to compensate for a change in core temperature. It was initially thought that the core temperature would start to rise immediately with inlet air temperature; then it was realized that it was still falling from the first flow of relatively cool air, as the core temperature had gone up several hundred degrees from nuclear heating before any air flow was started.
- (3) The reactor power had been left as low as possible in order to avoid core heating (and activation in case a shutdown and re-entry was necessary). As a result, the neutron detector signal supplied to the automatic control system was noisy from statistical variations in flux.
- (4) P_{AS} is stored air pressure, indicating amount of air available for remainder of test.
- (5) Temperatures 18 and 17 are core thermocouple readings displayed for the Nuclear Operator.
- (6) Thermocouples GA and GB read mixed air temperature downstream from the reactor.
- (7) This refers to ion chamber power level indications.
- (8) Copies of the procedure checklist issued for the test referred to the highest average core temperature as T_2 : see step 5.48 in checklist, Section 6.3.
- (9) Uthe notes here that the Nuclear Operator's exit air temperature readings are unreasonably low, and decides that this should be ignored.

- (10) Staker continued to comment on rising temperature for several seconds here, assisting Finke, whose attention was divided between power level and temperature indications.
- (11) Uthe had observed that there was a substantial difference between the direct ASCOP temperature readings and the temperature information being read by the Nuclear Operator, with a possibility of consequent excessively high temperatures in the core. By the time he was sure of this from comparing a number of channels, the high temperature plateau was almost over; he decided not to break into the program during the plateau, but to ask for an extra 100° reduction for safety on coming down from the plateau.
- (12) Barnett means that the hydraulic pressure fluctuation was probably a normal response to the fairly rapid sustained motion of the control vanes.
- (13) Finke's comments here refer to behavior of the vernier control rod. He was trying to decrease the power demand at a rate that could just be followed by vane motion, with the rod remaining about a quarter stroke off center in the "in" direction. By this time, the reactor was far sub-critical, with power being held up by delayed fission and γ -n neutrons. Consequently, the power level was very insensitive to rod position and slight variations in Finke's rate of turning the power demand dial were met by full-stroke deflections of the vernier rod. He finally gave up trying to continue this mode of operation, and completed shut down by pushing the scram button.
- (14) Uthe wished to let the core temperature drop farther before lowering the inlet air temperature, to avoid possible large temperature differences among various core components having unequal cooling rates.
- (15) This is recalled as having been spoken aside in answer to a question not recorded on the tape.
- (16) Residual power generation in the core was now low enough that an interruption in air flow could be tolerated. Switching to blower air supply would conserve remaining air in the tank farm for aftercooling in case of blower failure.

APPENDIX A

APPRAISAL OF AIRFLOW PARAMETER MEASUREMENTS

J. H. Moyer

Determination of reactor power and airflow rate at a selected point during each run, as described in Section 4.2.1, required knowledge of the various measureable airflow parameters at several stations along the duct. A discussion follows of the available measurements and how they were reduced to single best values at each station.

A-1 Mass Flow Rate

Two alternate flowmeters were available in the air supply system (in addition to the reactor inlet diffuser and exhaust nozzle, at which flow rate could also be determined). FM-2 was a small venturi meter which handled flows up to about 450 lb/sec, and operated choked. FM-1 was a larger, subsonic, venturi meter for high flow rates. Tank farm depletion was compared in each run with the integrated flow rate measurement, but could not be measured accurately enough to give more than general confirmation.

In the calculations of this report the errors in pressure and temperature measurements were estimated independently and propagated through the flow equations. The discharge coefficient used for FM-2 was 0.98, derived from water calibration. That used for FM-1 was obtained from blow-down experience and expressed as a function of flow rate.

IPT: The flow measurement was made at FM-2. After the run, gasket leaks were found in the temperature control valves, downstream from FM-2. From a previous comparison of FM-2 and diffuser flow measurements in non-nuclear runs, the leak was estimated to be about 5 lb/sec. Accordingly, the flow rate calculated from FM-2 pressure and temperature traces was reduced by this amount.

Thus eliminating a systematic error introduced a new random error, namely, how accurate was the 5 lb/sec estimate? The random error was taken to be ± 1.3 lb/sec, and this amount added to the uncertainty in flow rate derived from the flow rate error analysis. The flow rate used in power calculation, then, was 117.3 ± 3.5 lb/sec.

HP-1: Again FM-2 provided the measurement. The flowmeter temperature record in this run was ruined by a probe failure. Temperature was estimated from prior experience with the meter, and an appropriate uncertainty attached. The flow rate obtained, and used in power calculations, was 443.4 ± 5.8 lb/sec.

HP-2: Measurements were obtained from FM-1. The flow rate computed, and used in power calculations, was 661.4 ± 11.0 lb/sec.

HP-3: The measurement was from FM-1, which indicated 416.2 ± 9.0 lb/sec. This was the value used in power calculations.

A-2 Reactor Inlet Total Temperature, T_0°

Values used as T_0° were measured upstream from the diffuser. No temperature measurement was obtained between the diffuser and the reactor.

In IPT only one measurement was available, consisting of three chromel-constantan thermocouples paralleled to a Brown recorder. In HP-1 one other thermocouple was attached to an Offner recorder and agreed with the Brown. In HP-2 and -3 this thermocouple was attached to both Offner and ASCOP and produced unreliable results.

In this report the readings of the Brown recorder were used, with standard deviations estimated for each.

<u>T_0° values:</u>	IPT	$393 \pm 10^\circ\text{F}$
	HP-1	$394 \pm 15^\circ\text{F}$
	HP-2	$925 \pm 18^\circ\text{F}$
	HP-3	$402 \pm 10^\circ\text{F}$

A-3 Reactor Exit Total Temperature, T_2°

Three measurements were available from thermocouples on the exit pressure rake. Two of these were located close to the center of the duct ($r = 6$ in.) and one near the duct wall ($r = 17$ in.). In all runs the two near the center produced nearly identical readings, while the third was lower by 250°F or more. This was regarded as a strong indication of incomplete mixing inasmuch as the cool shroud air was discharged near the duct.

To derive numbers for calculation it was assumed that the outer thermocouple measured shroud air temperature and the other two measured the bulk mean temperature of all other air passing through the reactor. A bulk mean temperature for the combined air streams was computed by averaging the two center measurements and weighting this average and the reading of the outer thermocouple by their respective flow fractions:

$$\text{Bulk mean } T_2^\circ = \frac{\dot{w}_a}{\dot{w}_t} T_a^\circ + \frac{\dot{w}_b}{\dot{w}_t} T_b^\circ \quad (\text{A.1})$$

where

$\dot{w}_a, T_a^\circ$ refer to the center air stream,

$\dot{w}_b, T_b^\circ$ refer to the shroud air stream,

$\dot{w}_t$ = total mass flow rate from the reactor.

Flow fractions were obtained from theory alone and were used without any error assigned. Standard deviations in temperature were combined by the general expression for propagation of error in n independent variables:

$$\delta U = \left[\sum_{i=1}^n \left(\frac{\partial U}{\partial X_i} \right)^2 \delta X_i^2 \right]^{1/2} \quad (\text{A.2})$$

where

$U = U(X_1, X_2, \dots, X_i, \dots, X_n),$

δX_i = standard deviation of X_i .

The assumption made regarding incomplete mixing is itself somewhat inaccurate. To account for this effect, the difference between the value calculated in the above manner and the average of the two center thermocouples was noted. Half of this difference was then added to the calculated uncertainty, as an estimate of the inaccuracy in the mixing assumption.

The treatment of IPT data was an exception to the foregoing procedure. Here the edge thermocouple was not recorded, but the consistent behavior in the other runs indicated that some adjustment was in order. This was made by careful comparison with the other runs, and a slightly greater uncertainty was attached to the assumed value.

The results of this procedure are tabulated below, together with the raw data, identified by thermocouple number.

Run	T262 (°F)	T263 (°F)	T280 (°F)	T ₂ [°] , derived (°F)
IPT	1859 ± 30	1856 ± 30	--	1840 ± 34
HP-1	1579 ± 30	1579 ± 30	1263 ± 30	1558 ± 30
HP-2	1834 ± 30	1834 ± 30	1597 ± 30	1820 ± 27
HP-3	1743 ± 30	1755 ± 30	1409 ± 30	1726 ± 32

A-4 Reactor Inlet Total Pressure, p₁[°]

Two static and three total pressure measurements were available. All were converted to total pressure and a mean value used in calculation. The data, expressed as total pressure, and the mean values, are tabulated below.

Transducer	p ₁ [°] values: total pressure (psia)			
	IPT	HP-1	HP-2	HP-3
P-13	68.9	206.0	357.0	222.7
P-14	53.6	207.7	339.5	216.5
P-35	63	198	320	210
P-36	63	212	337	223
P-37	54.9	206	331.5	208.8
Mean	61	206	337	216

The uncertainty in these measurements has not been considered. The only use of p₁[°] in power calculations is in entering the tables of gas properties, where the pressure dependence is weak.

A-5 Reactor Exit Total Pressure, p₂[°]

Nine measurements were available, both static and total, and reported variously as gauge and absolute. All were converted to absolute total pressures, making use of approximately known flow rate to find the local Mach number. The Bourdon gauge readings were corrected from a postrun calibration.

To permit a comparison of estimated and experimental standard deviations, uncertainties were estimated for each measurement.

The raw data thus adjusted are tabulated below.

Trans-ducer	Recorded on	Measured as	IPT	Total pressure (psia)		
				HP-1	HP-2	HP-3
P-15	OFFNER	static	---	119.6 ± 8	188.6 ± 8	123.9 ± 8
P-16	ASCOP 3	static	18.5 ± 3	118.0 ± 3	179.6 ± 3	113.8 ± 3
P-17	ASCOP 3	total	28.6 ± 3	118.4 ± 3	187.0 ± 3	123.6 ± 3
P-18	ASCOP 3	total	31.4 ± 3	120.0 ± 3	183.7 ± 3	125.5 ± 3
P-19	OFFNER	total	32.2 ± 3	121.0 ± 5	184.0 ± 5	125.0 ± 5
P-21	ASCOP 3	total	25.7 ± 3	115.8 ± 3	179.5 ± 3	120.9 ± 3
P-22	OFFNER	total	33.1 ± 3	120.0 ± 5	184.0 ± 5	127.0 ± 5
P-23	OFFNER	total	29.2 ± 2	120.0 ± 5	187.0 ± 5	126.0 ± 5
Bour-don	VISUAL	total	34.9 ± 5	125.2 ± 5	195.2 ± 5	133.2 ± 5

The measurements were first combined to form the arithmetic mean, m , and the experimental standard deviation of the mean, S_m :

$$m = \frac{1}{n} \sum_{i=1}^n X_i \quad (A.3)$$

$$S = \left[\frac{1}{n} \sum_{i=1}^n (X_i - m)^2 \right]^{1/2}$$

$$S_m = \frac{S}{\sqrt{n}} \quad (A.4)$$

Then, for comparison, the measurements and their estimated uncertainties were combined in a weighted mean, m^w :

$$m^w = \frac{w_1 X_1 + w_2 X_2 + \dots + w_n X_n}{w_1 + w_2 + \dots + w_n} \quad (A.5)$$

where

$$w_i = \frac{1}{(S_i)^2}$$

The standard deviation of the weighted mean m^w may be found, by applying Eq. (A.2), to be

$$S_m^w = \frac{1}{(w_1 + w_2 + \dots + w_n)^{1/2}} \quad (A.6)$$

The experimental and estimated errors, computed by Eqs. (A.3) through (A.6), compare as follows:

Run	m	S_m	m^w	S_m^w
IPT	29.2	1.72	28.8	1.02
HP-1	119.8	0.80	119.0	1.27
HP-2	185.4	1.52	183.9	1.27
HP-3	124.3	1.59	122.8	1.27

For calculation of reactor power the experimental values m and S_m were used:

p_2° values (psia)	
Run	p_2°
IPT	29.2 ± 1.7
HP-1	119.8 ± 0.8
HP-2	185.4 ± 1.5
HP-3	124.3 ± 1.6

Use of the experimental rather than computed standard deviations is an arbitrary choice. An objection might be raised that it seems unreasonable to assign different values for different days' operations, but in fact the data system did seem to give more reliable information on some days than on others, partly because of different amounts of time available for adjustments and calibration before a run, or different length of standby period while waiting for weather clearance; and partly because of varying amounts of noise in the telemetering systems.

A-6 Diffuser Total Pressure, p_0°

Two applicable pressure measurements were made upstream from the diffuser in run HP-2, one static and the other total. These are tabulated

below, both expressed as total pressure, with estimated standard deviations shown. The mean value is given, with resultant standard deviation.

Identification	p_0° values (psia)		Estimated Std. deviation
	Measured as	Value	
p_{so}	static	392	± 2
p_{to4}	total	387	± 5
Mean		391	± 2

APPENDIX B
ANALYSIS OF FISSION PRODUCT LOSS IN TEST
OPERATION OF TORY II-A*

C. F. Donaghy[†] and E. Goldberg

During the period from May 14 to October 6, 1961, a total of four high power runs were conducted on the Tory II-A reactor at the Nevada Test Site. Gamma dose rate measurements during and after the runs included those made by:

- 1) A fixed detector net at about 1.3 miles from the test pad, at ground level.
- 2) Roving teams using portable radiation detection instruments at approximate distances of 5 and 8 miles from the reactor.
- 3) Aerial monitoring and cloud tracking with an L-20 aircraft employing portable monitoring equipment.

The results of the present analysis are based largely upon the gamma dose rates measured at the fixed detector net. The location of detectors in the net is shown in Fig. B-1. Satisfactory dose rates were recorded for the first three runs and a typical trace is shown in Fig. B-2. Note that the maximum dose rate recorded was 20 mr /h. The data from the October 6 (HP-3) run was faulty.

A knowledge of the fraction of the fission fragments which escaped with the effluent air is of considerable interest. Calculations of the expected escape fraction have been performed and it would be proper to compare the predictions to the field test data. However, certain difficulties are at once apparent: accurate verification of the escape fraction cannot be expected from field measurements alone because of the uncertainties involved in the measurements themselves, not to mention the gross uncertainty associated with the interim history of the fission products which escaped. The effluent cloud was invisible, and the paucity of data available on effluent cloud shape and height as well as the distribution of fission products within the cloud can produce errors in the calculated dose rate of an order of magnitude.

*Also appears as LRL report UCRL-6874-T Summary (SRD).

[†]Cdr., USN.

Fig. B-1. Perimeter monitoring system. Location of detector units 1 through 8.

Fig. B-2. Recorded radiation dose rates at detectors 1-5 of perimeter monitor net on Sept. 28, 1961 (run HP-1).

In view of the many uncertainties involved, a simple model has been used for our analysis. Holland's equation¹⁴ is used to determine the effective cloud height. Decay time is taken from the midtime of reactor peak power to the time of peak dose rate at a detector. All of the fission fragments which escaped are assumed to have decayed for this time interval. The effluent activity is treated as a point source rather than a volume source. Using this simple model, the analysis may be broken down into three steps, which will be separately discussed.

B-1 Calculation of the Expected Fission Fragment Escape Fraction

The mechanics whereby fission fragments may escape in the effluent include erosion, diffusion, and recoil. From past experiments on erosion of BeO at elevated temperatures by air containing water vapor,¹⁵ it was evident that erosion loss of fragments should be negligible compared to recoil loss. Experiments conducted at LRL¹⁶ indicate that diffusion losses, too, may be ignored in Tory II-A in comparison with recoil loss.

The recoil ranges of a number of fission fragments in a 90% BeO-10% UO₂ mixture have been experimentally determined at this laboratory.¹⁶ The gross fission fragment range so found is 3.4 mg/cm². It follows from simple considerations that the fraction of the fission fragments recoiling through the inner surface of a fuel tube into the coolant air is 0.0015. However, not all of the fragments entering the air stream will escape in the effluent. Those which have a favorable energy and direction will cross the intervening air gap to become inbedded in other parts of the fuel tube's inner wall. The probability of stopping in the air, once a fragment has entered it, has been calculated for different geometries.¹⁷ The product of the fraction escaping the inner BeO surface and the probability of being stopped in the air gives the fraction of fission fragments which stop in the air stream. This fraction has been used to do an integration over the length of the fuel-bearing region of Tory II-A, taking into account the axial variation of power density and air density.

B-2 Estimate of the Effluent Cloud Location

The effluent was invisible and there was little field information available for accurately positioning the cloud. However, dose rates were simultaneously observed at two or more detectors during cloud passage on

each of the test runs. Use of the observed dose rates and the $e^{-r/\lambda}/r^2$ attenuation yields a relationship between the distances to the cloud from the two detectors. If we knew the cloud height we could then position the cloud accurately and the detector-to-cloud distances could be found explicitly. We have used Holland's equation¹⁴ (with slight modification) to find the height of rise of the cloud. This has been taken as an approximation of the height at the time of passing the detectors in the fixed net. (For the May 14 run, a stable layer existed 1600 feet above the terrain which kept the exhaust below this altitude. The value from Holland's equation was 465 feet. The crude aerial tracking data for all runs was consistent with values of height employed.)

B-3. Calculation of the Expected Dose Rate and Comparison with the Observed Rate

The energy absorption rate in air for a point source emitting n_i photons per second of energy $h\nu_i$ (MeV) leads to a dose rate of

$$D(\text{mr/h}) = 5.5 \times 10^{-3} \sum_i (n_i h\nu_i) \left(\frac{\mu_a}{\rho} \right)_i \frac{e^{-(\mu_a)_i r}}{r^2}$$

where

$$\begin{aligned} (\mu_a/\rho)_i &= \text{mass attenuation coefficient for air (cm}^2/\text{gm), at} \\ &\quad \text{energy } E_i, \\ r &= \text{distance from source to detector (cm).} \end{aligned}$$

Using published values of $n_i h\nu_i$ per fission from Perkins and King,¹⁸ and taking $\overline{\mu_a/\rho} \approx 0.0255 \text{ cm}^2/\text{g}$, we find an effective mean free path of about 360 meters at Nevada Test Site (where the air density is about 0.00109 g/cm^3) for the decay times and detector-to-cloud distances involved. Then the dose rate is given by

$$D(\text{mr/h}) \approx 1.4 \times 10^{-4} \frac{N}{r^2} e^{-r/\lambda} \frac{1}{N} \sum_i (n_i h\nu_i)$$

where N is the number of fission events corresponding to the number of fission fragments which escape the reactor (i.e., $1/2$ the number of fission

fragments escaping), and $\frac{1}{N} \sum_i (n_i h \nu_i)$ is the rate of energy emission per fission $\left(\frac{\text{MeV/sec}}{\text{fission}} \right)$ from the cloud at the time for which the dose rate is being calculated.

The detector-to-source distances are then used with the point source model to calculate the expected dose rate based upon the calculated fission fragment escape corrected for decay time. For each run these calculations have been made using the effective cloud height as the height of source. In some cases the calculations were repeated for source heights which seemed to represent a reasonable upper or lower limit. These results are in Table B-1.

Table B-1. Comparison of observed and calculated peak dose rates

Date	Source height, h (ft)	Dose rate, D (mr./h)
May 14	465 ^a	0.7 obs 1.8 calc
Sept. 28	705 ^b	1.3 obs 1.0 calc
Oct. 5	1875 ^b	2.7 obs 0.7 calc

^a Estimated upper limit of h, 1000 ft.

^b Lower limit of h, 0 ft.

It is estimated that the error introduced by our simplified point source model is no more than a factor of two or three. Combining this with the errors due to uncertainty in cloud height, etc., one finds that the stated values of calculated dose rate are uncertain within a factor of ten. It is gratifying to note that the observed dose rates differ only within a factor of three from the calculated rates.

If the effluent cloud were more accurately positioned (therefore warranting use of a more accurate model) it is believed that the escape fraction could be verified within a factor of three. Current plans for the Tory II-C runs provide for a detector net with three times as many detectors (i. e., at 10° intervals on the 1.3-mile arc).

A fission fragment escape fraction of 0.001 is submitted as an average for the Tory II-A runs.

APPENDIX C
DATA SYSTEM
T. C. Varljen

C-1 Nuclear Instrumentation

C-1.1 General Description

The nuclear instrumentation system was designed to provide a useful measure of reactor neutron level over the entire operating range of the reactor, from shutdown to several times design point power. The detector coverage was divided into four ranges:

A. Startup Range (S1, S2). Startup instrumentation consisted of a pair of N. Wood type G20-12 BF_3 chambers mounted on a trolley running parallel to the axis of the reactor. These were initially positioned next to the reflectors during any test run, and later withdrawn remotely into the bunker. The neutron level was generally raised to give at least 10^3 counts per second on these chambers in the startup position. The chambers were then withdrawn to get a one-decade reduction in count rate, and neutron level was again increased to bring the count-rate of the "S" counters to 10^3 counts per second. At this point the next set of detectors was sufficiently on scale to transfer control to them.

B. Intermediate Range (I1, I2). Detectors in this range consisted of a pair of N. Wood type G20-12 BF_3 chambers positioned in a window in the test bunker disconnect room immediately behind the final turn in the reactor inlet air ducting. Upon transfer of control to these chambers, the start-up scrams were bypassed and then the "S" chambers were run completely into the bunker. Neutron level was again raised to get a count-rate of about 10^3 counts per second on the "I" chambers.

C. Low Power Range. In this range a pair of electrically compensated ion chambers were used. Channel C1 consisted of an Anton Type 807 CIC used with a linear micromicroammeter. This chamber was positioned directly beneath the "I" chambers inside the test bunker. The output of C1 was assumed to vary linearly with power level. This had to be qualified at

high airflow rates as will be noted below. The other sensor in this range was designated P4. This chamber was placed in a pipe running underneath the test vehicle between the railroad rails (directly behind chamber L2). This chamber was used with a Log-N amplifier. For the Intermediate Power Test an Anton Type 807 chamber was used. It was replaced by an equivalent Westinghouse Type 6377 for High Power Tests 1, 2, and 3. When control was transferred to these chambers, C1 generally read over 10^{-12} amp and P4 over 5×10^{-9} amp. The term "transfer of control" as used here refers to transfer of scram capability and operator attention; automatic servo control was not used in this range.

D. High Power Range. As many as six sensors were used in this range. The most important of these were designated P1, P2 and P3 and served primarily as feedback transducers in the automatic log power control system. These detectors were electrically-compensated ion chambers placed beneath the "I" chambers and C1 in the bunker window. For the Intermediate Power Test Anton Type 807 chambers were used but were later replaced by Westinghouse Type 6377 chambers. The outputs of P1, P2 and P3 went to logarithmic amplifiers.

Three fission chambers (L1, L2, and L3) were positioned beneath the reactor, in three pipes running between the railroad rails, to serve as linear power indicators. These chambers were designed to be used in the automatic linear power control loop, but log power control proved to be sufficient for all needs so linear power control was never used. Automatic log power control was quite safe and effective and all high power tests were conducted in this control mode. Since there was no way, a priori, to know what current the linear chambers would read at a given position, they were staggered in position beneath the test vehicle. Once the sensitivity of these detectors was established with respect to reactor power level they were placed in positions such that practically continuous linear level indication was provided in the high power region. The chambers used were Westinghouse Type 6941, dual-range, U^{235} -lined ionization chambers.

C-1.2 Operation of the System

The Intermediate Power Test gave the first good correlation of nuclear detector current with thermodynamic reactor power level measurements (up

operating power level as originally planned for run HP-1. Once the chamber response at about 150 MW with high airflow rate was established, more realistic level scram points could be set on the final two runs.

The fission chambers L1, L2, and L3 beneath the reactor appeared to give good data in their range of operation. These detectors were monitored continuously on the nuclear operator's console.

C-1.3 Scram System

Any combination of the detector-initiated scrams and resets could be bypassed, in any operating range, by the Nuclear Operator with the concurrence of the Test Conductor. Bypassing and unbypassing maneuvers were carefully checklisted to insure adequate scram coverage at all times.

It was never necessary to use an auxiliary neutron source during startup, even for the initial crits. The natural activity of the core (initially mainly due to U^{235} α 's and later supplemented by fission product decay γ 's interacting with beryllium) provided sufficient startup neutrons. For the early crits the background levels were sufficient to produce between 10 and 20 counts per second on the startup detectors, with the reflectors full-in and control elements in the scram condition.

In the first three operating ranges, exceeding any single scram set-point on any unbypassed detector channel was sufficient to initiate a scram. The scrams available were:

Startup Range (Channels S1 and S2):

S1 level (meter)	S1 level (recorder)
S2 level (meter)	S2 level (recorder)
S2 period (meter)	S2 period (recorder)

Intermediate Range (Channels I1 and I2):

I1 level (meter)	I1 or I2 level (recorder)
I1 period (meter)	I1 or I2 period (recorder)
I2 level (meter)	I2 period (meter)

Low Power Range (Channels C1 and P4):

C1 level (recorder)	
P4 level (meter)	P4 period (meter)

In the high power range (channels P1, P2, P3, L1, L2 and L3), two-out-of-three coincidence was necessary on any set of level or period meters.

P1, P2, and P3 level (3 meters)
P1, P2, and P3 period (3 meters)
L1, L2, and L3 level (3 meters)

C-1.4 Reactor Control

Low Power Region (power < 100 kW). In the Startup, Intermediate and Low Power ranges, reactor control was accomplished by manually programming control vane or rod servo position demand. Generally three of the control rods were positioned full-out as safeties, and one, selected as a vernier, was placed at mid-stroke.

Initial criticality during a run was achieved by moving the vanes. Once the reactor was critical, fine control could be accomplished with the vernier rod, with occasional small movements of the vanes.

High Power Region (power > 100 kW). When the reactor power level was high enough to produce a quiet current, on the order of 10^{-8} amp in the Log-N channels, control was transferred to the automatic log power mode. The reactor was put into automatic control prior to the start of significant core heat-up or the introduction of air through the core. Thus if a constant power level was demanded the vanes would automatically compensate for temperature, air poisoning, or other reactivity effects.

In this mode, power was increased by demanding an increase in ion chamber current. The rate of rise was pre-selected by the nuclear operator. Periods of 5 or 10 seconds were generally chosen: 5 seconds when large changes in power were required early in the run program, and 10 seconds at or near full power.

In the high power region, chambers P1, P2, and P3 provided two-out-of-three coincidence scram capability on Log-N level and period. Both period and level resets were used on the Intermediate Power Test but the level resets were bypassed for the three high power tests. In the event of trouble, it was felt, it would be preferable to scram rather than to be held at some elevated power by a level reset.

C-2 Air Process System

C-2.1 Flowmetering System

Two venturi meters were available (see Figs. 6-1 and 6-2). FM-1 was a 14-in. meter and FM-2 a 5.5-in. meter. Experience showed that FM-1 was reliable at flow rates above 250 pps and FM-2 reliable from 15 pps to 450 pps. Accurate data was more easily obtained from FM-2, since it operated in a choked condition. When the inlet diffuser operated under choked-flow conditions, air flow rates could easily be calculated at its station, as discussed in Section 4.2.1 and Appendix A.

C-2.2 Flow Control

The flow rate of air to the reactor was controlled by air valves AV-3 and AV-4. These are electro-hydraulically actuated, plug-type valves.

The choice of control valve used for a particular run depended on several factors, the primary considerations being the test duration and maximum flow rate desired.

AV-3, the larger valve, had been operated in facility shakedowns up to a flow rate of 1200 pps. Control modes available were automatic, position servo and manual override. In the automatic control mode the servo loop was closed on pressure measured upstream from the diffuser. A function generator could be used to supply air pressure demand as a function of time. In the position servo mode, control of the air flow rate was accomplished by setting the position demand of the valve (utilizing position feedback), while the operator monitored a pressure signal. Valve position was then adjusted to attain a pressure reading equivalent to the required flow rate. Manual Override Mode was open loop operation of the valve, suitable only for emergency shutdown.

The system automatically reverted to the override mode if an abnormal error signal was detected in the valve servo-control loop or whenever the pressure or rate of change of pressure at the diffuser inlet exceeded selected values. When the system was placed in override because of an accident condition or operator choice, the existing valve position was held or could be opened or closed at fixed rates.

AV-4 and FM-2 were in a 6-in. supply line and AV-3 and FM-1 in the main 24-in. line. The upper limit of AV-4 operation was about 400 to 500 pps,

attainable only at high air storage pressure. Control modes available for AV-4 were position servo and manual override.

AV-4 thus could not be used for a run which required design inlet air-flow conditions. However, a great many blowdowns and qualification runs were done at low flow rates, and required relatively accurate flow rate measurements, so that AV-4 was used frequently.

C-2.3 Air Temperature Control

The temperature of the air reaching the reactor was controlled by two electro-hydraulically actuated butterfly valves designated as AV-5A and AV-5B. These valves controlled the proportions of air which were allowed to pass through and around the heater. Cold and hot air was mixed in the plenum at the top of the heater and went from there to the test vehicle. The air temperature operator adjusted valve position while watching total reactor inlet air temperature (T_0°) indicators. During a run, as the energy stored in the heat mass was depleted, the operator sent an increasingly larger percentage of air through the heat mass in order to hold a constant T_0° value.

C-2.4 Inlet Air Pressure and Temperature Instrumentation

The placement of the various transducers involved in pressure and temperature measurements are shown in Figs. C-2 and C-3. Channels for measurement of p_0° designated as P_{TO1} , P_{TO2} , and P_{TO3} were primarily used as airflow control feedback transducers and were not recorded, to avoid possible interference with the control system. They were displayed on meters in front of the Air Flow Operator. The level and rate were limited by setpoints on these meters. P_{TO5} , P_{TO6} , and P_{TO7} were 0-150 psi transducers designed primarily to give information at low flow rates. They saturated during the high power tests since the pressure value generally exceeded their upper limits.

Three ports in the duct wall opened into a manifold for measurement of the average static pressure at the same station. This information went to a recorder.

Six thermocouples were utilized to obtain total inlet air temperature. The six probes projected about 2 in. into the airstream. The outputs of three chromel/constantan thermocouples went to a meter which the Air Temperature Operator watched. Two of the chromel/alumel thermocouple

Subscript s denotes static
T denotes total

Fig. C-2 Diagram of duct pressure measurement locations. General regions of the primary pressure and temperature values used for power and flow rate determination (Section 4.2.1) are indicated in boxes. Other references are to individual data channels.

Total inlet air pressure

Total inlet air temperature

GLL-638-2012

Fig. C-3 Location of pressure and temperature probes in duct upstream from inlet diffuser, for determination of p_0° and T_0° .

signals were paralleled and sent to a Brown recorder. The third chromel/alumel thermocouple output was recorded on both OFFNER (strip-chart) and ASCOP systems. A large discrepancy was noted between T_0° as recorded on the Brown recorder and as recorded on ASCOP and OFFNER for high power tests 2 and 3. The Brown data compared very well with other data, so that the ASCOP and OFFNER results are suspect. Physically the thermocouple in question did not appear damaged, and the proper calibration and reference baths were used. The problem seemed to be an interaction which occurred when the same signal was sent to two recording systems.

C-2.5 Duct Pressure Instrumentation

A total of 27 pressure measurements were made at various points in the test vehicle air ducting. Figure C-2 shows approximate axial locations of the various measurements. During early blowdowns the transducers were mounted directly on the ducting in an attempt to keep the air supply lines from duct to transducer as short as possible. However, much concern was felt over the possible effects of vibration and acoustical noise during high power operation, and as a result the transducers were remounted on the deck of the test vehicle on one of two panels whose approximate location is shown in Fig. C-2. The supply lines (1/4-in. o.d. aluminum tubing) were on the order of 12 feet long. The compressibility of the air in the tubing and the flow impedances introduced by the tubing and connectors caused pressure changes at the transducer diaphragm to lag pressure changes in the duct, but these transducers were chiefly intended to give steady-state pressure level indication and the loss in response time was considered less important than reduction of noise and vibration effects.

Two Bourdon gauges were set up next to the test vehicle to measure static pressure forward and aft of the core, connecting into the duct in the locations shown in Fig. C-2. These were monitored during the run by television. They provided pressure data for immediate post-run analyses and would have given last-ditch pressure information in the event of a telemetering system failure.

The duct pressure-temperature rake assemblies (PT2 and PT4) are shown in Fig. C-4, with the location of the various probes just upstream and downstream from the reactor. Unfortunately, several of the temperature channels failed before the high power runs.

Rakes upstream from reactor (p_1°)

Sp = spare

Rakes downstream from reactor (p_2°, T_2°)

Fig. C-4. Diagram of pressure-temperature rakes near the reactor inlet and exit faces.

C-3 Core Temperature Instrumentation

C-3.1 Basic System

Internal core material temperatures were measured by Pt/Pt-10% Rh thermocouples whose junctions were embedded in fuel elements or structural links. Fuel element thermocouple leads were carried into the core through columns of unfueled elements geometrically identical with the fuel elements. After leaving these columns by means of drilled wall holes, the leads terminated with their junctions cemented into holes in the ends of elements whose temperatures were to be measured. Leads for structural link thermocouples came into the core through the triangular gaps formed where fuel elements were adjacent to links. Figure C-5 shows a fuel element thermocouple after disassembly, with its junction still cemented in place. The last unfueled element of the lead-bearing column remains on the lead. Figure C-6 shows the core during assembly, with thermocouple leads emerging from the inlet end.

Core thermocouple signals, together with signals from other reactor thermocouples and from some miscellaneous pressure transducers, were transmitted to the Control Room by a multiplexing system (ASCOP), whose data display included a cathode-ray-tube continuous bar graph presentation of all data channels as well as the option of strip-chart recording or meter display of selected signals. In addition, all data coming through this system was recorded on magnetic tape, and remains accessible for later translation to strip-chart records, cathode-ray tube display, or digital form for computer treatment. Some temperature signals from the ASCOP system were displayed on meters at the Nuclear Operator's console.

C-3.2 Core Thermocouple Averaging System

During reactor operation at high power, the parameter of principal concern to the Nuclear Operator was core temperature, and it was very important that he should have a reliable, easily seen indication of this quantity. For the Intermediate Power Test the signals from six core thermocouples were displayed on individual meters at the nuclear console. One signal was also displayed on a strip-chart recorder, but this was too far removed from the operator's station to be of much use.

Fig. C-5. Fuel element thermocouple removed in reactor disassembly. The lead passes through the wall of an unfueled element into the end of fuel element K_a-158J.

Fig. C-6. Reactor core near completion of assembly, showing thermocouple leads emerging from inlet end.

The meter arrangement proved unsatisfactory because of the Nuclear Operator's difficulty in assimilating information from all six meters at once. This gave rise to a requirement for a system that would compute the average signal from six core thermocouples and display it on a recorder directly in front of the operator, with the capability of switching questionable signals out of the average-computing network. Individual temperature signals would also be displayed on meters at the nuclear console.

Such a system was built and successfully operated during the three high power tests. The ASCOP system served as a signal source, and provided a duplicate meter display which could be directly compared with the cathode ray tube display of all core temperatures by the Core Data Observers. It was their responsibility to detect and switch out questionable channels during a run.

In addition to the six core temperatures, two exit gas temperatures were also displayed on the Nuclear Operator's meters as a back-up in case of complete failure of all core thermocouples.

The average-temperature recording made during High Power Test 2 is reproduced in Fig. 4-4. During this test, three of the six thermocouples in the averaging network failed and had to be switched out. Note that the temperature trace maintained reasonable continuity throughout.

SECRET

-222-

REFERENCES

1. J. W. Hadley, "Tory II-A: A Nuclear Ramjet Test Reactor," UCRL-5484 (1959).
2. Pluto Quarterly Report No. 7, p. 13, UCRL-6376 (1961).
3. Pluto Quarterly Report No. 8, p. 11, UCRL-6516 (1961).
4. A. H. Shapiro, "The Dynamics and Thermodynamics of Compressible Fluid Flow," p. 85, Ronald Press (1953).
5. Reynolds Electric and Engineering Co., "Tory II-A Operation, On-Site Rad-Safe Report " (May 1961 and May 1962).
6. U. S. Public Health Service, Las Vegas, Nevada, "Tory II-A Report, Off-Site Radiological Safety " (Feb. 1962).
7. Data compiled at LRL by A. Lorenz and S. Kellman from various published sources, distributed internally only.
8. Pluto Quarterly Report No. 11, p.6, UCRL-6890 (1962).
9. E. Y. Robinson and W. M. Wells (Tentative title "Cracking in Tory IIA Fuel Tubes") to be published as UCRL-7369.
10. R. B. Meuser, "Temperatures and Thermal Stresses in Hexagonal Tubes and Pierced Plates with Internal Heat Sources," UCRL-5692 (1959).
11. W. B. Myers, W. M. Wells, and C. E. Walter, "On the Thermal Stress Behavior of Pluto Fuel Elements," UCRL-6877 (1962).
12. W. M. Wells and C. F. Cline, "Thermal Stress Tests of BeO and BeO- UO_2 ," UCRL-6852 (1962).
13. Pluto Quarterly Report No. 13, p. 150, UCRL-7079 (1962).
14. U. S. Weather Bureau/U. S. Atomic Energy Commission, "Meteorology and Atomic Energy," Government Printing Office, 1955. (AECU-3066).
15. W. A. Young, "Reactions of Water Vapor with BeO," NAA-SR-4446 (1960).
16. H. G. Hicks and J. C. Armstrong, Jr., "Escape of Fission Products from UO_2 -Loaded BeO Fuel Elements," UCRL-6148 (1961, SRD).
17. Pluto Quarterly Report No. 9, p. 5, UCRL-6625 (1961).
18. J. F. Perkins and R. W. King, "Energy Release from the Decay of Fission Products," Nuclear Sci. and Eng. 3, 726 (1958).

/rd

SECRET RESTRICTED DATA

— LEGAL NOTICE —

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission " includes any employee or contractor of the commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.