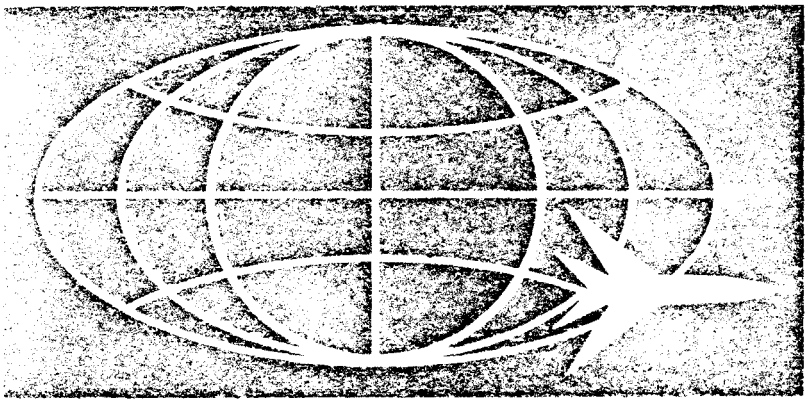
APEX-905

COMPREHENSIVE TECHNICAL REPORT

AIRCRAFT NUCLEAR PROPULSION PROGRAM

APEX-905



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HEAT TRANSFER REACTOR EXPERIMENT NO. 2

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This is one of twenty-one volumes summarizing the General Electric Company's direct-air-cycle aircraft nuclear propulsion program. Additional copies are available from the United States Atomic Energy Commission, Division of Technical Information Extension, Oak Ridge, Tennessee.

The APEX number and title of each volume in the series is shown in the following list.

APEX-901	Program Summary and References
APEX-902	P-1 Nuclear Turbojet
APEX-903	Reactor Core Test Facility
APEX-904	Heat Transfer Reactor Experiment No. 1
APEX-905	Heat Transfer Reactor Experiment No. 2
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APEX-905

COMPREHENSIVE TECHNICAL REPORT GENERAL ELECTRIC DIRECT-AIR-CYCLE AIRCRAFT NUCLEAR PROPULSION PROGRAM

HEAT TRANSFER REACTOR EXPERIMENT NO. 2

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Editor: R. R. GEHRING

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NUCLEAR MATERIALS AND PROPULSION OPERATION
(Formerly Aircraft Nuclear Propulsion Department)
FLIGHT PROPULSION LABORATORY DEPARTMENT
Cincinnati 15, Ohio

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ABSTRACT

This is one of twenty-one volumes summarizing the Aircraft Nuclear Propulsion Program of the General Electric Company. This volume describes Heat Transfer Experiment No. 2, a test reactor used to evaluate ANP fuel elements and solid moderator materials.

The reactor, a modification of Heat Transfer Reactor Experiment No. 1, had the seven center cells of the core removed, providing a hexagonal hole for test inserts.

The reactor and the inserts tested are described and the results of the various tests presented.

PREFACE

In mid-1951, the General Electric Company, under contract to the United States Atomic Energy Commission and the United States Air Force, undertook the early development of a militarily useful nuclear propulsion system for aircraft of unlimited range. This research and development challenge to meet the stringent requirements of aircraft applications was unique. New reactor and power-plant designs, new materials, and new fabrication and testing techniques were required in fields of technology that were, and still are, advancing very rapidly. The scope of the program encompassed simultaneous advancement in reactor, shield, controls, turbomachinery, remote handling, and related nuclear and high-temperature technologies.

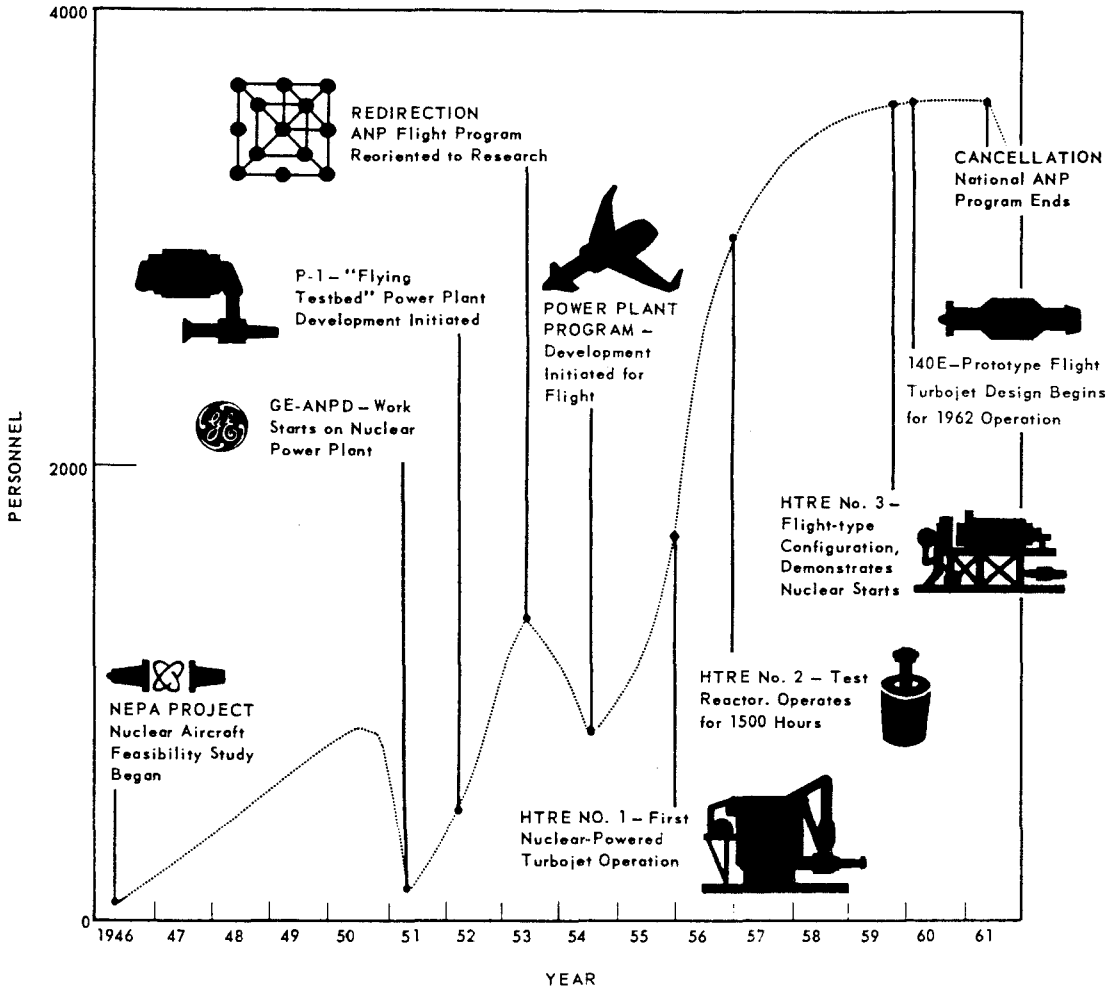
The power-plant design concept selected for development by the General Electric Company was the direct air cycle turbojet. Air is the only working fluid in this type of system. The reactor receives air from the jet engine compressor, heats it directly, and delivers it to the turbine. The high-temperature air then generates the forward thrust as it exhausts through the engine nozzle. The direct air cycle concept was selected on the basis of studies indicating that it would provide a relatively simple, dependable, and serviceable power plant with high-performance potential.

The decision to proceed with the nuclear-powered-flight program was based on the 1951 recommendations of the NEPA (Nuclear Energy for the Propulsion of Aircraft) project. Conducted by the Fairchild Engine and Airplane Corporation under contract to the USAF, the five-year NEPA project was a study and research effort culminating in the proposal for active development of nuclear propulsion for manned aircraft.

In the ensuing ten years, General Electric's Aircraft Nuclear Propulsion Department carried on the direct air cycle development until notification by the USAF and USAEC, early in 1961, of the cancellation of the national ANP program. The principal results of the ten-year effort are described in this and other volumes listed inside the front cover of the Comprehensive Technical Report of the General Electric Direct Air Cycle·Aircraft Nuclear Propulsion Program.

Although the GE-ANPD effort was devoted primarily to achieving nuclear aircraft power-plant objectives (described mainly in APEX-902 through APEX-909), substantial contributions were made to all aspects of gas-cooled reactor technology and other promising nuclear propulsion systems (described mainly in APEX-910 through APEX-921). The Program Summary (APEX-901) presents a detailed description of the historical, programmatic, and technical background of the ten years covered by the program. A graphic summary of these events is shown on the next page.

Each portion of the Comprehensive Report, through extensive annotation and referencing of a large body of technical information, now makes accessible significant technical data, analyses, and descriptions generated by GE-ANPD. The references are grouped by subject and the complete reference list is contained in the Program Summary, APEX-901. This listing should facilitate rapid access by a researcher to specific interest areas or



Summary of events – General Electric Aircraft Nuclear Propulsion Program*

*Detailed history and chronology is provided in Program Summary, APEX-901. Chronology information extracted from: Aircraft Nuclear Propulsion Program hearing before the Subcommittee on Research and Development of the Joint Committee on Atomic Energy, 86th Congress of The United States, First Session, July 23, 1959, United States Government Printing Office, Washington 1959.

sources of data. Each portion of the Comprehensive Report discusses an aspect of the Program not covered in other portions. Therefore, details of power plants can be found in the power-plant volumes and details of the technologies used in the power plants can be found in the other volumes. The referenced documents and reports, as well as other GE-ANPD technical information not covered by the Comprehensive Report, are available through the United States Atomic Energy Commission, Division of Technical Information Extension, Oak Ridge, Tennessee.

The Report is directed to Engineering Management and assumes that the reader is generally familiar with basic reactor and turbojet engine principles; has a technical understanding of the related disciplines and technologies necessary for their development and design; and, particularly in APEX-910 through APEX-921, has an understanding of the related computer and computative techniques.

The achievements of General Electric's Aircraft Nuclear Propulsion Program were the result of the efforts of many officers, managers, scientists, technicians, and administrative personnel in both government and industry. Most of them must remain anonymous, but particular mention should be made of Generals Donald J. Keirn and Irving L. Branch of the Joint USAF-USAEC Aircraft Nuclear Propulsion Office (ANPO) and their staffs; Messrs. Edmund M. Velten, Harry H. Gorman, and John L. Wilson of the USAF-USAEC Operations Office and their staffs; and Messrs. D. Roy Shoults, Samuel J. Levine, and David F. Shaw, GE-ANPD Managers and their staffs.

This Comprehensive Technical Report represents the efforts of the USAEC, USAF, and GE-ANPD managers, writers, authors, reviewers, and editors working within the Nuclear Materials and Propulsion Operation (formerly the Aircraft Nuclear Propulsion Department). The local representatives of the AEC-USAF team, the Lockland Aircraft Reactors Operations Office (LAROO), gave valuable guidance during manuscript preparation, and special appreciation is accorded J. L. Wilson, Manager, LAROO, and members of his staff. In addition to the authors listed in each volume, some of those in the General Electric Company who made significant contributions were: W. H. Long, Manager, Nuclear Materials and Propulsion Operation; V. P. Calkins, E. B. Delson, J. P. Kearns, M. C. Leverett, L. Lomen, H. F. Matthiesen, J. D. Selby, and G. Thornton, managers and reviewers; and C. L. Chase, D. W. Patrick, and J. W. Stephenson and their editorial, art, and production staffs. Their time and energy are gratefully acknowledged.

THE EDITORIAL BOARD:

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November 8, 1961

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TABLES

1. INTRODUCTION AND SUMMARY

The development of advanced reactor types for aircraft nuclear propulsion required an extensive in-pile test program of various improved fuel elements and moderators. Many of these tests could not be accommodated in the MTR or other conventional reactor installations because of test specimen geometry, or because of the nature of the test operation. For this reason, the basic HTRE No. 1 reactor design described in APEX-904 was modified to accommodate special test specimens of more advanced fuel element and moderator assemblies. The modified design was designated the HTRE No. 2 Reactor.

The HTRE No. 2 "parent core" was similar to the HTRE No. 1 core, except that the central seven air tubes were removed and replaced by a hexagonal void 11 inches across flats. A corresponding opening was made in the top shield plug so that sections of advanced reactors could be inserted into the HTRE No. 2 parent core without requiring removal of the core from the shield. The inserts were suspended from a small diameter shield plug, which filled the opening in the main shield plug. No special cooling air circuit was provided for the insert. The air was drawn from the common plenum chamber above the reactor.

Since it was expected that some of the inserts would contribute less to reactivity than the seven fuel cartridges which had been removed, an additional four inches of beryllium side reflector was added to the parent core to maintain an acceptable excess reactivity balance.

HTRE No. 2 was used principally for the testing of BeO ceramic fuel cartridges of the type planned for the XNJ140E-1, although some tests were performed using the metallic cartridges and hydrided zirconium moderator of the type used in HTRE No. 3.

The first test was conducted in HTRE No. 2 in July 1957. In the ensuing four years, a total of 16,628 megawatt hours were generated during 1,353 on-test hours.

This volume describes the reactor modifications; presents a brief description of each of the inserts including the L2 Parent Insert, which was used in nine of the thirteen tests conducted; and summarizes the results of the various tests.

2. GENERAL DESCRIPTION OF THE HTRE NO. 2

The HTRE No. 2 is a modification of HTRE No. 1 (described in APEX-903) and is used to test various moderator and fuel element configurations, designated "inserts."[1*] The principal modification consisted of removal of the center seven tubes to provide a hexagonal test hole, 11 inches across flats; and replacement of the 4-inch-thick beryllium reflector with an 8-inch-thick beryllium reflector. This portion of the reactor is designated "parent core." The supporting structure and shield is designated "parent plug." The inserts are lowered into and removed from the reactor through a hole in the parent plug. Figure 2.1 shows the arrangement of the parent core, the parent plug, and an insert.

The parent core and the inserts were mounted in the Core Test Facility (CTF) (described in APEX-903) where tests at substantial reactor powers (8 to 15 megawatts) were conducted. Figures 2.2 and 2.3 are a photograph and a schematic, respectively, of the CTF.

The HTRE No. 2 parent core utilizes a water moderator, and air cooled metallic fuel elements. The cooling air was produced by a modified J47 turbojet engine, operating on part nuclear and part chemical power. (The turbomachinery is described in APEX-903.)

The inserts, designed specifically for the purpose of obtaining experimental data, when combined with the parent core, formed a critical assembly with sufficient excess multiplication to allow a test program to be carried out. The criticality and excess multiplication were controlled by the insertion and removal of 29 boron control rods in the parent core. A 50- to 100-curie polonium-beryllium neutron source carried by a steel rod was used for initial startup. The control rods, of the shim-scram type, permitted quick shutdown of the reactor.

2.1 DESCRIPTION OF THE PARENT CORE

The parent core and parent plug of the HTRE No. 2 are essentially of the same construction and shape as the parent core and parent plug of the HTRE No. 1. The center seven fuel tubes of the HTRE No. 1 37-tube active core were removed to form the HTRE No. 2 test hole. The HTRE No. 2 30-tube bank is in a hexagonal array with radially varying tube spacings. The active portion of the array forms a regular hexagonal cylinder, 29.758 inches across flats, 34.361 inches across corners, and 29.125 inches long. The tubes extend 12.94 inches beyond each end of the active core. The tube bank is contained by tube sheets at either end, a 3/4-inch-thick aluminum wall forming a hexagonal hole 11.187 inches across flats at the center, and a cylindrical shell 3/8-inch-thick, 59 inches in diameter, and 55 inches long on the outside. (See Figures 2.4 and 2.5.) The core tank also contains the 8-inch-thick beryllium reflector, spaced approximately 1/8-inch from the outside tubes and supported by brackets which are welded to the core tube sheets. Thirty control-rod guide tubes are welded into the top tube sheet and are located by small spacer plugs which are welded to the top face of the bottom tube sheet. These tubes guide the travel of the control rod and provide passage for moderator water. The water flow in

*Superscripts refer to the reference lists that appear at the end of each section.

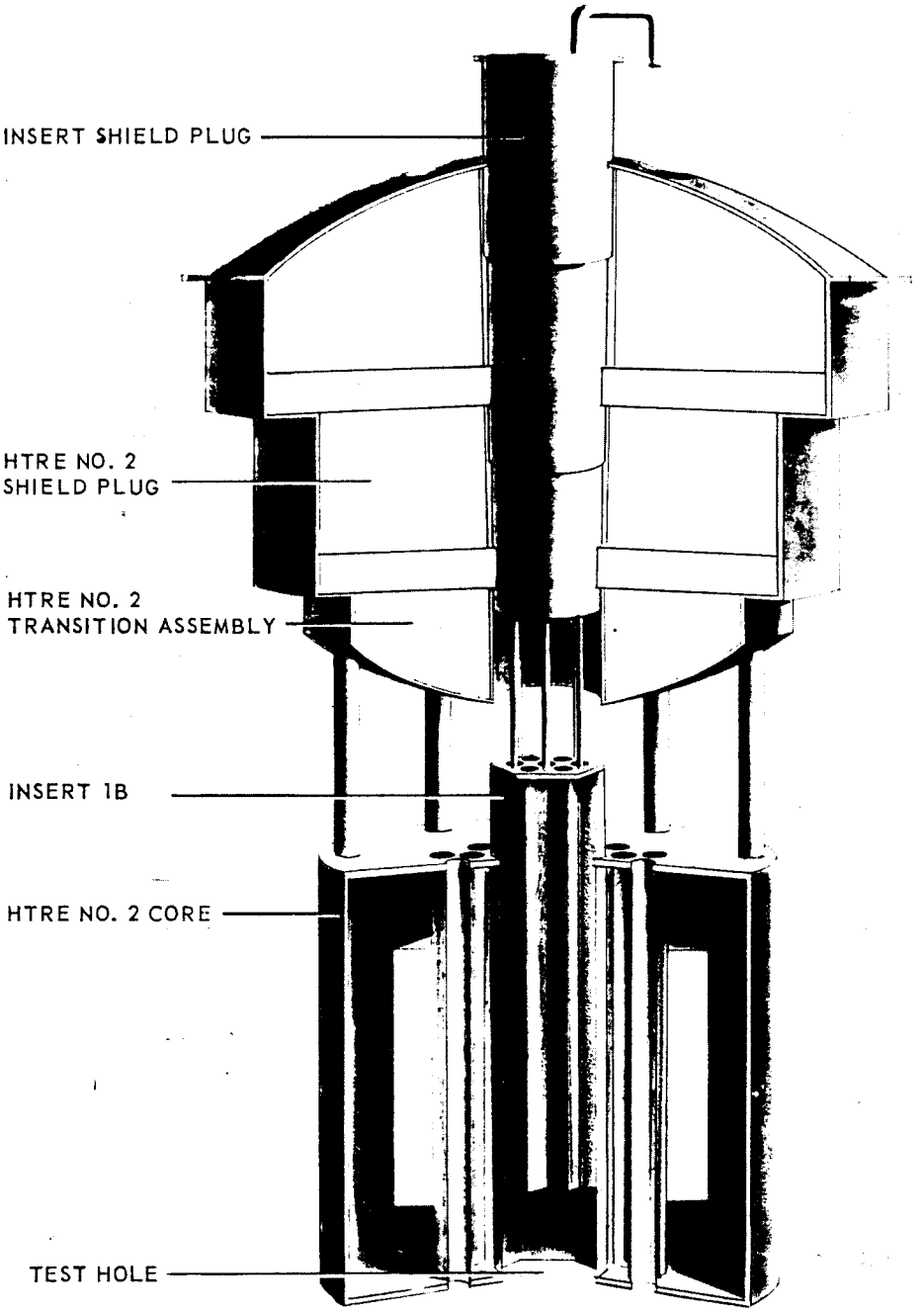

Fig. 2.1–Artist's conception of HTRE No. 2 parent core, shield plug and insert 1B (Dwg. 040-513)

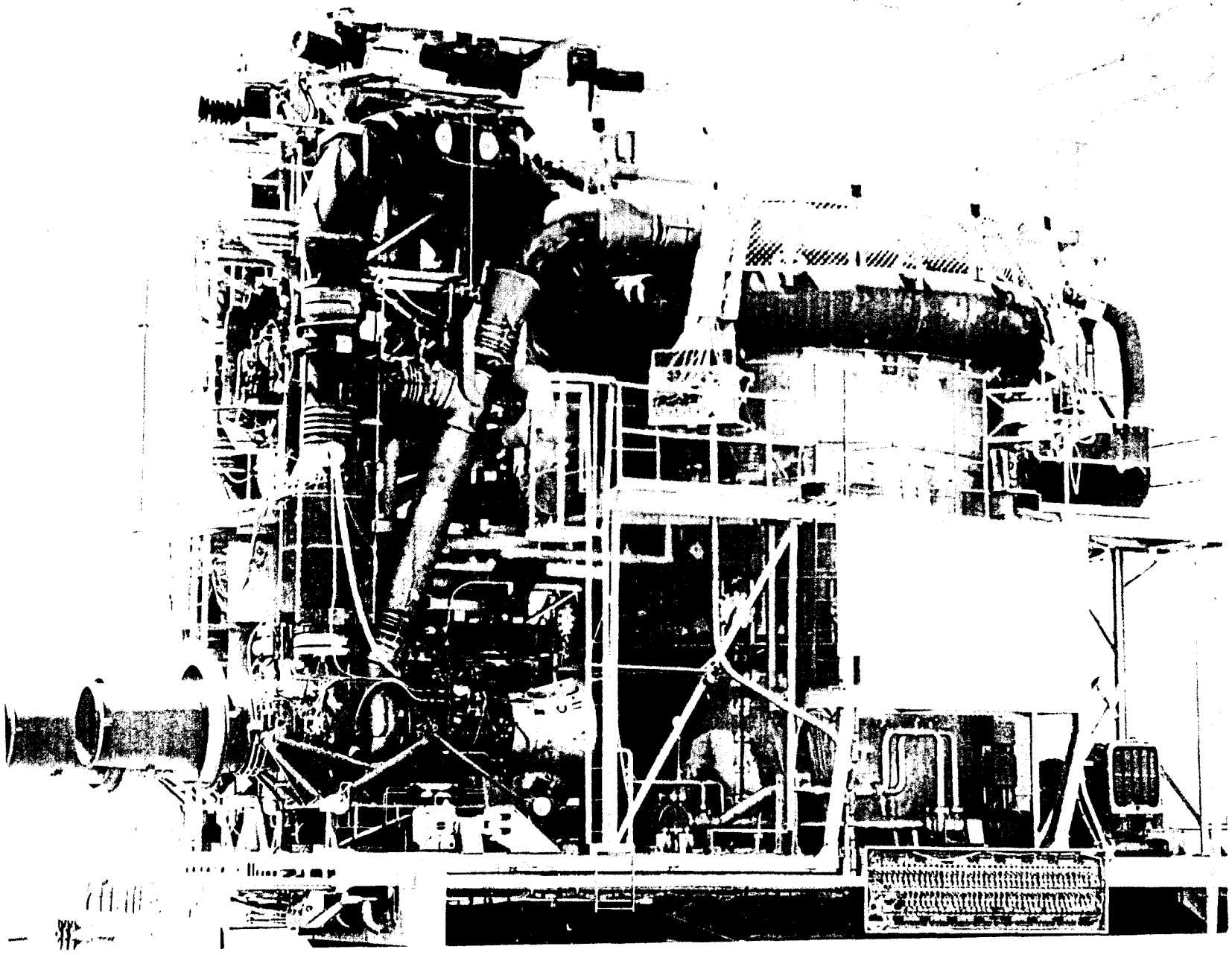

Fig. 2.2—Core Test Facility (CTF)

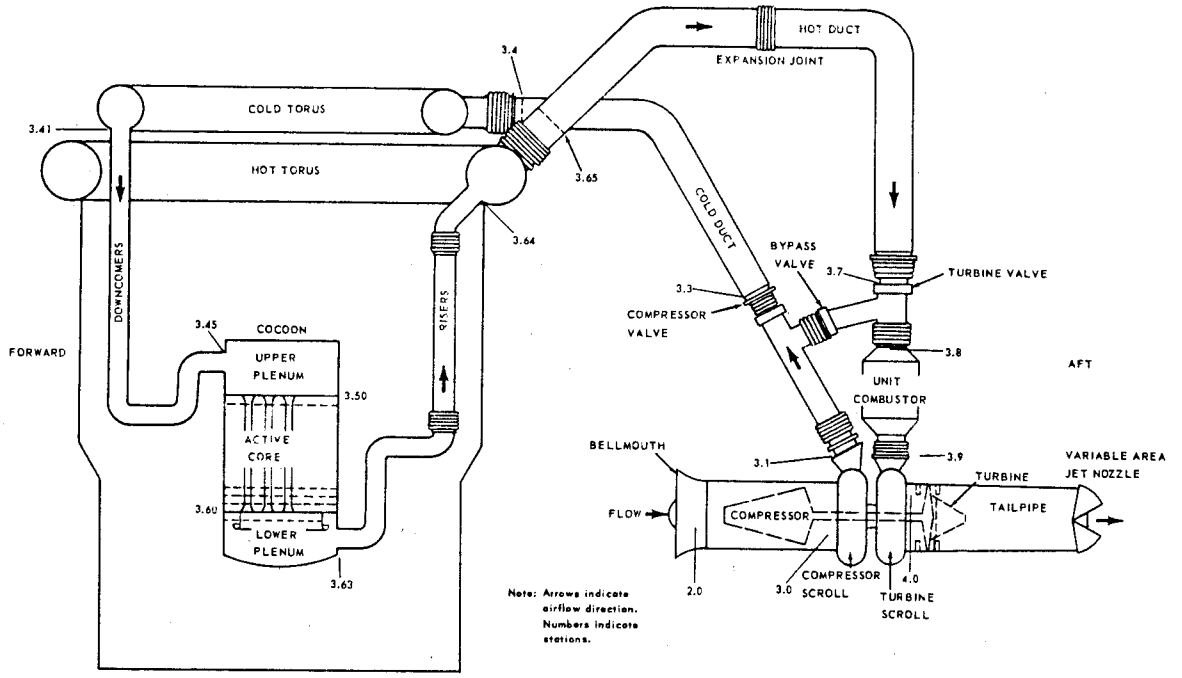

Fig. 2.3 – Core Test Facility (CTF)

the core is a two-pass system and serves as a moderator for neutrons and as a coolant for tubes and reflector. The water flows down the control rod guide tubes and splits, part going up along the fuel tubes and then to the inside of the beryllium reflector, and part going to cool the outside of the reflector. The split is 80 percent to the core and 20 percent to the reflector and is accomplished through the use of baffles. Figure 2.6 is a schematic of the water system.

2.1.1 SHIELD PLUG

The shield plug provides a barrier of dense materials for shielding the area above the active core from nuclear radiation, and provides support for the reactor core. The HTRE No. 2 plug has an opening at the center through which the inserts to be tested are inserted and removed.

The plug is constructed of stainless steel with passages for cooling water. A transition section, at the bottom of the plug, is filled with water and serves as a water outlet manifold. The parent plug also supports the parent core by means of the control rod guide tubes and core water exit tubes. The space between the core and plug form the air inlet plenum for the primary fuel element coolant. (See Figure 2.5.)

2.1.2 FUEL CARTRIDGES

HTRE No. 2 parent core fuel cartridges are identical in design to those constructed for use in the HTRE No. 1. A typical fuel cartridge is shown in Figure 2.7. Insulation liners house the cartridge and provide a thermal barrier to the moderator water. (See Figure 2.8.) The total weight of the fuel cartridge is 18.7 pounds. The cartridge is made up of 18 fuel element stages, a forward ring assembly, and an aft assembly. Of the 18 fuel stages, the first 11 are composed of 14 concentric rings, and stages 12 through 18 are composed of 16 concentric rings. Figures 2.9 and 2.10 show the front and rear view of a typical 16 concentric-ring stage.

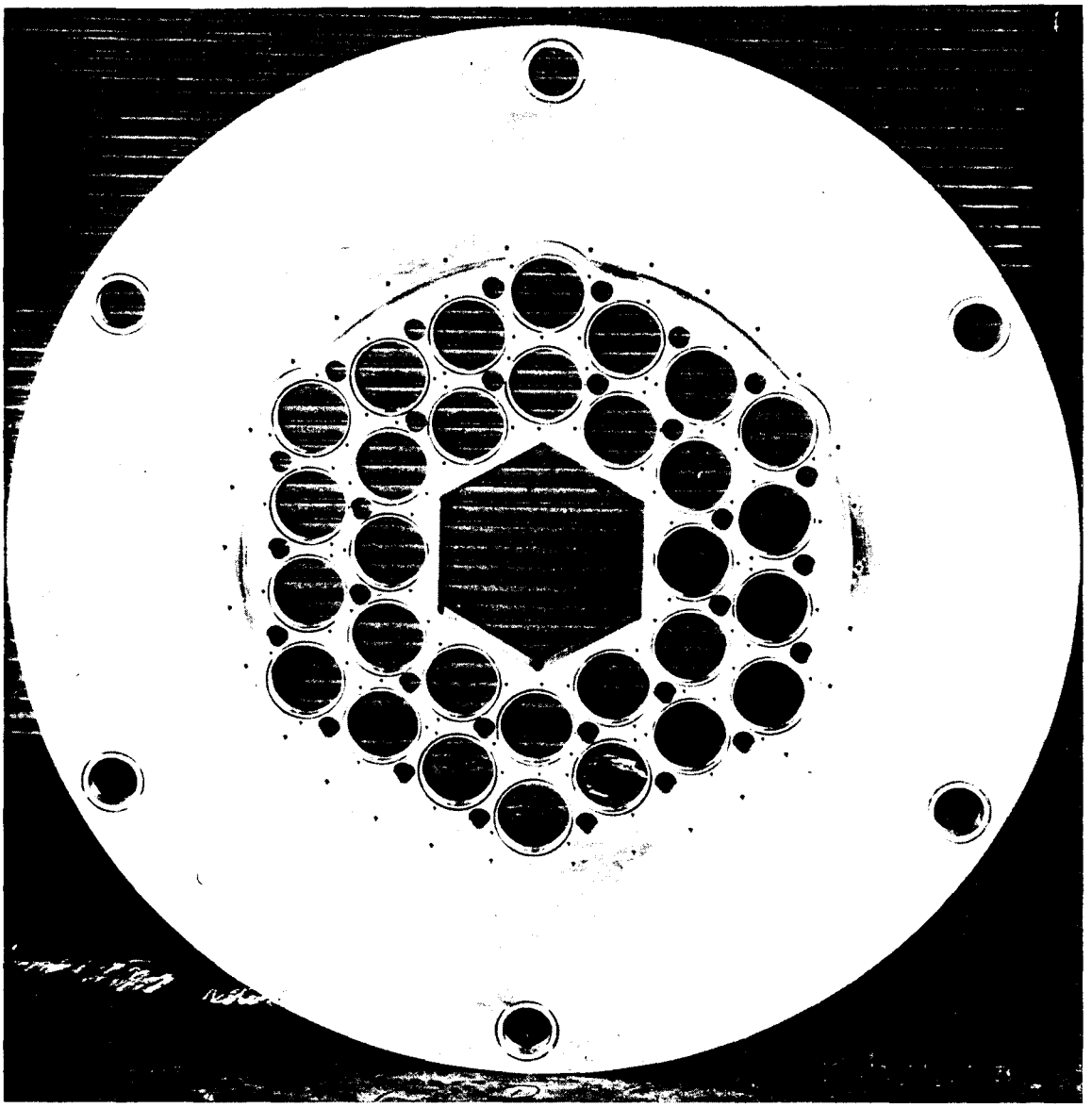

Fig. 2.4 – HTRE No. 2 top tube sheet (Neg. C03685)

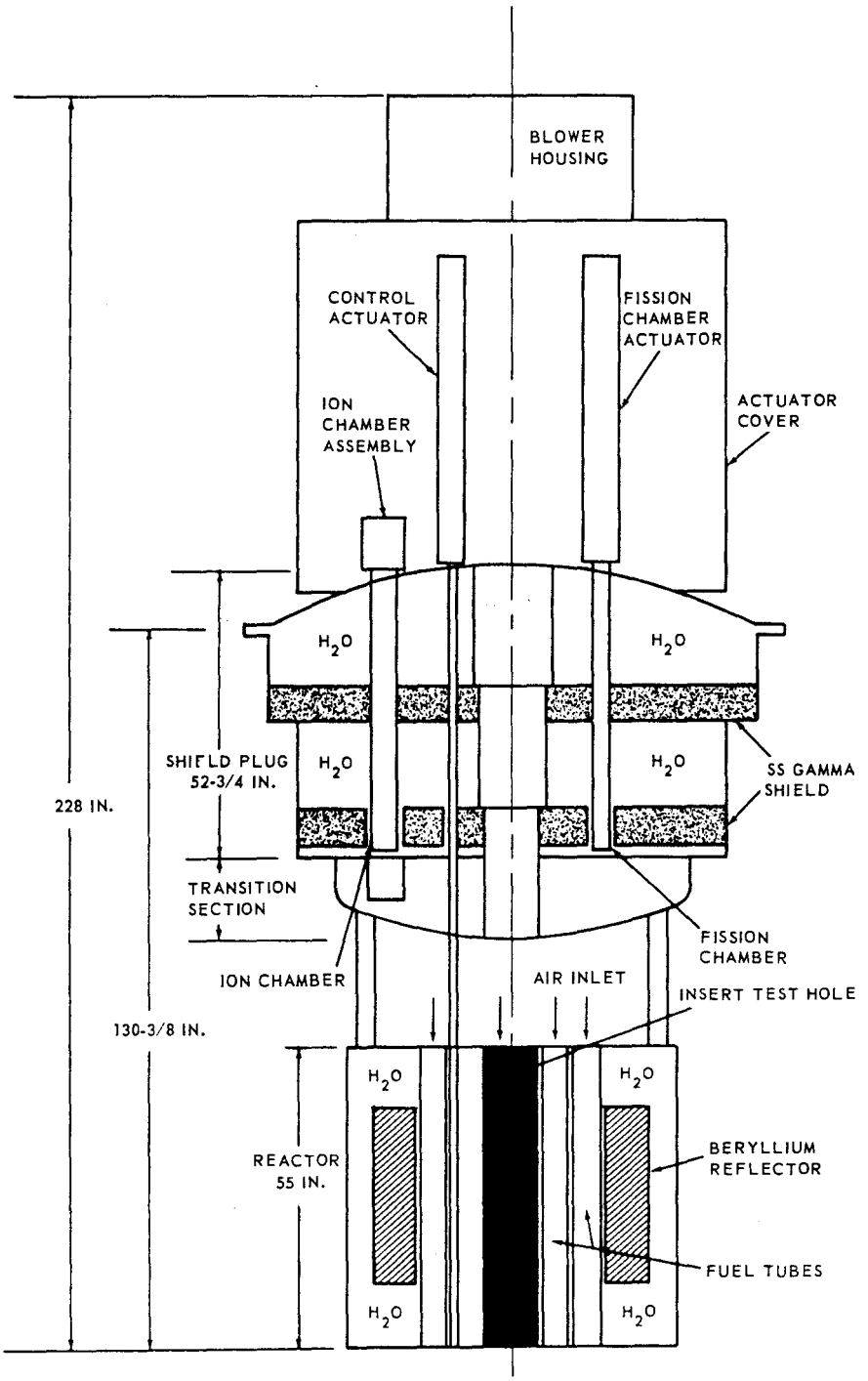

Fig. 2.5 – Schematic of HTRE No. 2 core components

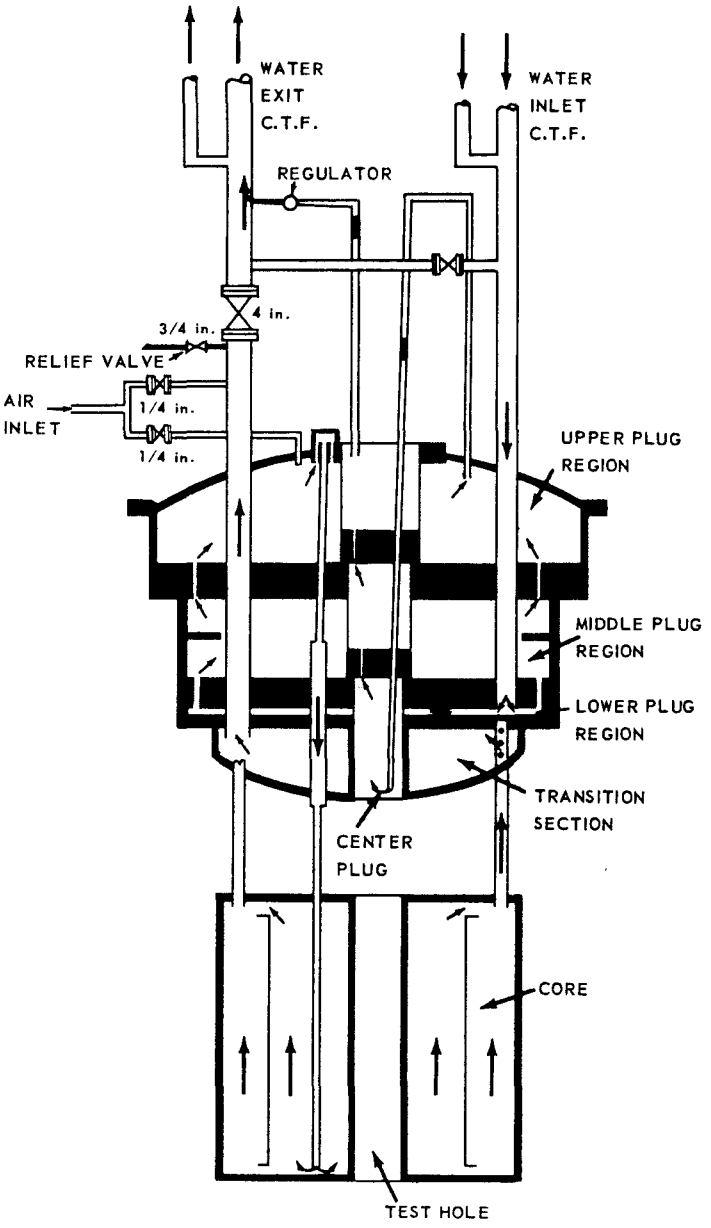

Fig. 2.6 – Schematic of HTRE No. 2 water moderator flow system

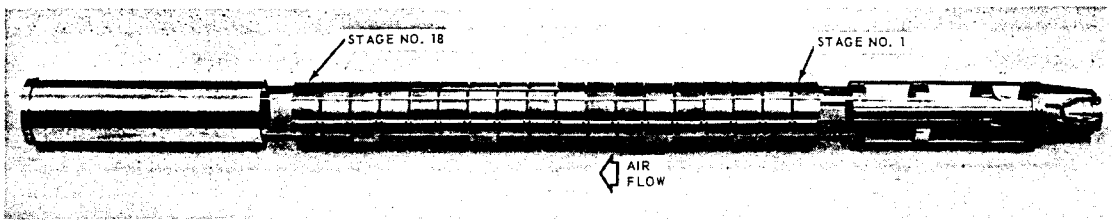

Fig. 2.7 – HTRE No. 2 parent core fuel cartridge (Neg. C04245)

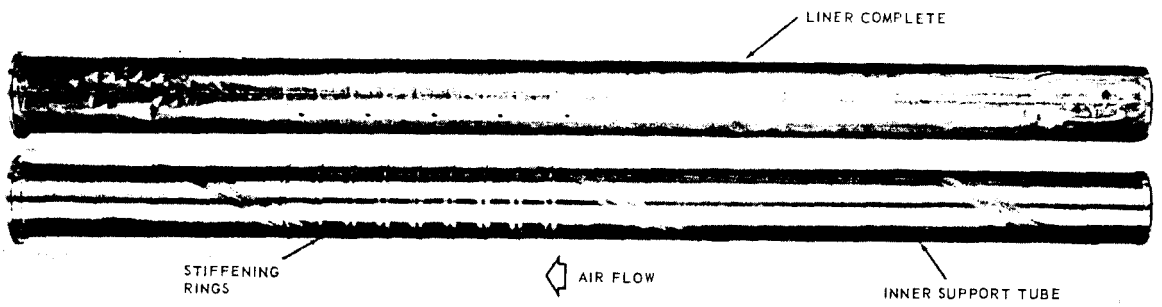

Fig. 2.8 – HTRE No. 2 fuel cartridge insulation liner (Neg. U34213)

Each ring is composed of a sandwich of uranium oxide mixed with a special 80Ni - 20Cr alloy forming the "meat" which is clad with 0.004 inch of 80Ni - 20Cr. The sandwich is sealed on the ends with braze-coated wire equal in diameter to the thickness of the fuel sandwich or ribbon. The weight ratio of the UO_2 in the "meat" mixture is 42 percent in all rings except in the innermost ring where the UO_2 weight ratio is 40 percent. The thickness of each ring varies from 0.021 inch in the inner rings to 0.013 inch in the outer ring. The variation is necessary to compensate for neutron flux depression. Detailed specifications for the fuel cartridge are given in Table 2.1.

The 18 fuel elements of the cartridges are held together by four rails to which they are spot welded. The forward ring assembly and aft assembly are welded to the four rails, forming the complete cartridge.

Connection and support of the cartridge in the core is automatic at the forward end, upon insertion. Disconnection for removal is accomplished by the insertion of a disconnect rod from the aft end, which opens a set of support fingers holding the cartridge in place. The disconnect rod does not remain in the core during operation.

2.1.3 CONTROLS AND INSTRUMENTATION[2]

HTRE No. 2 is equipped with the same type of control rods and fission and ion chambers as were used in HTRE No. 1. However, the number of rods and locations have been changed as has the mode of operation.

The primary reactivity control for HTRE No. 2 is a series of shim-scram rods, which can be arranged in four frames, or operated individually. Each frame contains a fixed number of rods, all of which move at the same time when a particular frame is activated. The HTRE No. 2 has 27 shim-scram rods which are arranged in frames as follows:

Frame 1 - 5 rods
Frame 2 - 6 rods
Frame 3 - 8 rods
Frame 4 - 8 rods

The frames are withdrawn in sequence, frame 1 being withdrawn first. Figure 2.11 shows the rod pattern and frame setup for HTRE No. 2.

Three ion chambers and three retractable fission chambers are mounted in the parent shield plug. The ion chambers provide signals to linear flux and automatic control channels; and the fission chambers provide signals for the log and period instrumentation. Other ion chambers located in the side shield water of the CTF provide signals for log flux and period circuits.

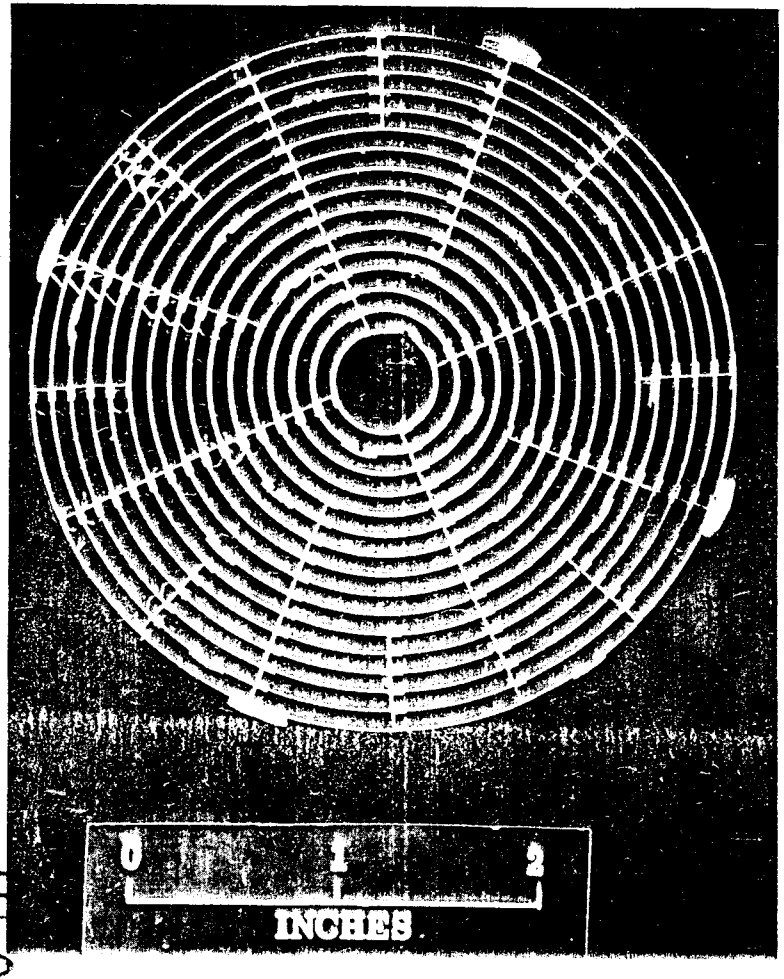

Fig. 2.9 – Stages 12 through 18 of HTRE No. 2 parent core fuel cartridge, front view (Neg. C03278)

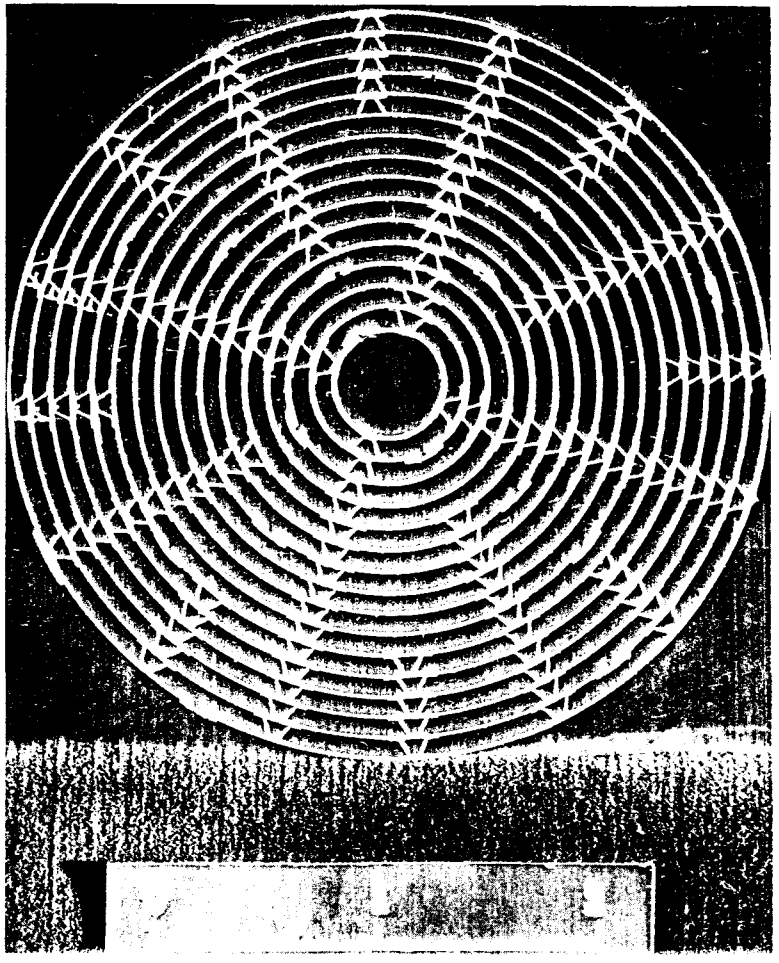

Fig. 2.10 – Stages 12 through 18 of HTRE No. 2 parent core fuel cartridge, rear view (Neg. C-03279)

TABLE 2.1

HTRE NO. 2 FUEL ELEMENT SPECIFICATIONS

STAGES 1 TO 11 INCLUSIVE

Ring No.	Loading Factor, grams UO_2 (Running inch, ±3%)	Outside Diameter, in.	Plate Thickness, in. (±0.0005)	Cut Length, in. (±0.015)	Core Weight Fraction, UO_2	Weight Ratio UO_2, g	Fueled Area, $in.^2$	Area Density, grams of $UO_2/in.^2$ of fueled area
1	1.1348	0.414	0.021	1.148	0.40	1.308	1.676	0.7773
2	1.1916	0.638	0.021	1.851	0.42	2.206	2.702	0.8162
3	1.1916	0.862	0.021	2.555	0.42	3.045	3.730	0.8162
4	1.1916	1.086	0.021	3.259	0.42	3.884	4.758	0.8162
5	1.1916	1.310	0.021	3.362	0.42	4.721	5.784	0.8162
6	1.1916	1.534	0.021	4.666	0.42	5.560	6.812	0.8162
7	1.1000	1.778	0.020	5.438	0.42	5.982	7.939	0.7534
8	1.0083	2.020	0.019	6.203	0.42	6.254	9.056	0.6906
9	0.9166	2.260	0.018	6.962	0.42	6.381	10.164	0.6278
10	0.8249	2.498	0.017	7.715	0.42	6.364	11.264	0.5650
11	0.7332	2.734	0.016	8.461	0.42	6.205	12.353	0.5022
12	0.6415	2.968	0.015	9.802	0.42	5.903	13.435	0.4394
13	0.5501	3.200	0.014	9.936	0.42	5.466	14.506	0.3768
14	0.5501	3.432	0.014	10.665	0.42	5.867	15.571	0.3768
			STAGES 12 TO 18, INCLUSIVE					
1	0.1348	0.524	0.021	1.493	0.40	1.694	2.180	0.7773
2	0.1916	0.726	0.021	2.128	0.42	2.536	3.107	0.8162
3	0.1916	0.928	0.021	2.762	0.42	3.291	4.032	0.8162
4	0.1916	1.130	0.021	3.397	0.42	4.048	4.960	0.8162
5	0.1000	1.330	0.020	4.030	0.42	4.433	5.884	0.7534
6	0.0083	1.528	0.019	4.658	0.42	4.697	6.801	0.6906
7	0.0083	1.726	0.019	5.280	0.42	5.324	7.709	0.6906
8	0.8249	1.920	0.017	5.899	0.42	4.866	8.612	0.5650
9	0.8249	2.114	0.017	6.509	0.42	5.369	9.503	0.5650
10	0.7332	2.306	0.016	7.117	0.42	5.218	10.391	0.5022
11	0.6415	2.496	0.015	7.719	0.42	4.952	11.270	0.4394
12	0.6415	2.686	0.015	8.316	0.42	5.335	12.141	0.4394
13	0.5501	2.874	0.014	8.912	0.42	4.903	13.012	0.3768
14	0.4583	3.060	0.013	9.501	0.42	4.354	13.871	0.3139
15	0.4583	3.246	0.013	10.086	0.42	4.622	18.726	0.3139
16	0.4583	3.432	0.013	10.670	0.42	4.890	15.578	0.3139

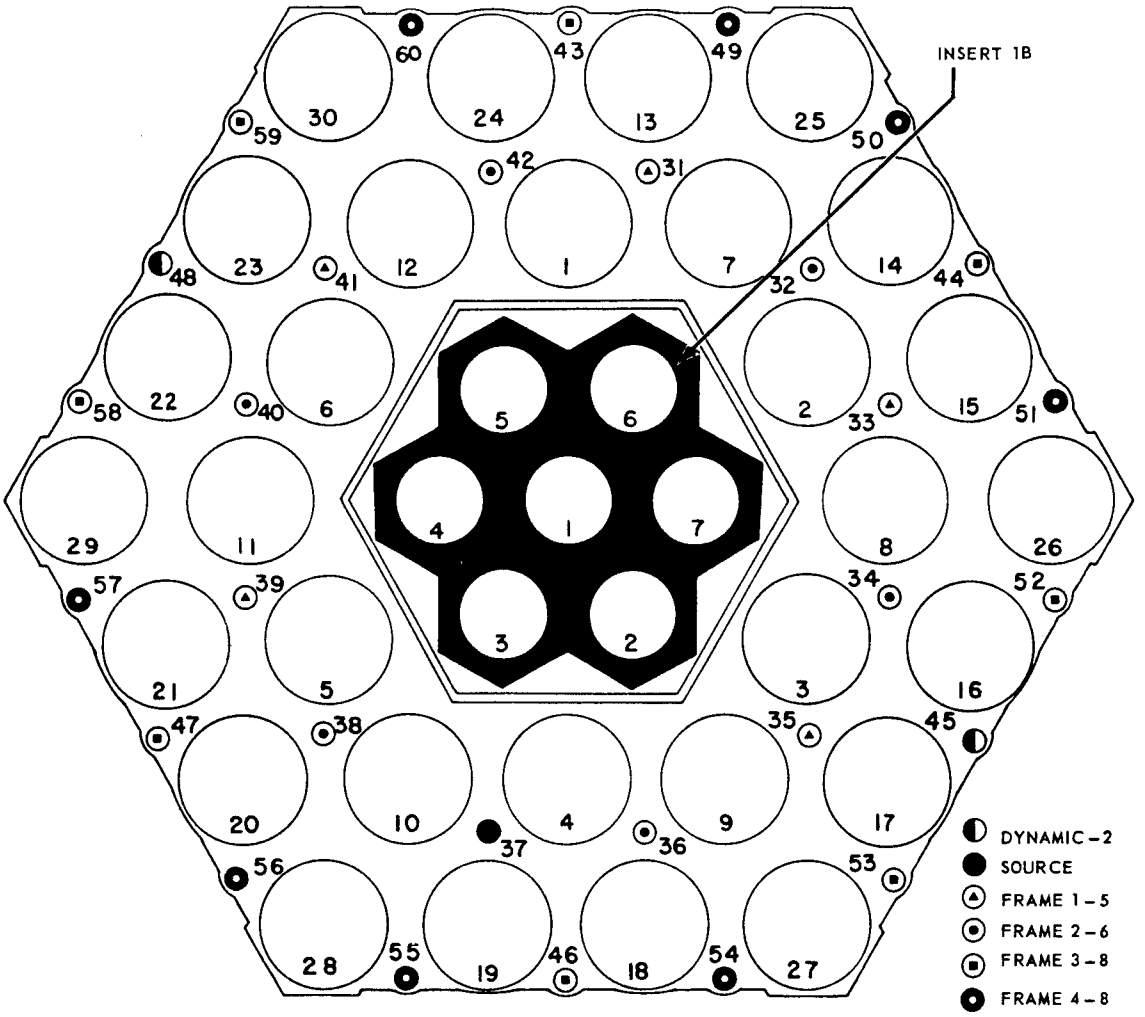

Fig. 2.11 – HTRE No. 2 control rod pattern with fuel tube and control rod numbers

2.2 PARENT CORE DESIGN DATA

Because of the strong nuclear interaction between the inserts and the parent core, it is impossible to describe the nuclear characteristics of the HTRE No. 2 system without some description of the insert configurations.

The nuclear design of the HTRE No. 2 core was predicted almost entirely on so-called insert 1B which was a model of a section from the HTRE No. 3 core, described in APEX-906. Consequently, in the description of the HTRE No. 2 reactor, reference will be made almost exclusively to insert 1B. The nuclear characteristics of other inserts subsequently tested resulted from the fixed design of the HTRE No. 2 and could not be predetermined. The nuclear characteristics of these other inserts are referenced in section 3.

2.2.1 REACTIVITY

The removal of the central seven tubes of the HTRE No. 1 and their replacement with different materials in various geometries resulted in a relatively large nuclear perturbation of the neutron flux. In general, the inserts had lower reactivity-worth than the

section removed. As a result, the possibility of very low or negative excess multiplication factors existed. It was, therefore, important in the design of the HTRE No. 2 to make allowance for a loss in excess reactivity, and if possible, to adjust the HTRE No. 2 configuration to increase the excess multiplication factor. Since time was a factor in the design, only relatively minor changes in HTRE No. 1 geometries were made. A critical experiment was impractical in view of the short design schedule. The fixing of design numbers was accomplished through knowledge obtained from HTRE No. 1 critical experiments and through machine calculations of HTRE No. 1 and HTRE No. 2 design configurations.

HTRE No. 1 critical experiments showed that moving the beryllium reflector 1/2 inch toward center resulted in an increase in excess reactivity of at least 3 percent. Calculations showed that increasing the thickness of the reflector from 4 to 8 inches would provide an additional increase of approximately 3 percent.

With this parent core geometry and the geometry and materials of insert 1B in the test holes, final calculations were made and a K_{ex} of 5 percent was computed for HTRE No. 2. The cross section of insert 1B is shown in Figure 2.12.

2.2.2 POWER DISTRIBUTIONS

Results of early machine calculations on the combination of the parent core and insert 1B showed undesirable fuel element power distribution characteristics in the insert. The specific power (power per gram of U^{235}) was lower than that of the parent core, and the power distribution was dished towards the center of the insert, producing a power gradient across the fuel cartridges in the outer ring of the insert assembly. This effect was attributed primarily to the fact that the hydrogen concentration per unit volume in the insert

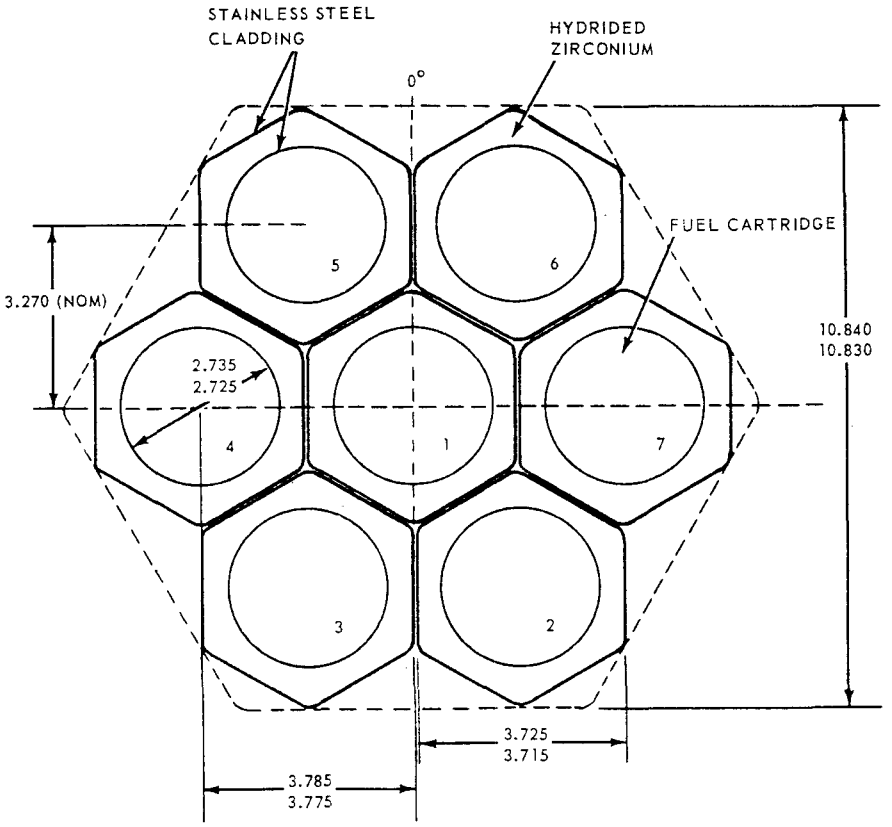

Fig. 2.12 – Cross section of insert 1B

was about two-thirds of that in the parent core. To assure that the insert would reach limiting temperatures before the parent core did, higher specific powers in the insert than in the parent core would have been desirable. In order to compensate for these effects, and to produce maximum possible reactivity, the specific fuel loading in the insert cartridges was increased by increasing ring thickness and making the moderator blocks as large as possible. (The fuel cartridge size was fixed arbitrarily to correspond to the HTRE No. 3 design.) Since adequate specific power distributions could not be obtained, airflow to the insert cartridges was orificed in order to maintain their temperatures at higher levels than those of the parent core cartridges. A significant power gradient remained across the diameter of the cartridges (in the outer ring) and this produced, later in the test operations, a rather large temperature gradient.

2.2.3 FINAL DESIGN

The over-all HTRE No. 2 design involved a compromise between reactivity, power distribution, and coolant flow distribution. This resulted in four major design considerations for HTRE No. 2 with insert 1B.

1. The beryllium reflector was moved in 1/2 inch from the position in HTRE No. 1 and increased in thickness from 4 inches to 8 inches.
2. The hydrided zirconium hex blocks of the insert were made as large as the test hole would permit, to increase the hydrogen volume fraction.
3. The insert fuel cartridge fuel loading was increased on a per unit volume basis by a factor of 1.33 times HTRE No. 1 loading.
4. The air passages were designed to assure that the ZrH moderator would get to temperature.

General Nuclear Data

Uranium Inventory (Parent Core)	79.9720 lb, (93.4% enriched)
Uranium Inventory (Insert 1B)	13.2902 lb, (93.4% enriched)
Total Uranium Inventory (Parent Core and Insert 1B)	93.2622 lb, (93.4% enriched)
Parent Core Average Thermal Neutron Flux at 68°F:	
Water	2.11×10^6 n/cm^2-sec-w
Fuel	4.63×10^5 n/cm^2-sec-w
Ratio (ϕm, ϕf)	4.56
Insert 1B Average Thermal Neutron Flux at 1500°F:	
Hydrided zirconium	1.38×10^6 n/cm^2-sec-w
Fuel	7.14×10^5 n/cm^2-sec-w
Ratio (ϕm, ϕf)	1.93
Calculated Excess in HTRE No. 2 with Insert 1B (95°F Moderator Temperature)	5%
Volume of Individual Insert Cell (Active Region)	359.55 in.^{3}
Volume of Parent Core (Active Region)	19,753.21 in.^{3}
Volume Inside Be Reflector (Active Region)	23,004.87 in.^{3}
Volume of Insert (Inside Aluminum Walls, Active Region)	3,251.66 in.^{3}

2.3 NUCLEAR ANALYSIS OF PARENT CORE AND INSERT 1B

The procedure followed to perform a nuclear analysis of HTRE No. 2 and the insert, involved the use of the IBM 650 and 701 computers, and Programs 1017, "C" and "F", as outlined in references 3 and 4.

Preliminary effort was directed to the reanalysis of the power distribution, temperature effects, and multiplication of HTRE No. 1, verification of which could be made by using experimental data on hand. This approach was deemed necessary in order to determine the correct two-group constants and to establish basic methods to be used in machine calculations for HTRE No. 2 with the various inserts.

The layout of the HTRE No. 2 core is shown in Figure 2.13. As previously described, the HTRE No. 2 core is the same as the HTRE No. 1 core, with the center seven tubes removed and with the 4-inch beryllium reflector replaced with an 8-inch reflector which was moved 1/2 inch closer to the tube bank. This portion of the reactor is defined as the "parent core." The void left by removing the center seven tubes is defined as the "test hole," and test pieces placed in this hole are defined as "inserts."

In the analysis of HTRE No. 2, it was assumed that:

1. The fuel cells surrounding the test hole had the same volume and equivalent radius as those used in the HTRE No. 1 analysis.
2. The set of fuel cells adjacent to the beryllium reflector occupied the remaining active volume of the parent core.

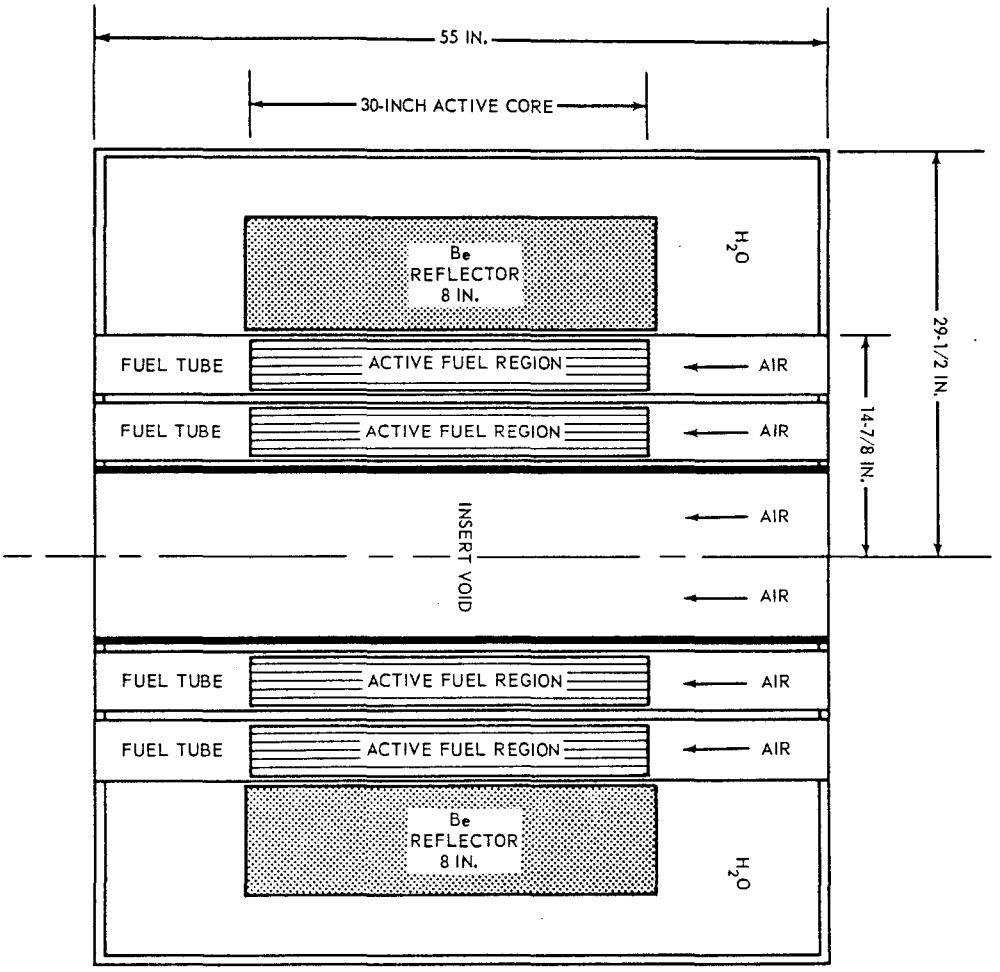

Fig. 2.13 – HTRE No. 2 core longitudinal cross section

3. The layer of water between the insert wall and nearest fuel cells could best be treated as a separate region. This assumption was based on the fact that in the HTRE No. 1 analysis, a separate water region next to the reflector provided the best correlation with experimental data.

Nuclear analysis of insert was made using the following criteria:

1. The insert, when combined with the parent core, makes the complete assembly critical, with enough excess multiplication to permit a reasonable test program.
2. When criticality is assured, materials are distributed, representing a portion of some full-scale reactor geometry, to give the best power profile.

The procedure for the analysis then, is to calculate excess multiplications for different volume fractions of the materials involved until the criteria are satisfied. For these calculations, materials were homogenized.

The HTRE No. 3 reactor was hydrided-zirconium-moderated and was made up of an array of 151 hexagonal fuel-moderator cells. The 1B insert represented a section of this array with only slight differences in the size of the individual fuel-moderator cells.

After a series of parametric studies, the volume fractions of moderator and void and fuel elements were fixed for the final calculations. Two-group diffusion theory constants with Behren's corrections and cell corrections for this loading can be found in reference 1. The radial one-dimension, multiregion criticality calculation using the materials volume fractions given in the reference produced power curves (Figure 2.14) and flux curves (Figure 2.15), and the evaluation that the excess reactivity of the HTRE No. 2 reactor was 5 percent.

These data were used to compute power generation in the individual fuel elements in the reactor, from which the thermodynamic characteristics of the reactor were computed.

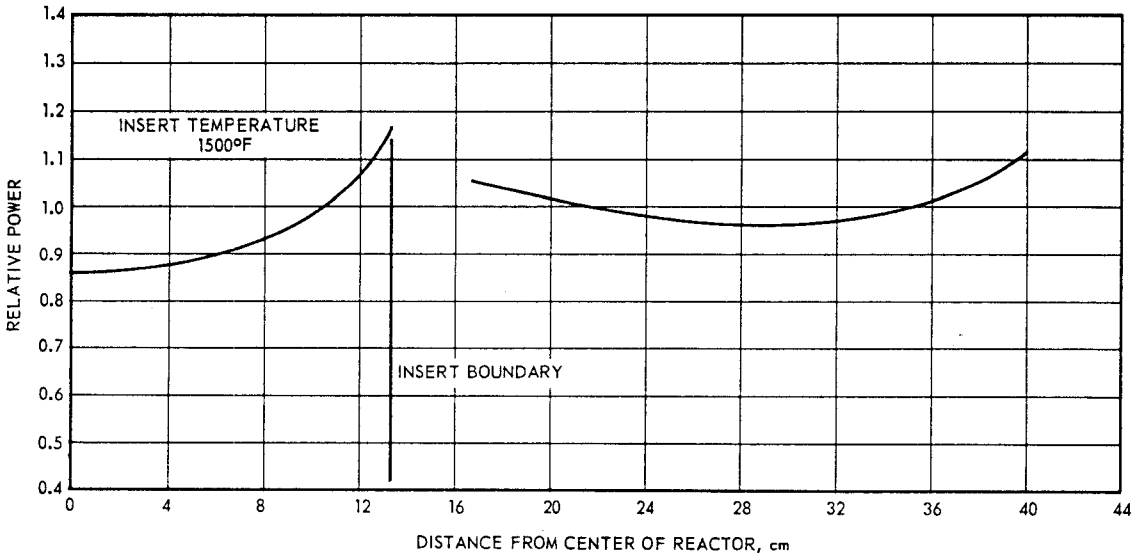

Fig. 2.14 – HTRE No. 2, gross radial power with insert 1B

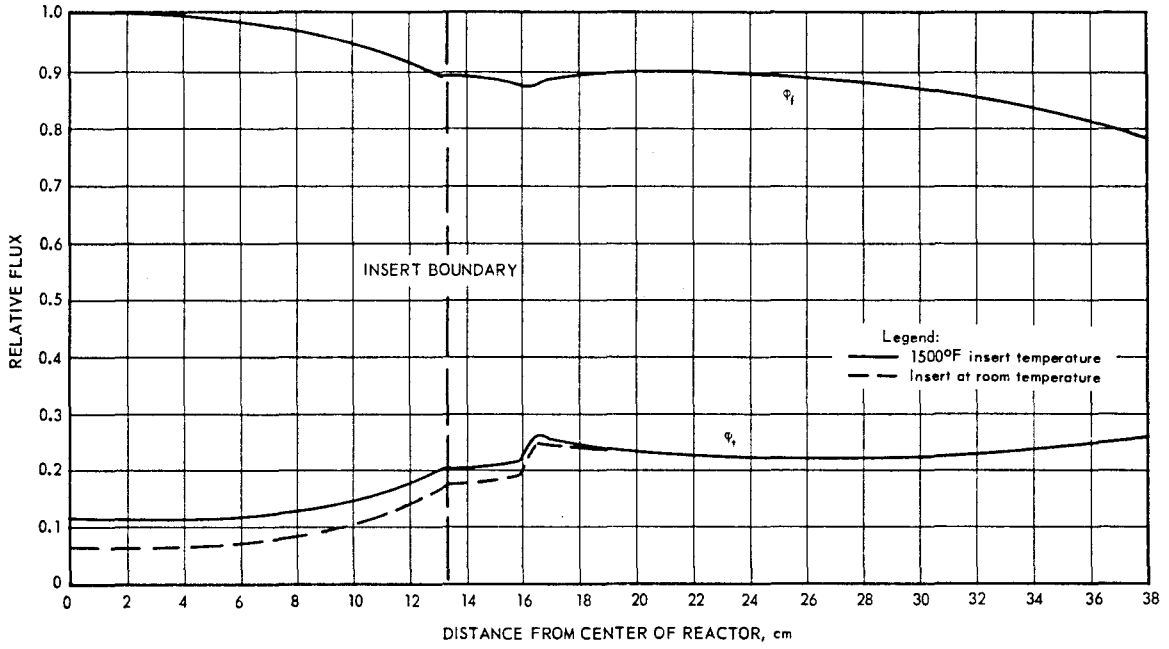

Fig. 2.15 – HTRE No. 2 flux curves with insert 1B

2.4 THERMODYNAMIC DATA

The thermodynamic design data includes data for the insert 1B because the parent core is a subcritical assembly without the insert. A method of analysis was established and carried through at 6 and 8 megawatt total reactor power. The reactor was subsequently operated at powers up to and including 15-megawatt total reactor power without complete documentation of the operating parameters; however, a close approximation of the design parameters can be obtained by extrapolating the data at the 6 and 8 megawatt levels to the higher reactor powers.

The primary purpose of the analysis was to predict maximum temperature distributions in the parent core with insert 1B. The determination of these temperatures involved a series of calculations and assumptions, in the areas of heat generation, flow distribution, and temperature calculation.

Heat generation data are given in Table 2.2. The longitudinal power distribution for the insert, taken from HTRE No. 1 gamma heating information,[5] is shown in Figure 2.16. Heat generation distribution figures for the parent core are shown in Table 2.3. The profile for insert fuel stage 18 was estimated from the curve for HTRE No. 1 by consideration of the relative amounts of end reflector in the respective regions. This probably constitutes one factor causing uncertainty in the relative temperatures between insert fuel and moderator. Neutron heating in the hydride was assumed to follow the given gamma heating distribution.

Determination of airflow and airflow distribution in the insert tubes was essential to calculations of surface temperatures. These calculations assumed that each insert fuel tube had three independent flow passages (i.e., fuel section, outer annulus, and centerhole through the fuel) with no cross flow between them. It is doubtful that this condition existed, but further refinements in methods was not practical. The flow distribution charts (Figures 2.17, 2.18, 2.19, 2.20 and 2.21) were employed to determine flow in the various passages.

TABLE 2.2

HTRE NO. 2 THERMODYNAMIC DATA

	Reactor Power, mw	
	8	6
Nominal Outer Annulus Spacing, in.	0.125	0.095
Engine Speed Air Force "Hot Day," rpm	7685	
Air Temperatures, °F		
Compressor discharge	447	
Reactor inlet	412	412
Reactor exit		
Insert, outer 6 tubes		
Fuel section	1257	960
Outer annulus	1147	1280
Parent core tubes	927	810
Mixed reactor exit	936	
Combustor inlet	811	
Turbine inlet	1256	
Flows, lb/sec		
Compressor discharge	58.42	58.42
Reactor		
Insert-outer 6 tubes (per tube)		
Outer annulus	0.19	0.10
Fuel section	0.56	0.68
Center hole in fuel	0.05	0.04
Parent core (per tube)	1.70	1.70
Core bypass leakage	1.71	1.71
Combustor fuel flow, lb/hr	1320	
Reactor Temperatures, °F		
Insert upper tube sheet	500	
Insert lower tube sheet	800	
Maximum parent core fuel	1100	940
Maximum insert outer tube fuel		
Inner rings	1440	1120
Outer rings	1420	1290
Maximum solid moderator		
Inner surface	1410	1440
Internal	1490	1490
Heat Generation, Btu/sec-mw of Power from Fuel		
Parent core fuel per tube	27.74	27.74
Outer insert fuel per tube	16.99	16.99
Center insert fuel per tube	14.22	14.22
Heat from moderator cell (per cell)	2.91	2.91
Heat from outer ring to outer annulus	1.56	0.96
Heat to air through outer insert fuel section per tube	15.20	15.80
Maximum heating rates in solid moderator, watts/gm-mw		
Gamma	0.147	0.147
Neutron	0.016	0.016

TABLE 2.3

FUEL ELEMENT AND MODERATOR POWER DISTRIBUTION

	P_{fn}[a]			ϕTE_n[b]	
Stage	Parent Core	Insert	Moderator	Parent Core	Insert
<1	-	-	0.0963	-	-
1	0.0437	0.0378	0.0385	0.888	0.984
2	0.0402	0.0382	0.0421	1.067	1.036
3	0.0469	0.0426	0.0457	1.075	1.058
4	0.0539	0.0504	0.0498	1.048	1.093
5	0.0595	0.0582	0.0529	1.035	1.054
6	0.0637	0.0637	0.0554	1.028	1.031
7	0.0673	0.0676	0.0571	1.018	1.019
8	0.0695	0.0700	0.0579	1.005	1.005
9	0.0701	0.0700	0.0578	1.001	0.998
10	0.0695	0.0693	0.0563	0.987	0.991
11	0.0673	0.0677	0.0545	0.980	0.983
12	0.0640	0.0655	0.0517	0.971	0.979
13	0.0596	0.0628	0.0489	0.962	0.972
14	0.0546	0.0588	0.0458	0.952	0.964
15	0.0486	0.0537	0.0427	0.943	0.952
16	0.0425	0.0469	0.0394	0.951	0.926
17	0.0387	0.0398	0.0361	0.968	0.920
18	0.0404	0.0373	0.0331	1.274	1.050
>18	-	-	0.0376	-	-

[a] P_{fn} = fraction of total power generated at stage n.
[b] ϕTE_n = ratio of power generation at trailing edge of stage n to average power in stage n.

Pressure losses in the venturi, insert inlet, and aft assemblies were calculated as a series of expansion, contraction, friction and diffusion losses. These appear to account for about 25 percent of the pressure loss across the insert tubes.

Friction factors for the outer annulus and the insert fuel were determined from the same Reynold's number relationship and multiplying factor. Calculations for hydraulic diameter included allowance for spacers, thermocouple leads, rails, etc. Flow in the outer annulus was determined with an assumed average "hot" spacing. This assumption was justified by final temperature and expansion calculations. Heat to the outer annulus air from the outer rings was likewise assumed, and was justified by final calculation of relative fuel and air temperature.

Insert tube sheet leakage was calculated as dependent on a series of expansion, contraction, and friction pressure losses through only those holes which were not plugged.

Fuel temperatures were calculated using standard film coefficient relationship. In the insert, temperature calculations were made for three regions: moderator surface, outer ring, and inner fuel rings. Radiant heat transfer was determined to be negligible. The internal temperature rise in the solid moderator was subject to uncertainties in both heating rates and thermal conductivity (assumed 10 Btu/hr-ft^2-^{o}F/ft). Because of some heat loss due to leakage air and radiation to the parent core, the outer moderator cells did not exhibit uniform peripheral temperatures. Cladding contact resistance for the internal moderator surfaces was assumed negligible. Figure 2.22 presents longitudinal temperature profiles for the insert and parent core at 8 megawatts to air.

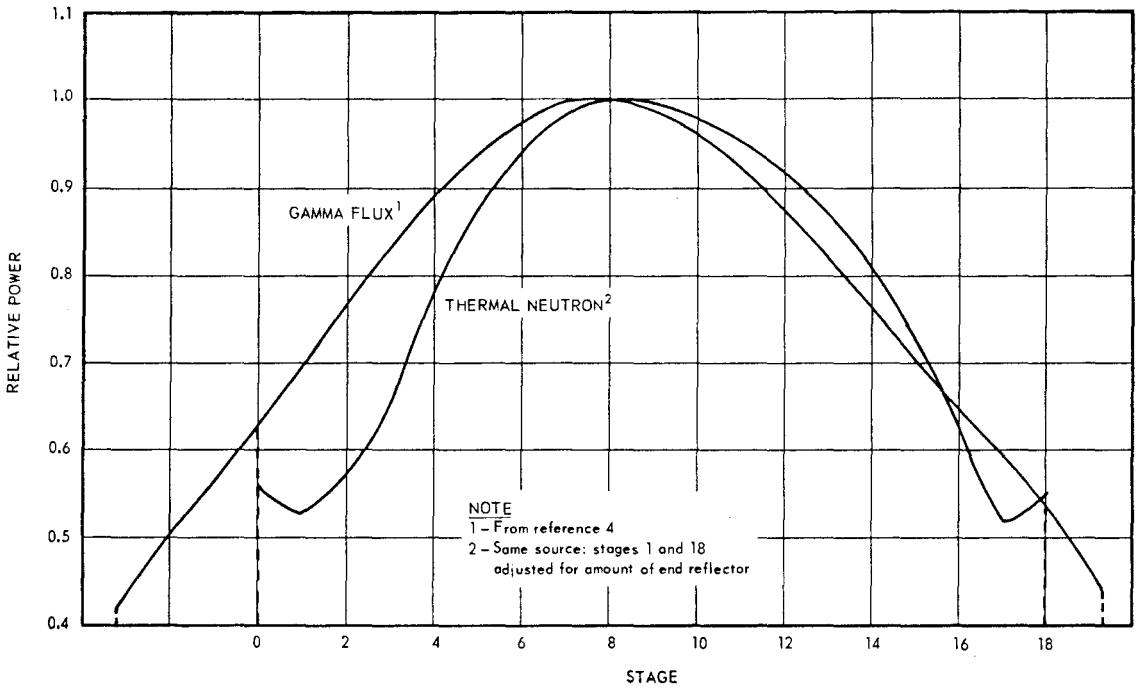

Fig. 2.16 – Insert 1B longitudinal flux distribution in HTRE No. 2

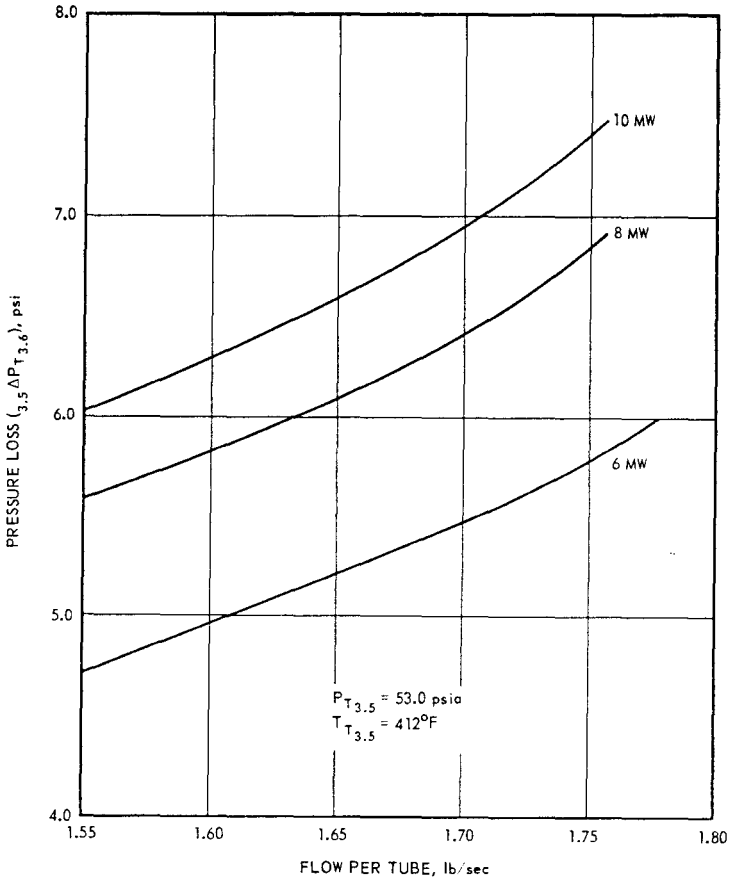

Fig. 2.17 – Insert 1B pressure loss through parent core

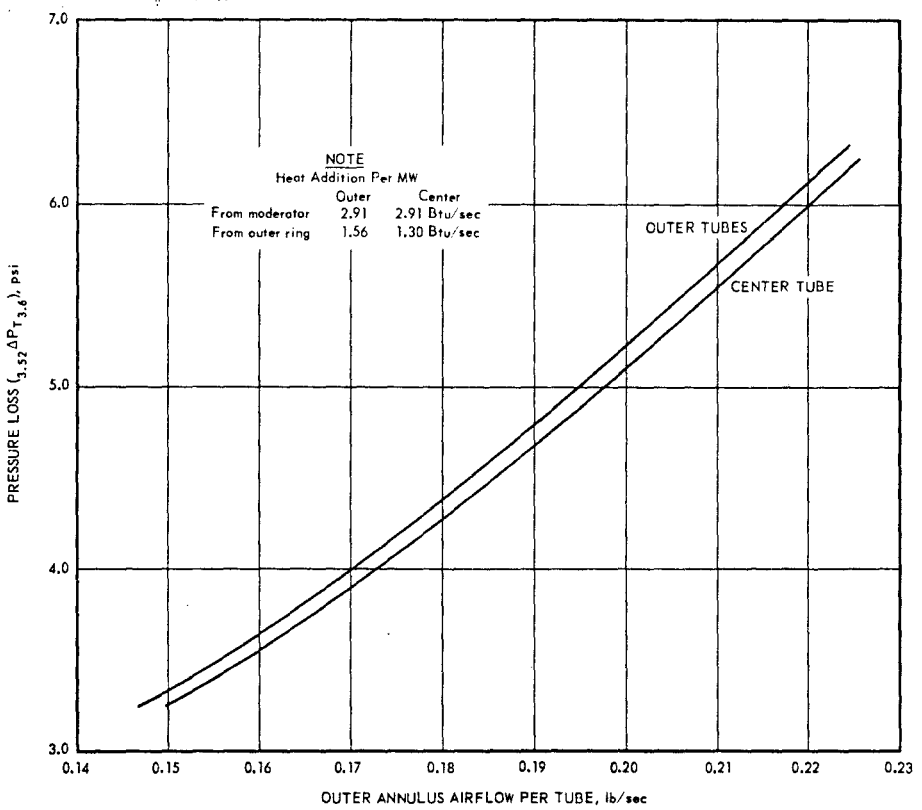

Fig. 2.18 – Insert 1B pressure loss in outer annulus at 8 mw to air

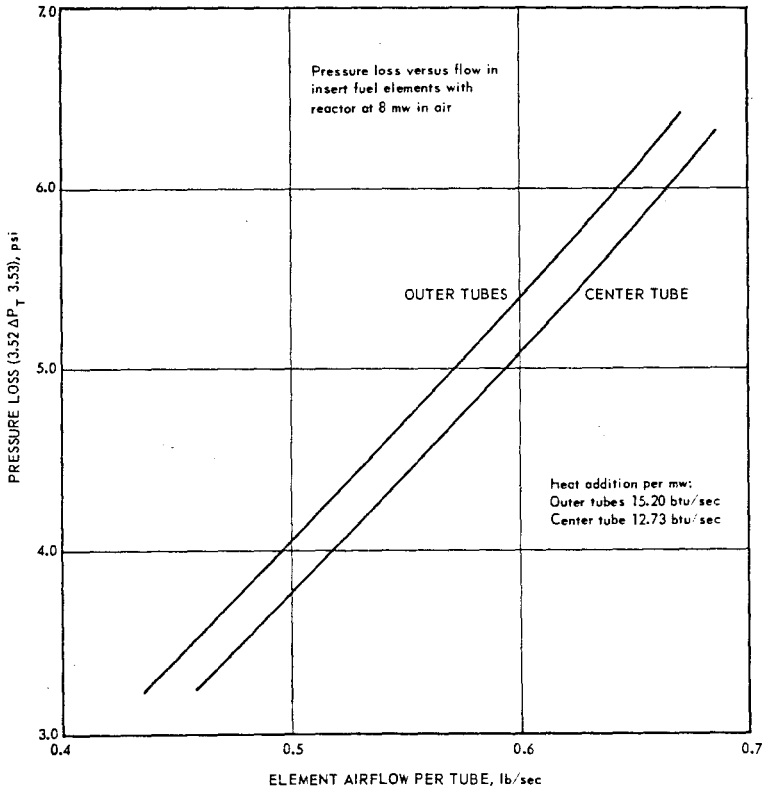

Fig. 2.19 – Insert 1B pressure loss for fuel element at 8 mw to air

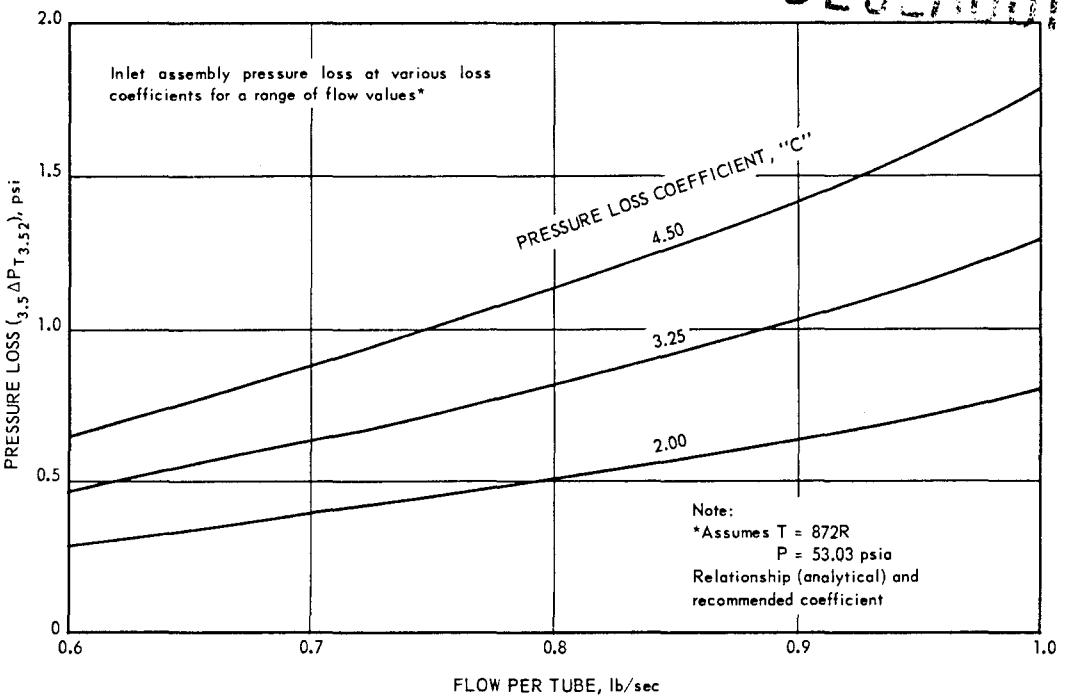

Fig. 2.20 – Insert 1B pressure loss for inlet assembly of fuel cartridge

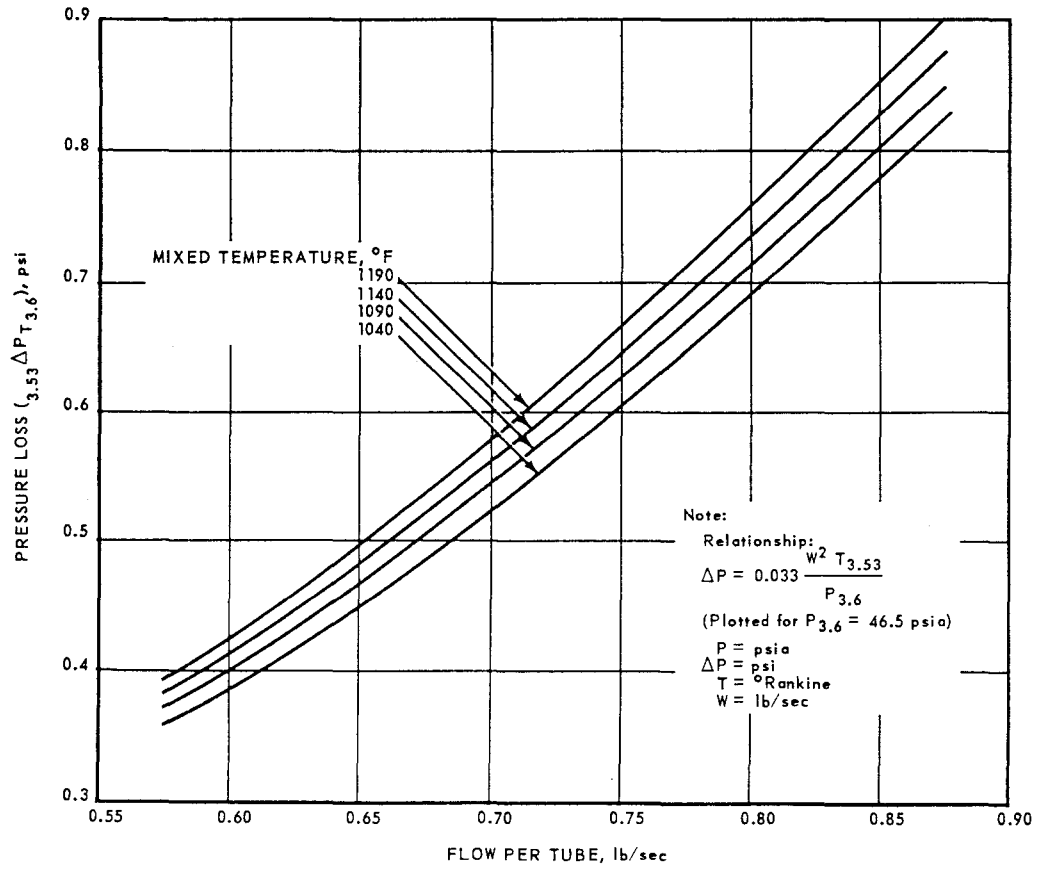

Fig. 2.21 – Insert 1B pressure loss for fuel cartridge aft assembly

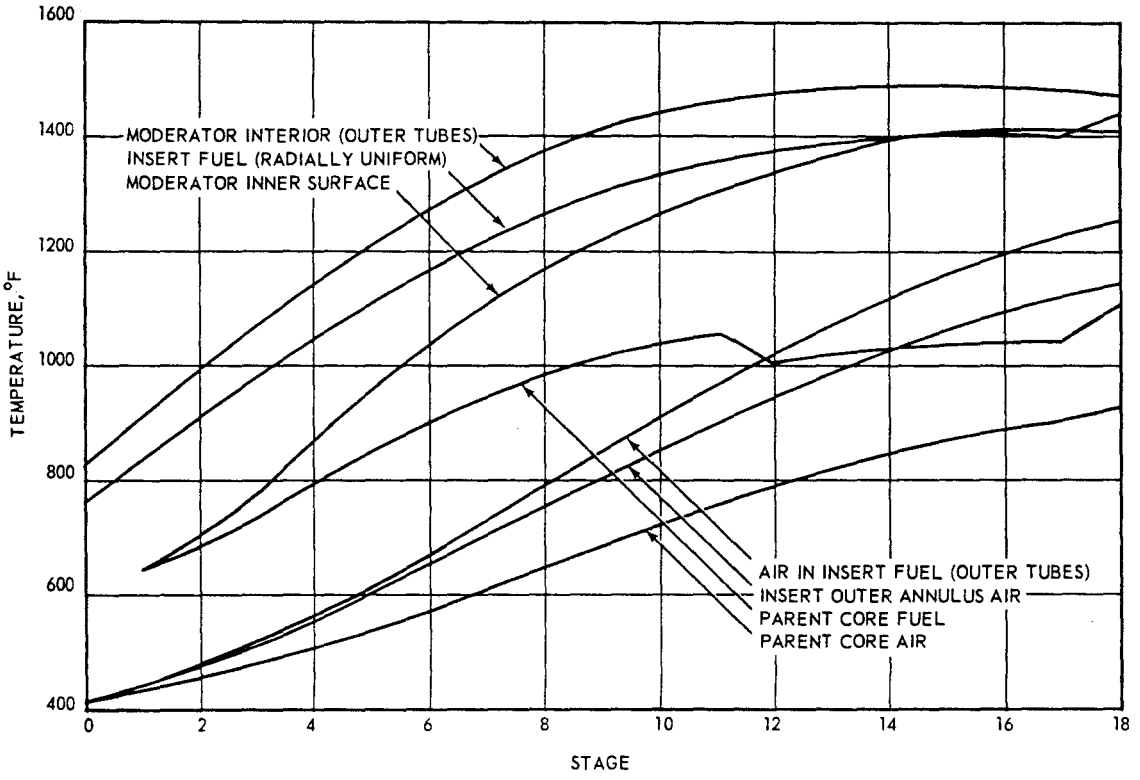

Fig. 2.22 – HTRE No. 2 and insert 1B longitudinal temperature profiles, 8 mw to air

2.5 REFERENCES

1. Evans, R. C., "HTRE No. 2 Design Data Book," GE-ANPD, XDC 56-8-104, August 1956.
2. Smith, R. H., and Clavell, G. H., et al., "Handbook HTRE No. 2 Controls and Instrumentation Systems," GE-ANPD, XDC 60-1-1, December 10, 1959.
3. Selengut, D. S., "Criticality Calculations on the 701 Computer. (Program A)." GE-ANPD, XDC 55-3-18, February 17, 1955.
4. Selengut, D. S., "Summary of the GE-ANPD Reactor Machine Computation Program," GE-ANPD, XDC 55-4-44, April 6, 1955.
5. Rosenthal, H. B., "HTRE No. 1 Gamma Heating (TRA III)," GE-ANPD, DC 55-8-26, August 1955.

3. METALLIC TEST INSERTS

The initial power tests performed in the HTRE No. 2 utilized seven-cell metallic insert assemblies.[1] Three different metallic inserts - 1B, 1C, and 1D - were tested. Insert 1A was a mockup assembly used in critical experiments to determine experimental values of excess reactivity and power generation. All three insert assemblies, operated at power, were essentially identical mechanically and had similar nuclear characteristics.

3.1 INSERT DESCRIPTION AND TEST OBJECTIVES

A brief description of the metallic test inserts, together with the primary objective of the testing program in which each was involved, is given in the following section. Detailed test procedures and results for each insert appear in the reference listed with the particular insert, and test results are summarized in section 6 of this volume.

3.1.1 INSERT 1B

Insert 1B consisted of seven stainless-steel-clad, hydrided zirconium, hexagonal tubes as moderating material with concentric ring metallic fuel elements in the hole of the moderator. This insert, a model of the section from the HTRE No. 3 core, served as the basis from which most of the nuclear design of the HTRE No. 2 core was predicted. The primary purpose of the insert 1B tests was to evaluate clad hydrided zirconium as a moderating material capable of withstanding temperatures of 1650°F. Figure 3.1 shows the insert after test. Test procedures and results for this cartridge are described in reference 2.

3.1.2 INSERT 1C

Insert 1C consisted of seven unclad, slotted, hydrided zirconium hexagonal tubes as moderating material. The insert was modified after initial operation by replacing the 11-ring fuel cartridges with 10-ring cartridges and plugging four additional holes in each of the moderator cell rear transition pieces. An over-all view of the insert is shown in Figure 3.2. Primary purpose of the test was to evaluate mechanical and materials characteristics of the unclad, hydrided zirconium as a moderating material. Test procedures and results are more thoroughly described in reference 3, and section 6.

3.1.3 INSERT 1D

Insert 1D, similar to 1B and 1C, consisted of seven hexagonal hydrided zirconium tubes. The six outer tubes contained similar fuel cartridges of the metallic concentric ring design. The center tube contained a beryllium bar to boost the flux in the center of the insert and to reduce the flux gradient across the diameter of the six outer tubes. Two of the outer tubes contained individually, remotely operable pneumatic valves at the inlet end to reduce the airflow to the fuel elements during operation. The test was devised to evaluate the hazard of a "one tube melt" in a full size reactor similar in design and containing the same fuel element and moderator materials as insert 1B. Complete test procedures and results are treated in reference 4, and section 6.

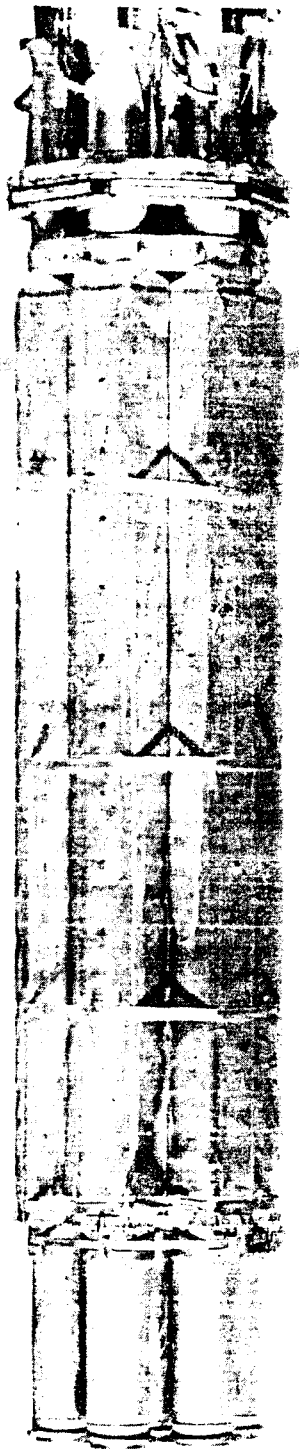

Fig. 3.1 – Insert 1B after test (Neg. C13512)

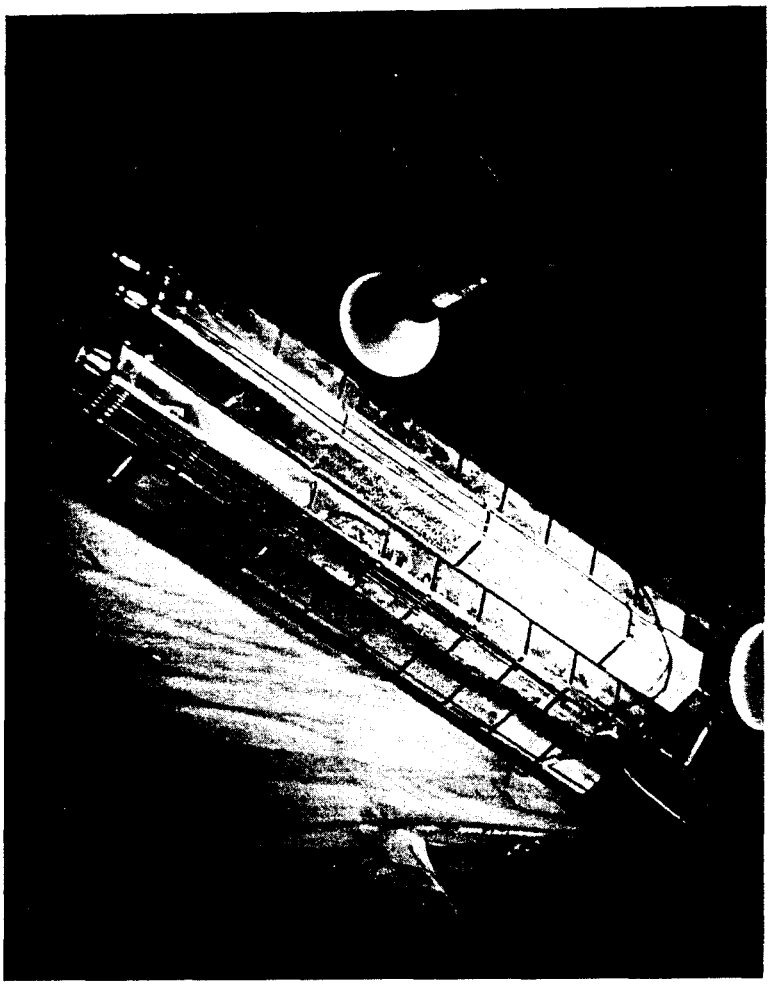

Fig. 3.2 – Insert 1C after test (Neg. C13630)

3.2 SUMMARY OF INSERT 1 NUCLEAR CHARACTERISTICS

3.2.1 FUEL CARTRIDGE DESIGN

In the preliminary thermodynamic analysis, a set of fuel-cartridge ring-loadings and ring-spacings were specified and the heat fluxes for the individual plates were computed. The heat fluxes were used to calculate temperatures in insert cells. If these calculated temperatures did not satisfy design specifications, cartridge-loadings and spacings were changed and the thermodynamic analysis repeated until the desired results were obtained.

A review of the nuclear and thermodynamic analyses showed that a loading of 93.4 percent enriched uranium of 2 pounds ± 0.2 pounds would be required for each insert 1 cartridge.

Using this loading as a basis, a fine radial power distribution was calculated for a homogenized equivalent-center insert 1 cell, with an assumed hydrided zirconium moderator temperature of 1500°F. This fine radial curve is shown in Figure 3.3.

Using this calculated fine radial power curve, an iteration process was carried out to determine the individual fuel-ring thicknesses, which produced the flattest fine radial

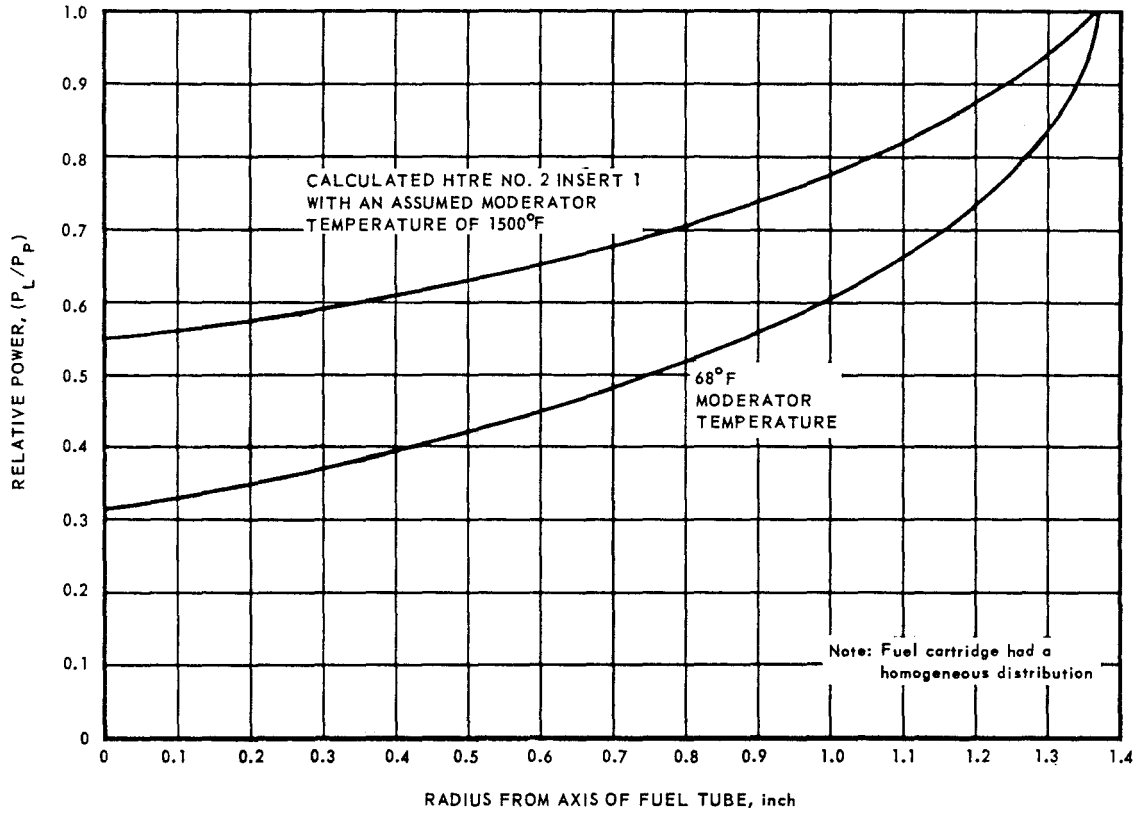

Fig. 3.3 – Insert 1B fine radial power distribution for fuel cartridge

heat flux. This was done: (1) by deciding the number of fuel rings per stage needed to contain the uranium and to provide heat-transfer surface, (2) by assuming some plate thickness (uranium loading), (3) by using plots (Figure 3.4) of the homogeneous uranium loading and relative power, shown in Figure 3.5, as a function of stage-radius to plot the non-homogeneous power distribution, and (4) by using this distribution to calculate the heat fluxes. Table 3.1 gives the specifications of insert 1B fuel elements by ring number.

Figure 3.5 shows the resulting fine radial specific power curve which was used to calculate the relative heat flux (ratio of local-to-average heat flux). Figure 3.6 shows a plot of relative heat flux versus distance from the axis of the cartridge. Table 3.1 presents the specifications for insert 1B fuel cartridges. Table 3.2 presents the data used to calculate relative heat fluxes.

3.2.2 TOTAL POWER

Absolute heat fluxes in the various stages and rings of the representative insert 1 fuel cartridges were needed to calculate insert cartridge temperatures. These heat fluxes were computed by using:

1. The gross radial-power curve from which the weighted-average relative specific power were obtained (Figure 3.7).
2. The longitudinal relative power curve (Figure 3.8) which is presented as a ratio of the local-average specific power to the tube-average specific power. This curve was corrected for the reduced thickness of the end reflector of insert 1.
3. The fine radial relative (Q_L/Q_A) heat flux curve shown in Figure 3.6. In using this curve, it was assumed that the same relative heat-flux distribution holds for all 18 stages of the insert fuel cartridge.

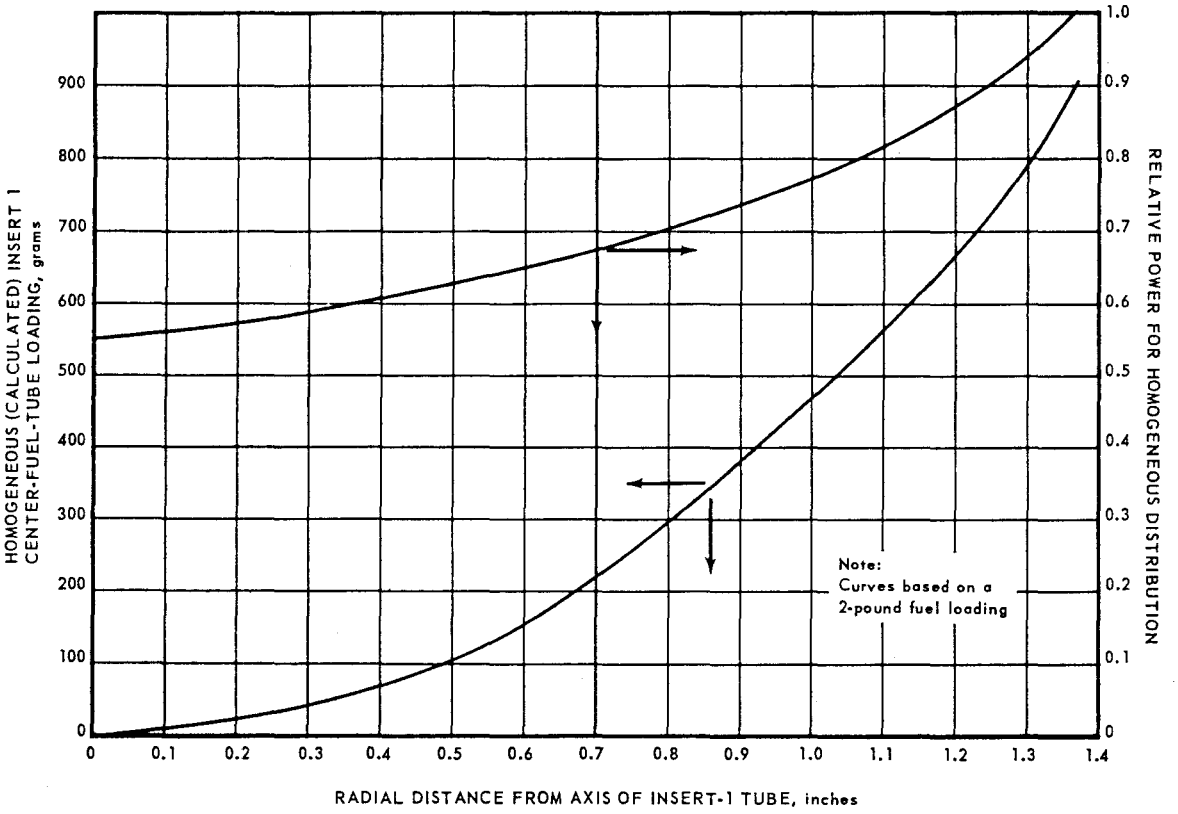

Fig. 3.4 – Insert 1 relative power for homogeneous distribution

TABLE 3.1

INSERT 1B FUEL ELEMENT SPECIFICATIONS

Ring No.	Plate Thickness, in.	Outside Diameter, in.	Cut Length, in.	Core Weight Fraction, percent	Weight Of UO_2, g	Fueled Area, $in.^2$	Fueled Volume, $in.^3 \times 10^{-2}$	Area Density, $g/in.^2$	Loading Factor, g/in. UO_2
1	0.021	0.468	1.3173	0.40	1.4950	1.9233	2.5003	0.7773	1.1703
2	0.021	0.670	1.9519	0.42	2.3260	2.8498	3.7047	0.8162	1.2347
3	0.021	0.872	2.5865		3.0822	3.7763	4.9092	0.8162	1.2346
4	0.021	1,074	3.2211		3.8384	4.7028	6.1136	0.8162	1.2346
5	0.021	1,276	3.8557		4.5946	5.6293	7.3181	0.8162	1.2346
6	0.021	1,478	4.4903		5.3508	6.5558	8.5225	0.8162	1.2346
7	0.020	1,678	5.1238		5.6360	7.4808	8.9770	0.7534	1.1397
8	0.020	1,878	5.7521		6.3271	8.3981	10.0777	0.7534	1.1397
9	0.019	2.076	6.3793		6.4321	9.3138	10.2452	0.6906	1.0448
10	0.018	2.272	7.0002		6.4163	10.2203	10.2203	0.6278	0.9447
11	0.018	2.468	7.6159	0.42	6.9806	11.1192	11.1192	0.6278	0.9447

NOTE:

Weight of UO_2 stage = 52.4791

Weight of 18 stages = 944.6238 g of UO_2

or = 2.0825 lb of UO_2

or = 1.8326 lb of U^{235}

Nominal core width = 1.460 in.

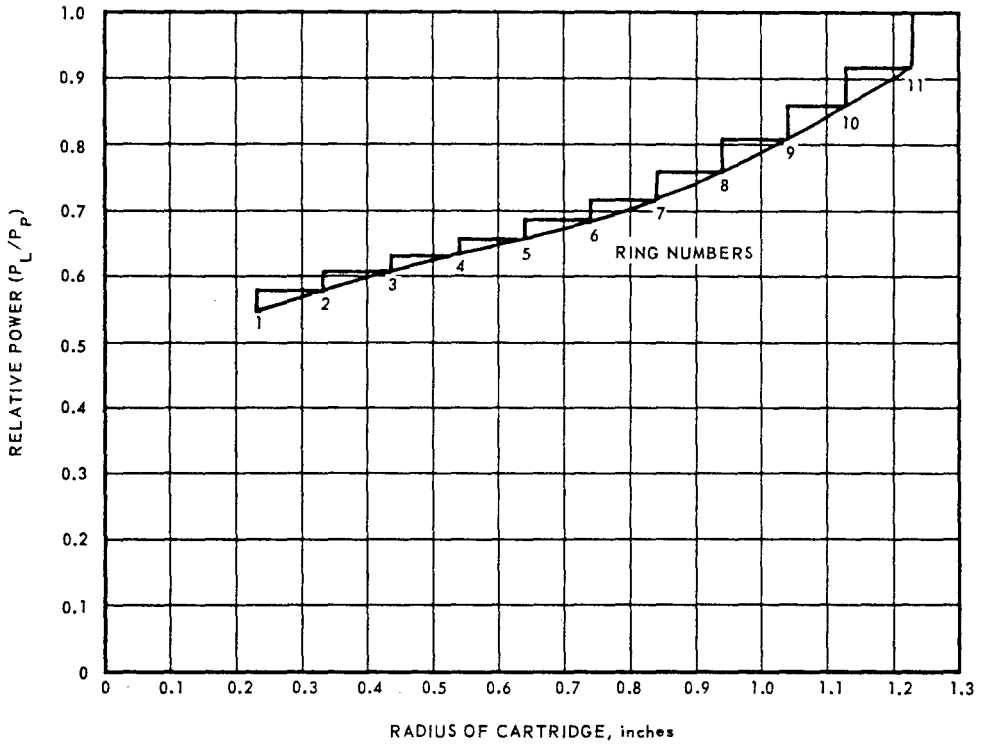

Fig. 3.5 – Insert 1 eleven-ring fuel cartridge fine radial power distribution

TABLE 3.2

RELATIVE HEAT FLUX

Ring No.	P_L/P_p	ρ Area Density, g/in.^{2}	$P_L/P_p \times \rho$	A Area	$P_L/P_p \times \rho \times A$	$\frac{P_L/P_p \times \rho}{\Sigma P_L/P_p \times \rho \times A} = \frac{Q_L}{Q_A}$
11	0.96	0.6278	0.6027	11.1192	6.7015	1.0711
10	0.89	0.6278	0.5587	10.2203	5.7100	0.9929
9	0.838	0.6906	0.5787	9.3138	5.3899	1.0284
8	0.788	0.7534	0.5937	8.3981	4.9860	1.0551
7	0.741	0.7534	0.5583	7.4808	4.1765	0.9922
6	0.705	0.8162	0.5754	6.5558	3.7722	1.0226
5	0.676	0.8162	0.5518	5.6293	3.1062	0.9806
4	0.650	0.8162	0.5305	4.7028	2.4948	0.9428
3	0.624	0.8162	0.5093	3.7763	1.9233	0.9051
2	0.595	0.8162	0.4856	2.8498	1.3839	0.8630
1	0.565	0.7773	0.4392	1.9233	0.8447	0.7805
				Σ71.9695	Σ40.4890	

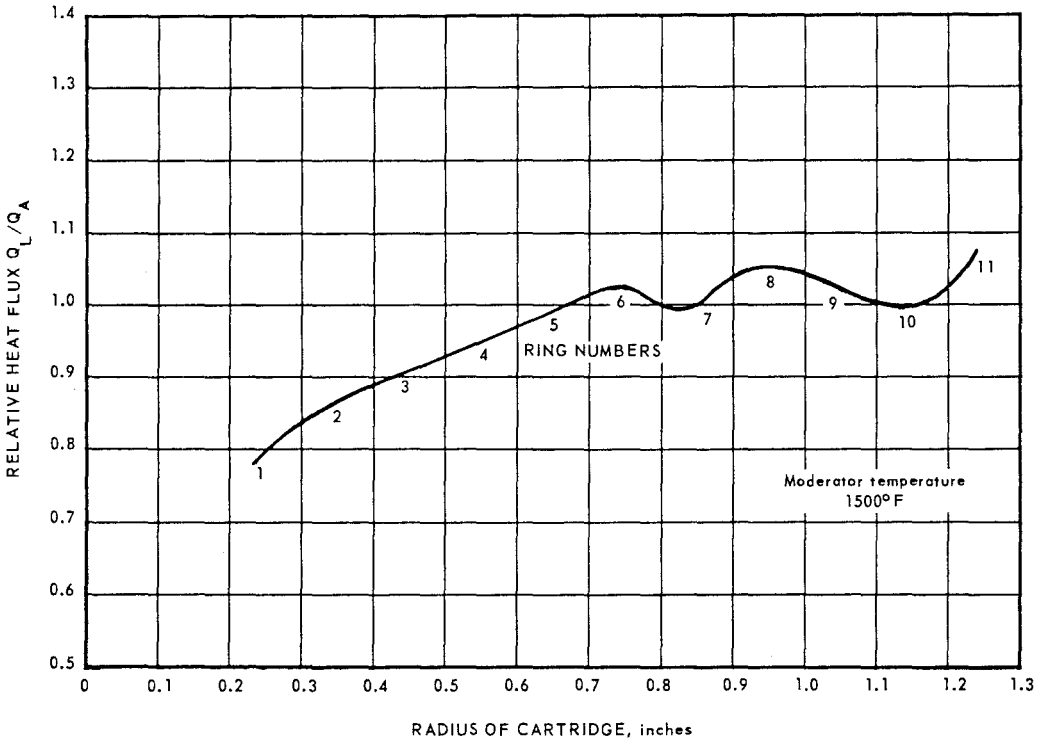

Fig. 3.6 – Insert 1 relative (local to average)

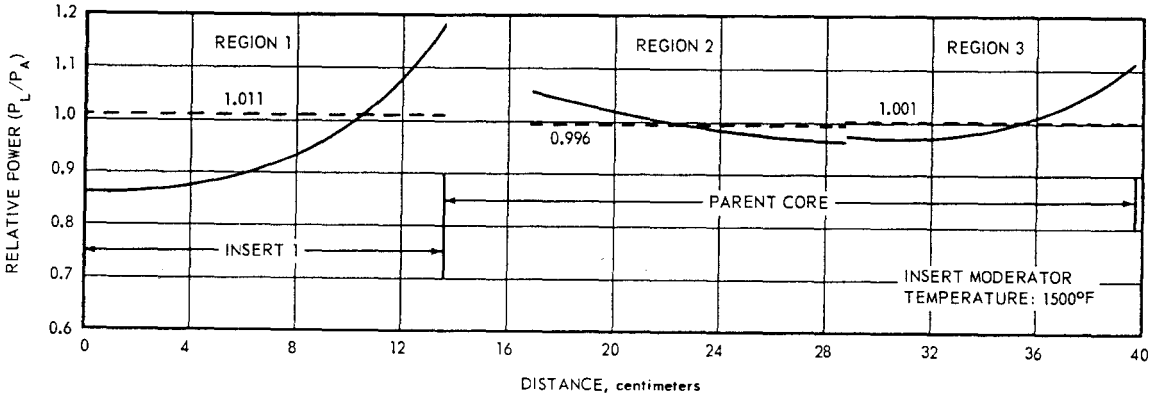

Fig. 3.7 – HTRE No. 2 and insert 1 gross radial-power

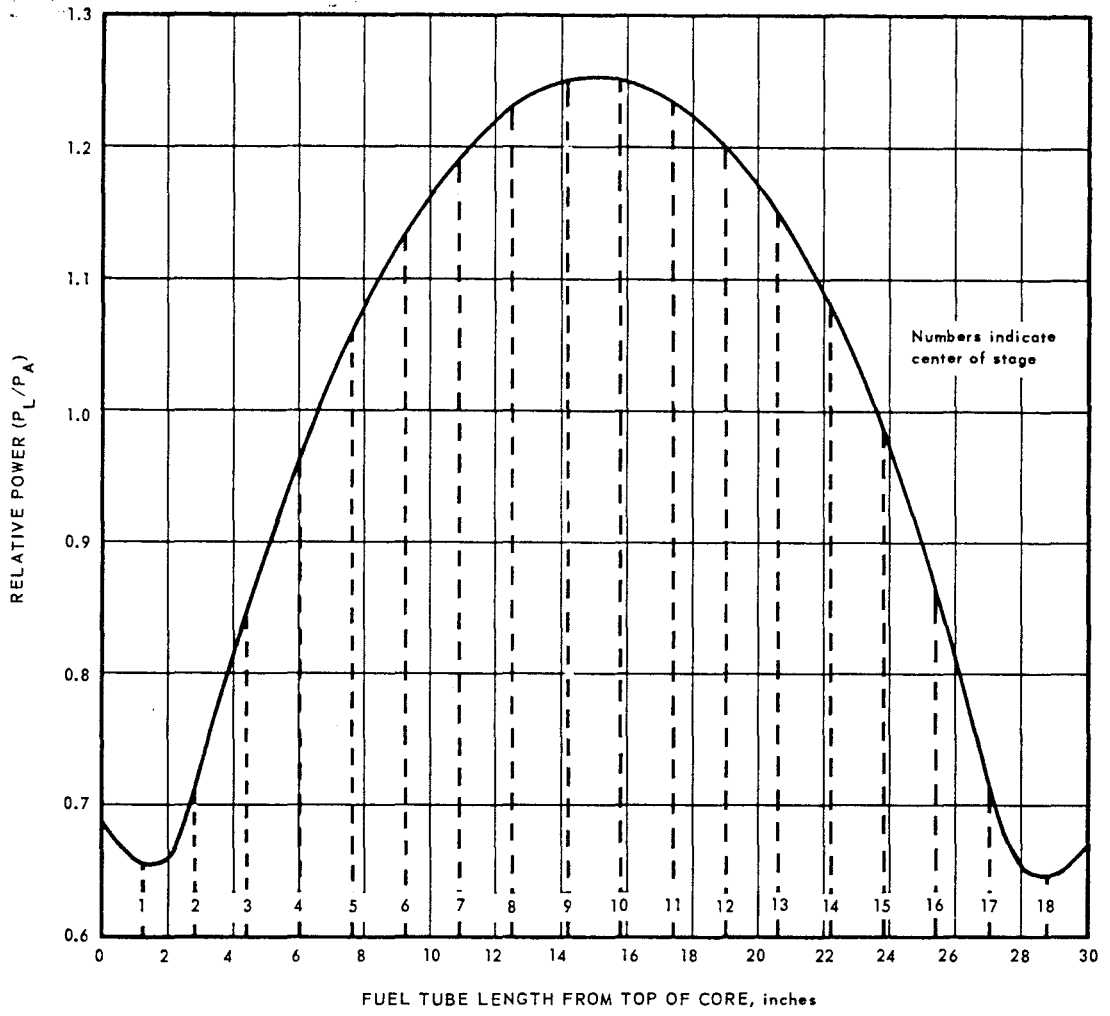

Fig. 3.8 – Insert 1 longitudinal relative power versus fuel tube length

The procedure used to determine the absolute heat fluxes involved the determination of the individual stage absolute-average heat flux per unit of reactor power, and the application of the relative heat flux curve Q_L/Q_A for the particular ring.

The absolute-average heat flux per unit of reactor power for the individual stages was determined and presented in four steps: (1) total insert power, (2) insert tube powers, (3) insert tube stage powers, and (4) heat fluxes.

From the regional data, obtained in the nuclear analysis, and from the gross radial power curve (Figure 3.7), a conversion constant was computed which allows the determination of the absolute powers in the regions of the reactor (three regions were used in the analysis of HTRE No. 2). This constant is given by:

$$k = \frac{P}{[\overline{P}i\ Vi + \overline{P}_1\ V_1 + \overline{P}_2\ V_2]} \qquad (1)$$

where P is the total reactor power $\overline{P}i$, $\overline{P}_1$, $\overline{P}_2$ are the weighted average relative powers in the particular region of the reactor (the regions are defined by the nuclear analysis). Vi, V_1, V_2 are the volume of the regions in cubic inches.

The total insert power is then given by $Pi = k\ \overline{P}i\ Vi$.

3.2.3 TUBE POWER

In order to calculate individual insert fuel-tube and stage powers, it is necessary to know the weighted-average relative specific power in the portion of the insert where the fuel cell is located.

In the actual insert No. 1, there were seven fuel tubes among which the total insert power had to be apportioned. To determine this apportionment, the following factors were used: (1) the geometry of the insert (radii of fuel-tube regions), (2) a curve of mass-of-fuel versus the radius of the insert used in the calculations (Figure 3.9), and (3) the assumption that the outer six tubes of the insert were each generating the same total power (see Figure 3.10 for the HTRE No. 2 and insert 1 layout).

Figure 3.9 shows that for a two-pound loading (907 gms), assumed for the nuclear analysis, the center tube had an equivalent radius of 5.08 centimeters in the homogenized insert. The equivalent radius of the entire insert hexagonal was 13.338 centimeters. With these radii as boundaries, the insert was considered to be divided into two annular regions, the center region containing tube No. 1, and the outer region containing tubes No. 2 through 7 (see Figure 3.10). From the gross radial-power curve (Figure 3.7), the weighted-average relative specific power for these two regions is given by:

$$\text{Center Cell (1)} \quad \frac{\int_0^{5.08} P_i dr_i}{\int_0^{5.08} dr_i} = \overline{P}i_1 = 0.8748 \qquad (2)$$

$$\text{Outer Cells (2 thru 7)} \quad \frac{\int_{5.08}^{13.338} P_i dr_i}{\int_{5.08}^{13.338} dr_i} = \overline{P}i_2 = 1.0275$$

The ratios

$$\frac{Pi_1}{Pi} \text{ and } \frac{Pi_2}{Pi}$$

then, give the relative amounts of power generated per unit volume in each region of the insert 1. If the average absolute specific power of the insert 1 is given by:

$$|\overline{P}i| = k\ \overline{P}i \quad k \text{ from (1)} \qquad (3)$$

then the average absolute specific power (power-per-unit volume) in each insert 1 region is:

$$|\overline{P}i_1| = |\overline{P}i|\ \frac{\overline{P}i_1}{\overline{P}i} = \frac{P \times \overline{P}i}{\overline{P}i \times Vi + \overline{P}_1 \times V_1 + \overline{P}_2 \times V_2} \qquad (4)$$

$$\times \frac{\overline{P}i_1}{Pi} = C \times P \times \overline{P}i_1$$

$$|\overline{P}i_2| = |\overline{P}i|\ \frac{\overline{P}i_2}{Pi} = C \times P \times \overline{P}i_2 \qquad (5)$$

where

$$C = \frac{1}{\overline{P}i\ Vi + \overline{P}_1\ V_1 + \overline{P}_2\ V_2}$$

and is constant.

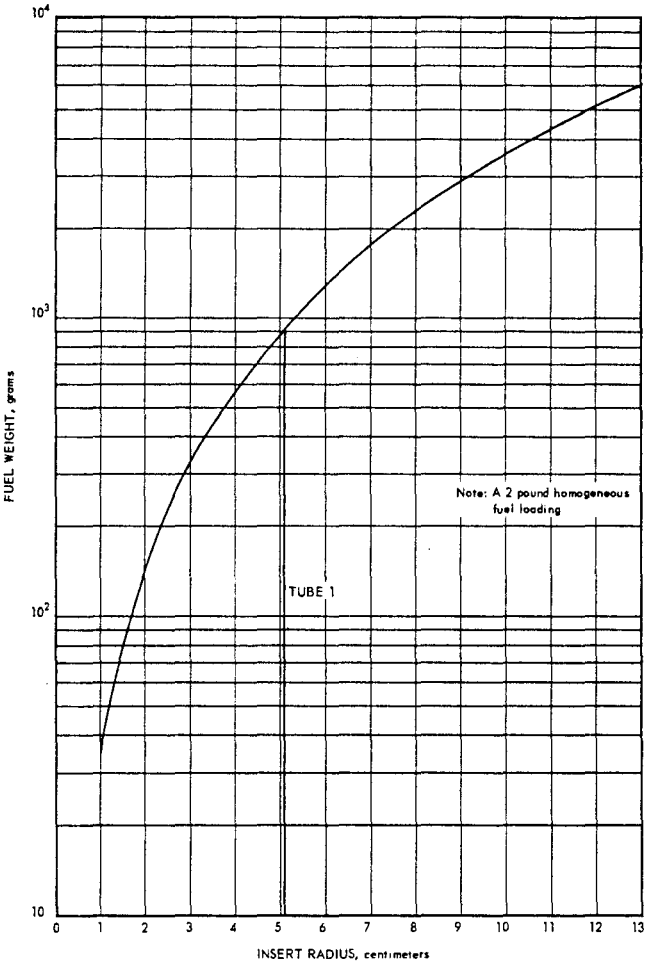

Fig. 3.9 – Insert 1 fuel loading versus insert radius

3.2.4 STAGE POWER

Applying the longitudinal power distribution curve shown in Figure 3.8 directly to equation (4) and (5) gives the stage absolute-average specific power.

$$|\overline{P}i_{1m}| = C \times P \times \overline{P}i_1 \times B_m \qquad m = \text{stage } 1,\ 2,\ 3,\ \text{etc.} \tag{6}$$

$$|\overline{P}i_{2m}| = C \times P \times \overline{P}i_2 \times B_m \tag{7}$$

where the B_m's are the values taken from Figure 3.8 and represent the ratios of the individual stage average specific power to the cartridge average specific power, values which have been determined experimentally in the HTRE No. 1 critical mockup.

The total power per stage is then obtained by multiplying equations (6) and (7) by the stage value (V_S), giving:

$$|Pi_{1m}| = C \times P \times \overline{P}i_1 \times B_m \times V_S \tag{8}$$

$$|Pi_{2m}| = C \times P \times \overline{P}i_2 \times B_m \times V_S \tag{9}$$

NOTE:
Numbers 1 through 7 indicate fuel cells of Insert 1B

Numbers 1 through 30 indicate fuel cells of HTRE No. 2

Numbers 31 through 60 indicate control rod positions

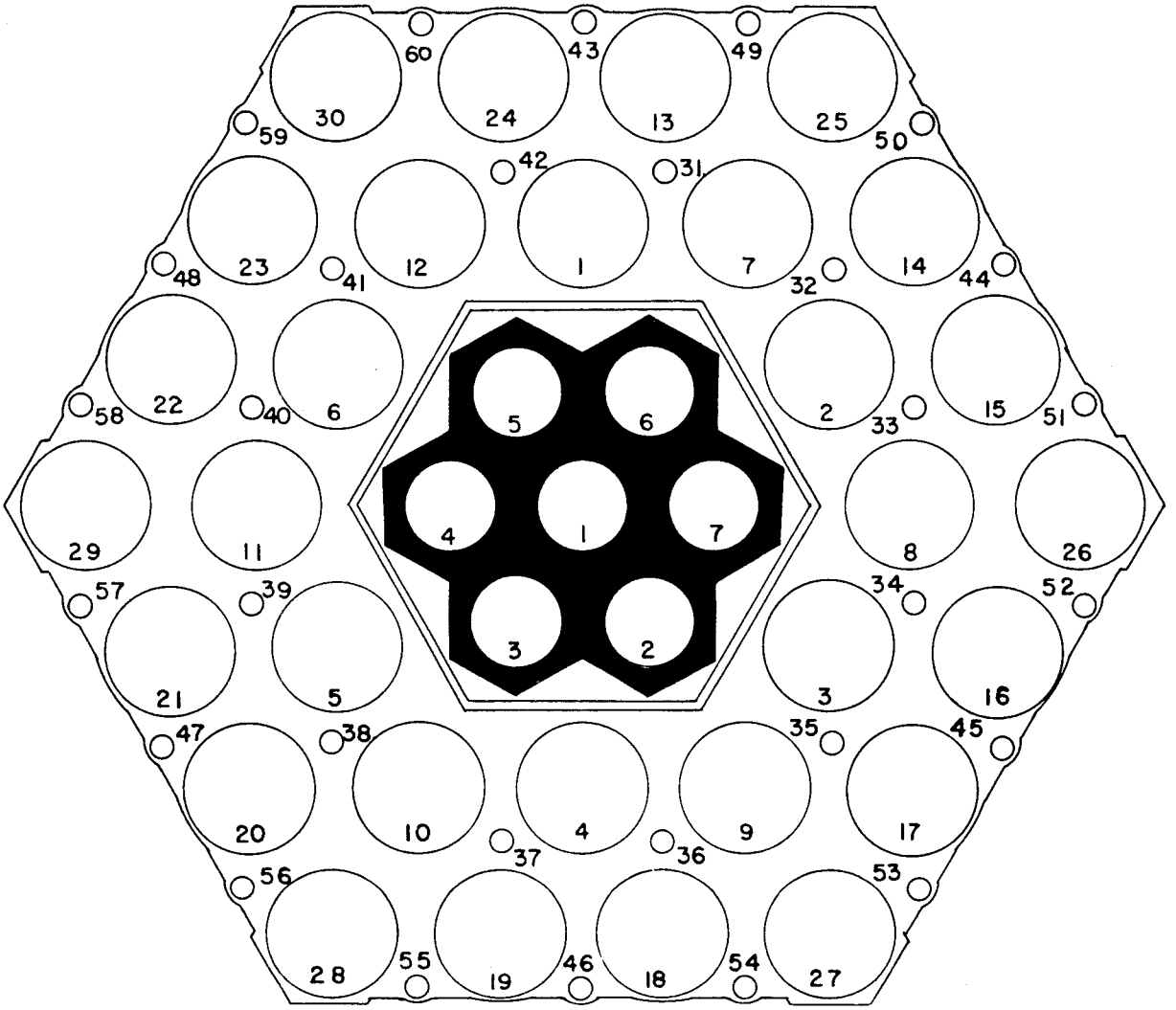

Fig. 3.10 – HTRE No. 2 and insert 1B core cross section

3.2.5 HEAT FLUX

The average heat flux (Q_A) per stage is then obtained by dividing equations (8) and (9) by the total area for heat transfer for a stage, giving:

$$Q_{a1} = \frac{C\ P\ \bar{P}i_1\ B_m\ V_s}{A} \tag{10}$$

in units of power per unit area

$$Q_{a2} = \frac{C\ P\ \bar{P}i_2\ B_m\ V_s}{A} \tag{11}$$

Finally, the heat flux for the particular ring of the stage is found by applying the numbers (D_n) obtained from Figure 3.6, which are ratios of the local (ring) heat flux to the stage average heat flux, to equations (10) and (11), giving:

$$Q_{1n} = \frac{Q_{a1} \times Q_{1n}}{Q_{a1}} = Q_{a1} \times D_n = \frac{C\ P\ \bar{P}i_1\ B_m\ V_s\ D_n}{A} \tag{12}$$

$$Q_{2n} = \frac{Q_{a1} \times Q_{2n}}{Q_{a1}} = Q_{a1} \times D_n = \frac{C\ P\ \bar{P}i_2\ B_m\ V_s\ D_n}{A} \tag{13}$$

where for,

$m = 1$, n = ring No. 1, 2, etc.

$m = 2$, n = ring No. 1, 2, etc.

$m = 18$, n = ring No. 1, 2, etc.

In equations (12) and (13), for the HTRE No. 2 with insert 1,

$$C = 2.832 \times 10^{-6}$$

$$V_s = 343.209\ cm^3 = 20.9\ in.^3$$

Values of B_m, and D_n for fuel cartridge are listed in Table 3.3. Values of heat flux for the ring, stages, and tube are given in Table 3.4, and are given as a constant which must be multiplied by the total reactor operating power (P) to give "Q" in watts per square inch. Detailed specifications for the fuel cartridge are given in Table 3.1. For comparison Table 3.5 shows values of heat flux from the solid moderator. These values are based on the assumption that all heat generated in the hydrided zirconium was removed from the inner surface.

The heat flux numbers used in a detailed thermodynamic evaluation of the HTRE No. 2 parent reactor and insert 1 were previously published.[2] The results of this analysis are presented in graphical form in Figures 3.11 and 3.12. The temperatures are presented as a function of the longitudinal fuel element position and are computed for a total reactor power of 8 megawatts. The outer annulus gap between fuel element and moderator at temperature was calculated to be between 0.122 and 0.128 inch.

TABLE 3.3

VALUES OF B_m AND D_n

Stage No.	Ratios Of Individual Stage Average Specific Power To Cartridge Average Specific Power, (B_m)	Ring No.	Ratios Of Local (Ring) Heat Flux To The Stage Average Heat Flux, (D_n)
1	0.66	1	0.785
2	0.711	2	0.865
3	0.85	3	0.915
4	0.96	4	0.945
5	1.053	5	0.985
6	1.13	6	1.035
7	1.19	7	0.992
8	1.23	8	1.055
9	1.251	9	1.025
10	1.250	10	0.990
11	1.23	11	1.055
12	1.2		
13	1.145		
14	1.072		
15	0.98		
16	0.86		
17	0.71		
18	0.64		

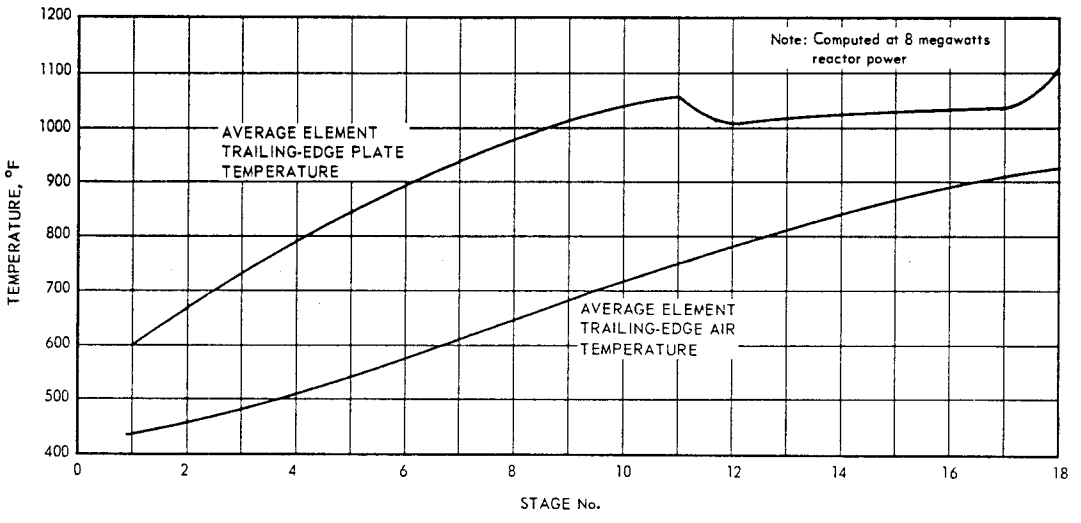

Fig. 3.11 – Comparison of HTRE No. 2 fuel element and air temperatures

[redacted]

TABLE 3.4

INSERT 1 HEAT FLUX VALUES FOR FUEL ELEMENTS

Ring No.	Area, sq. in.^{2}	Stage 1	Stage 2	Stage 3	Stage 4	Stage 5	Stage 6	Stage 7	Stage 8	Stage 9
1	1.9233	3.060 x 10^{-6}	3.2970 x 10^{-6}	3.9415 x 10^{-6}	4.4516 x 10^{-6}	4.8828 x 10^{-6}	5.2400 x 10^{-6}	5.5180 x 10^{-6}	5.7035 x 10^{-6}	5.8010 x 10^{-6}
2	2.8498	3.372 x 10^{-6}	3.6328 x 10^{-6}	4.3432 x 10^{-6}	4.9052 x 10^{-6}	5.3800 x 10^{-6}	5.7735 x 10^{-6}	6.0785 x 10^{-6}	6.2850 x 10^{-6}	6.3920 x 10^{-6}
3	3.7763	3.567 x 10^{-6}	3.8431 x 10^{-6}	4.5942 x 10^{-6}	5.1890 x 10^{-6}	5.6915 x 10^{-6}	6.1100 x 10^{-6}	6.4315 x 10^{-6}	6.6455 x 10^{-6}	6.7620 x 10^{-6}
4	4.7028	3.684 x 10^{-6}	3.9690 x 10^{-6}	4.7448 x 10^{-6}	5.3590 x 10^{-6}	5.8775 x 10^{-6}	6.3075 x 10^{-6}	6.6430 x 10^{-6}	8.8665 x 10^{-6}	6.9835 x 10^{-6}
5	5.6293	3.840 x 10^{-6}	4.1367 x 10^{-6}	4.9457 x 10^{-6}	5.5855 x 10^{-6}	6.1270 x 10^{-6}	6.5750 x 10^{-6}	6.9240 x 10^{-6}	7.1570 x 10^{-6}	7.2785 x 10^{-6}
6	6.5558	4.035 x 10^{-6}	4.3470 x 10^{-6}	5.1965 x 10^{-6}	5.8690 x 10^{-6}	6.4375 x 10^{-6}	6.9090 x 10^{-6}	7.2755 x 10^{-6}	7.5200 x 10^{-6}	7.6485 x 10^{-6}
7	7.4808	3.867 x 10^{-6}	4.1663 x 10^{-6}	4.9808 x 10^{-6}	5.6250 x 10^{-6}	6.1705 x 10^{-6}	6.6220 x 10^{-6}	6.9730 x 10^{-6}	7.2080 x 10^{-6}	7.3305 x 10^{-6}
8	8.3981	4.113 x 10^{-6}	4.4309 x 10^{-6}	5.2970 x 10^{-6}	5.9825 x 10^{-6}	6.5620 x 10^{-6}	7.0425 x 10^{-6}	7.4160 x 10^{-6}	7.6650 x 10^{-6}	7.7960 x 10^{-6}
9	9.3138	3.996 x 10^{-6}	4.3050 x 10^{-6}	5.1465 x 10^{-6}	5.8125 x 10^{-6}	6.3755 x 10^{-6}	6.8415 x 10^{-6}	7.2053 x 10^{-6}	7.4475 x 10^{-6}	7.5745 x 10^{-6}
10	10.2203	3.859 x 10^{-6}	4.1580 x 10^{-6}	4.9708 x 10^{-6}	5.6140 x 10^{-6}	6.1580 x 10^{-6}	6.6080 x 10^{-6}	6.6030 x 10^{-6}	7.1930 x 10^{-6}	7.3160 x 10^{-6}
11	11.1192	4.1131 x 10^{-6}	4.4309 x 10^{-6}	5.2970 x 10^{-6}	5.9825 x 10^{-6}	6.5620 x 10^{-6}	7.0425 x 10^{-6}	7.4160 x 10^{-6}	7.6650 x 10^{-6}	7.7960 x 10^{-6}
1	1.9233	3.5946 x 10^{-6}	3.8724 x 10^{-6}	4.6295 x 10^{-6}	5.2285 x 10^{-6}	5.7350 x 10^{-6}	6.1545 x 10^{-6}	6.4810 x 10^{-6}	6.6990 x 10^{-6}	6.8135 x 10^{-6}
2	2.8498	3.9610 x 10^{-6}	4.2670 x 10^{-6}	5.1012 x 10^{-6}	5.7615 x 10^{-6}	6.3190 x 10^{-6}	6.7815 x 10^{-6}	7.1395 x 10^{-6}	7.3820 x 10^{-6}	7.5080 x 10^{-6}
3	3.7763	4.1900 x 10^{-6}	4.5140 x 10^{-6}	5.3965 x 10^{-6}	6.0945 x 10^{-6}	6.6850 x 10^{-6}	7.1740 x 10^{-6}	7.5540 x 10^{-6}	7.8055 x 10^{-6}	7.9420 x 10^{-6}
4	4.7028	4.3273 x 10^{-6}	4.6618 x 10^{-6}	5.5730 x 10^{-6}	6.2940 x 10^{-6}	6.9035 x 10^{-6}	7.4085 x 10^{-6}	7.8025 x 10^{-6}	8.0650 x 10^{-6}	8.2020 x 10^{-6}
5	5.6293	4.5105 x 10^{-6}	4.8588 x 10^{-6}	5.8090 x 10^{-6}	6.5605 x 10^{-6}	7.1960 x 10^{-6}	7.7225 x 10^{-6}	8.1330 x 10^{-6}	8.4060 x 10^{-6}	8.5490 x 10^{-6}
6	6.5558	4.7394 x 10^{-6}	5.1060 x 10^{-6}	6.1040 x 10^{-6}	6.8935 x 10^{-6}	7.5610 x 10^{-6}	8.1150 x 10^{-6}	8.5455 x 10^{-6}	8.8325 x 10^{-6}	8.9835 x 10^{-6}
7	7.4808	4.5426 x 10^{-6}	4.8935 x 10^{-6}	5.8500 x 10^{-6}	6.6070 x 10^{-6}	7.2475 x 10^{-6}	7.7775 x 10^{-6}	8.1905 x 10^{-6}	8.4660 x 10^{-6}	8.6100 x 10^{-6}
8	8.3981	4.8310 x 10^{-6}	5.2045 x 10^{-6}	6.2220 x 10^{-6}	7.0270 x 10^{-6}	7.7075 x 10^{-6}	8.2715 x 10^{-6}	8.7105 x 10^{-6}	9.0030 x 10^{-6}	9.1570 x 10^{-6}
9	9.3138	4.6936 x 10^{-6}	5.0565 x 10^{-6}	6.0450 x 10^{-6}	6.8270 x 10^{-6}	7.4885 x 10^{-6}	8.0360 x 10^{-6}	8.4630 x 10^{-6}	8.7472 x 10^{-6}	8.7473 x 10^{-6}
10	10.2203	4.5334 x 10^{-6}	4.8838 x 10^{-6}	5.8385 x 10^{-6}	6.5940 x 10^{-6}	7.2330 x 10^{-6}	7.7615 x 10^{-6}	7.7555 x 10^{-6}	8.4485 x 10^{-6}	8.5930 x 10^{-6}
11	11.1192	4.8310 x 10^{-6}	5.2045 x 10^{-6}	6.2220 x 10^{-6}	7.0270 x 10^{-6}	7.7075 x 10^{-6}	8.2715 x 10^{-6}	8.7105 x 10^{-6}	9.0030 x 10^{-6}	9.1570 x 10^{-6}

		Stage 13	Stage 14	Stage 15	Stage 16	Stage 17	Stage 18	Stage 10	Stage 11	Stage 12
1	1.9233	5.3095 x 10^{-6}	4.9709 x 10^{-6}	4.5443 x 10^{-6}	3.9878 x 10^{-6}	3.2923 x 10^{-6}	2.9677 x 10^{-6}	5.7965 x 10^{-6}	5.7035 x 10^{-6}	5.5645 x 10^{-6}
2	2.8498	5.8505 x 10^{-6}	5.4775 x 10^{-6}	5.0075 x 10^{-6}	4.3942 x 10^{-6}	3.6278 x 10^{-6}	3.2702 x 10^{-6}	6.3865 x 10^{-6}	6.2850 x 10^{-6}	6.1315 x 10^{-6}
3	3.7763	6.1890 x 10^{-6}	5.7940 x 10^{-6}	5.2970 x 10^{-6}	4.6482 x 10^{-6}	3.8375 x 10^{-6}	3.4592 x 10^{-6}	6.7565 x 10^{-6}	6.6455 x 10^{-6}	6.4860 x 10^{-6}
4	4.7028	6.3915 x 10^{-6}	5.9840 x 10^{-6}	5.4705 x 10^{-6}	4.8006 x 10^{-6}	3.9634 x 10^{-6}	3.5726 x 10^{-6}	6.9775 x 10^{-6}	6.8665 x 10^{-6}	6.6985 x 10^{-6}
5	5.6293	6.6620 x 10^{-6}	6.2370 x 10^{-6}	5.7020 x 10^{-6}	5.0040 x 10^{-6}	4.1311 x 10^{-6}	3.7238 x 10^{-6}	7.2730 x 10^{-6}	7.1570 x 10^{-6}	6.9820 x 10^{-6}
6	6.5558	7.0005 x 10^{-6}	6.5540 x 10^{-6}	5.9915 x 10^{-6}	5.2580 x 10^{-6}	4.3408 x 10^{-6}	3.9128 x 10^{-6}	7.6420 x 10^{-6}	7.5200 x 10^{-6}	7.3365 x 10^{-6}
7	7.4808	6.7090 x 10^{-6}	6.2815 x 10^{-6}	5.7425 x 10^{-6}	5.0395 x 10^{-6}	4.1604 x 10^{-6}	3.7504 x 10^{-6}	7.3245 x 10^{-6}	7.2080 x 10^{-6}	7.0320 x 10^{-6}
8	8.3981	7.1355 x 10^{-6}	6.6810 x 10^{-6}	6.1070 x 10^{-6}	5.3595 x 10^{-6}	4.2470 x 10^{-6}	3.9884 x 10^{-6}	7.7900 x 10^{-6}	7.6650 x 10^{-6}	7.4785 x 10^{-6}
9	9.3138	6.9325 x 10^{-6}	6.4905 x 10^{-6}	5.9335 x 10^{-6}	5.2070 x 10^{-6}	4.2988 x 10^{-6}	3.8750 x 10^{-6}	7.5685 x 10^{-6}	7.4475 x 10^{-6}	7.2650 x 10^{-6}
10	10.2203	6.6960 x 10^{-6}	6.2690 x 10^{-6}	5.7310 x 10^{-6}	5.0290 x 10^{-6}	4.1520 x 10^{-6}	3.7427 x 10^{-6}	7.3100 x 10^{-6}	7.1930 x 10^{-6}	7.0175 x 10^{-6}
11	11.1192	7.1355 x 10^{-6}	6.6810 x 10^{-6}	6.1070 x 10^{-6}	5.3595 x 10^{-6}	4.4247 x 10^{-6}	3.9884 x 10^{-6}	7.7900 x 10^{-6}	7.6650 x 10^{-6}	7.4785 x 10^{-6}
1	1.9233	6.2360 x 10^{-6}	5.8385 x 10^{-6}	5.3375 x 10^{-6}	4.6840 x 10^{-6}	3.8670 x 10^{-6}	3.4857 x 10^{-6}	6.8080 x 10^{-6}	6.6990 x 10^{-6}	6.5355 x 10^{-6}
2	2.8498	6.8715 x 10^{-6}	6.4335 x 10^{-6}	5.8815 x 10^{-6}	5.1610 x 10^{-6}	4.2610 x 10^{-6}	3.8410 x 10^{-6}	7.5015 x 10^{-6}	7.3820 x 10^{-6}	7.2020 x 10^{-6}
3	3.7763	7.2690 x 10^{-6}	6.8055 x 10^{-6}	6.2215 x 10^{-6}	5.4595 x 10^{-6}	4.5074 x 10^{-6}	4.0630 x 10^{-6}	7.9360 x 10^{-6}	7.8055 x 10^{-6}	7.6180 x 10^{-6}
4	4.7028	7.5070 x 10^{-6}	7.0285 x 10^{-6}	6.4255 x 10^{-6}	5.6385 x 10^{-6}	4.6552 x 10^{-6}	4.1962 x 10^{-6}	8.1955 x 10^{-6}	8.0650 x 10^{-6}	7.8680 x 10^{-6}
5	5.6293	7.8250 x 10^{-6}	7.3260 x 10^{-6}	6.6975 x 10^{-6}	5.8775 x 10^{-6}	4.8522 x 10^{-6}	4.3738 x 10^{-6}	8.5425 x 10^{-6}	8.4060 x 10^{-6}	8.2010 x 10^{-6}
6	6.5558	8.2225 x 10^{-6}	7.6980 x 10^{-6}	7.0375 x 10^{-6}	6.1755 x 10^{-6}	5.0985 x 10^{-6}	4.5958 x 10^{-6}	8.9760 x 10^{-6}	8.8325 x 10^{-6}	8.6170 x 10^{-6}
7	7.4808	7.8805 x 10^{-6}	7.3780 x 10^{-6}	6.7450 x 10^{-6}	5.9190 x 10^{-6}	4.8867 x 10^{-6}	4.4050 x 10^{-6}	8.6035 x 10^{-6}	8.4660 x 10^{-6}	8.2590 x 10^{-6}
8	8.3981	8.3810 x 10^{-6}	7.8470 x 10^{-6}	7.1735 x 10^{-6}	6.2950 x 10^{-6}	5.1970 x 10^{-6}	4.6846 x 10^{-6}	9.1495 x 10^{-6}	9.0030 x 10^{-6}	8.7835 x 10^{-6}
9	9.3138	8.1425 x 10^{-6}	7.6235 x 10^{-6}	6.9695 x 10^{-6}	6.1160 x 10^{-6}	5.0490 x 10^{-6}	4.5514 x 10^{-6}	8.8895 x 10^{-6}	8.7472 x 10^{-6}	8.5340 x 10^{-6}
10	10.2203	7.8650 x 10^{-6}	7.3635 x 10^{-6}	6.7315 x 10^{-6}	5.9070 x 10^{-6}	4.8768 x 10^{-6}	4.3960 x 10^{-6}	8.5860 x 10^{-6}	8.4485 x 10^{-6}	8.2425 x 10^{-6}
11	11.1192	8.3810 x 10^{-6}	7.8470 x 10^{-6}	7.1735 x 10^{-6}	6.2950 x 10^{-6}	5.1970 x 10^{-6}	4.6846 x 10^{-6}	9.1495 x 10^{-6}	9.0030 x 10^{-6}	8.7835 x 10^{-6}

[redacted]

TABLE 3.5

INSERT 1 HEAT FLUX VALUES FOR SOLID MODERATOR

Stage No.	Watts Watt, in.^{2}
1	3.983×10^{-6}
2	4.292×10^{-6}
3	5.129×10^{-6}
4	5.793×10^{-6}
5	6.351×10^{-6}
6	6.829×10^{-6}
7	7.181×10^{-6}
8	7.422×10^{-6}
9	7.543×10^{-6}
10	7.543×10^{-6}
11	7.422×10^{-6}
12	7.241×10^{-6}
13	6.909×10^{-6}
14	6.472×10^{-6}
15	5.914×10^{-6}
16	5.190×10^{-6}
17	4.284×10^{-6}
18	3.862×10^{-6}

Note: Area of stage = 11.724 in.^{2}

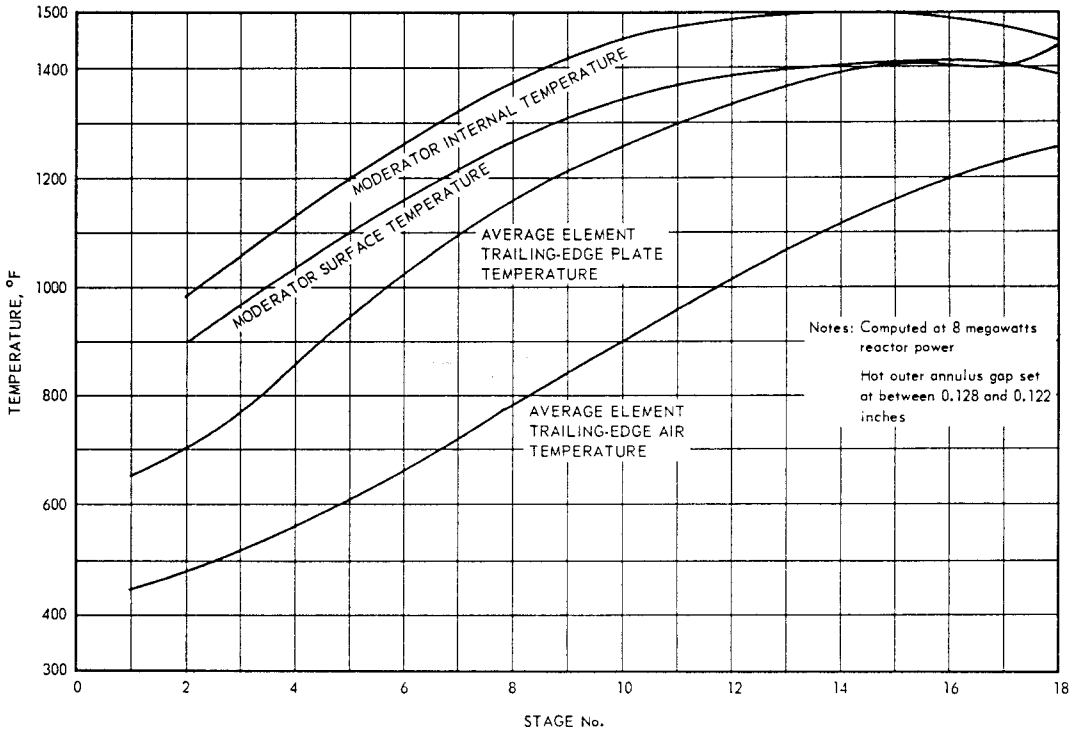

Fig. 3.12 – Temperature profiles of insert 1

3.3 REFERENCES

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2. Evans, R. C., "HTRE No. 2 Insert 1B Operations Report," GE-ANPD, XDC 57-10-26, October 1957.
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4. Pincock, G., "Operation 'BOOT' Test Results," GE-ANPD, DC 58-7-728, July 27, 1958.
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4. CERAMIC TEST INSERTS

Following the tests of the metallic inserts, tests of ceramic inserts were begun. Results of the first of these tests (insert 2B) indicated the need for higher power density and thus led to the subsequent design and testing of the parent insert L2 in which nine cartridges (eight ceramic and one metallic) were tested. Table 4.1 is a summary of ceramic tests performed in the HTRE No. 2.

TABLE 4.1

SUMMARY OF CERAMIC TESTS IN HTRE NO. 2 REACTOR

Insert No.	Fuel Tube, I.D. Clad	Nominal Maximum Insert Temperature, °F	Maximum Power Density, Btu/in^3-sec[a]	Test Time, hr
2B	bare	2550	1.86	25
		2830		100
L2A1	bare	2500	2.57	100
L2A2	bare	2700	2.51	94
L2E1	Al_2O_3	2500	2.69	106
L2E2	Al_2O_3	2500	2.75	46
		2600		99
L2E3	ZrO_2	2500	3.37	102
		2600		102
L2E4	b	4400	3.15	(10 min.)
L2E5	bare	3700	3.22	2
L2E6	ZrO_2	2500		50
		2600		54
		2650	2.82	56
		2700		193
		2750		3

[a]Based on nuclear analysis.

[b]Bare, ZrO_2 and Al_2O_3 clad tubes were used.

4.1 INSERT 2B

Insert 2B was devised to evaluate the use of ceramics as reactor core components adaptable to nuclear propulsion aircraft. It was designed to fit into the hexagonal test hole of the HTRE No. 2 parent core.

4.1.1 DESCRIPTION AND TEST OBJECTIVE

Insert 2B was divided into six triangular sections, with each section subdivided longitudinally into 11 layers. Seven support rods were attached at the top of the insert to a perforated stainless steel support plate. The insert was attached to the insert plug by six hangar rods. The fueled tubes consisted of BeO, UO_2, and Y_2O_3. The stacked insert is shown in Figure 4.1.

Primary purpose of the insert 2B tests was to evaluate the insert materials for mechanical stability after the materials had been subjected to fuel element temperature of 2750°F. Complete test procedures and results are described in reference 1, and section 6.

4.1.2 INSERT 2B NUCLEAR ANALYSIS

A series of critical experiments using mockups of insert 2B was performed to establish design characteristics with reference to power generation and gamma secondary heat generation in the insert. The data obtained were used to perform the thermodynamic analysis for this insert and were used particularly to specify orificing of the fuel tubes to produce a flat radial temperature profile in the insert. Primary considerations in the critical experiment effort were the determination of the power distribution which would exist in the insert fuel tubes. The power distribution data were corrected from nuclear analysis to establish distribution data which would exist in the insert at temperature. The total power generation in the insert used in the nuclear analysis was 10 percent of HTRE No. 2 power generation.

The data resulting from the nuclear analysis of the HTRE No. 2 with insert 2B is given in reference 1.

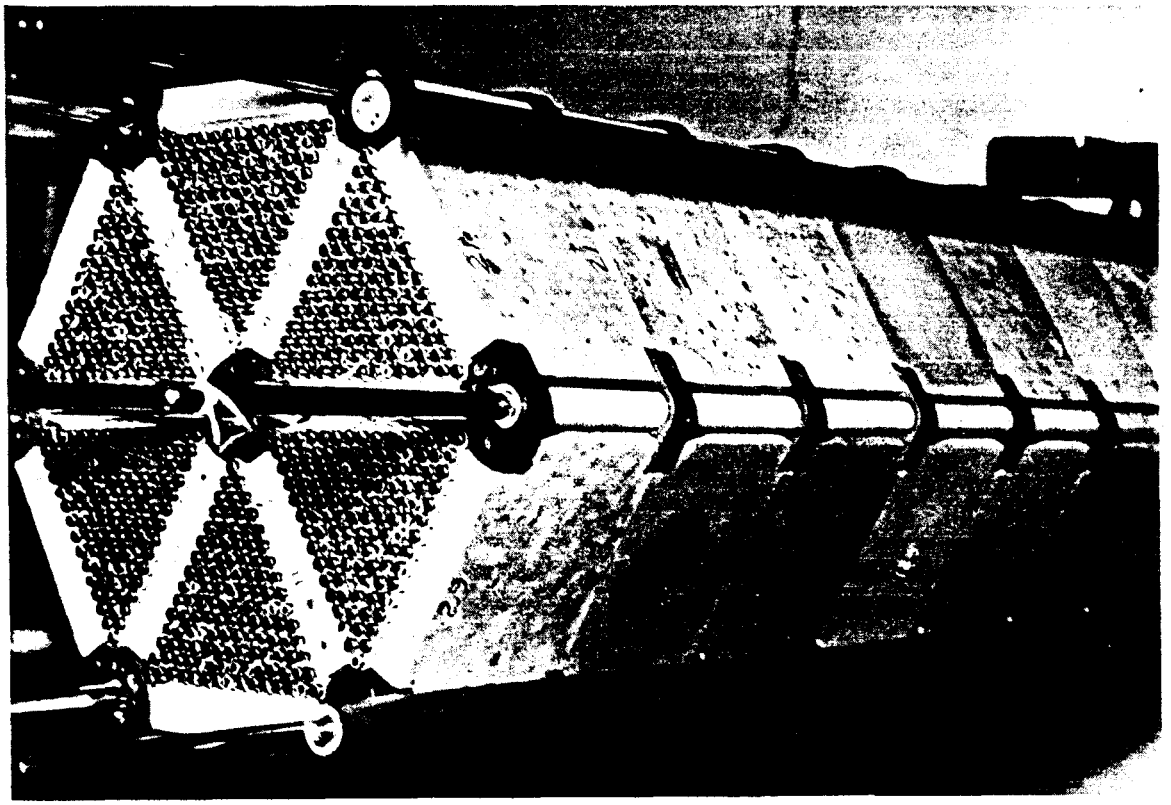


Fig. 4.1 – Insert 2-B completely stacked (Neg. CO9589)

4.2 PARENT INSERT L2

The operation of insert 2B made apparent the necessity of increased power densities.[2] During operation of 2B, considerable flow restriction was required to reach insert design temperatures. The resulting low fuel-tube air-weight flows represented a variable in the post-operational analysis which was difficult to evaluate.

Following the testing of insert 2B, a "parent insert" designated insert L2 was designed. This insert served as a vehicle to test several fuel cartridge configurations using various materials and was subsequently used for nine different cartridge tests. Figure 4.2 shows the insert, with parent plug and instrumentation section.

The parent insert L2 consisted of two concentric hexagonal aluminum cans with the central can forming the test hole. Between the two were positioned hexagonal layers of beryllium and water as moderator. Figures 4.3 and 4.4 show the design layout and cross section of the L2 insert. The outer-can wall measured 11.062 inches across flats and was 0.250-inch thick. The inner wall had an inside-across-flats nominal distance of 4.8625 inches and was 0.1875-inch thick. The cans were 52.300 inches long and were sealed together at the bottom with a 0.250-inch-thick aluminum plate. The top seal plate was provided with inlet and exit plenums for the water circulation system. The center beryllium slab, which extended to contact the top plate, produced a two-pass water circulation system. Figure 4.5 shows the water circulation system. The cooling water was supplied from the insert plug through flexible lines at a rate of approximately 15 gallons per minute. The volume between the two aluminum cans contained 90 percent beryllium and 10 percent water. These proportions, chosen from the results of the critical experiments, gave the best compromise between excess multiplication of the HTRE No. 2 insert assembly and power density in the ceramic fuel cartridge. The L2 can structure was supported from the insert plug by six stainless steel support rods which were spring-loaded at the plug end for alignment purposes.

This basic aluminum-beryllium structure with the central hexagonal test hole was used for many tests in which the test hole was loaded with the same or different configurations of fuel and moderator.

4.2.1 PARENT INSERT L2 NUCLEAR ANALYSIS

The design of parent insert L2 was based on the concept that a concentration of moderating material in an insert around a centrally located test hole would boost the fission flux and consequently boost the power generated in a fuel cartridge placed in the test hole. Several configurations for the basic L2 design were evaluated by nuclear analysis[3] and by critical experiments. The analysis and experiments involved the variation of materials contents of the insert to produce a final configuration which was the best K_{ex}-power density compromise. In the analysis, the center cell of the insert was loaded with BeO and U^{235}. Calculations were performed with the remaining regions of the insert-can filled with 70 percent beryllium and 30 percent water; 90 percent beryllium and 10 percent water; and 90 percent beryllium and no water (10% void). The calculations predicted the excess multiplication and the power density in the fueled region. A plot of these results appears on page 119 of reference 4.

To establish the final design of the L2 configuration a series of critical experiments was performed in the Low Power Test Facility at the Idaho Test Station. A mockup of the HTRE No. 2 core and reflector was installed in a rectangular water tank, and a mockup of the insert, with provisions for varying the materials contents, was installed in the HTRE No. 2 test hole. This structure was similar in final configuration to the L2 insert.

The test program called for the evaluation of three moderator (Be and H_2O) configurations, three configurations of the ceramic tube fuel cartridge, and an advanced metals

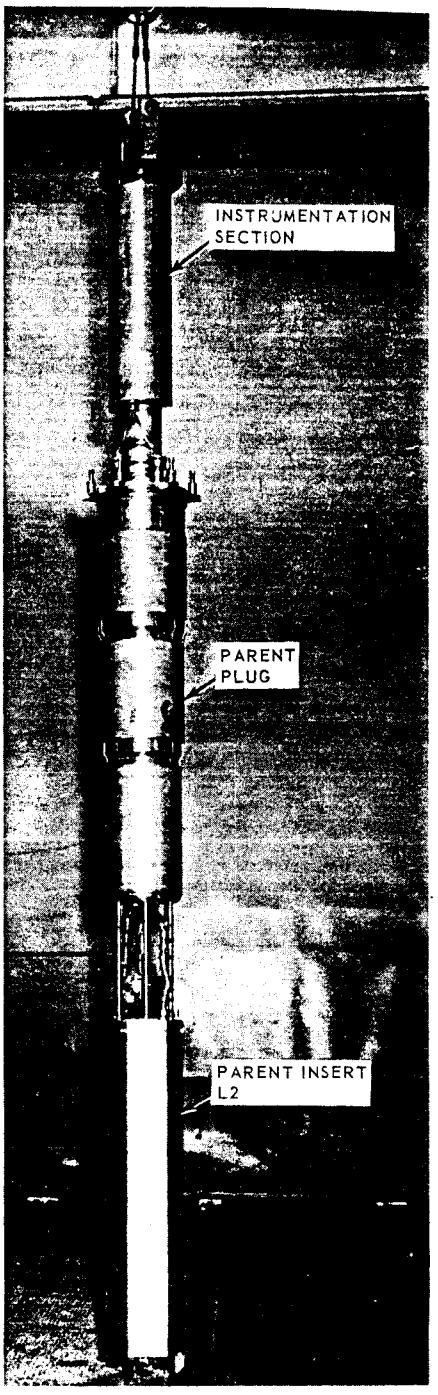
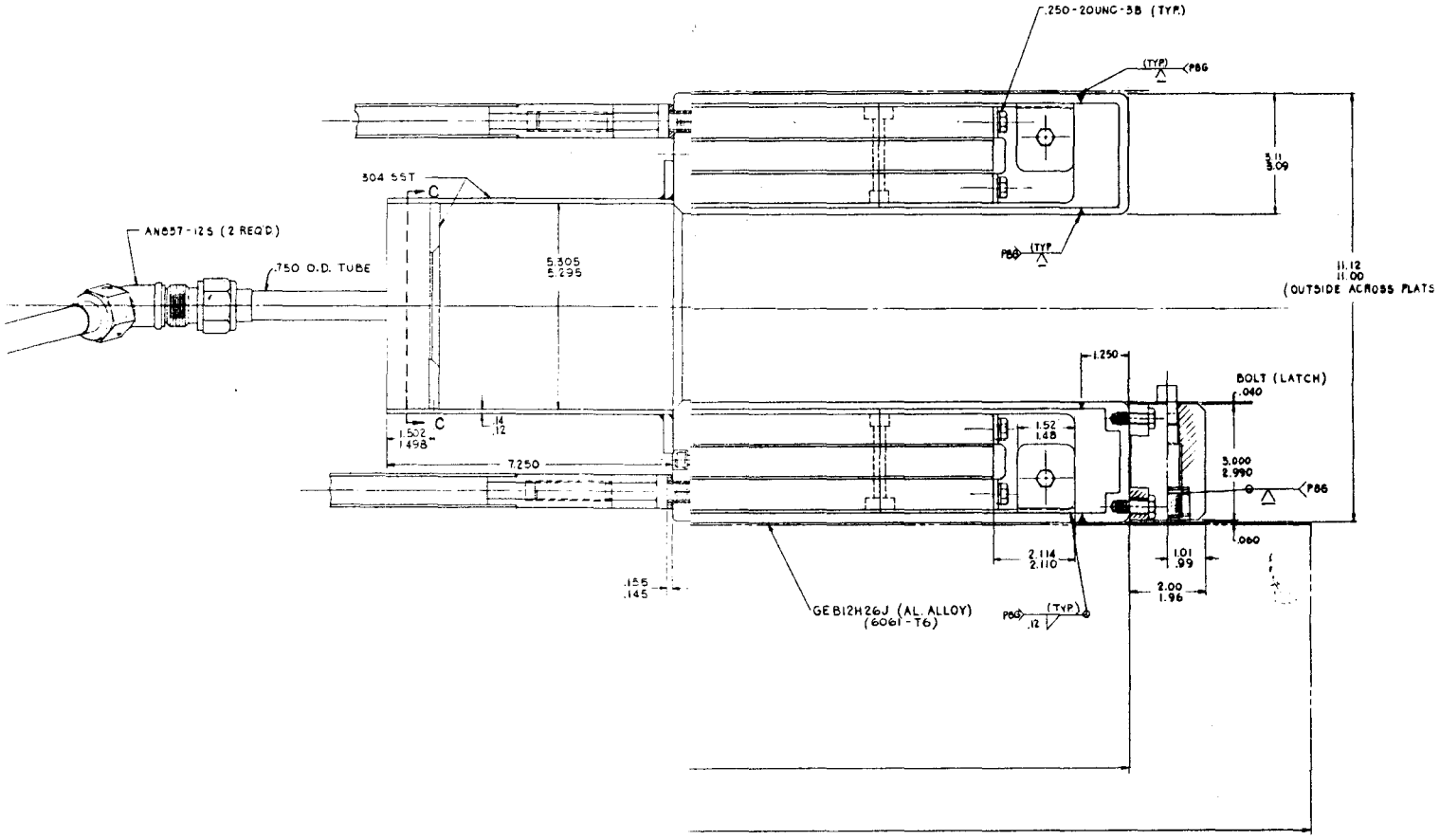


Fig. 4.2 – Parent insert L2 (Neg. C-14581)


.250-20UNC-3B (TYP.)
(TYP) PBG
304 SST
C
AN837-12S (2 REQD.)
.750 O.D. TUBE
5.305
5.295
5.11
5.09
PBG (TYP)
11.12
11.00
(OUTSIDE ACROSS FLATS
1.250
BOLT (LATCH)
.040
1.52
1.48
3.000
2.990
PBG
C
1.502
1.498
.14
.12
7.250
.060
2.114
2.110
1.01
.99
2.00
1.96
.155
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GE B12H26J (AL. ALLOY)
(6061-T6)
PBG (TYP)
.12

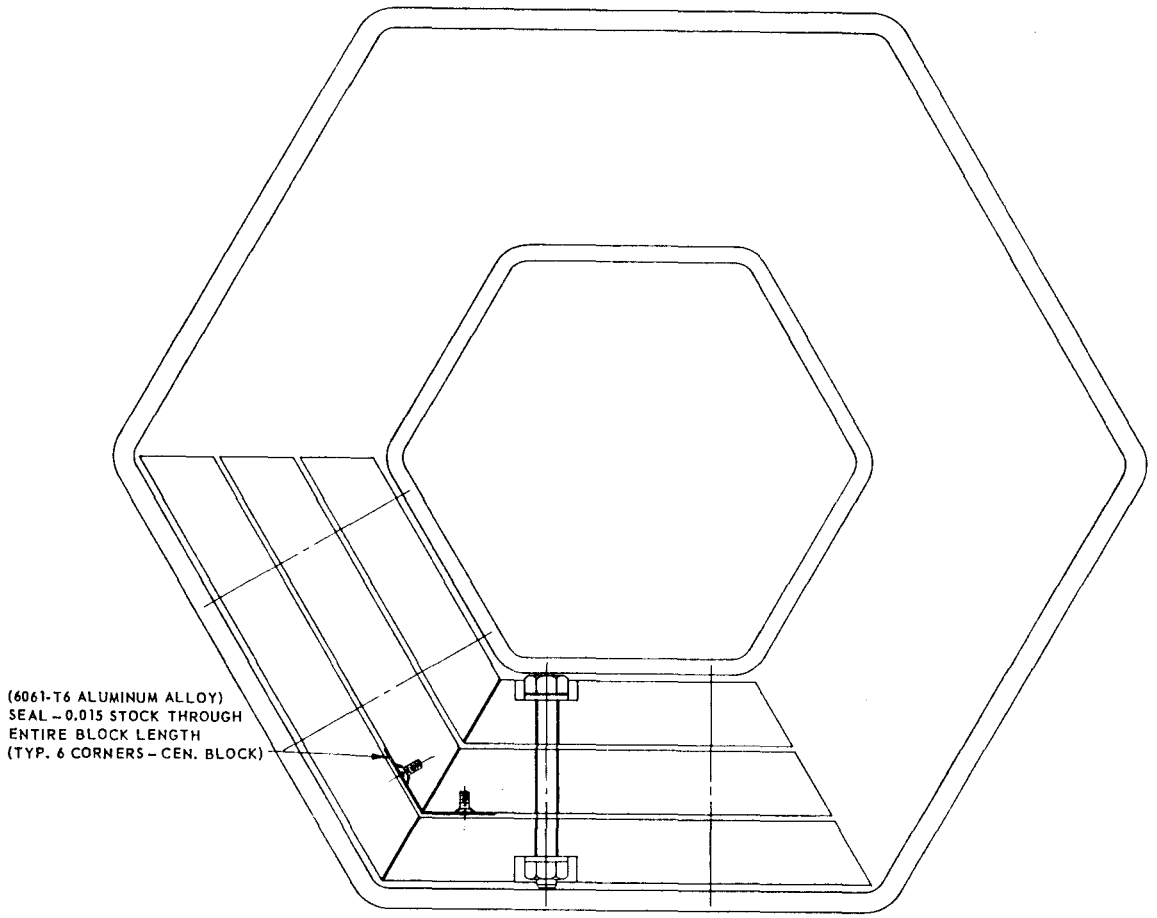

Fig. 4.4 – Parent-insert L2 cross section

fuel cartridge. In the configurations, the beryllium-water ratio and the fuel-tube content were varied to determine the best compromise between excess reactivity and power density in the fueled tubes.

From the data obtained, a final L2 configuration was established, with the reflector region containing 90 percent beryllium and 10 percent water and the ceramic fuel cartridge containing 61 fuel tubes. The K_{ex} for this configuration was measured 1.14 percent at 950°F with an average power density of 3.4 Btu/in^3-sec. Although the K_{ex} as measured was low, additional reactivity was obtained for operation by increasing the parent core water-moderator temperature to a nominal 150°F. At this temperature, the K_{ex} was predicted to be 1.6 percent which was sufficient to override the xenon developed during continuous operation.

The transverse, radial, and longitudinal relative power distributions measured in the insert fuel cartridge for the configuration chosen are presented on pages 121, 122, and 123, respectively, of reference 4. These distributions were applied to a thermodynamic analysis. The over-all core power distribution and the K_{ex} and power density as a function of beryllium are shown on pages 124 and 125, respectively, of reference 4.

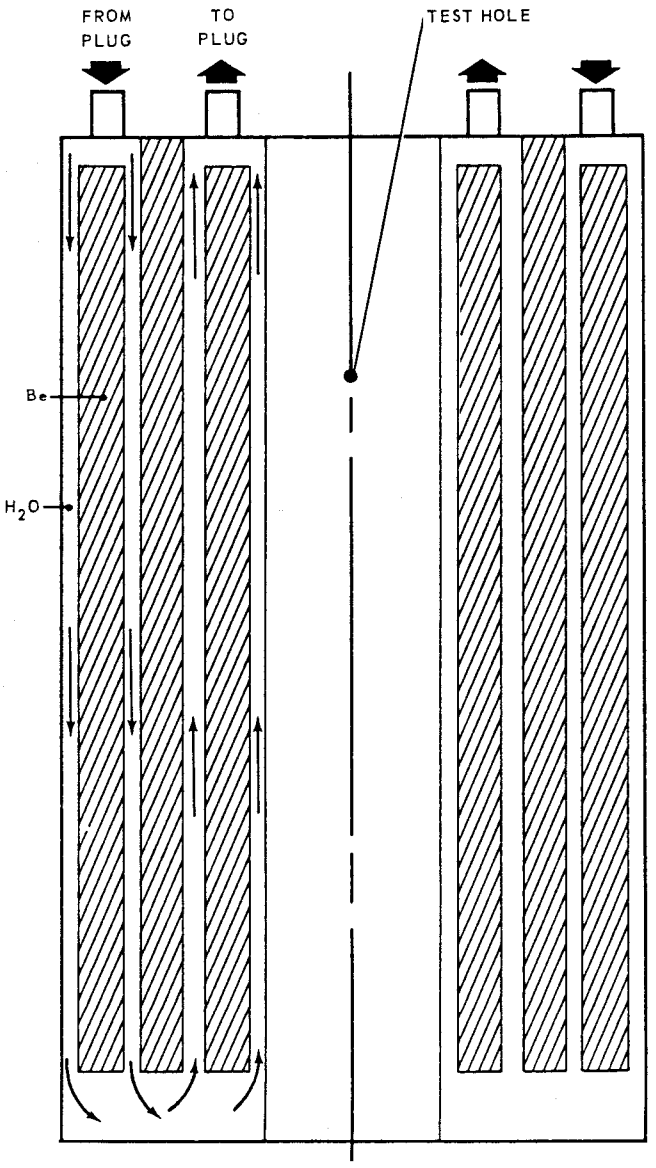

Fig. 4.5 – Water circulation system (insert L2)

4.3 PARENT INSERT TEST CARTRIDGES

Insert L2 was used for testing nine different insert cartridges: L2A1, L2A2, L2C1, L2E1, L2E2, L2E3, L2E4, L2E5, and L2E6.

4.3.1 CARTRIDGE L2C1

All of the cartridges were ceramic, with the exception of cartridge L2C1, which was a metallic concentric-ring fuel element cartridge. Cartridge L2C1 had a fuel sheet of a chromium-UO_2-titanium core clad with an iron-chromium-yttrium alloy. The fuel sheet was formed into 18 stages and positioned in concentric rings around a central hydrided zirconium moderator rod. Primary purpose of the cartridge L2C1 tests was to evaluate new materials and their performance in advance fuel cartridge designs. Details of the test are given in reference 5.Test results are summarized in section 6 of this report.

4.3.2 CARTRIDGE L2A1

The type A cartridges, both of essentially the same mechanical design, contained round fueled tubes of BeO and enriched U^{235}.

Cartridge L2A1 was a ceramic cartridge designed to make possible the evaluation of operational effects of water vapor corrosion on fueled BeO tubes. Such corrosion had been observed in insert 2B and in laboratory tests in which the fueled tubes operated at temperature for a prolonged period of time. Purposes of the cartridge test included the establishment of a realistic operating temperature for ceramic tubes in light of the corrosion problem, the evaluation of crystal growth effect on system performance, and the determination of flow variation with time and wall temperature. The secondary purpose of the test was to observe fission-product release from the fuel BeO tubes as a function of operating time in an attempt to establish a pattern for fission-product release for a full-scale ceramic reactor. Test procedures and results are more fully treated in reference 1, and in section 6 of this report.

4.3.3 CARTRIDGE L2A2

Cartridge L2A2 consisted of uncoated fueled and unfueled tubes. The tubes, cylindrical in shape and the same size as the L2A1 tubes, were placed into the parent insert L2.

Primary purposes of the cartridge tests were:

1. To provide a suitable source for the evaluation of the effectiveness of the electrostatic precipitator as a fission product filter. A material was sought which, when injected upstream of the precipitator, would give 90 percent removal efficiency.
2. To obtain further information pertaining to the release of fission products as a function of temperature from uncoated BeO fueled tubes.
3. To study atmospheric diffusion of fission products released from the IET stack under various meteorological conditions.

Section 6 summarizes the test results.

4.3.4 CARTRIDGE L2E1

The only other insert cartridge that used round fueled tubes was cartridge L2E1. It differed from the L2A1 and L2A2 cartridges in that the fueled tubes were coated on the inner surface with aluminum oxide (Al_2O_3).

This cartridge consisted of a bundle of round fueled and unfueled BeO (ceramic) tubes in a hexagonal pattern. The complete L2E1 cartridge was composed of a front reflector of unfueled BeO tubes, a fueled region, and a rear reflector of unfueled BeO tubes. Tubes in the fueled region were staggered so that no two tube joints lined up. An instrumentation disconnect, capable of handling 24 thermocouples, was fitted to the front end of the cartridge. An air orificing plate which measured and metered airflow was built into the cartridge disconnect. Complete test objectives, procedures, and results are described in reference 7, and in section 6.

4.3.5 CARTRIDGE L2E2

Cartridges L2E2, L2E3, L2E4, L2E5, and L2E6 all had hexagonal fueled tubes with round coolant holes through the center. The across-flats and inside diameter dimensions of the tubes were different as was the type of cladding used in each cartridge. The mechanical structure was essentially the same for these cartridges, with differences occurring only in the amount of insulation used and the type of aft retainer plate employed.

The L2E2 cartridge was similar to the L2E1 cartridge except that the fueled BeO tubes were hexagonal in shape. The fueled tubes were coated on the inside diameter with aluminum oxide. Thirty-nine thermocouples were used to measure fueled tube temperatures, discharge air temperature, and insulation linear temperature. Primary purposes of the cartridge tests were to evaluate the materials characteristics of the Al_2O_3 coating, to evaluate the over-all performance of the cartridge, and to determine fission product release as a function of temperature. Reference 1 provides a thorough description of test procedures and results, and section 6 summarizes the results.

4.3.6 CARTRIDGE L2E3

Cartridge L2E3 was similar to L2A1 and L2E1 except for two changes in the fueled tubes. The L2E3 tubes were hexagonal in shape with an inside diameter coating of zirconia instead of alumina. Primary purpose of the cartridge test was to further evaluate the ceramic tube (BeO) as carrier for fuel in nuclear reactors. Procedures and results of the test are given in reference 1. See section 6 for a summary of test results.

4.3.7 CARTRIDGE L2E4

Cartridges L2E4 and L2E5 were designed to permit hazard testing. General purpose of these tests was to determine the results of restricting or completely stopping the cooling air supply to a reactor under full nuclear power.

Cartridge L2E4 contained hexagonal fueled and unfueled BeO tubes. The fueled tubes had ZrO_2 and Al_2O_3 coatings. The center 18 tubes, not including the centermost tube, were plugged to prevent any flow of cooling air. In the test, the ceramic insert cartridge was operated for 10 minutes at a power level sufficient to cause portions of the plugged fuel region to melt. Purpose of the test was to evaluate the nature and propagation of this melt and to verify the ability to predict such phenomena. Reference 8 shows complete test procedures and results. Section 6 of this report summarizes test results.

4.3.8 CARTRIDGE L2E5

Cartridge L2E5 also had the center 18 tubes, not including the centermost tube, blocked. In this cartridge, however, approximately 10 percent of the normal airflow was permitted to pass through the tubes. Purposes of the cartridge test included determination of the effect of the partial airflow block, measurement of the release of fission products, and measurement of the downwind diffusion of these products under specific meteorological conditions. Details of the test are given in reference 9, and section 6.

4.3.9 CARTRIDGE L2E6

Cartridge L2E6 was a ceramic BeO fuel cartridge with hexagonally-shaped tubes. The tubes were coated on the inner surfaces with ZrO_2. The tests were conducted in order to evaluate the ZrO_2 coating at temperatures above ACT design conditions, and to better understand the fission-product-release behavior as a function of time and temperature. Reference 10 provides complete test procedures and results. Also, see section 6.

The nuclear characteristics of the L2 series of insert cartridges were essentially the same for all the cartridges. Small variations occurred in the radial power profiles depending upon the size of the fuel bundle and the void volume.

4.4 REFERENCES

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8. Pincock, G. D., "Test Results D101 - L2E4," GE-ANPD, DC 60-10-720, October 12, 1960.
9. Pincock, G. D., "Test Results D101 - L2E5," GE-ANPD, DC 60-11-719, November 23, 1960.
10. Field, J. H., and Foster, D. C., "Test Results D101 - L2E6," GE-ANPD, DC 61-6-704, June 12, 1961.

5. PARENT CORE OPERATION HISTORY

This section presents a summary of the HTRE No. 2 tests with emphasis on the data accumulated for the parent core.[1]

The power operation of HTRE No. 2 began in July 1957, and continued until March 1961. A total of 13 different inserts were tested. The test data from each reactor run (designated Initial Engine Test or IET) is presented in Table 5.1.

TABLE 5.1

TEST DATA

IET	Insert Test Assembly	Total Test Time, hr	Maximum Temperature, °F		Total Reactor Power Above 0.1 mw, mw-hr
			Parent Fuel	Parent Air	
8	1B	38	1400	970	7,785
10	2B	125	1750	1125	7,785
11	1C	100	1200	850	7,785
12	1D	2	1400	970	7,785
14	L2A1	100	1550	1040	1,356.71
15	L2C1	81	1350	930	975.93
17	L2E1	106	1400	970	1,191.62
19	L2E3	204	1700	1100	3,025.14
20	L2E2	145	1550	1040	1,830.63
21	L2A2[a]	48	1450	1000	348.35
22	L2E4	0.17	1550	1040	7.77
23	L2A2[a]	46	1450	1000	258.48
24	L2E5	2	1400	970	25.36
26	L2E6	356	1200	900	3,347.00
		1353.17			16,627.73

[a]Same test cartridge.

5.1 FUEL ELEMENTS

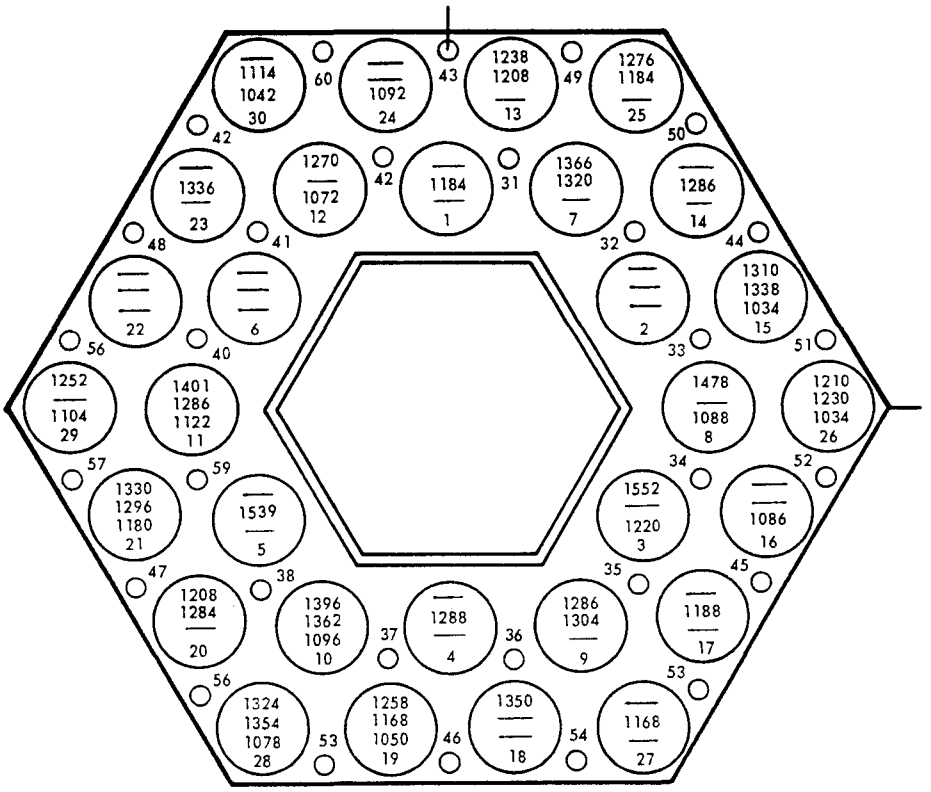
Following the insert 1D test, fuel tubes Nos. 10 and 11 from the parent-core were removed to determine the effect of the operations on the life of the fuel elements. Studies of these tubes indicated that, after 265 hours operation, less than one-third of the useful life of the fuel elements had been expended. This was based primarily on the oxide stringer penetration of the fuel element cladding material. After a total on-test operating time of

552 hours, which was after the insert L2E1 test, 25 of the 30 parent core fuel cartridges were replaced with new cartridges. Fuel burnup and samarium poisoning had caused the excess reactivity to fall low enough so that xenon poisoning caused the operation to become inefficient, necessitating the replacement. The parent core and shield plug were also replaced at this time because a moderator water leak had developed at the junction of one of the support tubes and the core. Because of the radiation level of the core, repairing the cracked weld was impractical.

After accumulation of a total of 997 on-test hours (445 hours since the partial refueling) thirteen of the parent core fuel cartridges were replaced. These cartridges included the five that had operated since the initial power test.

Following the second reloading of fuel, the parent core, with insert L2E6, was operated for 356 hours. This was the last test performed in HTRE No. 2.

Although the objective in the design of HTRE No. 2 was to obtain a gross flat radial temperature profile in the parent core during operation, there were variations from tube to tube. Based on the analysis of a normal operation, the maximum temperatures for each cartridge occurred at the 11th and/or 18th stage, and thermocouples were attached at these locations. Figure 5.1 shows a typical temperature distribution. The gross temperature variation between the inner and outer rows of fuel cartridges was attributed to the radial variation in power generation. The particular insert, which was being tested, did



Note:
Peak insert temperature = 2500°F.
Upper entry is 18th stage, 15th or 16th ring fuel temperature.
Middle entry is 11th stage, 14th ring fuel temperature.
Lower entry is discharge air temperature.

Fig. 5.1 – Gross parent core temperature survey during L2A1 test

not affect this significantly. A portion of the circumferential temperature variation within a row was also attributed to the same reason, as well as the fabrication tolerance effects on the fuel element ring spacing. The fuel and air temperature relationship for increasing power steps, the total HTRE No. 2 reactor power and exit air relationship, and the parent core fuel element and exit air temperature measurements during an endurance operation are presented on pages 132, 133, and 134, respectively, of reference 1. Although the data is shown as continuous, this particular operation was performed with approximately 25 shutdowns.

As mentioned earlier, five fuel cartridges had operated for a total of 997 hours on test or 13,280.73 megawatt-hours, total, above 0.1 megawatt. No deteriorating effects were observed, either during operation or upon post-test examination. Two of the five cartridges were sectioned after testing. Figures 5.2, 5.3, 5.4, and 5.5 show the 11th and 18th stages of these cartridges.

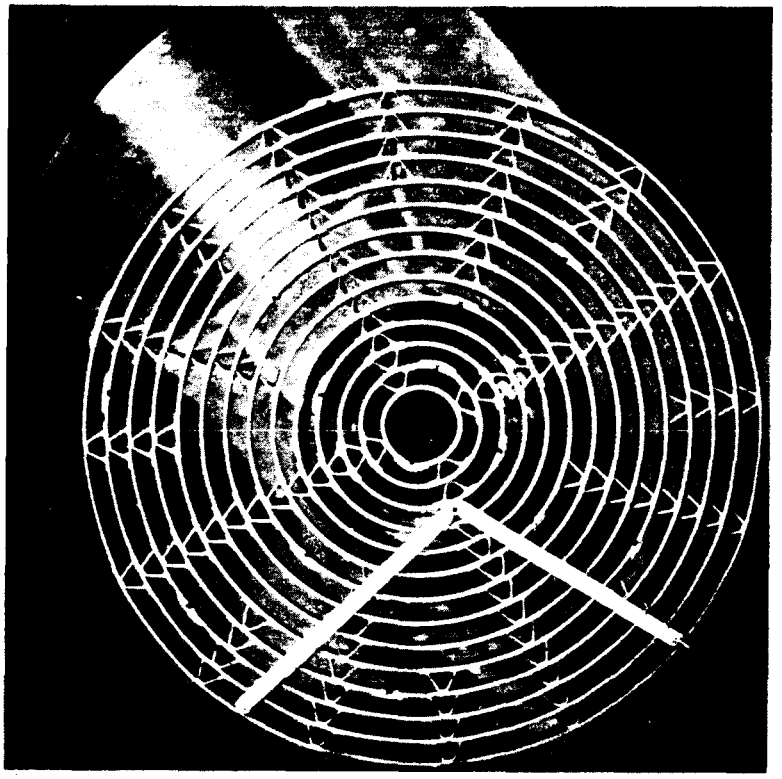


Fig. 5.2 – Downstream face of stage 11, cartridge 441 after 997 hours on test (Neg. U4199-28)

5.2 EXCESS REACTIVITY

The continuous operation of the HTRE No. 2 during a test averaged approximately 6 hours per day or 30 hours per week. This provided the most efficient utilization of manpower in view of the marginal value of excess reactivity for testing with the L2 parent insert. Typical plots of measured and computed values of fission product poison and fuel depletion are shown in Figures 5.6 and 5.7. Two typical days' operation are shown together with the applicable power histories. Agreement between measured and computed values were generally good except for the initial critical reading for each run.

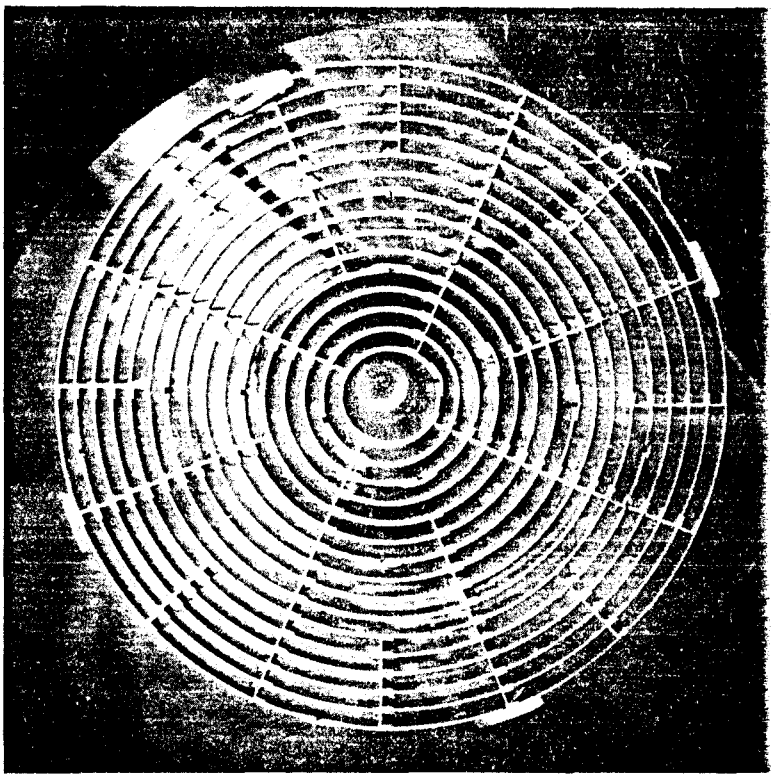


Fig. 5.3 – Upstream face of stage 18, cartridge 411 after 997 hours on test (Neg. U4199-41)

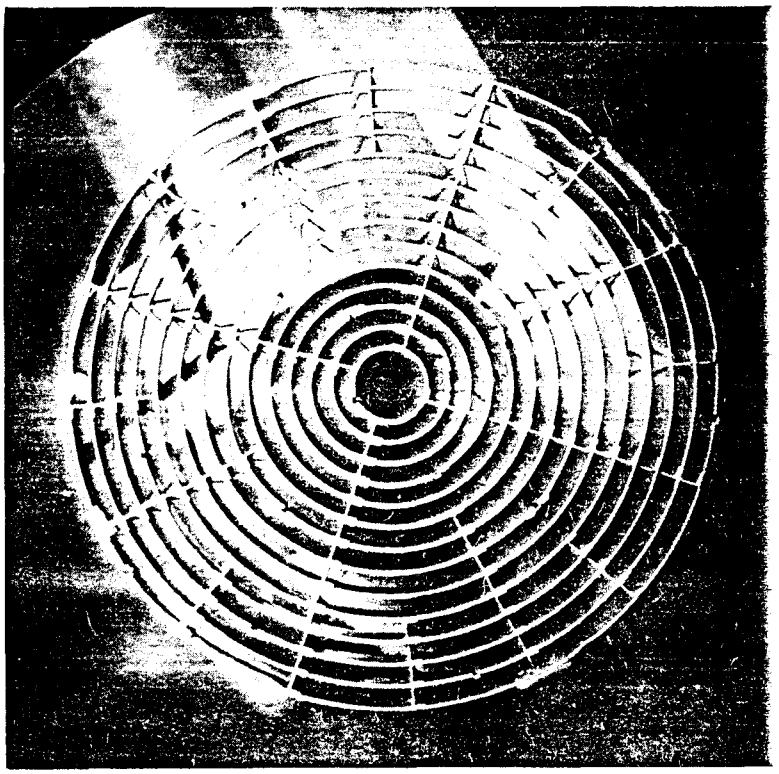


Fig. 5.4 – Upstream face of stage 11, cartridge 452 after 997 hours on test (Neg. U4199-39)

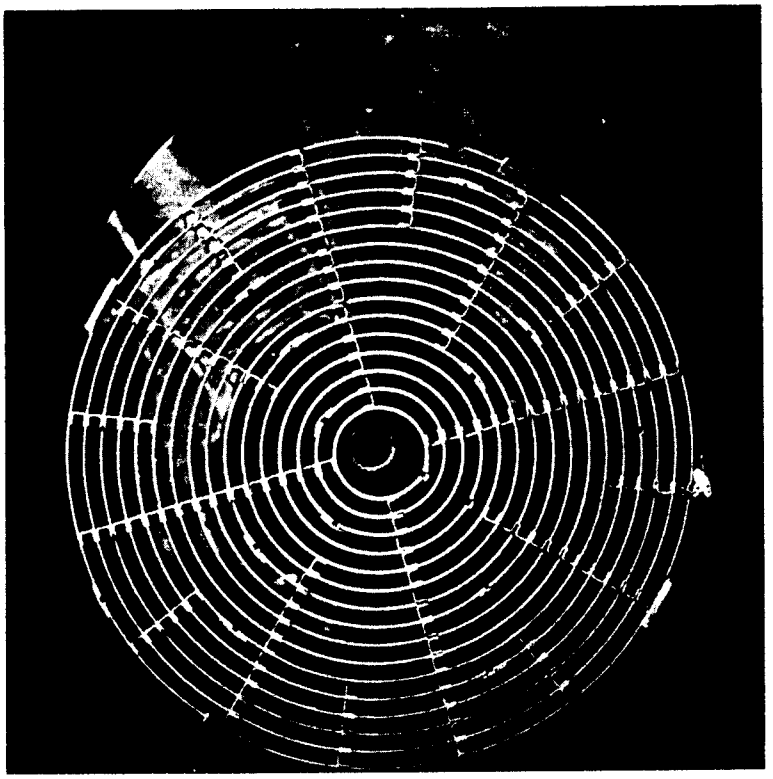


Fig. 5.5 – Upstream face of stage 18, cartridge 452 after 997 hours on test (Neg. U4198-44)

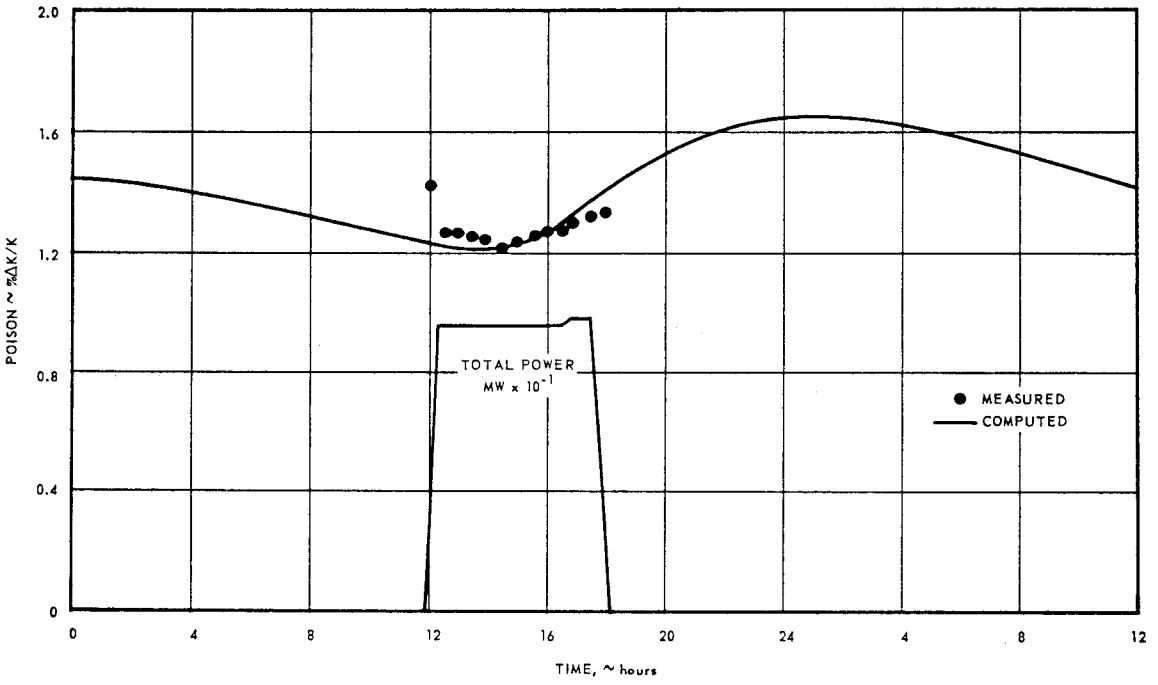


Fig. 5.6 – Xenon poisoning versus time during L2E6 cartridge test

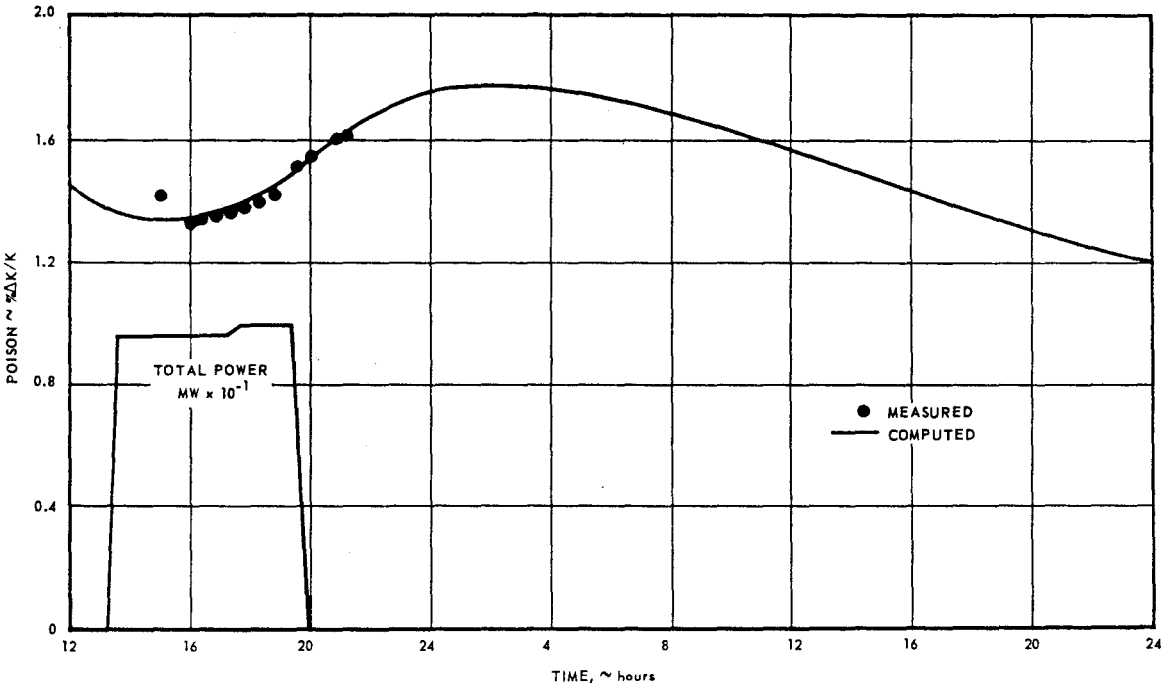


Fig. 5.7 – Xenon poisoning versus time during L2E6 cartridge test

5.3 REFERENCES

1. Blake, J. C., Flagella, P. N., et al., "HTRE No. 2 Design and Operational Summary," GE-NMPO, APEX-731, July 19, 1961.

6. SUMMARY OF TEST RESULTS

6.1 INSERT 2B

The primary purpose of the insert 2B experiment was to operate the HTRE No. 2 reactor system at powers and airflows sufficient to produce a maximum fuel-element temperature of 2750°F, and in a postoperational examination to evaluate the insert 2B materials for physical and mechanical stability. The following design specifications were established:

Maximum insert fuel tube temperature	2750°F
Maximum insert exit air temperature	2300°F
Moderator slab temperature	2000° to 2600°F
Support rod temperature, nominal	1200° to 1600°F
Fuel tube uranium concentration by weight	6 percent

The insert 2B test was the first in a series of ceramic tests. Very little was known, at the time of this test, about the operational characteristics of ceramics. The manufacturing and design analysis techniques were likewise in early stages of development.

6.1.1 POWER TESTING

Some difficulty was encountered in obtaining the design temperature in the insert at the specified power level. During the power operations, parent-core fuel-cartridge temperature limits were reached on the initial attempts to obtain insert design conditions. As a consequence, the HTRE No. 2 had to be returned to the hot shop for adjustment of flow to the insert, by plugging holes in the front orifice plate. Later, the insert was operated at a temperature of 2830°F maximum for the first time with an exit air temperature of 2300°F. The insert was operated at these design conditions for periods of time totalling 100 hours, after which the insert was returned to the hot shop for disassembly and examination of the insert.

As the insert was unstacked in the hot shop a white substance was observed on the inner diameter of the BeO tubes. This substance appeared in the largest quantity at the leading and trailing edges of the tubes, and was observed first on the trailing edge of the sixth stage. It increased slightly in quantity as the insert was unstacked, with a large increase occurring at the trailing edge of stage 10, where deposits apparently completely blocked flow passage in some tubes. Approximately 50 percent of the tenth-stage tubes of the insert showed large deposits with each of the six cells exhibiting the same uniformity of deposit. A number of the fueled tubes in the tenth stage were white in appearance indicating that UO_2 fuel had been lost from the tube.

The deposit of BeO crystals was not anticipated in preoperational predictions. No such deposits had been observed on fueled tubes tested in LITR and MTR under similar temperature conditions. The deposits observed in insert 2B were caused by a BeO hydrolysis produced by the action of water vapor in the air stream and BeO in the fueled tubes. The water air vapor picked up BeO in the cooler regions of the insert and carried it into the hot regions of the insert where it was deposited in the end sections of the fueled tubes.

The bleaching caused by UO_2 fuel loss in some of the fueled tubes indicated that temperatures on the order of 3000° to 3200°F were reached, probably near the end of the test.

6.1.2 FISSION FRAGMENT RELEASE FROM INSERT 2B

Throughout the testing of the 2B insert, the release of fission products was monitored and measured in various stack sampling points.

A general trend of increasing beryllium release existed throughout the 100 hours of endurance testing, with the steepest slopes in the data existing during the last 50 hours of testing. The increase was the result of excess temperature and surface corrosion.

After 25 hours of testing, the insert temperature was increased to approximately 2500°F, and the percent releases were essentially flat and varied from 0.005 percent to approximately 0.03 percent. When the temperature of the insert was increased to 2850°F, the release rate of several of the isotopes increased by factors of 2 to 3.

The short-lived I^{135} did not increase substantially, indicating a boiling-off or corrosion of stored long-lived iodines. Throughout the 100 hours of endurance testing, the gross stack release as measured at the vault increased by a factor of approximately 30.

6.2 INSERT 1C

Insert 1C was designed for a reactor experiment to evaluate unclad, slotted, hydrided zirconium as a core neutron-moderating material.

The insert was mounted in the HTRE No. 2 reactor core (A-4) which had previously logged approximately 160 hours at design powers (10 to 12 megawatts, 1760 megawatt hours) with the same set of fuel elements. K_{ex} had been measured at 2.5 percent.

The test program called for the gradual increase in reactor power and temperature until design conditions were reached. When the reactor power was increased to produce a 700°F insert moderator temperature, the stack gas radiation monitoring equipment recorded fresh fission products in the effluent, with a level of discharge of 700 curies per hour (measured 17 seconds after release) or 14 curies per hour (measured 10 minutes after release).

This unexpected release was probably the result of uranium oxide, deposited in the lower cocoon during the insert 2B operation, fissioning in the neutron-flux field below the reactor core. To assure that a fuel cartridge had not ruptured, the CTF was returned to the hot shop for examination and cleaning. In the hot shop, the cocoon was flushed with nitric acid and water, removing a total of 8.4 grams of U^{235}. It was established that no fuel-cartridge rupture had occurred.

While the CTF was in the hot shop, the 11-ring insert fuel cartridges were replaced with new 10-ring* fuel cartridges, and the four additional holes in each of the moderator cell rear-transition pieces were uniformly plugged. These changes were made because extrapolation of the preliminary power data indicated that parent core fuel-element temperatures would exceed 1750°F and the insert fuel cartridges would exceed 1900°F before the solid moderator reached the design temperature of 1200°F. When testing was resumed the reactor power level was increased to 11 megawatts to produce the 1200°F moderator temperature. At this power level, the release was measured at 380 curies per hour (measured 17 seconds after release) or 7.7 curies per hour (measured 10 minutes after release).

*The 10-ring elements have better performance characteristics than the 11-ring and also run 50° to 100°F cooler for the same total reactor power level.

The reactor was subsequently operated in intervals totalling 100.24 hours.

A series of reactor controls tests were performed during the 1C run to prove out the XD servosystem. Ramp functions of a 600°F change in fuel-element temperature, and 1-megawatt step inputs in power were introduced into the system with the servo controlling the reactor. The servosystem operated satisfactorily.

During the 100 hours of power testing, various thermocouple temperature data were recorded. Over the period of operation, the total reactor power required to maintain the design moderator temperature gradually dropped, as did the insert fuel-cartridge exit-air temperature and plate temperature. This indicated that the total power generated in the moderator was increasing or that an oxide film, known to build up on the moderator cooling slot surfaces, was altering the heat transfer coefficient. The resulting reduction in power necessary to maintain moderator temperature did not affect the operation procedure.

Moderator cell longitudinal temperature data were much as expected. The moderator-cell seventeenth-stage circumferential-temperature data indicated a circumferential temperature scallop of the order of 18 percent. Since temperature scallops of the order of 30 percent were observed in insert 1B, the center moderator bar of insert 1C effectively flattened the circumferential temperature distribution.

Fuel cartridge power was 16.5 ± 1.5 kilowatts per megawatt. The critical experiment had predicted 18 kilowatts per megawatt.

For the most part, the operation of insert 1C proceeded without incident. With the exception of the initial high release rates, due to uranium fissioning in the cocoon, no unusual observations were made throughout the 100 hours of endurance running.

A complicating feature of the operational data analysis was the fact that a 1200°F average moderator temperature was not achieved, although test conditions were satisfied by the 1200°F average temperature of the three hottest thermocouples. Since hydrogen loss may be a strong function of temperature level, evaluation of this test must take into account that the actual average temperature of the hot stage of the moderator was closer to 1100°F than 1200°F.

The lower-than-predicted average hot-stage temperature was attributed to the fact that the magnitude of the secondary or gamma heating is a strong function of the local fuel cartridge power.* Thus, a 20 to 25 percent circumferential fuel-cartridge power scallop could be reflected as a 15 to 20 percent scallop in gamma heating and consequently in moderator temperature. The temperature scallop was measured at approximately 18 percent.

6.3 INSERT 1D

Insert 1D was similar to inserts 1B and 1C in that it consisted of seven hexagonal hydrided-zirconium blocks. The six outer blocks contained similar fuel cartridges of the metallic concentric ring design. The center block contained a beryllium bar for the purpose of boosting the flux in the center of the insert and reducing the flux gradient across the diameter of the six outer blocks. Two of the outer blocks contained remotely

*During early critical experiment work with the HTRE No. 2 and insert 1A, health physics gamma sensitive films were exposed in the reactor. This film was shielded with aluminum and lead to reduce beta fluxes. The results indicated that where local circumferential scallops in fuel cartridge power existed there also existed a proportional scallop in the gamma flux.

operable pneumatic valves at the inlet end, for the purpose of reducing the airflow to the fuel elements during operation.

The purpose of the test was to evaluate the hazard of a "one-tube melt" in a full size reactor similar in design and containing the fuel element and moderator materials of insert 1D.

6.3.1 OPERATION

A controlled amount of fission-product activity was required for this experiment. The reactor was operated at power, not exceeding 1750°F on the insert fuel cartridge, until approximately 23 megawatt hours had been accumulated. The resulting fission product activity was then permitted to decay for approximately 48 hours. During the rise to power of this fission-product storage run, the airflow restrictor valves were operated in the opened and closed positions, and the temperature data were recorded.

Following the decay period of approximately 48 hours, all CTF, exhaust duct, and field sampling devices, as well as weather instruments, were made ready for the meltdown.

The reactor was again taken to a power level which produced a maximum fuel-cartridge temperature of 1750°F on tube 6_i.* After a period of 5 minutes, the airflow restrictor valve on tube 6_i was closed, causing the fuel cartridge to overheat and melt. This phase of the test ended with an automatic reactor scram resulting from short periods of increased reactivity recorded by the fission chambers. The short period was presumably caused by the delay neutrons present in the hot ducting due to the meltdown. The fission chambers were in the withdrawn position, virtually unshielded from any neutrons present in the hot torus region.

Pertinent data curves obtained during the actual insert testing, substantiated by many data points, were taken from oscillograph traces which provided an accurate time scale. Other data were taken from Brown Strip Chart Recordings, which result in considerably more time error. The instant of reactor scram, as indicated by the oscillograph trace of Rod Frame No. 3, is shown at 15 seconds.

All ninth-stage thermocouples on tube 6_i came back into service after the scram, when the tube temperature had dropped. This would indicate that the ninth stage had remained intact. Although the temperature had exceeded the limit of the recorders, they were not destroyed as were the eighteenth-stage thermocouples. The discharge air temperature ($T_{3.54i}$) thermocouple under tube 6_i also exhibits this phenomena.

The solid-moderator temperature at the eighteenth stage of tube 6_i showed practically no change throughout the extent of the test, indicating that the meltdown created no dangerous temperatures in the solid-moderator bar.

6.3.2 EXCESSIVE REACTIVITY

As a means of detecting the loss of fuel caused by the meltdown, K_{ex} values were obtained before and after the test. Since it was determined during the critical experiment that an insert fuel cartridge in tube 6 was worth approximately 1.24 percent $\Delta k/k$, the loss in K_{ex} could be a measure of the extent of damage or loss of fuel due to the meltdown. Three typical control rods were calibrated but each rod worth was 17 percent below the values obtained during the critical experiment. In spite of these differences, it was felt that the K_{ex} values obtained would serve as a rough evaluation of the extent of meltdown.

A comparison of like conditions before and after the test indicate that there was a loss of from 0.154 to 0.262 percent $\Delta k/k$. This represents 12 to 21 percent of the total worth of the fuel cartridge.

*Tube No. 6 of insert.

Visual examination of the fuel cartridge in the Radioactive Materials Laboratory indicated that approximately 40 percent of the fuel cartridge left the active core region. A considerable amount of the melted portions of the fuel elements was deposited in the tail cone section. This deposit could not have contributed significantly to the K_{ex}.

The K_{ex} results showed definite indication that there was a loss of fuel.

6.3.3 FISSION PRODUCT RELEASE

The fission-product release values were obtained from radiochemical separations on the carbon traps located at various points in the CTF and IET exhaust system.

The average total uranium release as indicated by the first 5 inches of the duct loop carbon traps was 1.47 grams with a mean deviation of ± 0.51 grams. This release represents approximately 3 percent of one fuel stage. Postoperation examination indicated that more uranium was lost and undoubtedly passed through the exhaust system. There was some indication from the detector equipment that possibly there were two bursts of activity. It could be that the majority of the uranium was released in chunks of slag and was not sampled by the sampling loops.

Studies of the distribution of the uranium deposition down the length of the carbon traps indicated that at least 95 percent of the uranium was found in the first 5 inches of the carbon.

As mentioned previously, it appeared that approximately 40 percent of the fuel cartridge left the active core region. This should not be construed to mean that 40 percent of the fuel cartridge actually melted. There is quite conclusive evidence from a close examination of the photographs of the deposits in the tail cone of tube 6_i that there are many rings of fuel in this deposit which did not melt. These unmelted portions would not have contributed to the release of fission products or uranium.

While the insert was in the hot shop, a piece of slag was taken from the deposit in the lower reflector region of tube 6_i and sent to the Radiochemical Laboratory and Counting Laboratory for analysis. At about this same time, the basket at the exit of tube 6_i was removed from the instrument harness. In the basket was a portion of one of the fuel stages, presumably stage 18. Three samples of the melted fuel element were taken from this basket and were also sent to the Radiochemical Laboratory for analysis. Radiochemical separations were performed on I^{131} and Zr^{95}, and the uranium content was determined for the various samples.

While the insert was in the Radioactive Materials Laboratory, four more samples of the slag were removed from the deposits in the lower reflector region, and analyzed as above. Following disassembly, a portion of the outside fuel ring of the eighteenth stage, which seemed to be intact, was removed and three pieces were cut from this ring of fuel and analyzed for I^{131}, Zr^{95}, and uranium.

During the disassembly of the fuel cartridge the slag deposit was removed from the tail cone region and weighed. The total weight was 798 grams. The remains of each stage, except stage 18, were also weighed as they were cut from the fuel cartridge.

A summation of all the weights of the slab deposit in the tail cone and the remains of stages 10 through 18 inclusive, assuming the remains of stage 18 to weigh 40 grams, account for approximately 1100 grams. This does not include the contents of the basket. If the weight for stage 9 (227.15 grams) is the weight of a typical fuel element, then initially there were approximately 2000 grams of fuel element in the last nine stages. This indicates that there was a loss of some 900 grams of fuel element through the exhaust system.

6.4 TESTING IN L2 PARENT INSERT

The L2 parent insert is described in section 4. The following sections describe the operation of the various test cartridges tested in the L2 parent insert.

6.5 CARTRIDGE L2A1

The first test cartridge designed for the L2 insert was the L2A1 cartridge. It was designed to evaluate operational effects of water-vapor corrosion on fueled BeO tubes and to establish a realistic operating temperature for ceramic tubes in light of the corrosion problem observed in the insert 2B tests. It was designed also to determine the flow variation with time and wall temperature. A secondary purpose of the test was to observe fission product release from the fuel BeO tubes as a function of operating time, in an effort to establish a pattern for fission product release for a full-scale ceramic reactor.

The L2A1 was operated at a temperature of approximately 2500°F, or 250 degrees lower than insert 2B. Test data indicated that at 2500°F the hydrolysis or rate of material redeposition in the hot region of the insert would not be as great. In the course of performing this test, it was hoped to establish a realistic operating temperature for a BeO ceramic reactor consistent with future system requirements. The specifications for the operation of the HTRE No. 2 reactor with the L2A1 insert are as follows:

1. Maximum total reactor power	10 to 16 mw
2. Maximum insert power	250 to 300 mw
3. Insert maximum temperature (initial) hottest couple	2500°F
4. Maximum average temperature (average of 3 hottest)	2500° to 2600°F
5. Insert exit air temperature average	1600° to 2100°F
6. Parent core fuel cartridge temperature (average 3 hottest)	1300° to 1500°F
7. Parent core maximum fuel element temperature (single couple)	1750°F
8. Parent core fuel cartridge exit air temperature (average 3 hottest)	1000° to 1100°F
9. Insert water moderator temperature (inlet)	100° to 170°F
10. Insert beryllium moderator temperature	100° to 200°F
11. Insert unfueled temperature	2000° to 2600°F

The L2A1 cartridge was tested in HTRE No. 2 power reactor for a total of 100.25 hours at a maximum temperature of 2500°F. In general, the operation proceeded without unexpected incidents. Following a preliminary checkout and calibration of facility instrumentation and associated equipment, the reactor was made critical for the first time in a test series with a K_{ex} of 0.41 percent at a parent-core water-moderator temperature of 93°F. At a water temperature of 150°F the K_{ex} was 1.55 percent. The critical experiment predicted K_{ex} of 1.78 for 150°F.

At the conclusion of the endurance testing, the insert was removed from the HTRE No. 2 and given a thorough visual examination. Examination of the rear support plate showed no evidence of plugging. However, oxidation on the metal in the fuel region holes was observed, whereas holes in the unfueled region were shiny. This observation indicated that much cooler air was flowing down the unfueled regions of the insert. Examination of the first three stages of unfueled tubes showed no tube deterioration or crystal growth. On the trailing edge of some of the fourth stage tubes (first fueled stage) a substance having an appearance of white powder was observed. The upstream face of stage 5 also shows this white substance. The first crystal growths or deposits were observed on the trailing edge of stage 6 (third fueled stage, 25.037 inches below the top grid plate). Crystal deposits were

observed to be at a maximum on the downstream edge of stage 7 (fourth fueled stage 29.281 inches below the top grid plate). Deposits were observed in decreasing amounts on stages 8, 9, and 10. Small amounts of deposits were observed in the upstream edge of the eleventh (unfueled) stage. In general, the deposits appeared to be heaviest in the central region of the insert, suggesting the higher temperature operation of that region and the influence of higher temperature operation on the formation of the deposit. The amount of deposit was much smaller than that observed in the insert 2B operation, suggesting that the deposit is a sensitive function of temperature in the range 2400° to 2750°F. The fact that the largest deposits occurred at the tube ends suggests that some cooling of the end of the tube is taking place, causing the beryllium oxide hydrolyzed into the airstream to condense at these points. Cooling at the ends can be accomplished either by crossflow at the joint or by misalighment of the tube causing edges.

6.5.1 EFFLUENT DATA FOR L2A1

During the tests, effluent sampling was carried out using carbon trap and filter paper techniques. The iodine, barium, and strontium were chemically separated from the carbon. These chemically separated fission products were counted using the 256-channel gamma ray analyzer. Isotopes for which analyses were made were iodine-131, iodine-132, iodine-133, iodine-134, iodine-135, iodine-139, iodine-140, strontium-91, and strontium-92. The amount of beryllium in the carbon trap samples taken from the 80-foot stack sampling point was also determined. The beryllium analysis indicated an average release of 3×10^{-5} grams per second. Comparison of the data collected at the stack and at the CTF harness located beneath the insert indicate general agreement in trend, however, the insert probe consistently recorded higher releases. The lower releases at the stack could have been the result of iodine plated-out in the ducting leading to the stack, or the higher release of the insert probe could have been due to previous contamination of the insert sampling line. Data indicated a definite increase in I^{131} activity over the endurance testing and a slight increase in I^{133} and I^{135}. This increase is believed to have been a function of some corrosion pehnomena caused by the hydrolysis of surface material.

6.6 CARTRIDGE L2C1

The L2C1 cartridge was designed to the specifications for an early design version of the XMA-1 power plant, with fuel-ring loadings adjusted to HTRE No. 2 L2 insert power distributions.

The power operation of the cartridge in the HTRE No. 2 was conducted to evaluate performance of new materials in advanced fuel cartridge designs. The L2C1 cartridge was of metallic concentric-ring design, with fuel sheet made of a Cr-UO_2-Ti core clad with an Fe-Cr-Y alloy. The fuel sheet was formed into 18 stages and positioned in concentric rings around a central hydrided zirconium moderator rod. Each stage was nominally 1.50 inches long with an interstage separation of a nominal 0.125 inch. The active length of the cartridge was 29.905 inches.

6.6.1 POWER OPERATION

The HTRE No. 2 reactor with the L2C1 cartridge was operated at design power conditions in successive operations for a summed total of 80.75 hours. The peak indicated-fuel-plate temperature was 2091°F.

Pertinent data recorded for the on-test condition were:

1. Peak moderator bar temperature 1792°F
2. Peak fuel cartridge discharge air temperature 1599°F

3. Peak parent core fuel plate temperature	1352°F
4. Average parent core discharge air temperature	986°F
5. Total engine weight flow	53.09 lb/sec
6. Insert weight flow (total)	1.20 lb/sec
7. Total power to air	8.14 megawatts
8. Total insert power to air	0.40 megawatt
9. Adapter tube temperature	730°F

The K_{ex} was measured at 2.54 percent $\Delta k/k$ at a water moderator temperature of 150°F. (Predicted K_{ex} was 2.35 percent $\Delta k/k$.)

During the operation, the gross effluent activity remained relatively constant for the first 30 hours. After 30 hours the activity climbed sharply by a factor of 1.23 and after 50 hours again climbed sharply by a factor of 1.6, and thereafter continued to gradually increase to the conclusion of the operation. The gross effluent activity was interpreted in terms of fuel sheet rupture and became the reason for the shutting down of the operation.

6.6.2 EFFLUENT ANALYSIS

During approximately 30 hours of endurance testing the release of the iodine isotopes was more or less constant at 10^{-3} percent of the formation. This release was due primarily to the fissioning of uranium deposited in the lower cocoon from previous HTRE No. 2 experiments. This conclusion is substantiated by data recorded during the power operation of the 1C insert (seven unclad hydrided zirconium moderator cells and six insert type fuel cartridges) which indicated gross stack releases, at 11 megawatts total reactor power, of 7.7 curies per hour, aged 10 minutes.

Upon the return of the CTF to the hot shop the L2C1 fuel cartridge was removed from the parent insert for visual examination. This examination showed no damage to the outermost fuel sheet of the cartridge. The individual stages were then separated from the cartridge and the rings examined. From this examination it was concluded that stages 3, 4, 5, 10, 11, 12, 13, 14, 15, 16, 17, and 18 had fuel sheet blisters which in some instances had ruptured.

6.7 CARTRIDGE L2E1

The L2E1 insert cartridge was made up of a bundle of round fueled and unfueled BeO (ceramic) tubes arranged in a hexagonal pattern. The tubes were stacked one on top of the other to form a complete fuel cartridge composed of a front reflector of unfueled BeO tubes 12.305 inches long, a fueled region 29.743 inches long, and a rear reflector of unfueled BeO tubes 4.435 inches long. The tubes in the fueled region were staggered so that no two tube joints lined up. The unfueled tubes in the front and rear reflectors had nominal lengths of 3.435 and 4.435 inches, and fueled tubes had a nominal length of 4.249 inches, except for tubes required to adjust for tube staggering which had lengths of 1.417 inches and 2.832 inches.

The fueled tubes of the L2E1 have a nominal outside diameter of 0.368 inch and a nominal inside diameter of 0.277 inch. The inside diameter includes a 0.0015-inch-thick coating of Al_2O_3 on the fueled tubes. The tube bundle is contained in an insulation liner-support tube made up of layers of Hastelloy X, Thermoflex, and 304 stainless steel. The Hastelloy liner is 0.005 inch thick and is designed to operate at a maximum temperature of 2000°F, the Thermoflex is 0.250 inch thick and the stainless steel liner-support tube is 0.012 inch thick.

The fuel loading of the L2E1 tubes was varied in accordance with a nuclear analysis for the purpose of flattening the radial power generation. The tube bundle contains 127 tubes in cross section, 56 of which are fueled and 5 of which are instrumentation tubes.

6.7.1 POWER OPERATION

The following is a list of the predicted parameters for the L2E1 operation:

1. Maximum total reactor power (dependent on ambient conditions)	10 to 14 mw
2. Design reactor power	12 mw
3. Maximum insert cartridge power	200 to 300 kw
4. Maximum insert temperature (initial) single reliable thermocouple	2515°F
5. Insert average exit air temperature	1650°F
6. Parent core fuel cartridge temperature	1300° to 1500°F
7. Parent core maximum fuel element temperature	1750°F
8. Parent core fuel cartridge exit air temperature	1000° to 1100°F
9. Compressor discharge air weight flow at 7500 rpm	59.9 lb/sec
10. Reactor inlet pressure	54.2 psia
11. Reactor inlet temperature	344°F
12. Reactor pressure drop	8.0 psi
13. Pressure drop across tube bundle	4 psi
14. Pressure drop across orifice plate	4 psi
15. Insert total weight flow (including 10% leakage)	0.75 lb/sec

After the performance characteristics of the insert had been established the operation proceeded to accumulate time on the reactor at the design maximum indicated (thermocouple) temperature of 2515°F.

At the conclusion of 106 hours of endurance testing, the reactor was returned to the hot shop where the insert was removed and examined. Visual examination of the cartridge assembly immediately after removal showed the structure to be in excellent condition. The cartridge was then removed to the Radioactive Materials Laboratory (RML) where half of the insulation liner was removed and the fuel tubes examined.

A general visual examination (through the RML periscope) of the fueled tubes showed no crystal growths or white powder (BeO) deposits. Some of the tube ends had a whitish appearance but this was attributed to the coating process. The inside diameter of some of the tubes appeared to be rougher than others with variations in the degree of blackness also being noted. The outside surface of some of the tubes was streaked with a substance having a tan color. The presence of these streaks suggests flow down the interstices and possibly across the edges.

6.7.2 EFFLUENT ACTIVITY

During the L2E1 operation samples of the effluent were taken from several locations along the duct and stack system. The most consistent set of data was obtained from the sampling probe located at the 80-foot level of the stack. This sampling line terminates in a shielded vault at the base of the stack which is habitable during power operations making possible monitoring of the equipment and the taking of several samples at various times.

The fractional release of I^{131}, I^{135}, Ba^{140}, and Sr^{91} was measured by carbon filters. In general the carbon trap I^{131} data indicated a reasonably flat release rate when all the points were averaged. Data from an analysis of the carbon traps for beryllium indicated

an average release of 8.5 micrograms per second. The data indicated that the Al_2O_3 inside diameter coating was to some extent effective in delaying the release of certain fission products.

6.8 CARTRIDGE L2E3

The L2E3 insert cartridge was devised to further evaluate the BeO ceramic tube as the carrier for fuel in nuclear reactors. The design of the fueled tube differed from the L2E1 and the L2A1 cartridges in that the tubes were of hexagonal shape and the inside diameter coating was zirconia instead of alumina. The tube bundle had seven fueled stages and contained 52 fueled tubes per stage, 54 unfueled tubes per stage in the seven fueled stages, and 106 in the four unfueled stages. The fueled and unfueled tubes measured 0.366 inch across flats and had a nominal inside diameter of 0.264 inch. The fueled tube length was 4.155 inches and the unfueled tube lengths are varied to account for 1/4 inch of the tube staggering. The fueled tubes are formed in a coextrusion process resulting in a coating of approximately 0.005 inch of zirconia applied to the inside diameter of the tube. The fueled and unfueled tubes of the cartridge are contained in an insulation liner.

The primary purposes of L2E3 cartridge tests were:

1. To evaluate the effectiveness of the ZrO_2 inside diameter coating in resisting BeO hydrolysis and reducing the release of fission products by operating the L2E3 fuel cartridge at peak temperatures of 2500°F and 2600°F for 100 hours or more at each temperature.
2. To determine fission-product release as a function of temperature.

6.8.1 POWER OPERATION

Design and operating conditions established for the L2E3 test were:

1. Reactor Inlet Conditions:
   Air temperature, 340°F
   Air pressure, 53 psia
   Engine speed, 7500 rpm
2. Pressure drop across insert assembly, 7.25 psia.
3. Insert friction factor, $F = 0.053\ Re^{-0.2}$.
4. Insert heat transfer coefficient, $\frac{hD}{K} = 0.0205\ Re^{0.8}\ Pr^{0.4}$.
5. Maximum radial average longitudinal power density, 2.61 Btu/sec/cubic inch at 10 megawatts.
6. Unfueled tube average heat generation rate, 0.045 watt/gram/megawatt.
7. Maximum tube wall temperature, 2500°F and 2600°F.
8. Radial temperature to be flattened by using flow restricting orifices.

As the L2E3 was taken to power for the first time, the parent core fuel element temperature limit was reached before on-test conditions of 2500°F in the insert cartridge could be reached. This necessitated the insertion of a flow-restricting orifice plate at the front of the insert cartridge. In subsequent power operations, test conditions were reached at 15.1 megawatts total power (a factor of 1.8 over predictions). Average heat generation rate of 0.155 Btu per second per cubic inch per megawatt of total HTRE No. 2 power was calculated. (This value was based on a matching of wall temperatures and total pressure drops taken from test data, and is 38 percent lower than predicted heat generation rate.) The reactor was operated a total of 101.67 hours at 2500°F design

point and at a 2600°F design point for 102.05 hours. The calculated average value of 0.164 Btu per second per cubic inch per megawatt was used to determine performance curves for the operational characteristics of the insert.

As a result of the pretest analysis, no insert orifice plate had been installed to restrict insert weight flow (which was usually done to allow design temperatures in the insert to be reached without exceeding maximum parent core temperature). The increase in flow rate with increased reactor power (15 compared to the predicted 8.5) limited the peak insert temperature to 2400°F at the maximum allowable parent core temperature. The installation of the first orifice plate allowed the attainment of 2500°F insert temperature. A second orifice plate was installed to allow a 2600°F insert temperature. All but 40.3 hours of the test were accumulated with this latter configuration.

6.8.2 EFFLUENT ANALYSIS

During the L2E3 power testing, samples of the effluent were periodically collected and analyzed to determine the fraction of fission products escaping from the stack. Data were gathered at various maximum insert temperatures as well as during approximately 100 hours of endurance testing at nominal 2500°F and 2600°F maximum insert fueled temperatures.

Data collected at temperatures lower than 2500°F indicate no significant temperature effect on the release of fission products. Release fractions at lower temperatures were essentially the same as those measured during the endurance testing at 2500°F. Initial values of the I^{135} release were in the neighborhood of 1×10^{-4} and showed a decreasing trend for about the first 30 to 40 hours. The average value after the first 30 hours of testing was 4.0×10^{-5}. I^{131} data from the same effluent samples exhibited no continuing trend during the 100 hours at 2500°F and the average value of the release fraction was 3.2×10^{-4}. Analysis of the samples for barium isotopes and strontium isotopes indicate that their average release fractions were as follows: barium, 1.4×10^{-5}; strontium, 91×10^{-5}; and strontium, 92×10^{-6}.

The data on barium and strontium release indicate that release during the endurance testing was essentially constant or very slightly decreasing. Following the 100 hours of testing at approximately 2500°F, an additional 100 hours of endurance testing were accumulated at approximately 2600°F maximum insert fueled temperature. Fractional release of I^{131} and I^{135} determined from effluent samples collected during these tests, was somewhat higher (a factor of 1-1/2 to 2) than at the 2500°F point. The average value for the water content in the air was 2.88×10^{-3} grams H_2O per gram of air. Analysis for other isotopes indicate that the fission product release was essentially the same at 2600°F and 2500°F.

All release fractions are as measured and have not been corrected for sampling line plate-out. Previous experience with plate-out under the conditions in the IET sampling system would indicate the correction for plate-out is probably between a factor of 2 and 5.

6.8.3 POSTOPERATIONAL EXAMINATION

Upon completion of the test, the insert tubes were removed a stage at a time. Examination showed no blockage or gross crystal formations.

The hottest stages in the insert (Stages 8 and 9) were examined for crystal formations on the inside-diameter surfaces at the tube ends. No crystal formations were visible. Individual tubes were also bisected and examined under high magnification (10 x to 30 x). Minute crystal-like growths were observed, but were of no consequence in regard to any

effect on the insert thermodynamic performance. No cracked or broken tubes were found which could be attributed to the power operation.

A large number of tubes showed a reddish color on the ends of the tubes. This color was more intense in the hotter stages. This same discoloration has been observed on ZrO_2 coated tubes run in the MTR tests. There were also a few fueled tubes which appeared to have a small blister between the cladding and the fueled tube.

6.9 CARTRIDGE L2E2

The L2E2 cartridge was similar to L2E1 except that the fueled BeO tubes were hexagonal in shape. The L2E2 was formed of tubes stacked and staggered in layers to form a fueled cartridge approximately 45 inches long. The tube bundle was made up of 1936 fuel-size tubes (728 of which were fueled and 55 of which were instrumentation rods) and 330 unfueled solid half-hex rods. The tube bundle contained 104 fueled tubes in cross section. Tubes measured 0.295 inch across flats and had a nominal inside diameter of 0.225 inch. The fueled tubes were coated on the inside diameter with a nominal 0.0015-inch-thick layer of aluminum oxide (Al_2O_3). The length of each fueled tube was 4.155 inches and the length of each unfueled tube was 4.155 inches except for unfueled tubes at either end of the cartridge which varied in length to compensate for fuel tube staggering. The length of the fueled region was 29.008 inches and was composed of seven stages of fueled tubes. The individual tubes in the fueled region of the cartridge were orificed to produce a flat radial temperature profile. Past data from the operation of the L2A1 and the L2E1 were used to adjust the radial profile of the L2E2. (The L2A1 and L2E1 profiles, although calculated flat, indicated 200° to 300°F gradients.) Fueled tubes were composed of 6 percent by weight UO_2, 7.33 percent Y_2O_3, and the remainder was BeO. The tube density after firing for fueled tubes was 97 percent of theoretical (minimum) and for unfueled tubes was 95 percent (minimum).

The following is a list of design and operating specifications for the L2E2 insert:

1.	Maximum total reactor power	10 to 14 mw
2.	Insert cartridge total power	200 to 300 kw
3.	Maximum insert temperature	2500°F and 2600°F
4.	Average insert exit air temperature	1650° to 1750°F
5.	Parent core fueled cartridge temperature	1300° to 1500°F
6.	Parent core maximum temperature	1750°F
7.	Parent core fueled cartridge exit air temperature	1000° to 1100°F
8.	Compressor discharge air weight flow, X39-5 engines at 7070 rpm	58 lb/sec
9.	Reactor inlet pressure	53 lb/psia
10.	Reactor inlet temperature	300°F
11.	Parent core pressure drop nominal at temperature and power	8 psi
12.	Pressure drop across insert fuel bundle	6 psi
13.	Pressure drop across orifice plate	2 psi
14.	Insert total flow	0.7 lb/sec
15.	Tube friction factor	$F = 0.053\ Re^{-2}$
16.	Tube heat transfer coefficient	$\frac{hD}{K} = 0.0205\ Re^{0.8}\ Pr^{0.4}$

Unfueled tube average longitudinal power (gamma and neutron heating) equals 0.045 watt/gram/megawatt.

6.9.1 POWER TESTING

Primary purposes of the L2E2 test were:

1. To evaluate the materials characteristics of the Al_2O_3 coating and the over-all performance of the cartridge by operating the Al_2O_3-coated hexagonal ceramic fueled tubes at a peak fueled temperature of 2500°F and 2600°F for 100 hours or more at each level.
2. To operate the insert fueled cartridge at various temperature levels at specified intervals during the endurance testing to determine fission product release as a function of temperature.

Excess reactivity when the reactor was made critical was measured at 1.65 percent $\Delta k/k$ at an average parent core moderator temperature of 150°F and at a core pressure with one jet engine operating. Consideration of samarium poisoning and fuel depletion from previous tests resulted in a clean K_{ex} of 2.71 $\Delta k/k$ at 150°F moderator temperature for this reactor assembly. The measured cold cleaned K_{ex} at LPTF for this assembly was 2.45 percent at a moderator temperature of 150°F.

A total of 45.68 hours at the 2500°F maximum indicated temperature, and 99.16 hours at the 2600°F maximum indicated temperature, was accumulated with the L2E2 in the insert.

The L2E2 insert cartridge test was very successful in that a flat radial temperature profile was obtained. This was the first time in a reactor materials test that a large bundle of tubes was operated under more or less uniform radial temperature conditions. The data obtained, therefore, are more representative of full scale reactor operations.

The examination of the tubes after the test and a review of the beryllium release data showed that BeO hydrolysis was at a minimum. Essentially no crystal deposit was observed in the tubes. The fact that, for the first time, cracks were observed in the fueled tubes after test indicates that stress limits might have been surpassed in certain regions of the cartridge. That observation, however, might be qualified by the fact that an inferior grade of fuel tube was accepted for the L2E2 test. Such tubes may have had lower yield points than tubes of acceptable quality.

The iodine release fractions were a factor of 5 to 10 higher than that measured during the L2E3 test. The fact that the I^{131} release fraction increased during the test may be indicative of tube cracking and corrosion.

It is to be emphasized that, although cracking probably occurred during testing, no change in cartridge thermodynamic performance was observed.

6.9.2 EFFLUENT DATA

During the test of the L2E2 insert, regular effluent sampling program was in effect. The average I^{135} release fraction at 2500°F test point was 6×10^{-5} and at the 2600°F test point the average was approximately 2.0×10^{-4}, higher by a factor of 2. No significant increasing trend was noticeable in the I^{135} data, but the I^{131} release fraction definitely exhibits an increasing trend starting at about 6×10^{-4} for the first 10 or 15 hours and increasing to about 4×10^{-3} after 60 hours of testing.

The stack release of gross fission products as measured by extrapolating the activity measured on paper samples of the total effluent showed an increasing trend during testing at 2600°F and also indicated a marked temperature dependence between 2500°F and 2600°F.

6.10 CARTRIDGE L2A2

This section contains the test results from the Fuel Element Effluent Test, (FEET).

The fuel cartridge consisted of uncoated fueled and unfueled BeO tubes. The tubes were cylindrical in shape and were the same size as the L2A1 tubes.

The primary purposes for performing this series of tests were:

1. To provide a source suitable for evaluating the effectiveness of the electrostatic precipitator, which is in the side loop at IET, as a fission product filter. The test was aimed at finding a material which when injected upstream of the precipitator would give 90 percent removal efficiency.
2. To obtain further information pertaining to the release of fission products as a function of temperature from uncoated BeO-fueled tubes.
3. To study atmospheric diffusion of fission products released from the IET stack under various meteorological conditions.

Initially, the cartridge was scheduled for only 50 hours at a peak insert temperature of approximately 2700°F. After 47.66 hours, the CTF was returned to the hot shop and the insert fuel cartridge removed and examined. The examination showed that the cartridge was still in good condition. Following the LIME experiment, (L2E4, described in section 6.11), it was decided to continue the FEET testing, since further information was needed to evaluate the electrostatic precipitator. Testing continued for an additional 45.99 hours.

The data obtained from the electrostatic precipitator showed that by injecting a solution of $AgNO_3$ upstream of the precipitator, 90 percent of the gross fission products and 90 percent of the iodine isotopes could be removed from the effluent.

During each day's operation, samples were taken of the effluent to evaluate the fission product release rate. Near the end of the operating period, the fuel temperature was increased to a peak of 2800°F. This test was performed to obtain field data and also to determine the effect of temperature on the fission product release rate. This was the highest temperature attained during the test.

The majority of the reactor operation was at low powers (approximately 3 megawatts) with the two aftercooling blowers providing the forced air coolant. This was done to eliminate the engine combustion products from the effluent, which had an effect on the removal efficiency of the precipitator.

6.10.1 POWER OPERATION - FEET 1

During the FEET 1 test period a total of 348.35 megawatt hours were accumulated on the reactor assembly. The reactor was operated for 86.95 hours above 1 percent (approximately 10 kilowatts).

The majority of the test hours were accumulated with blower operation only. The two aftercooling blowers were operating on high speed to provide the reactor cooling air. This type of operation was a deviation from the test program and was adopted as a means of eliminating the engine combustion products from the effluent which had definite effects on the precipitator performance. Approximately the same concentration of fission products was maintained in the side loop under this type of operation as with the engine operation.

Four main types of material were injected into the system during the test. They are as follows:

1. Lamp black, fly ash, and converter fumes which were injected in a dry state.
2. Solutions of copper sulphate and silver nitrate.

3. Silver iodide acetone solution injected by a cloud seeding device.
4. Motor oil.

Tests were also performed to determine the effects of engine combustion carbon, industrial water, and demineralized water.

Considerable difficulty was encountered during the entire test series with the various injection devices, the side loop effluent sampling system, and the precipitator power supply. The experience obtained with the side loop and associated equipment during FEET 1 proved to be very valuable in making FEET 2 a very successful test. Although very little precipitator efficiency data were generated, the test was worthwhile.

6.10.2 POWER OPERATION - FEET 2

During the FEET 2 test period, a total of 258.48 megawatt hours were accumulated with 85.42 hours of operation above 1 percent.

The majority of the test time was accumulated with the two aftercooling blowers providing the coolant air. This was the same procedure used during FEET 1. From the experience gained from FEET 1 testing, a number of modifications were made to the side loop equipment. These changes coupled with the prior experience resulted in a very successful test.

A new method of injecting solids was used during the testing. Reliable rates down to 3 pounds per hour of $AgNO_3$ could be obtained. A vibrating box and ramp were provided to supply the materials to a jet pulverizer where the material was ground by air friction then drawn into the exhaust duct. Satisfactory operation was obtained as long as the air supplied to the pulverizer was dry and the injected material could be kept dry.

An AgI generator of improved design was constructed and operated during this test series. This generator provided a hotter flame, a more constant and reliable flow of AgI solution, and more complete burning of the solution without the formation of a crystal on the AgI nozzle.

The sampling system located in the IET service room for the "LIME" test was used so four samples could be taken on the side loop. This allowed a shorter run time and provided samples which were simpler to correlate.

Due to failure of the secondary voltage resistors of the precipitator, runs were made with no resistors in the circuit for the front plates. These runs provided efficiencies greater than 90 percent; therefore, the remainder of the test was run without resistors and efficiencies greater than 90 percent were obtained consistently.

Washing of the precipitator was held to a minimum because of difficulties encountered from clogged wash nozzles. The precipitator was washed only if the performance of the precipitator had dropped below minimum values.

6.10.3 TEST TEMPERATURE DATA

A comparison of FEET 1 peak insert temperature versus insert air temperature rise reveals a smaller rise associated with engine operation which was attributed to a higher inlet temperature condition. No discontinuity in temperature relationships was apparent between engine and blower operation, indicating that turbulent flow relationships were applicable for both flow conditions despite the six-fold change in total weight flow. The insert temperature relationships were the same for FEET 1 and 2.

Temperature distribution data for both tests with blower operation show that operation with aftercooling blowers resulted in a steeper longitudinal temperature profile in the insert. This curve change is attributed to the decreased core inlet air temperature which produced a relative depression of the upstream tube temperatures and a slight rearward

shift of the longitudinal peak. In addition, a greater temperature spread was noted in the three instrumented tubes during blower operation indicating a possible redistribution of weight flow toward the outer tubes.

A considerably flatter profile was evident for FEET 1 with engines than with blowers. This would seem to support the contention that a redistribution of air flow occurred during blower operation. A comparison of the insert air temperature rise profiles indicates a flattened temperature distribution in the case of blower operation.

The insert was operated with a nominal peak fuel tube temperature of 2676° ± 22°F for approximately 16 hours with engines and 2700° ± 15°F for approximately 32 hours with aftercooling blowers. These temperatures were obtained from arithmetic averages of peak temperatures recorded during on-test operation. No significant change in the insert temperature traces was observed with the exception of the division at the point where the shift was made from engine to blower operation. The lowering of the average surface temperature during blower operation is a result of the steeper radial temperature profile. During FEET 2, the insert was operated with a nominal peak fuel tube temperature of 2687°F ± 4.2 hours and 2809°F ± 3°F for an additional one hour with engines. Blower operation totaled 37.3 hours at a nominal temperature of 2688°F ± 13°F.

During FEET 1, several test limits were set on insert temperatures, but only two of these appeared likely to restrict operation. A limit of 50 hours was set at a peak insert temperature of 2700°F or more, but not to exceed 3000°F. Operation at 3000°F was limited to 10 hours with a two hour maximum run at any one time. Initially the rear grid plate temperature was limited to 2000°F; however, this limit was later raised to 2100°F. The highest insert temperature recorded during FEET 1 was 2740°F at the 35-inch depth on tube number 7. The FEET 2 was operated for an additional 50 hours of testing above 2400°F. Not more than 10 of these hours were to be at a peak insert temperature in excess of 2700°F and no operation was to exceed 3000°F. Extrapolation of recorded insert temperatures on FEET 2 indicated a peak temperature of 2700°F or more for approximately 5.9 hours with an average peak of 2683°F for the remainder of the test. The highest insert temperature recorded during FEET 2 was 2812°F at the 30.5-inch depth on tube number 9. Approximately 46 hours were recorded at a peak insert temperature in excess of 2400°F. A total of 24 thermal cycles were recorded at a temperature between 2000°F and 2100°F on the rear grid plate. The 2100°F rear grid limit was not reached during the entire period of testing.

6.10.4 NUCLEAR RESULTS

As the reactor was made critical, the excess reactivity of the system was calculated to be 0.95 percent $\Delta k/k$ at an average parent moderator temperature of 133°F. Application of the suggested reactivity coefficient of 0.021 percent $\Delta k/k$ per °F for moderator temperature resulted in a K_{ex} of 1.31 percent $\Delta k/k$ at 150°F moderator temperature, as compared to the predicted figure of 1.22 percent $\Delta k/k$. Initial critical reading for FEET 2 resulted in a K_{ex} of 1.33 percent at 150°F moderator temperature. This is considered good agreement in light of the inaccuracies involved in rod-position observation and the estimates made in arriving at the predicted value.

A comparison of measured and computed values of fuel-depletion and fission-product poisoning was kept throughout the test. Good correlation between these values was difficult to obtain due to the short duration of significant power operation. Agreement between computed and measured values was generally good except for the initial critical reading for each run. The observed deviation of the measured poison from that predicted at reactor startup was presumed due to a higher effective fuel cross section with increased temperatures, as a result of the so-called Doppler broadening effect.

Some inconsistencies in the measured poison values are probably due to reactivity effects of insert cooling water temperature and insert fuel temperature. Coefficients to be applied for these effects were not established due to the lack of the extended operation necessary for conclusive results and the observed inability to accurately repeat a reactivity measurement. During blower operation at low power, very little variation in poison level was noted. At the close of the test, an estimated increase of only 0.094 percent $\Delta k/k$ in total poison and burnup was calculated (0.055 percent from FEET 1 plus 0.039 percent from FEET 2).

6.10.5 FEET 1 RESULTS

Materials investigated as additives were: carbon black (4 forms), catalytic fines, fly ash, copper sulphate (in solution), converter fines, silver nitrate (in solution), silver iodide (dry), and combustion products.

With some of the additives, it was very difficult to obtain stable precipitator operation, with others re-entrainment was a severe problem.

The significant results from the tests conducted are reported in Table 6.1 which gives the removal efficiency, additive, and additive injection rate. Each reported removal efficiency is an average value obtained from 2 to 4 independent measurements under the same conditions. The probe used for each sample was installed clean prior to each run and was removed with the charcoal trap. The probe was leached and the plate-out activity (gross and isotopic) was added to the trap activity in determining the reported efficiencies. The values marked with an asterisk are exceptions; they do not include a plate-out correction.

TABLE 6.1

REMOVAL EFFICIENCY OF ELECTROSTATIC PRECIPITATOR* - FEET 1

Additive	Additive Rate, hr	Removal Efficiency, percent				
		I^{131}	I^{133}	I^{135}	I, average	Gross
None[a]	-	36	22	38	32	70*
H_2O	0.29[b]	61	61	54	59	80
$AgI + H_2O$	0.44	68	66	68	69	80
Combustion products $+ H_2O$	0.58	69*	72*	79*	73*	92*
$AgNO_3$	19.5	84	84	82	83	87

[a]Demineralized H_2O used for spray cooling to 300°F.
[b]This number represents the solid content of the industrial water.
*Indicates no plate-out correction.

To better mockup the eventual application of a precipitator to a full nuclear powered jet engine, these tests were conducted using the aftercooling blowers as a coolant supply to eliminate combustion products from the effluent. The exception to this in Table 6.1 is number 4 which was a run with engine operation to determine the effect of combustion products on removal efficiency.

In general, removal efficiency appeared, from this data, to be a function of mass injection rate, and $AgNO_3$ appeared quite promising as an additive. None of the unreported materials showed sufficient improvement over the no-additive removal efficiencies to warrant further investigation.

6.10.6 FEET 2 RESULTS

The sampling system at IET was completely revamped and greatly improved prior to this test. During this test, three sets of samples (one upstream sample and one downstream sample of the effluent per set) were collected simultaneously during each test. Careful design and operation of this sampling system plus the increase in data collected yielded much more reliable data.

In addition to the three sets of charcoal trap data collected, there were three sets of sampling probes available for collecting paper filter samples and samples with Anderson Air Samplers (cascade impactors for particle size distribution studies).

The effect of contact or mixing time of the additive with the effluent prior to entry into the precipitator was investigated by injecting the additive through spray nozzles in the side loop and then through another set of nozzles located in the tailpipe of the jet engine. Injection at the latter location provides about 10.1 seconds of mixing time compared to approximately 1.4 seconds when the additive is injected into the side loop with normal (12,500 cfm) side loop flow.

Over a range of two decades, no significant increase in efficiency was noted due to a greater amount of $AgNO_3$ being injected into the effluent. The injection rate was calculated as a weight concentration of the additive in the effluent entering the precipitator. Time prohibited a further investigation of the lower end of the curve to determine when the efficiency begins to drop due to insufficient additive.

6.10.7 STACK RELEASE DATA

At least one charcoal trap sample of the effluent was collected from a sampling at the 80-foot level of the stack during each run. During FEET 1, the iodine release fractions all showed some tendency to increase as the test progressed. The I^{131} release showed the largest increase. The FEET 2 results showed little or no increase in the iodine release fractions with time; however, there is considerable data-scatter, which is true of both tests, which virtually discredits any definite conclusions with respect to increasing or decreasing trends.

6.11 CARTRIDGE L2E4

The L2E4 cartridge contained fueled and unfueled BeO tubes which were hexagonal in shape. The tubes had the following nominal dimensions: across flats, 0.295 inch; inside diameter, 0.203 inch; length, 4.155 inch.

The fueled tubes contained 6 percent UO_2 by weight in the matrix excluding the cladding. The insert contained 198.6 grams of UO_2.

All coating material was on the inner surface only. The unfueled tubes were uncoated. The ZrO_2 coating had a nominal thickness of 0.003 inch. The Al_2O_3 coating was 0.001 inch thick.

There were eleven stages of tubes in the fuel cartridge, seven of which contained fueled tubes. The first, third, and eleventh stages were all unfueled tubes. Stages 4 through 10, inclusive, contained 59 fueled tubes, 74 unfueled tubes, and 24 half-hex of smaller pieces per stage.

The tube joints were staggered 1/4 inch by varying the unfueled tube lengths on the first and eleventh stages. The staggering arrangement was intended to reduce lateral flow of air and to maintain tube alignment.

The center 18 tubes, not including the centermost tube, were plugged to prevent any flow of cooling air through these tubes. The flow was plugged by placing solid unfueled rods in the 19 positions on stage 3. The downstream end of the plugged section was open. The next two rings of fueled tubes were orificed with a 0.068-inch orifice plug. All of the tubes in the unfueled region were orificed to 0.077 inch.

The tube bundle was surrounded by two insulation liners. The outer liner was essentially the same as has been used for the previously tested L2 ceramic insert fuel cartridges. It consisted of two half-hex pieces which were held together by pinned hinges. The inner liner was in six pieces which were held in place by the outer liner. The insulating material was potassium titanate (KT).

The purpose of the test was to operate the ceramic insert cartridge for 10 minutes at a power level sufficient to cause portions of the plugged fuel region to melt, and to evaluate the nature and propagation of such a melt and verify the ability to predict such phenomena.

6.11.1 INCREASING POWER OPERATION

Insert temperature steps were initiated with a reactor flow rate of 57.5 pounds per second. Reactor power was raised to produce peak surface temperatures of approximately 1500°, 2000°, and 2500°F in the blocked fuel region of the insert. On the following day, reactor power was increased in approximately 40 seconds from 10 kilowatts to 4 megawatts. A peak insert temperature of 2800°F was predicted.

The equilibrium temperature in the blocked fuel tube region fell below the temperature lines established. For the 3.65-megawatt condition, the peak recorded insert temperature on the same thermocouple was 2467°F, whereas, for the 4-megawatt run, the peak recorded temperature on the same thermocouple was only 2390°F. The probable explanation for this is a shift in the radial temperature distributions across the insert. The exact reason for the shift is not clearly defined: however, a decrease in the tube contact resistance; i.e., increased thermal conductivity, would account for the shift in radial distribution.

6.11.2 INSERT MELT OPERATION - 12 MEGAWATTS

A total of 7.77 megawatt hours were accrued on the reactor at power in excess of 100 kilowatts prior to the melt run. The power generated in the insert was determined by the relationship: 0.0183 x reactor power-to-air (megawatts). The calculated iodine inventory in the insert at clock time 18 hours and 30 minutes is given below.

Iodine Isotope	Activity
131	9.8 curies
132	27.0 curies
133	29.0 curies
134	22×10^{-4} curies
135	1.1 curies

Thirty-three percent of target reactor power, 4 megawatts, was reached in approximately 253 seconds and the 12-megawatt point, 90 percent linear flux settings, was attained in approximately 295 seconds. Reactor scram occurred at 850 seconds elapsed time.

The temperature in the center of the plugged fuel tube sections increased in a somewhat linear manner until the 322-second point. After about 329 seconds, the leads of the five thermocouples opened. All the thermocouples failed at the same instant although the temperatures ranged from 2600° to 3250°F. This would indicate that the thermocouple leads reached melting temperature ($\sim 3200^{\circ}$F) and opened at some longitudinal location, probably

in the fourth tier of the fuel tubes. The individual thermocouple bundle is too small to offer much resistance to heat flow and the wires probably reached the same temperature as the surrounding fuel tubes.

The measured temperature rise at the time of thermocouple failure was about 2150°F per minute which meant that if the temperature slope remained constant, the plugged tubes reached 4400°F some 64 seconds after reactor power had stabilized. A temperature of 7200°F was the theoretical equilibrium temperature at the 12-megawatt point excluding melting, burnup, and deformation.

Only two reliable thermocouples were recording on tube 8-3 during the test and thermocouple number 8-3-31 went erratic 145 seconds after maximum power was attained. The thermocouple failed at the recorded temperature of 3150°F, which is about the limit for platinum wire. A thermocouple located at the 25-inch depth on the same tube read throughout the test and recorded a peak temperature of 3240°F. A definite discontinuity in the temperature curve was noted 130 seconds after reaching 12 megawatts. The increase in temperature after approaching an apparent equilibrium state was attributed to a change in the tube geometry. This particular tube was a solid unfueled ceramic tube located against the outer ring of plugged fuel tubes with thermocouple leads running down slots on three corners adjacent to the outer row of fueled tubes with flow.

Each of the 13 thermocouples located on unfueled tubes recorded continuously throughout the melt experiment. The maximum recorded surface temperature was approximately 1810°F on tube number 9-1 at the 31-inch depth.

6.11.3 NUCLEAR RESULTS

As the reactor went critical, the measured reactivity of the core and L2E4 insert was 0.241 percent Δk/k at a moderator water temperature of 143°F or 0.388 percent Δk/k at a corrected water temperature of 150°F.

Prior to the melt, reactivity measurements were made with rod frames 2 and 4 fully withdrawn and the rods in frame 3 equally withdrawn. The measured reactivity in rod worth was approximately 0.25 percent Δk/k at an average moderator water temperature in the parent core of 143°F. Corrected to a water temperature of 150°F gives a value of 0.40 percent Δk/k. Xenon poison in the core was computed and the worth amounted to less than 0.01 percent Δk/k at the start of run 6.

After the core had been made critical and before power increase, rod number 203 in hole number 36 dropped. Since the worth of this particular rod was 0.42 percent Δk/k, the moderator water temperature was increased approximately 20°F to bring the excess reactivity up to 0.25 percent Δk/k.

After a cooling period of approximately 30 minutes following shutdown, the reactor was made critical and K_{ex} measurements were made. The excess reactivity in rod worth amounted to 0.543 percent Δk/k at a water temperature of 157.5°F. Correcting this value for a 150°F moderator water temperature gave 0.38 percent Δk/k which reflected little or no loss in reactivity.

6.11.4 POSTOPERATION EXAMINATION

The outer liner, the vanguard and instrumentation disconnect, and the visible upper thermocouple leads appeared to be in excellent condition.

The center of the aft grid plate corresponding to the area of the blocked tubes had been melted or burned away and the web supporting the discharge air thermocouples was gone.

All the thermocouples in the center solid tube 8-6 were melted off 15 inches from the forward grid. Examination of all other thermocouples pulled from the cartridge showed

the leads to be whole, clean and smooth, with only a slight loss of luster toward the downstream end. All air thermocouples downstream of the rear grid melted away along with the supporting web.

Before removing the insulation liner, steel rods were inserted through the outer tube layers extending through the forward and rear grid plates to hold tubes in position. The housing assembly was bent back, along the 270-degree hinge and the upper insulation pads were removed exposing the ceramic tubes.

The outer layer of nonfueled tubes showed some variation in color but there was no evidence of excessive heat, fusing or breakage of tubes. As the removal of the outer layer of nonfueled tubes started, many of the tubes of both layers of nonfueled tubes tumbled from their original positions. Further inspection showed that all of the stages, 4 through 11, were fused together end-to-end in the fueled region and all of the fueled tubes below stage 6 were fused. About 50 percent of the outer layers of fueled tubes in stages 4 and 5 were not fused.

Nonfueled tubes from stages 1, 2, 3, and 4 were whole and apparently undamaged. On stage 5, four broken tubes were found in the second layer. The number of broken tubes increased considerably through stages 6 and 7. It was also noted that these tubes exhibited a pattern of cracking or shattering and appeared to break and crumble easily.

Beginning with stage 8 and increasing through stages 9 and 10, a fusing of the second layer of nonfueled to the adjacent fueled layer was noted. The bond between fused tubes (both fueled and nonfueled) was strong so that in attempts to separate the tube, the tubes broke rather than separating.

After removing all loose tubes, it was found that the fused fueled portion was separated by a cross section between stages 7 and 8. The separation occurred on a relatively smooth plane, without regard to the 1/4-inch stagger of the tubes, and has been attributed to the heat of the cartridge rather than handling damage.

From the over-all appearance of the cartridge, it was evident that melting temperatures had indeed been reached in the center of the cartridge but that the damage had almost entirely been contained in the blocked fuel region. Further investigation indicated that some migration of molten fuel had evidently caused the melting of the lower grid plate and subsequent blockage of the center 19 tubes at the eleventh stage. The tubes outside the blocked region were all open.

Examination of the forward half of the cartridge showed that all tube rows were open. Toward the hot end, the walls of the tubes appeared to be thinner than normal but there was no indication of blocking or distortion of tube geometry.

6.12 CARTRIDGE L2E5

The L2E5 test utilized a ceramic (BeO) insert fuel cartridge with the center 18 rows of tubes, not including the centermost tube, blocked so that approximately 10 percent of normal flow passed through the blocked tubes. Normal airflow was provided in the fueled tubes surrounding the center 18 tubes. The arrangement and number of tubes in the L2E5 was exactly the same as for the L2E4 insert.

The primary objective of the L2E5 test was to determine the effect of partially blocking the airflow to 18 fuel columns in a ceramic fuel cartridge.

The pretest analysis indicated that melting of the center tubes would not occur.

6. 12. 1 POWER OPERATION

The test was performed in two parts. The first part required the operation of the reactor at 2800°F in the reduced-flow region of the insert. During this period of operation, transient and steady-state data were obtained to better predict insert conditions during the second part of the test. The second part of the test required that 2230°F be attained in the normal flow region of the insert and that this temperature level be maintained for as long as 2 hours or until specified limits were reached.

The fueled tube temperatures in the limited-flow region of the insert increased at essentially a constant rate until approximately 560 seconds. At this time the rate of temperature rise began to decrease until the thermocouples opened at about 590 seconds. The fact that all the thermocouples except the one at the 25-inch depth opened at the same time indicates that the leads melted at approximately the 29-inch depth. The thermocouple at the 25-inch depth appeared to have reached equilibrium temperature at 86 percent linear flux and failed shortly after the power was increased to 90 percent.

The relations between the 25-inch temperature and the temperatures at the other depths remained essentially constant during the temperature transient. If it is assumed that these relations remained the same until temperature equilibrium was reached, the maximum temperature attained in the limited flow region was probably 3800° to 3900°F, based on an estimated equilibrium temperature of 3400°F at the 25-inch depth at a linear flux reading of 92 percent.

Had reactor power been left at 86 percent, which corresponded to the predicted power of 9. 3 megawatts required to attain 2230°F, it appeared that a measured temperature of 2100°F would have been reached at the 37-inch depth. Consideration of the predicted 4 percent temperature difference across the fourth-row tubes would result in a peak fourth-row fueled tube temperature of 2190°F at 86 percent (9. 3 megawatts), which is in very close agreement with the predicted temperature of 2230°F at this power level. The subsequent power increase to 92 percent to give a maximum recorded temperature of 2230°F in the fourth row resulted in an extrapolated temperature of 2320°F. The temperature increase of only 130°F for a power increase of 6 percent (1. 4 megawatts) is probably explained by the higher pressure drop across the core with increased power and the resultant higher weight flow through the insert tubes.

6. 12. 2 NUCLEAR TEST RESULTS

The excess reactivity of the system was computed as 0. 63 percent $\Delta k/k$ at a parent core moderator water temperature of 168°F or 0. 24 percent $\Delta k/k$ at 150°F. This K_{ex} measurement was taken at a very low reactor power so that the insert cooling water temperature was also 168°F. At high reactor powers, the temperature rise in the insert cooling water was considerably higher than in the parent core moderator water and this seemed to have a positive effect on reactivity. At a reactor power of 10 megawatts, the parent core water temperature rise was from 8° to 10°F whereas the insert parent core water temperature increased approximately 50°F. By applying a previously estimated positive reactivity coefficient of approximately 0. 005 percent $\Delta k/k$ per degree increase in insert cooling water temperature, a satisfactory correlation was made between the reactivity measurements obtained before, during and after the test. No loss in reactivity was observed during the test except for the loss attributable to the accumulation of fission-product poisons.

6. 12. 3 POSTOPERATION EXAMINATION

Examination of all the tubes in the cartridge showed that all tubes were open and there was no evidence of melting. The center 18 tubes of the fueled region had a heavy deposit

of white crystals on the inner surface. An X-ray diffraction analysis showed the crystals to be BeO and Y_2O_3.

The outer liner, the vanguard and instrumentation disconnect, and the visible upper thermocouple leads appeared to be in excellent condition.

The rear grid plate was intact but the downstream surface was heavily oxidized.

The thermocouples in the center tube were melted off approximately 27 inches from the top of the front grid plate. All other thermocouples pulled from the cartridge appeared to be in excellent condition. The air thermocouples downstream of the rear grid were intact and in good condition.

The outer layers of nonfueled tubes showed no evidence of excessive heat, fusing or breakage of tubes. As the removal of the outer layer of nonfueled tubes started, many of the tubes of both layers of nonfueled tubes tumbled from their original position. Further inspection showed that stages 6 through 11 were fused end to end in the fueled region and all tubes in the first three rings of stages 8, 9, and 10 were fused.

All nonfueled tubes in stages 1 through 10 were whole and apparently undamaged. Nonfueled tubes in stage 11 that were fused just below the fueled region exhibited cracking or shattering and appeared to break and crumble easily. All fueled tubes that did not fuse were whole and apparently undamaged.

After removing all loose tubes, it was found that the fused fueled section was separated by a cross-section in stage 7. All fused tubes had numerous transverse and cross-sectional cracks. Many of the cracks ran the length of the tube. During handling, stages 6, 7, and 11 broke in many pieces while the remainder of the tube bundle separated at stage 9.

6.13 CARTRIDGE L2E6

The L2E6 insert consisted of a ceramic (BeO) fuel cartridge with hexagonal tubes coated on the inner surface with approximately 0.003 inch of ZrO_2.

The primary objectives of the test were as follows:

1. Evaluate the ZrO_2 coating at high temperatures.
2. Operate the insert at a fuel temperature ranging from 2500° to 2800°F in order to better understand the fission product release behavior as a function of time and temperature.

The arrangement and number of tubes were exactly the same as for the L2E5 cartridge.

6.13.1 POWER OPERATION

The insert was operated at four different on test conditions: 2520°F, 2620°F, 2720°F and 2820°F.

A total of 3346.70 megawatt hours were accumulated during the 398.97 hours of operation above 1 percent power. One percent power for this test series was approximately 130 kilowatts. The peak reactor power to air during any of the runs in this test series was 9.25 megawatts. A total of 352.39 hours of testing was obtained at or above a peak insert fuel element temperature of 2520°F. Of this time, 3.08 hours were accumulated at 2750°F, 196.71 hours at 2700°F, 102.08 hours at 2600°F, and 50.52 hours at 2520°F.

As the number of operating hours increased it became increasingly difficult to obtain criticality due to fuel burnup and the daily increase in xenon poison following shutdown.

This problem accentuated the low K_{ex} value built into the insert, resulting at times in required moderator temperatures greater than 200°F to obtain criticality at startup. The necessity of operating with abnormally high moderator temperatures undoubtedly contributed to the increased number of control rod failures experienced during this time.

6.13.2 NUCLEAR RESULTS

When the reactor was made critical, the excess reactivity of the system was calculated to be 0.73 percent Δk/k with an average moderator temperature of 149°F. Application of the suggested reactivity coefficient of 0.021 percent Δk/k°F for moderator temperature and addition of 0.32 percent Δk/k due to samarium poison and fuel depletion in the old parent fuel elements, resulted in a "clean" K_{ex} of 1.07 percent Δk/k at 150°F moderator temperature. This was in excellent agreement with the predicted 1.08 percent Δk/k.

6.13.3 EFFLUENT SAMPLING RESULTS

Comparison of the data at a given test condition with the inventory of the corresponding isotope plotted on the same time scale gave no indication that the release was proportional to inventory. The inventory of I^{131}, for instance, was relatively constant after 100 hours and actually showed a decreasing trend after 250 hours.

Also apparent from the data is a definite temperature effect on the release above 2600°F. Effluent samples collected at 2500°F and 2600°F show little or no difference in release fraction. In general, a change from 2600°F to 2700°F increased the release fraction about a factor of 2 for the iodine isotopes. A further increase to 2800°F (nominal) appeared to produce a further increase of nearly a factor of 10 in I^{131} and a factor of 4 to 6 in I^{133} and I^{135}. Sufficient data were not available to permit a great deal of confidence in these values; however, a longer operation at the 2800°F condition might have produced significantly different results.

The strontium data indicated no trend. The Ba^{140} data indicated a steady increase.

6.13.4 POSTOPERATION EXAMINATION

Initial inspection of the L2E6 revealed a piece of glass insulation tape lodged on the upper grid plate. The origin of the material is not known but the presence of the material could explain an apparent shift in temperature distribution which occurred after the initial approach to power early in the endurance testing. The appearance of the material indicates that it was there for a considerable length of time. The effect of this obstruction was not as apparent as might be expected due to the fact that there is about 1/2-inch space between the top of the tubes and the grid and because a large part of the pressure drop is taken by the orifice in each tube column.

General appearance of all stages was exceptionally good. There were no broken tubes or anything at all unusual in any of the unfueled tubes. Blisters in the clad material were observed in the center tubes of stages 10 and 9 and to a somewhat lesser extent in stages 8 and 7. Almost all the outside row of fueled tubes were broken in half or thirds in stages 10 through 6. Except for a few tubes adjacent to the outside row of tubes these were the only broken or cracked tubes.

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