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IRRADIATION-CAPSULE STUDY OF URANIUM MONOCARBIDE

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Small cylindrical specimens of enriched uranium carbide were irradiated in the MTR as part of a program to evaluate for Atomics International the usefulness of uranium carbide as a high-temperature fuel for stationary power reactors. Detailed thermal and nuclear analyses were made to arrive at an appropriate capsule design on the basis of target specimen center-line temperature (~1500 F), specimen surface temperature (1100 F), specimen composition (uranium -5 w/o carbon), and a capsule OD of 1.125 in.

Temperature data from thermocouples inside the capsules indicated that five of the six capsules irradiated had operated at close to the design conditions. The sixth gave temperatures lower than anticipated, presumably because the irradiation was conducted where the unperturbed neutron flux was lower than the design value of 1×10^{11} nv. Irradiation periods for individual capsules were varied to give burnups ranging from 1,000 to 20,000 mwd/t of uranium. Preliminary evidence indicates that this range of burnups was achieved, even though a detailed analysis of uncertainties shows that sizable prediction errors are possible.

By using temperature and heat-flux data from the actual irradiations to estimate effective in-pile specimen thermal conductivities, it was found that the conductivity did not appear to vary during the exposures.

INTRODUCTION

A radiation-effects program on unclad, enriched, uranium-carbon alloys at and near the monocarbide (4.8 w/o carbon) composition is being conducted at Battelle Memorial Institute for Atomics International (AI). As-cast uranium-carbon specimens, prepared at Battelle as unclad cylindrical slugs, were encapsulated and are being irradiated in the Materials Testing Reactor (MTR). The burnups desired are 1,000 to 20,000 MWD/T (2000 lb) of uranium. Capsules are identified and the specimens and irradiation program are described further in Table 1.

TABLE 1. SUMMARY OF UC IRRADIATION PROGRAM

Capsule	Nominal Carbon Content of Specimen, w/o	Desired Burnup, MWD/T of uranium	MTR Position	MTR Cycles of Irradiation[(a)]
BMI-23-1	5.0	1,000	A28NE	1
BMI-23-2	5.0	5,000	A28NE	6
BMI-23-3	5.0	10,000	A28SE	12
BMI-23-4	5.0	20,000	A27SE	24
BMI-23-5	4.6	5,000	A30NE	6
BMI-23-6	4.8	5,000	A13NE	6

(a) Number of cycles is based on an unperturbed thermal-neutron flux density of 1.0×10^{14} neutrons/(cm^2)(sec).

In designing the capsule, temperatures were assumed to be 1100 F and 1500 F at the specimen surface and specimen central core, respectively. Thermocouples were incorporated into each capsule to provide data for estimating these temperatures, fission-heat generation rates, and specimen thermal conductivity under irradition conditions. By puncturing the capsule, the total fission-gas released can be collected and measured after irradiation. Dosimetry is provided to determine the integrated thermal-neutron flux received during irradiation.

This report presents various details of the program from capsule design to interpretation of capsule data, including in-pile thermal performance monitored by thermocouples and integrated specimen dosage as estimated from reactor-quoted flux information and dosimetry. The purpose of carrying out the various data analyses is to provide a background of information to establish, with as much precision as possible, the several in-pile conditions that must be considered in the final evaluation of irradiation damage to the specimens.

PRINCIPAL CAPSULE-DESIGN INFORMATION

The capsule design is shown in Figure 1. Figure 2 is a view of one capsule partially assembled. As can be seen, the capsule design provides for a single-wall structure with NaK-immersed test specimens stacked along the capsule axis. As discussed in the ensuing paragraphs, the factors governing design are compatibility of materials, heat-transfer and nuclear considerations, firm positioning of the unclad specimens, adequate use of thermocouples and dosimeters, and ease of disassembly in the hot cell.

Preirradiation Experiments

At the outset of the capsule-design phase, a number of potential difficulties were recognized. If the UC specimens were unusually brittle, they might easily fracture as a result of thermal or mechanical shock. NaK could not be used with confidence as the heat-transfer medium until there was some assurance that there would be no interaction with the UC and the structural metals. There was also the question of whether or not common solvents, like acetone and carbon tetrachloride, would react with the UC during pre- and postirradiation specimen cleaning.

Since little authoritative information could be found on these items, simple experiments were conducted. In determining compatibility, UC specimens were placed in molybdenum and in Type 304 stainless steel baskets, sealed into NaK-loaded stainless steel capsules under a helium atmosphere, and heated in an electric furnace. Test conditions and findings are summarized in Table 2.

Specimens exposed in these experiments remained intact but showed slight discoloration at points of contact with the baskets. They experienced small weight losses (a few mg in 10-g samples), as shown in Table 2. However, no measurable change in specimen density was observed, and there was no evidence of NaK penetration into the specimen surfaces.

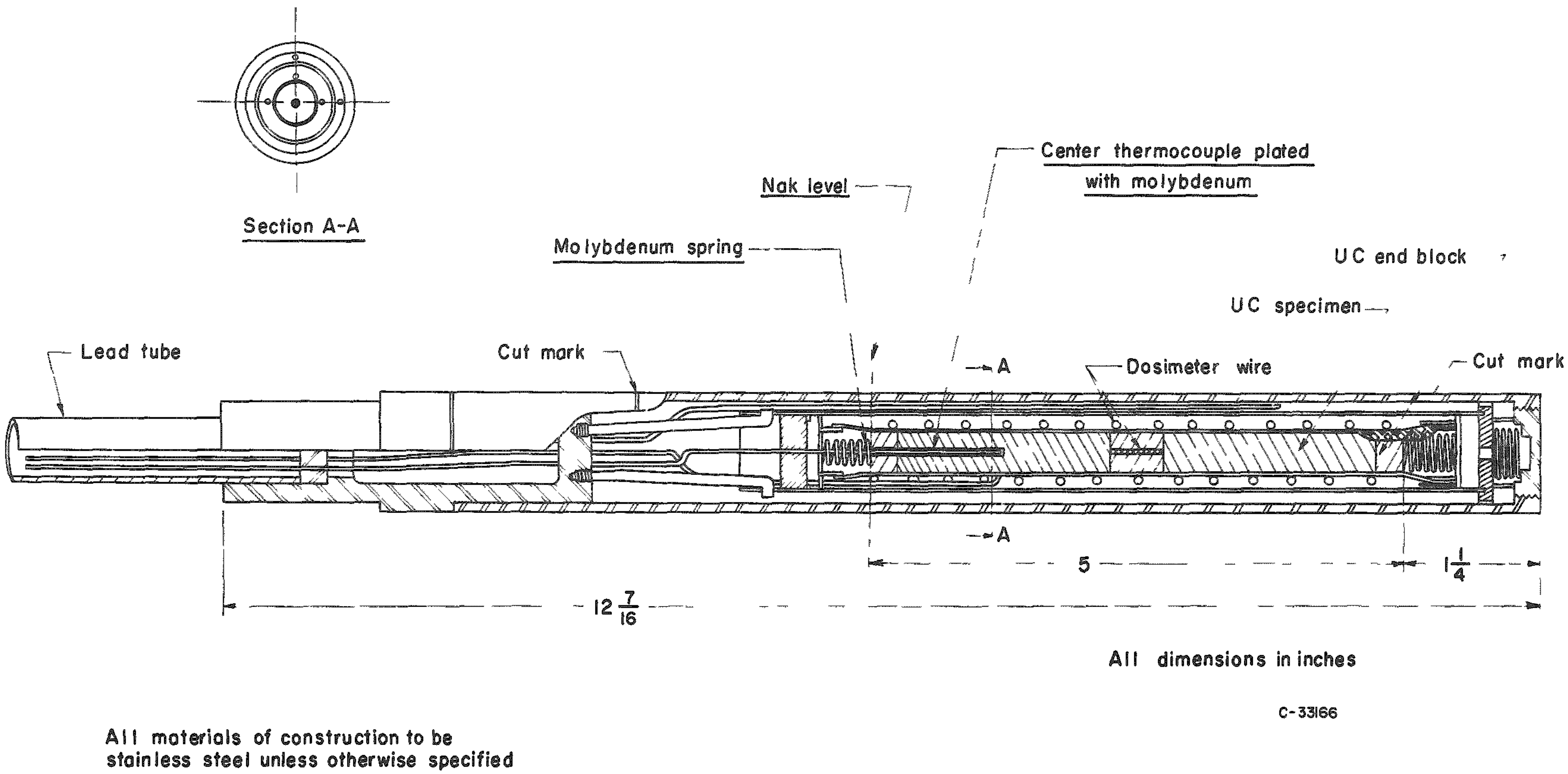

FIGURE 1. BASIC DESIGN OF CAPSULE FOR THE IRRADIATION OF URANIUM MONOCARBIDE

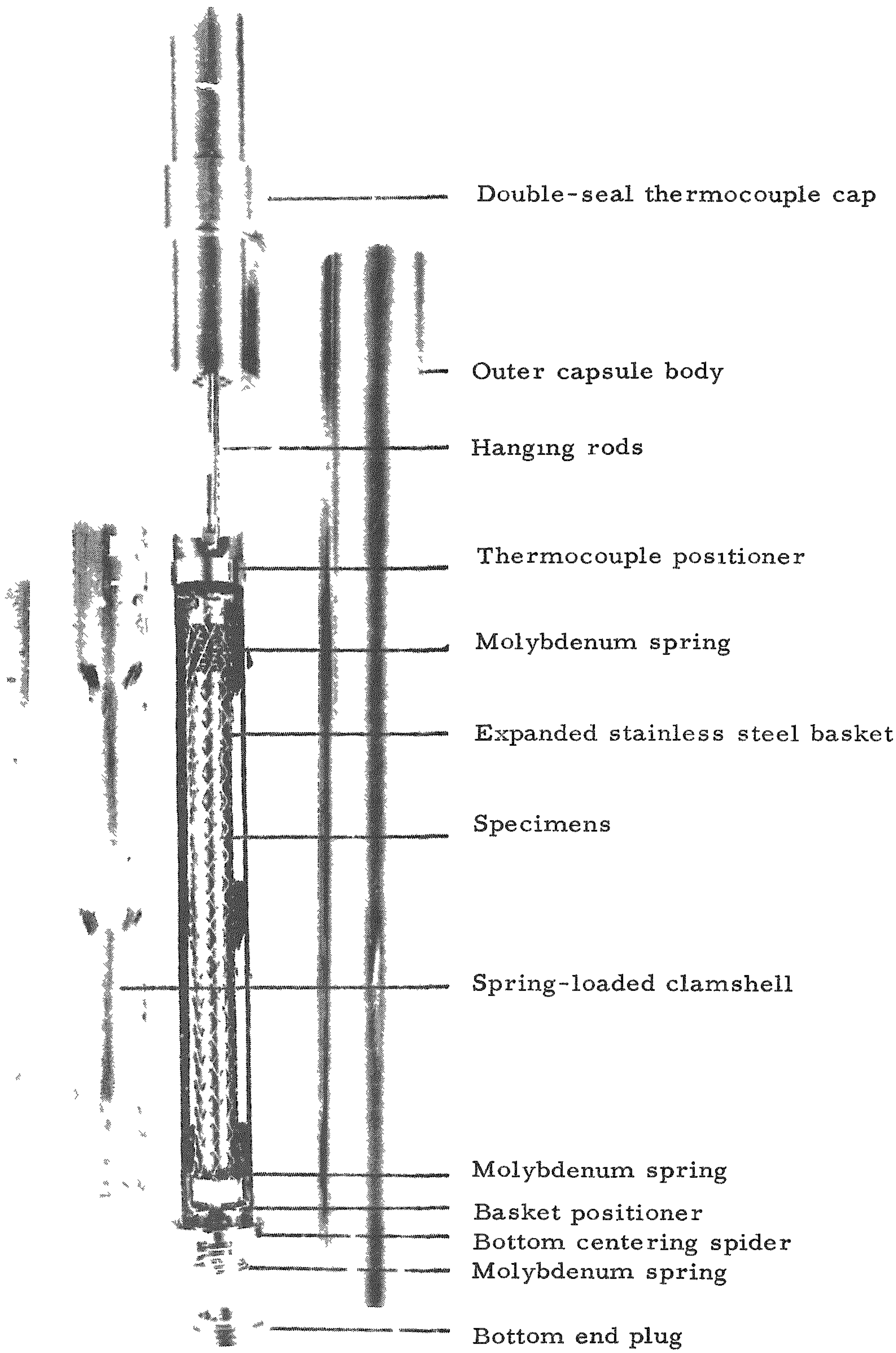

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FIGURE 2. VIEW OF PARTIALLY ASSEMBLED CAPSULE

TABLE 2. UC-NaK COMPATIBILITY TESTS

Test	Basket Material	Exposure, weeks	Temperature, F	Weight Change, mg Basket	Weight Change, mg Specimen
1	Type 304 stainless steel	2	1100	-0.4	-3.6
2	Molybdenum	2	1100	-0.3	-4.5
3	Molybdenum	2	1300	+0.3	-5.5
4	Molybdenum	6	1300	0.0	-5.6
5	Type 304 stainless steel	12	1100	(a)	-13.6
6	Molybdenum	12	1100	-2.0	-22.4

(a) Stainless steel baskets became brittle and fragments were lost so that weight changes could not be assessed. Except for embrittlement, the baskets appeared to be unchanged.

Specimen baskets were essentially unchanged except for the embrittlement of the Type 304 stainless steel baskets during the 12-week exposure at 1100 F. To explore this embrittlement, metallographic examinations were made of the UC specimens from the 12-week experiments and of samples from the steel basket. Results indicated that the embrittlement of the steel was probably caused by a carbide precipitate (Cr_4C) at the grain boundaries and possibly by the formation of an unknown phase at these boundaries and along strain lines within the grains. It is possible that the unknown phase may have been sigma or some other embrittling phase resulting from diffusion of one of the elements from the testing environment. However, the possibility that the UC contributed to the situation is tempered somewhat by the fact that the microstructure of the surface of the UC specimen examined appeared to be normal.

In general, such embrittlement of unstabilized stainless steel is not surprising since the test temperature, 1100 F, is in the middle of a sensitizing (carbide precipitation) range, 900 to 1400 F. The use of a stabilized stainless steel such as Type 347 and the elimination of cold working would, of course, minimize such a tendency; however, Type 304 was used throughout the program.

In addition to the experiments with NaK, the compatibility of UC with various liquids was examined to determine those suitable for use in preirradiation and postirradiation examination. Compatibility was determined by observing UC specimens before, during, and after exposure and by comparing pretest and posttest specimen weights. Tests were run at room temperature for periods of 1, 3, and 5 hours with water, acetone, methyl alcohol, ethyl alcohol, butyl alcohol, benzene, xylene, kerosene, CCl_4, and Zyglo.

Reactions were observed only with water. During the 1-hour exposure gas bubbles were evolved from the specimen, and during the 3-hour exposure some flaking occurred. The specimens' change in weight with liquids other than water was insignificant (less than 1 mg in 10-g samples). Also, weight changes were insignificant after allowing some UC specimens to stand in a Drierite desiccator for 2 days, for 2 months, and also after heating for 2 hours and 17 hours in a vacuum oven at 180 F and 550 microns of pressure.

In drop tests and thermal-shock tests, UC specimens were spring loaded in a stainless steel expanded metal basket, immersed in a liquid medium, and contained in a stainless steel capsule, as they would be during irradiation in the MTR. Acetone at room temperature was the liquid medium in the drop test because its density and viscosity are similar to those of NaK at 1100 F. The capsule was dropped with its axis horizontal onto a concrete floor from heights of 1, 2, and 3 feet. Then the capsule was dropped with the axis vertical. The specimens did not fracture during any of these mechanical shocks.

In the thermal-shock test, good specimen shock resistance was also demonstrated. Two capsules with specimens immersed in NaK were heated in an electric furnace to 1300 F and then plunged into room-temperature water. This cycle, during which the capsules cooled to room temperature in about 2 minutes, was repeated ten times. Radiographs taken after the first, fifth, and tenth cycle indicated that the specimens did not fracture.

Thermal and Nuclear Design

These nominal values of capsule parameters were regarding as fixed for thermal and nuclear design:

Specimen surface temperature – 1100 F

Specimen composition – uranium-5.0 w/o carbon*

Experimental density of uranium-5.0 w/o carbon – 13.4 g/cc**

Capsule shell, OD (water-contacting side) – 1.125 in.

Capsule shell, ID – 0.938 in.

In addition to these fixed specifications, it was desired to maintain the specimen center-line temperature near 1500 F during irradiation.

The following paragraphs describe the analyses made to arrive at capsule specifications satisfying these various initial stipulations. A summary of design constants finally adopted is as follows:

Specimen diameter – 0.375 in.

Specimen uranium-235 content – 8.3 w/o

Average fuel loading – 1.9 g/in.

Effective neutron flux – 0.23 x 10^{14} nv

*Specimen compositions of uranium-4.5 w/o carbon and uranium-4.8 w/o carbon were also employed.
**The theoretical density of uranium-5.0 w/o carbon is 13.63 g/cc.

Fission-heat generation rate – 6.4 x 10^3 Btu/hr per in. of specimen

Specimen center-line temperature – 1530 F.

Fission-Heat Production

Initial trial heat-transfer calculations and other considerations including feasibility of specimen fabrication, indicated that a specimen 3/8 in. in diameter would provide an acceptable starting point for the thermal design. With specimen-surface temperature (1100 F) and specimen diameter (3/8 in.) fixed, and an assumed sink temperature of 120 F, a nominal fission-heat production was found by use of the standard equation for radial heat flow in cylindrical geometry.

$$T_A - T_B = \frac{Q_{A-B} \ln \frac{r_B}{r_A}}{2\pi k_{A-B}}$$

where

T = temperature level at boundary of interest

Q = rate of radial heat flow

r = boundary radius

k = thermal conductivity across path A-B.

In using this general equation, certain refinements were desirable. These are essentially as follows:

(1) The capsule cross section was divided into four annular regions (specimen surface to reference thermocouple located adjacent to the specimens, reference thermocouple to inner wall of capsule shell, shell wall, and outer shell surface to coolant). See Figure 3. Heat-transfer calculations for each region were based on mean constants for that region.

(2) The possible effects of nonradial heat flow were considered in a simplified way by assigning an arbitrary value to "e", which is the ratio of pure radial heat flow through the NaK (from reference thermocouple to the thermocouple near the inner capsule wall) to fission-heat generation rate, i.e., $T_2 - T_4 = \dfrac{e\, Q_{fission} \ln \frac{r_4}{r_2}}{2\pi k_{2-4}}$

The assignment of nominal values for e posed a vexing question, since the complexity of the heat-flow situation precluded rigorous treatment. After appraisal of several factors, including (1) possibilities for axial heat loss by both conduction and convection, (2) the fact that the main experimental specimens had a fairly high L/D ratio, and (3) the presence of partial specimens above and below the main specimens to inhibit axial heat flow, a nominal value

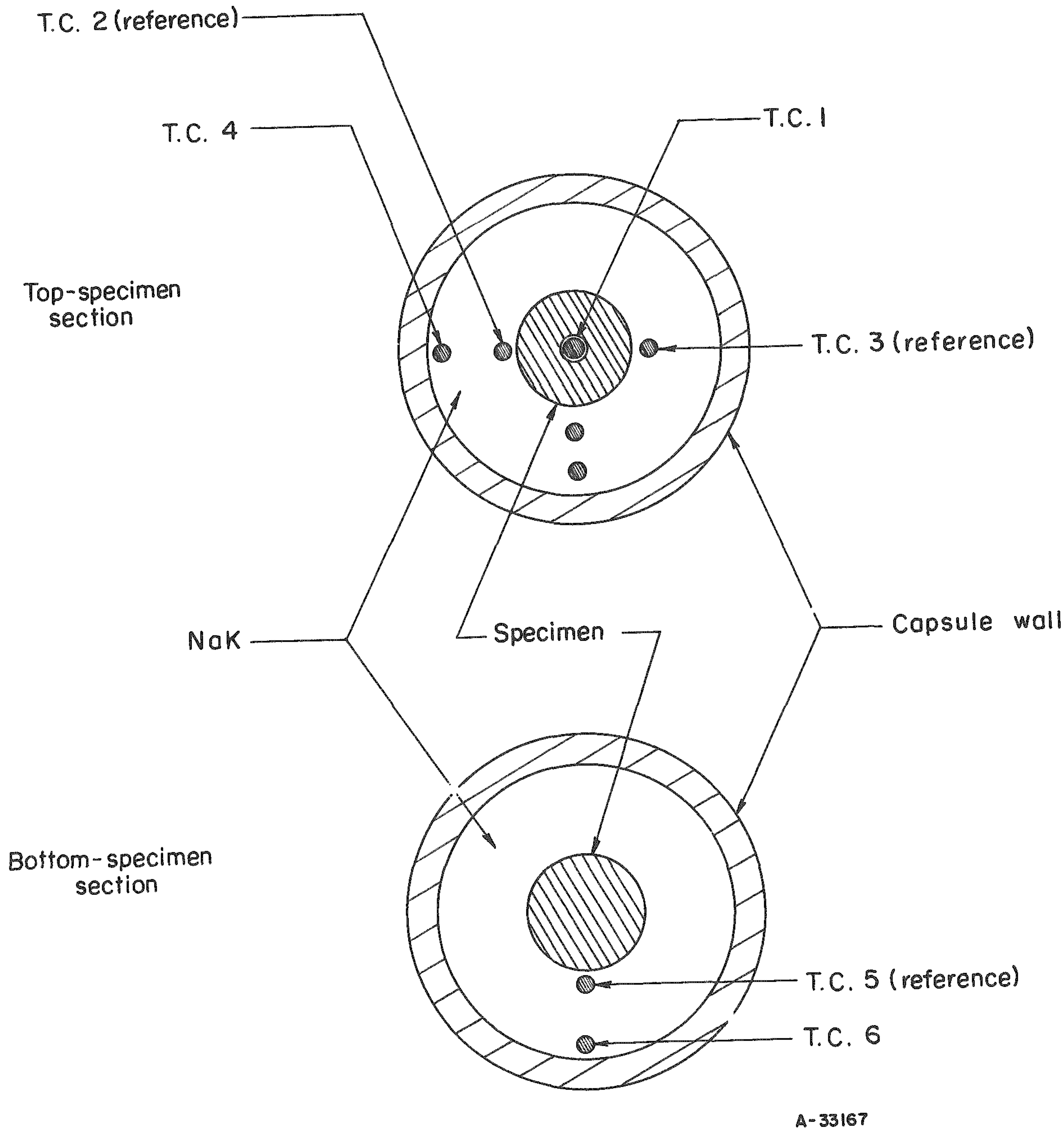

FIGURE 3. SCHEMATIC DIAGRAM OF CAPSULE SECTIONS AT TOP AND BOTTOM SPECIMEN LOCATIONS

of 0.9 was selected for design purposes. As the irradiations have progressed and performance data have become available, it has appeared that the value of 0.9 may have been too high. Actual values during irradiation, however, have been extremely difficult to evaluate. This situation is described later as a part of the interpretation of irradiation data.

Applying the radial-heat-flow equation with appropriate constants for the approximately annular regions, the relationship shown in Figure 4 between nominal fission-heat generation and the temperature sensed by the reference thermocouple located near the specimen surface was established. Similarly, the relationship between the nominal specimen-surface temperature (T_s) and reference thermocouple reading was obtained; it is plotted in Figure 4 also.

Specimen Center-Line Temperature

In designing the capsule, intraspecimen temperatures were estimated on the basis of a thermal conductivity of 14 Btu/(ft)(hr)(F) for uranium-5 w/o carbon. This is an out-of-pile experimental value* found for a specimen maintained at 1000 to 1500 F. For the design fission-heat generation rate of 6400 Btu/(hr)(in.), 1100 F specimen-surface temperature, and a conductivity of 14 Btu/(hr)(ft)(F), the heat-transfer equation ($Q = 4\pi k \Delta T$) for internal-heat-generating cylinders yielded a specimen center-line temperature of 1530 F. This temperature was regarded as fulfilling the original specifications. Figure 4 also includes the reference temperature-nominal specimen center-line temperature (T_c) relationship developed on the basis described.

Specimen Uranium-235 Content and Effective Flux in the Specimen

With the fission-heat generation rate of 6400 Btu/(hr)(in.) established, the remaining parametric adjustment was that between the specimen's uranium-235 content and effective flux in the specimen. Essentially, the governing relationship is

$$Q_f = \phi_e \times E \times A,$$

where

Q_f = fission-heat generation rate

ϕ_e = specimen effective flux

E = uranium-235 content

A = constant incorporating specimen weight, uranium-UC molecular-weight ratio, Avogadro's Number, fission cross section of uranium-235 (592 barns per atom), and heat generation per fission (2.67×10^{-14} Btu per fission or 175 Mev/fission).

*Secrest, A. C., Jr., Foster, E. L., and Dickerson, R. F., "Preparation and Properties of Uranium Monocarbide Castings", BMI-1309, January 2, 1959.

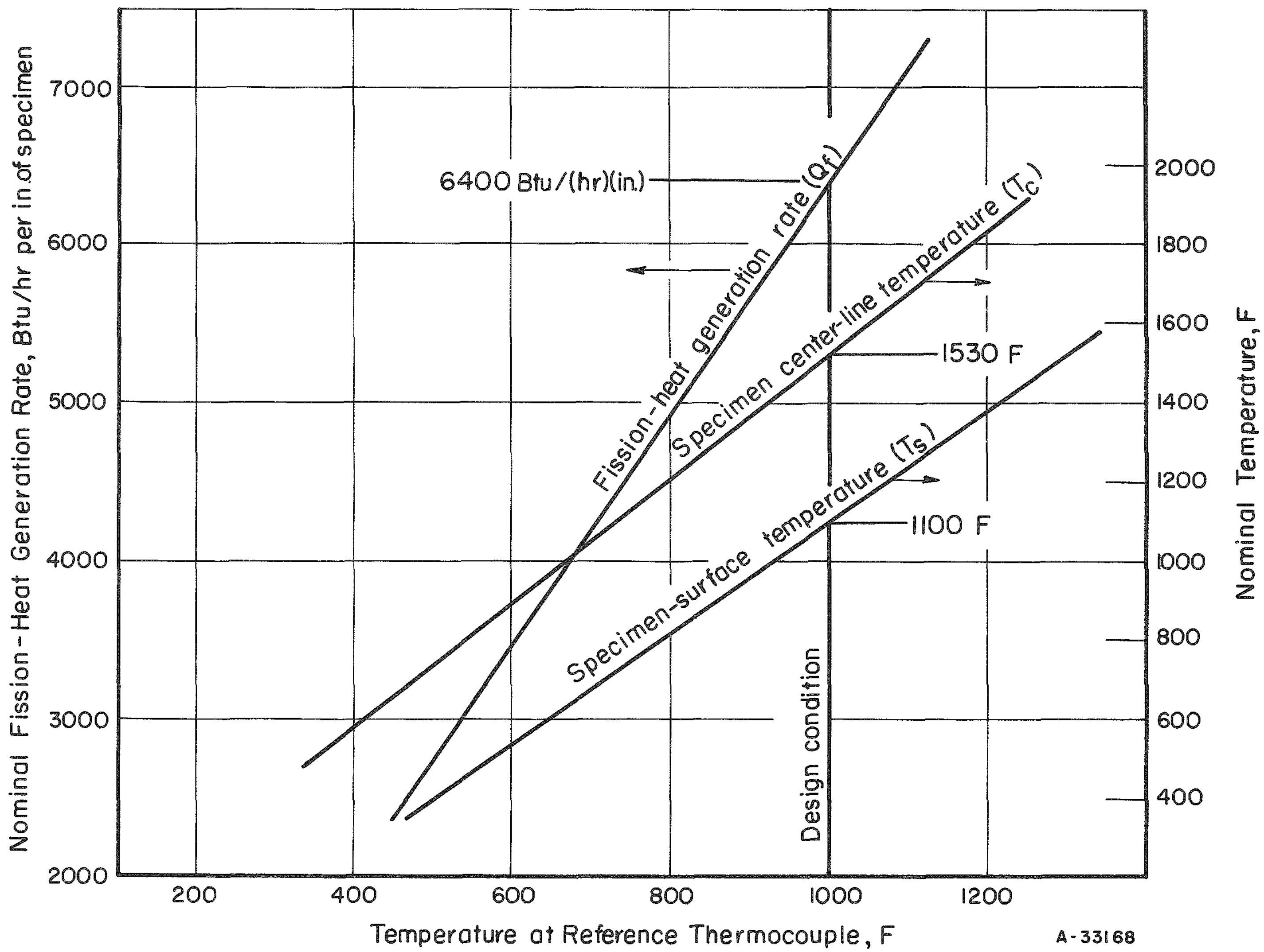

FIGURE 4. NOMINAL PARAMETERS VERSUS TEMPERATURE AT REFERENCE THERMOCOUPLE

Several factors led to the selection of specific design values for ϕ_e and E from this relationship. The principal requirement was that the effective flux in the specimen be acceptable from the viewpoint of the availability of high-flux reactor space. This led to the selection of a specimen flux level of 0.23×10^{14} nv; the corresponding uranium-235 content was 8.3 w/o, giving a nominal fuel loading of 1.9 g of uranium-235 per inch of specimen.

Unperturbed Flux Required

A required unperturbed flux slightly greater than 1.0×10^{14} nv was obtained from the selected effective flux, 0.23×10^{14} nv, through use of the following combination of perturbation factors:

$$\phi_e = \phi_u \times f \times e^{-\Sigma_a x} \times 0.7,$$

where

ϕ_e = effective flux

ϕ_u = unperturbed flux

f = Brad Lewis perturbation factor* (0.33 for the specimens chosen)

$e^{-\Sigma_a x}$ = capsule attenuation factor based on slab theory (0.95 for the capsule conditions chosen)

0.7 = correction factor based on Battelle experience with capsules irradiated at the MTR.

Mechanical Design and Assembly

Specimen Assembly

As shown in Figure 1, the fuel stack in each capsule consists of five unclad UC specimens. Two of these, the main experimental samples, are 2 in. long and are located in the central zone of the capsule. A specimen 1/4 in. long is placed at each end of the stack to minimize thermal and nuclear end effects and a specimen 1/2 in. long separates the two full-length pieces. The 1/2-in. specimen is hollowed out to accommodate a coil of 18-mil-diameter dosimeter wire. The upper 1/4-in. specimen piece and the upper full-length specimen are also hollowed out to accommodate a 1/16-in.-OD thermocouple for measuring the specimen central-core temperature. The holes were drilled ultrasonically.

An expanded-stainless steel basket is employed to retain the specimen stack. The basket assembly is positioned radially in the capsule by a system of hanging rods, centering spiders, and two, thin, half-cylinder basket shrouds, as shown in Figure 2. Adequate clearance is provided between the basket and the specimens to allow unrestrained

*Lewis, W. B., "Flux Perturbations by Material Under Irradiation", Nucleonics (October, 1955).

radial growth of the specimen during irradiation. As indicated in Figure 1, the specimen stack is spring loaded (molybdenum springs) at the ends for firm positioning, yet growth in the axial direction can be accommodated.

The basketed array also provides cushioning against physical damage; however, the stack is not disrupted if one or more of the specimens happens to fracture. Also of importance is the easy access to the specimens during disassembly of the capsule after irradiation.

Thermocouples

Six thermocouples are provided in each capsule to monitor temperatures at varying distances from the center line of the specimens. The thermocouples are 1/16-in. OD, are sheathed with stainless steel, and are insulated with MgO between the 10-mil sheath and the 10-mil Chromel and Alumel wires. The thermocouple junctions are prepared by fusing the wires into the sheath in a manner which forms a sealed tip as well as a grounded junction.

As shown by the schematic drawing in Figure 3, the thermocouple locations are as follows:

T. C. 1 – In the hollow of the upper full-length specimen, giving an indication of specimen central-core temperature (referred to throughout the report as T_1). Because fuel was removed to accomodate this thermocouple, this temperature is slightly lower than the analytically derived specimen center-line temperature referred to as T_c.

T. C. 2 – In the NaK bath as close as possible to the surface of the upper full-length specimen, about halfway along its length.

T. C. 3 – Similar to but 180 deg from T. C. 2. During the irradiations, the readings of T. C. 2 and T. C. 3 were substantially the same.

T. C. 4 – In the NaK bath adjacent to T. C. 2 but against the capsule wall.

T. C. 5 – Similar to T. C. 2 but facing the lower full-length specimen.

T. C. 6 – Similar to T. C. 4 but adjacent to T. C. 5.

As indicated previously, T. C. 's 2, 3, and 5 are regarded as the reference thermocouples, and the temperatures they monitored are used in the various analyses of capsule thermal performance.

Dosimetry

Small-diameter (16 to 25-mil) dosimeter wires are incorporated into the capsule assemblies in three locations:

(1) Axially along the water-contacting surface of the capsule body

(2) Wound spirally around the expanded metal basket

(3) Coiled into the hollow of the 1/2-in.-long center specimen.

Dosimeter compositions were titanium-0.58 w/o cobalt, nickel-0.55 w/o cobalt, and aluminum-0.63 w/o cobalt.

Loading and Assembly

The capsule top, with the basketed specimen assembly hanging from it, was lowered into the capsule body. The bottom of the capsule body section contained a threaded plug provided for NaK loading in a drybox which had been pre-evacuated to about 1 micron and then filled with helium.

Main closures were Heliarc welded. The thermocouple sheaths were brazed into the headers with a nickel-alloy braze having a melting temperature in the 1800 F range. Leak checks were carefully made of all welds and brazes by using a helium mass-spectrometer leak detector.

PLAN OF PRESENTATION OF CAPSULE DATA

The information derived from the capsule-irradiation data is presented as follows:

(1) Pertinent primary data are given. These include central-core temperatures (T_1) and reactor-quoted flux levels during individual irradiation cycles and available dosimeter-derived intracapsule fluxes.

It should be noted that the mass of data from the reference thermocouples (T_2 and T_5) are not presented herein, since these data do not represent actual specimen temperatures of interest but are useful only as starting points in the various analyses.

(2) Reference-thermocouple data are used to estimate fission-heat generation rates (Q_f) and, as a sequel, final specimen-burnup levels. This involves radial heat transfer in the capsule regions between the reference-thermocouple locations and the reactor-coolant-water heat sink. Final specimen-burnup levels are also estimated on the basis of reactor-quoted fluxes and dosimeter data.

(3) Using (a) estimated heat-generation rates (from Item 2), (b) the reference thermocouple data, and (c) the heat-transfer characteristics of the narrow annulus of NaK separating the reference couple and the specimen surface, estimates are made of specimen-surface temperatures.

(4) By using (a) the estimated specimen-surface temperatures (Item 3) and (b) the temperature (T_1) monitored by the central-core thermocouple

located in the top specimen in each capsule, the top-specimen temperature gradients are estimated. This, along with the estimated fission-heat generation rate, permits an estimate of effective thermal conductivity (for top specimens only).

(5) Bottom-specimen center-line temperatures are estimated on the basis of (a) heat generation rate, (b) specimen-surface temperature and (c) effective specimen thermal conductivity. Since the bottom specimens did not contain a central-core thermocouple, the value for effective thermal conductivity used is that calculated as outlined in Item 4 for the top specimen in the capsule.

Included in the discussion of Items 2, 3, 4, and 5 above are estimates of the effects of various uncertainties that can be associated with the governing parameters.

CAPSULE-IRRADIATION DATA

The irradiations of Capsules BMI-23-1 and -23-2 were completed early in 1959, those of Capsules BMI-23-3 and -23-5 were completed in September, and Capsule BMI-23-6 was discharged in December, 1959. Capsule BMI-23-4 is scheduled for discharge in June, 1960.

Capsule-Irradiation-Temperature History

Capsule BMI-23-1

Capsule BMI-23-1 was loaded with 3/8-in.-diameter uranium-5.0 w/o carbon specimens, as indicated in Table 3. The capsule was inserted into MTR Position A28NE during the shutdown for Cycle 107, the cycle lasting 22 days. The capsule was located below the midplane of the reactor core, with the capsule bottom 29 in. from the top of the lattice.

TABLE 3. SPECIMEN DATA FOR CAPSULE BMI-23-1

Specimen[a]	1A	3 (Top)	1B	2 (Bottom)	5
Position in Capsule From the Top	1	2	3	4	5
Specimen Length, in.	0.251	1.800	0.489	2.000	0.251
Fuel Loading, g/in.	1.80	1.83	1.76	1.92	1.92
Density by Immersion, g/cc	13.38	13.36	13.36	13.37	13.36

(a) Atomics International designation.

Nominal in-pile temperature levels for this capsule, as derived from reference-thermocouple data, are given in Table 4. Since the recorded temperature levels fluctuated together, the general temperature history during the cycle can be described by tracing the temperatures indicated by the thermocouple inserted into the top specimen

(TC 1). At the start of the cycle, this temperature was 1390 F but decreased steadily to about 1100 F at midcycle shutdown. During this shutdown, the capsule was raised 3 inches to increase operational temperature. At startup, the monitored central-core temperature was 1600 F, but this had steadily dropped to 1100 F by the end of the cycle.

TABLE 4. SUMMARIZED NOMINAL TEMPERATURE LEVELS FOR CAPSULE BMI-23-1 DURING CYCLE 107

Derived Surface Temperature of Top Specimen[(a)], F			Derived Surface Temperature of Bottom Specimen[(a)], F			Measured Central-Core Temperature of Top Specimen[(b)], F			Derived Center-Line Temperature of Bottom Specimen[(c)], F		
Max	Min	Mean	Max	Min	Mean	Max	Min	Mean	Max	Min	Mean
880	720	770	620	470	540	1600	1100	1300	1200	800	980

(a) As indicated in the text, T_s (nominal) is derived from T_2 (or T_5 in the case of the bottom specimen) and the T_s-T_2 relation in Figure 4.

(b) Measured by T. C. 1.

(c) As indicated in the text, T_c (nominal) is derived from T_5, estimated heat generation rate, and an effective thermal conductivity calculated from the thermal performance of the top specimen.

Capsule BMI-23-2

Specimen data for Capsule BMI-23-2 are listed in Table 5. This capsule was inserted into the A28NE position during the shutdown for Cycle 110 and located just above the midplane of the reactor core. The capsule bottom was 18 in. from the top of the reflector.

TABLE 5. SPECIMEN DATA FOR CAPSULE BMI-23-2

Specimen	31 A	26 (Top)	31 B	37 (Bottom)	36 A
Position in Capsule From the Top	1	2	3	4	5
Specimen Length, in.	0.249	1.996	0.479	1.897	0.254
Fuel Loading, g/in.	1.76	1.81	1.74	1.92	1.96
Density by Immersion, g/cc	13.36	13.34	13.39	13.43	13.52

Nominal temperature data for this capsule are presented in Table 6. Again, the general temperature history may be traced by referring to the recorded central-core levels. At the beginning of Cycle 110 the core temperature was approximately 1550 F; as the cycle progressed, this temperature rose until at the end of the cycle it reached 1830 F. To lower the core temperature, the capsule was raised 1 in. at shutdown for Cycle 111. During Cycle 111, the central-core temperature increased steadily from an initial 1160 F to a final 1540 F. Subsequently, the capsule position was slightly modified (raised about 1/2 in.), and during Cycle 112 the central-core temperature ranged from 1150 to 1380 F. There was no change in position for Cycle 113, and during this cycle the central-core temperature increased slowly from 920 F at the start to 1490 F at the end. During Cycle 114 shutdown, the capsule was lowered 1 in.; central-core temperatures were 1260 to 1540 F during the cycle. The capsule remained in this position during

Cycle 115, the sixth and last cycle. The central-core-temperature range of 1270 to 1450 F during the final cycle was slightly lower than that during the previous cycle.

TABLE 6. SUMMARIZED NOMINAL TEMPERATURE LEVELS FOR CAPSULE BMI-23-2[a]

	Derived Surface Temperature of Top Specimen, F			Derived Surface Temperature of Bottom Specimen, F			Measured Central-Core Temperature of Top Specimen, F			Derived Center-Line Temperature of Bottom Specimen, F		
Cycle	Max	Min	Mean	Max	Min	Mean	Max	Min	Mean	Max	Min	Mean
110	1070	920	1010	900	650	790	1830	1400	1600	1620	930	1340
111	970	790	890	770	640	720	1540	1160	1360	1250	940	980
112	910	760	830	710	600	660	1380	1150	1260	1120	960	1060
113	950	630	830	730	500	650	1490	920	1280	1130	760	1050
114	980	810	910	780	650	710	1540	1260	1420	1210	990	1150
115	890	830	880	710	640	680	1450	1270	1340	1100	960	1090

(a) Refer to footnotes for Table 4.

Capsule BMI-23-3

Table 7 presents specimen data for Capsule BMI-23-3, which was inserted into Position A28SE of the MTR at the shutdown for Cycle 116 for a total irradiation of 12 cycles. The capsule was located just above the midplane of the core with the capsule bottom being 18 in. from the top of the reflector.

TABLE 7. SPECIMEN DATA FOR CAPSULE BMI-23-3

Specimen	47	28 (Top)	33A	29 (Bottom)	366
Location in Capsule From the Top	1	2	3	4	5
Specimen Length, in.	0.244	1.991	0.494	1.999	0.253
Fuel Loading, g/in.	1.79	1.88	1.80	1.94	1.94
Density by Immersion, g/cc	13.50	13.45	13.50	13.46	13.48

Core temperatures, as indicated by the thermocouple located in the center of the top 2-in. specimen, were below the design temperature of 1500 F during Cycle 116, ranging from 900 to 1360 F. The capsule was lowered 2-1/2 in. at the end of the cycle, and core temperatures for the next cycle ranged from 1400 to 1550 F. To allow for a shift in the regulating rods, the capsule was lowered another inch at the end of Cycle 117 and remained in that position for the remainder of the irradiation. Central-core temperatures for Cycle 118 were again in the 1120 to 1350 F range, but they ranged from 1160 to 1430 F during Cycle 119. Core temperatures for the remaining cycles were low and were near 1100 F. The thermocouple located in the center of the top specimen failed during Cycle 121, and thereafter, core temperatures were estimated from the readings of thermocouples still operating. Maximum, minimum, and mean core and surface temperatures (where available) for the top and bottom 2-in. -specimens for each cycle are given in Table 8.

TABLE 8. SUMMARIZED NOMINAL TEMPERATURE LEVELS FOR CAPSULE BMI-23-3[a]

	Derived Surface Temperature of Top Specimen, F			Derived Surface Temperature of Bottom Specimen, F			Measured Central-Core Temperature of Top Specimen, F			Derived Center-Line Temperature of Bottom Specimen, F		
Cycle	Max	Min	Mean	Max	Min	Mean	Max	Min	Mean	Max	Min	Mean
116	800	550	720	710	520	580	1360	900	1190	1640	870	1000
117	890	840	860	680	660	670	1550	1400	1460	1440	1240	1310
118	840	720	790	640	570	620	1350	1120	1220	1090	920	1010
119	870	760	820	660	610	640	1430	1160	1270	1210	1020	1100
120	820	450	760	660	410	610	1300	610	1170	1120	570	1030
121[b]	880	830	860	620	670	680	1420	1300	1340	1150	1000	1120
122	780	780	750	640	550	620	--	--	1150	1110	960	1070
123[c]	800	600	720	--	--	--	--	--	1100	--	--	--
124[d]	780	680	700	--	--	--	--	--	1100	--	--	--
125	--	--	--	--	--	--	--	--	1100	--	--	--
126	--	--	--	--	--	--	--	--	1100	--	--	--
127	--	--	--	--	--	--	--	--	1100	--	--	--

(a) Refer to footnotes for Table 4.
(b) Thermocouple located in center of top 2-in. specimen failed during the cycle. Mean values for following cycles are estimates from readings of the other operating thermocouples.
(c) Thermocouple near surface of bottom specimen failed.
(d) Thermocouple near surface of top specimen failed.

Capsule BMI-23-5

Capsule BMI-23-5 was loaded with 3/8-in. -diameter uranium-4.6 w/o carbon specimens, as indicated in Table 9. The capsule was inserted into Position A30NE of the MTR at the shutdown for Cycle 122, for a total irradiation of 12 cycles. The capsule was located at about the midplane of the core with the capsule bottom being 24 in. from the top of the reflector. The capsule remained in this location for the entire irradiation of six cycles.

TABLE 9. SPECIMEN DATA FOR CAPSULE BMI-23-5

Specimen	49 B	42 (Top)	44 A	50 B (Bottom)	50 A
Location in Capsule From the Top	1	2	3	4	5
Specimen Length, in.	0.252	1.996	0.619	1.866	0.250
Fuel Loading, g/in.	1.93	1.92	1.94	1.96	1.99
Density by Immersion, g/cc	13.84	13.74	13.83	13.83	13.82

The core temperature, as indicated by the thermocouple located in the center of the top 2-in. specimen, was about 1380 F for the entire Cycle 122. Core temperature during the first part of Cycle 123 was also about 1380 F, reaching a maximum at 1420 F. However, at this point, the thermocouple failed. The core temperature for the remainder of the cycle appeared to be about 1400 F, as estimated from the thermocouples near the surface of the top specimen. Core temperatures for Cycle 124 were estimated to be in the 1250 to 1360 F range, which was lower than previous temperatures. However, core temperatures during Cycle 125 rose from a low of 1270 F to a high of about 1370 F. Core temperatures dropped in Cycle 126 with a maximum of about 1360 F. The core temperature was fairly constant during Cycle 127 with a short peak to about 1400 F at midcycle. Maximum, minimum, and mean core and surface temperatures for the top and bottom 2-in. specimens for each cycle are given in Table 10.

TABLE 10. SUMMARIZED NOMINAL TEMPERATURE LEVELS FOR CAPSULE BMI-23-5[(a)]

	Derived Surface Temperature of Top Specimen, F			Derived Surface Temperature of Bottom Specimen, F			Measured Central-Core Temperature of Top Specimen, F			Derived Center-Line Temperature of Bottom Specimen, F		
Cycle	Max	Min	Mean	Max	Min	Mean	Max	Min	Mean	Max	Min	Mean
122	900	860	890	820	760	800	1380	1330	1360	1440	1240	1330
123[(b)]	990	940	980	850	820	860	1420	1370	1400	1260	1220	1250
124	880	820	860	740	720	730	1360	1250	1300	1230	1170	1200
125	980	930	840	830	740	800	1370	1270	1300	1380	1130	1270
126	910	760	840	780	660	730	1360	1200	1300	1230	1110	1180
127	930	820	880	790	690	730	1400	1200	1300	1340	1210	1260

(a) Refer to footnotes for Table 4.

(b) Thermocouple located in the center of the top specimen failed. Values reported for core temperatures are estimated on the basis of readings of surrounding thermocouples.

Capsules BMI-23-4 and -6

The irradiation of Capsules BMI-23-4 and -6 was carried out in the MTR in Positions A27SE and A13NE, respectively. Maximum, minimum, and mean core and surface temperatures for the top and bottom specimens are given in Table 11.

TABLE 11. SUMMARIZED NOMINAL TEMPERATURE LEVELS FOR CAPSULES BMI-23-4, BMI-23-6(a)

Cycle	Derived Surface Temperature of Top Specimen, F			Derived Surface Temperature of Bottom Specimen, F			Measured Central-Core Temperature of Top Specimen, F			Derived Center-Line Temperature of Bottom Specimen, F		
	Max	Min	Mean	Max	Min	Mean	Max	Min	Mean	Max	Min	Mean
						Capsule BMI-23-4						
115	790	750	760	620	570	590	1320	1190	1250	1110	980	1040
116	740	690	720	670	510	590	1280	1120	1190	1410	840	1090
117	750	710	930	580	510	530	1260	1200	1230	1170	920	980
118	630	570	610	460	440	450	1030	940	1000	860	770	820
119	710	660	690	540	500	520	1170	1100	1160	990	910	940
120	710	390	670	530	300	490	1200	600	1120	980	450	900
121	750	730	740	560	530	540	1280	1230	1250	1120	1030	1070
122	720	670	690	550	500	510	1210	1110	1140	960	860	880
123	690	660	680	520	490	510	1120	1040	1100	910	830	870
124	660	630	650	500	480	490	1020	1080	970	820	780	810
125	700	610	670	530	480	510	1120	970	1070	930	810	870
126	670	620	630	520	490	500	1030	990	1010	900	820	840
127	680	660	670	540	520	530	1110	1060	1080	950	910	930
						Capsule BMI-23-6(b)						
126	1030	890	980	810	620	700	--	--	--	--	--	--
127	930	850	900	690	600	670	--	--	--	--	--	--

(a) Refer to footnotes for Table 1.

(b) Thermocouple located in hollow of top specimen failed at startup.

Flux History for Discharged Capsules

Exposure in terms of MWD and reactor-quoted unperturbed thermal fluxes for Capsules BMI-23-1, -23-2, -23-3, and -23-5 are given in Table 12.

TABLE 12. FLUX HISTORY FOR CAPSULES BMI-23-2, BMI-23-3, AND BMI-23-5

Cycle	Reactor MWD	Days	Reactor-Quoted Unperturbed Flux, nv x 10^{14} Top 2-In. Specimen	Reactor-Quoted Unperturbed Flux, nv x 10^{14} Bottom 2-In. Specimen
			Capsule BMI-23-1	
107a	499	12.6	0.89	0.76
107b	371	9.3	0.96	0.89
			Capsule BMI-23-2	
110	585	14.6	0.85	0.97
111	608	15.2	0.77	0.93
112	554	13.8	0.77	0.93
113	572	14.3	0.69	0.93
114	568	14.2	0.77	0.93
115	585	14.6	0.77	0.93
			Capsule BMI-23-3	
116	673	17.4	0.70	0.85
117	624	15.9	0.95	1.05
118	557	14.7	1.00	1.10
119	587	15.5	1.00	1.10
120	633	16.8	1.00	1.10
121	631	16.5	1.00	1.10
122	640	16.8	1.00	1.10
123	599	16.2	1.00	1.10
124	666	17.2	1.00	1.10
125	566	15.7	1.00	1.10
126	552	14.1	1.00	1.10
127	560	15.5	1.00	1.10
			Capsule BMI-23-5	
122	640	16.8	0.75	0.80
123	599	16.2	0.75	0.80
124	666	17.2	0.75	0.80
125	566	14.1	0.75	0.80
126	552	14.1	0.75	0.80
127	560	15.5	0.75	0.80

Available Dosimetry Results

Postirradiation dosimetry has been completed for Capsules BMI-23-1 and BMI-23-2, and results are given in Table 13.

TABLE 13. DOSIMETRY DATA FOR CAPSULES BMI-23-1 AND BMI-23-2

Wire Composition	Wire Location	Indicated Average Thermal-Neutron Flux, nv x 10^{13}
	Capsule BMI-23-1	
Nickel-0.62 w/o cobalt	In hollow of center piece	1.95
Titanium-0.58 w/o cobalt	At surface of specimen stack-A[a]	3.71
	-B	3.48
	-C	4.23
	-D	5.89
	Capsule BMI-23-2	
Titanium-0.58 w/o cobalt	In hollow of center piece	2.26
Titanium-0.58 w/o cobalt	At surface of specimen stack[b]	
	Surface of upper specimen	4.34
	Surface of center piece	3.68
	Surface of bottom specimen	3.45
Titanium-0.58 w/o cobalt	At surface of capsule shell	6.02
Aluminum-0.63 w/o cobalt	At surface of capsule shell	3.65
Nickel-0.55 w/o cobalt	At surface of capsule shell	4.67
Nickel-0.55 w/o cobalt	At surface of capsule shell	4.99

(a) The wire spiraled around the specimen stack was cut into four equal sections. The value of the flux at the surface of the center piece, 3.85 x 10^{13} nv, is the average of the fluxes of wires B and C

(b) The wire spiraled around the specimen stack, in this case, was cut into three sections. Each section was cut approximately from the wire opposite the two specimens and the center piece.

ANALYSIS OF SPECIMEN BURNUP

Nominal Levels

The in-pile capsule temperatures, reported flux history, and postirradiation dosimetry results provide the basic information for estimating nominal specimen-burnup levels by several analytical methods. Five methods were employed:

Method I. Fission-heat generation rate obtained from the design relation between reference-thermocouple reading and heat flux, shown in Figure 4.

Method II. Effective flux derived from reactor-quoted unperturbed fluxes by the combination of perturbation factors (Brad Lewis factor, Battelle experience factor, and the capsule-attenuation factor) presented previously.

Method III. Effective flux derived from the integrated flux level at the specimen surface, provided by dosimetry, in combination with the Brad Lewis specimen-attenuation factor Y only.[a]

Method IV. Effective flux derived from the specimen-surface dosimeter flux in combination with a specimen-attenuation factor provided by IBM-650 calculation with the P-3 code.[b]

Method V. Effective flux derived from specimen-surface dosimeter fluxes in combination with a specimen-perturbation factor calculated from the dosimetry at the specimen surface and in the core of the central 1/2-in.-long specimen. An approximation to the diffusion equation was used to obtain the specimen-perturbation factor.[c]

(a) The Brad Lewis attenuation factor, f, is a product of F (a ratio which represents the flux depression from the unperturbed level to the level at the specimen surface) and Y (a ratio representing the depression from the specimen surface inward). Y can be obtained directly from the geometric and nuclear characteristics of the specimen, using the Brad Lewis curve. Multiplying the surface-dosimeter flux by Y gives the mean specimen flux, ϕ_e.

(b) "The P-3 Program for the IBM 650 Computer", DC 56-7-30, May 31, 1956.

(c) Gladstone, S., and Edlund, M. G., "The Elements of Nuclear Reactor Theory", D. Van Nostrand Company, Inc. (1952).

Table 14 presents nominal burnup results for specimens from Capsules BMI-23-1 and BMI-23-2 as derived by the methods outlined above. Preliminary data obtained by Method II for recently opened Capsules BMI-23-3 and BMI-23-5 are also given. It may be seen that there is some disagreement among the burnup levels yielded by the various methods. To obtain some insight into the discrepancies, various sources of uncertainty were analyzed. In essence, the uncertainties in the nominal relation established for fission-heat generation rate and reference temperature (Method I) were examined by a statistical approach. Also, uncertainty limits were assigned to the major independent parameters in Methods II through V, and the net uncertainties were estimated. In these latter cases, an attempt was made to include in the analysis commonly accepted sources of error and uncertainties that would represent a typical rather than an all-inclusive situation. Lastly, the burnup levels with the estimated uncertainty ranges were compared.

TABLE 14. ESTIMATED NOMINAL BURNUP LEVELS DERIVED BY VARIOUS METHODS

Method of Determining Effective Flux	Top Specimen: Effective Flux, nv x 10^{13}	Top Specimen: Burnup[a] a/o Uranium (Fission Only)	Top Specimen: Burnup[a] MWD/T of Uranium	Bottom Specimen: Effective Flux, nv x 10^{13}	Bottom Specimen: Burnup[a] a/o Uranium (Fission Only)	Bottom Specimen: Burnup[a] MWD/T of Uranium
Capsule BMI-23-1						
Method I	1.52	0.153	1,150	1.15	0.116	870
Method II	2.12	0.213	1,590	1.89	0.190	1,420
Method III	1.93	0.195	1,460	1.54	0.156	1,160
Method IV	2.90	0.290	2,170	2.31	0.232	1,740
Method V	3.12	0.315	2,360	2.48	0.249	1,870
Capsule BMI-23-2						
Method I	1.61	0.59	4,400	1.40	0.52	3,900
Method II	1.67	0.61	4,600	2.01	0.74	5,500
Method III	1.50	0.55	4,100	2.15	0.81	6,100
Method IV	2.26	0.82	6,100	3.19	1.10	8,300
Method V	2.48	0.90	6,800	3.49	1.28	9,600
Capsule BMI-23-3						
Method II[b]	2.17	1.75	13,000	2.39	1.93	14,300
Capsule BMI-23-5						
Method II[b]	1.63	0.66	4,900	1.74	0.70	5,300

(a) Based on 175 Mev/fission.

(b) Opening of capsules BMI-23-3 and -23-5 was completed too recently for inclusion of dosimeter results in this report. Method II was used as an expedient way of obtaining preliminary burnup data for this table.

The following paragraphs involve a discussion of the various steps and results yielded by this approach.

Uncertainties

Uncertainties in Method I

Figure 5 is essentially a reproduction of the curve in Figure 4, with the addition of a derived uncertainty band.* The individual uncertainty limits are listed in Table 15 below. Of these limits, that associated with thermocouple location is the most important, not only because it is relatively large but also because it is applicable in a high-temperature-gradient zone.

TABLE 15. ASSUMED UNCERTAINTY LIMITS FOR METHOD I

Parameter	Nominal Value	Assumed Uncertainty Limits, per cent
Distance of T_2, T_3, and T_5 from specimen center line	0.219 in.	±14 (equivalent to ± 1/32 in.)
Radius of outer shell	0.563 in.	±0.4
Radius of specimen	0.188 in.	±1
Thermal conductivity of NaK	15.1 Btu/(hr)(ft)(F)	±5
Thermal conductivity of stainless steel	9.4 Btu/(hr)(ft)(F)	±5

The application of these individual uncertainties leads to an estimated uncertainty of +25, -15 per cent in the fission-heat generation rate, and hence in the estimated burnup. The asymmetry of the uncertainty stems from the logarithmic term in the radial heat-flow equation.

Uncertainties in Method II

A net uncertainty of ±28 per cent is estimated for this method. This represents a combination of ±20 per cent in reactor-quoted flux and ±20 per cent in capsule-perturbation factor. The fairly standard limit of ±20 per cent for reactor-quoted fluxes is generally satisfactory, to account for factors such as fuel depletion, the experimental load, and control rod programming.

The constant 0.7 is employed in Method II as part of the array of factors used to estimate effective fluxes from reactor-quoted fluxes. This constant is based on a

*The individual tolerances were combined by the following equation based on a Taylor series:

$$P_y = \frac{1}{\bar{y}} \sqrt{\sum_{n=1}^{i} \left(\bar{x}_i \frac{dy}{dx_i} P_{xi} \right)^2}$$

where P_y and P_x are independent fractional deviations, and $\bar{y}$ and $\bar{x}$ are nominal values.

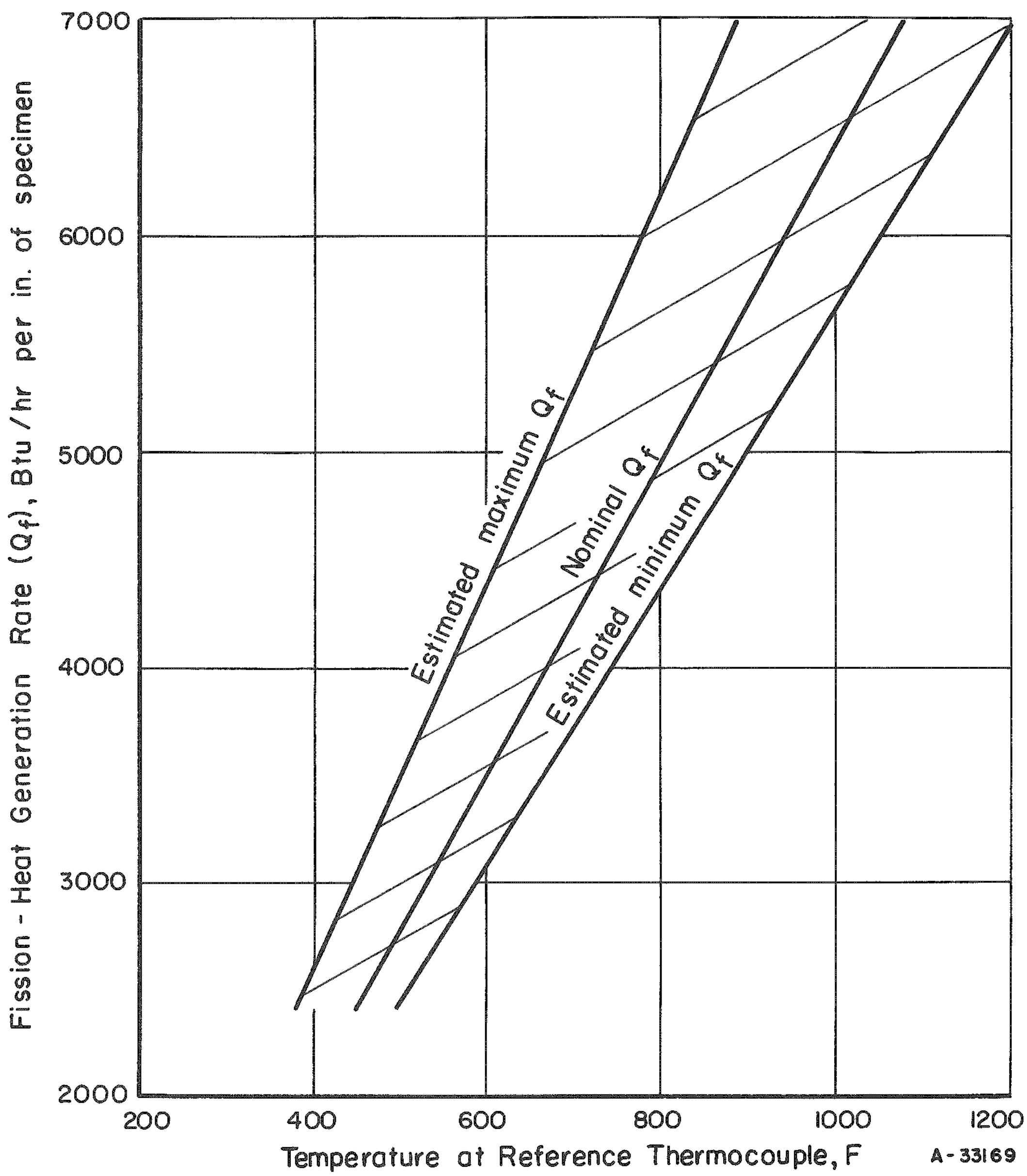

FIGURE 5. FISSION-HEAT GENERATION RATE VERSUS TEMPERATURE AT REFERENCE THERMOCOUPLE READING WITH ESTIMATED UNCERTAINTY BAND

variety of Battelle irradiation data from MTR which indicate that the application of a factor of 0.7 is justified. However, it seems safe to say that the uncertainty in the total factor is still ±20 per cent and in some cases probably higher.

At this time, adequate information is not available to pinpoint the factor or combination of factors which this constant is aimed at correcting. It is believed, however, from a limited number of comparisons with net perturbations derived by machine calculations, that the correction may be primarily to account for a component of the perturbation offered by the capsule body itself which is not accounted for by the $e^{-\Sigma_a x}$ slab-theory attenuation factor.

Uncertainties in Methods III, IV, and V

These methods are based on a common dosimeter-derived integrated flux at the specimen surface. Nominally, ±15 per cent is associated with the analysis of dosimeter wires to obtain integrated fluxes. However, these limits presume no error stemming from sources such as dosimeter composition and dosimeter sampling technique. During hot-cell operations, precise sampling usually proves to be a problem. While dosimetry data can usually be viewed with confidence, they can also introduce vexing questions. For example, the capsule-shell dosimeters from Capsule BMI-23-1 indicate flux variations of nearly 100 per cent at the same level around the capsule periphery. Such a discrepancy cannot be rationalized easily.

While the uncertainty associated with dosimeter-derived integrated flux can be readily estimated, the uncertainties associated with the nominal specimen-attenuation factors* employed in Methods III, IV, and V are difficult to appraise. For the purpose of this analysis, each of these is assigned a limit of ±15 per cent. When combined with the ±15 per cent for dosimetry, net uncertainty is about ±20 per cent in each case.

In the case of the Brad Lewis Method, the assignment of an uncertainty limit to the Y component depends in part on the limit assigned to the total specimen-capsule perturbation factor. Unfortunately, there is no acceptable basis to judge the apportionment of errors among the various components. If, however, ±20 per cent is assigned to the total, as was done in connection with Method III, it is not unreasonable to assign ±15 per cent to the Y component alone.

The machine calculations used for Method IV can be expected to yield fairly valid information about the net perturbation through a given geometry, and the assignment of ±15 per cent to the calculations would appear to be safe. The chief limitation with this type of machine calculation is the difficulty of predicting quantitative neutron-flux levels without complete knowledge of source levels.

The uncertainty associated with the specimen-attenuation factor as derived in Method V is particularly difficult to assess, even though the underlying data are furnished entirely by measured dosimeter activities. The technique employed to translate specimen-boundary integrated flux to a specimen-attenuation factor is based on the assumption that the flux distribution through the specimen will follow a diffusion-type function, amenable to a Bessel solution. In essence, an effective neutron diffusion length

* As a matter of interest, these factors are 0.45 (Brad Lewis Y factor for Method III), 0.67 (obtained from machine calculation, Method IV), and 0.74 (derived from surface and central-core dosimeters, from Capsule BMI-23-2, Method V).

was calculated from boundary flux levels rather than estimated on the basis of physical-property data. It seems apparent that these manipulations can be subject to an uncertainty of ±15 per cent and probably higher.

Comparison of Estimated Burnups Considering Uncertainties

The bar charts in Figures 6 and 7 present comparisons of burnup levels resulting when the foregoing uncertainty limits are factored into the nominal burnup data given in Table 14. In general, the burnup ranges thus established do not consistently overlap, thus indicating that the uncertainty factors considered are not of sufficient magnitude or that unknown sources of error are significant.

There is probably enough evidence to be fairly certain that the burnup levels yielded by Method I are somewhat low. In examining this, it is apparent that a value of the nonradial component of the heat flow larger than the 10 per cent ($e = 0.9$) employed for design and calculational purposes would bring the Method I results into closer agreement with the others. However, the data do not permit the selection of a specific value for e because (1) there is no basis for choosing which of Methods II through V should be the standard of comparison and (2) nonradial heat flow effects appear to have been different in Capsules BMI-23-1 and -23-2, even though the capsules were similar in design and construction.

Qualitatively, this latter point can be illustrated by the information in the following compilation:

Capsule	Axial Location in Irradiation Position	Ratio of Top- to Bottom-Specimen Flux According to Position	Ratio of Top- to Bottom-Specimen Burnup Based on Flux Considerations (and Confirmed by Dosimetry)	Ratio of Top- to Bottom-Specimen Burnup as Calculated by Method I
BMI-23-1	Below peak-flux plane	>1	>1	>1
BMI-23-2 (as well as -23-3, and -23-5)	Above peak-flux plane	<1	<1	>1

As indicated, Capsule BMI-23-2 was located in the reactor above the reported peak-flux plane. There is, therefore, good reason to believe that the rate of heat generation in the bottom specimen was greater than in the top specimen. However, the bottom-specimen temperature (and hence the burnup based on Method I) was lower. Unlike this, Capsule BMI-23-1 was located below the peak-flux plane. In this case, as might be expected, the temperature of the upper specimen was higher than that of the lower.

The apparent inconsistency for Capsule BMI-23-2 must be related to the fact that the opportunity for heat escape from the bottom zone of the capsule is somewhat greater than from the top zone; the latter is insulated by the blanket gas above the NaK. Even so, this explanation is not totally satisfactory, considering the thermal performance of Capsule BMI-23-1. In this case, the bottom specimen might be expected to have resided at

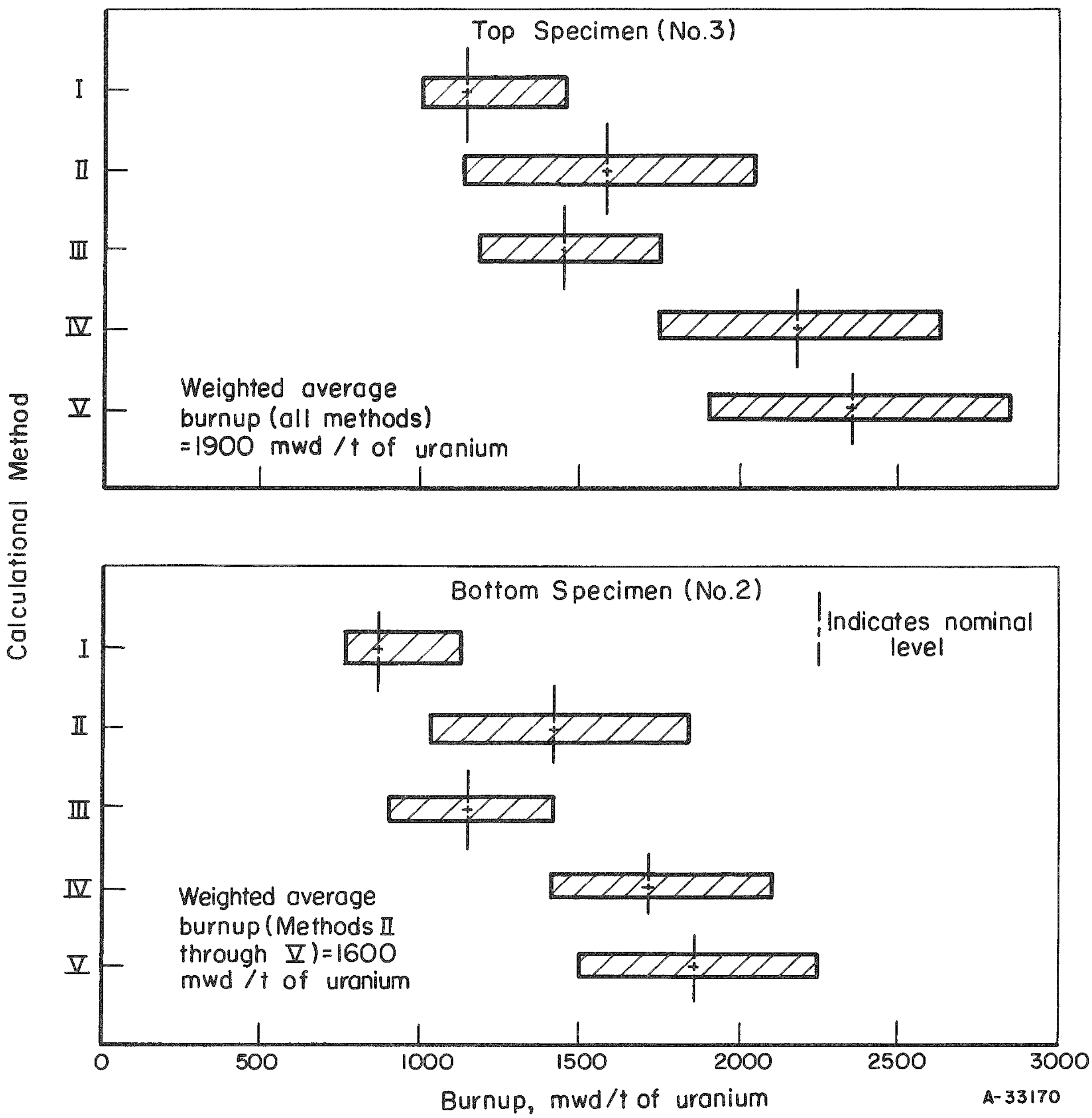

FIGURE 6. SPECIMEN-BURNUP DATA FOR CAPSULE BMI-23-1 WITH ESTIMATED UNCERTAINTY BANDS

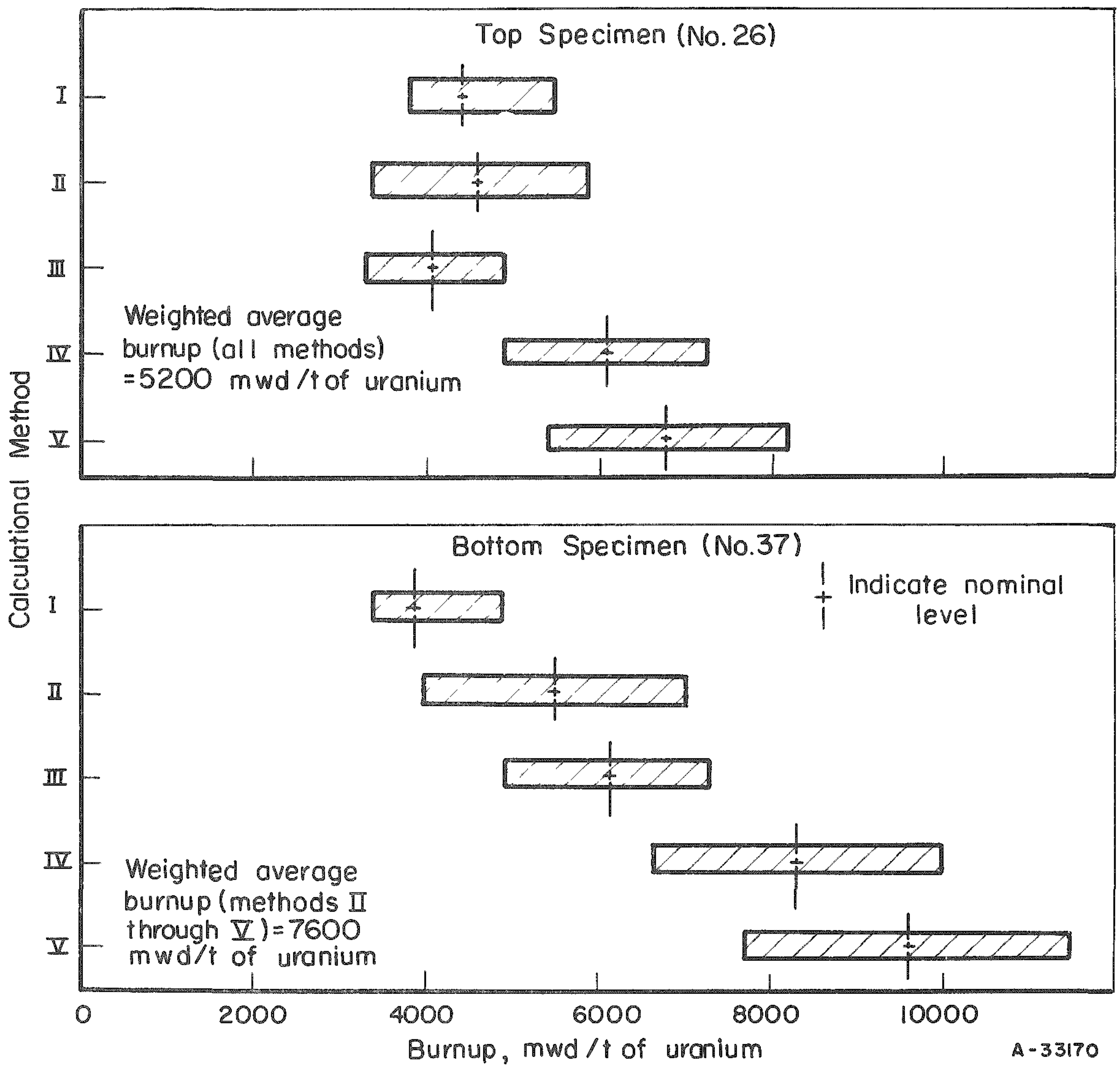

FIGURE 7. SPECIMEN-BURNUP DATA FOR CAPSULE BMI-23-2 WITH ESTIMATED UNCERTAINTY BANDS

a lower temperature than was actually the case, if, in this capsule, a top zone-bottom zone heat-loss inequality had existed in the same proportion as indicated for Capsule BMI-23-2.

While the analysis in the above paragraphs leaves much to be desired, the reasoning on which it is based would point to the conclusion that a factor of e = 0.9 for Method I yields a lesser error in computing top-specimen burnup than in computing bottom-specimen burnup. Unfortunately, as stated before, there appears to be no acceptable way to assign specific values to e which fit the various cases.

Estimates of Mean Burnup Levels

It is believed that the foregoing discussion demonstrates some of the difficulties associated with the interpretation of capsule data in terms of specimen-burnup levels. This entire area, of course, needs considerable intensive study before the state of the art is sufficiently advanced to meet the needs of reactor fuel technology.

In the case of all irradiation experiments involving fueled material, a desired end product of an analysis of capsule data is the best possible prediction of specimen-burnup levels. The mean levels indicated in Figures 6 and 7 are included to fulfill this requirement. The calculation of these levels was based on the following expression:

$$\beta_{wm} = \frac{\Sigma(R_i\, \beta m_i)}{\Sigma R_i},$$

where

R_i = the range of burnup (based on uncertainties) for the ith method.

βm_i = the average burnup given by the ith method

β_{wm} = the weighted mean burnup for i methods.

In the case of the top specimens, all five burnup-estimating methods are included in the weighting; however, for reasons outlined in the previous section, Method I values are omitted in weighting the bottom-specimen levels.

ANALYSIS OF SPECIMEN THERMAL PERFORMANCE BASED ON METHOD I HEAT-GENERATION RATES

As indicated previously, fission-heat generation rates can be employed in the estimation of specimen-surface temperatures, effective specimen thermal conductivity in pile, and specimen center-line temperatures. These factors are discussed in the following sections on the basis of heat-generation rates derived by Method I. Since it would appear that these heat-generation rates may be somewhat lower than actually existed during the irradiations, the calculated values which appear in various tables and plots carry the following qualifications:

(1) The specimen-surface temperatures may be slightly low by virtue of the direct influence of Q_f in the calculation of temperature gradient between the reference thermocouple and the specimen surface (T_s - T_2 or T_5).

(2) In similar fashion, the effective specimen thermal conductivities may be low since they are calculated essentially on the basis of the ratio $Q_f/(T_1 - T_s)$. Since the denominator is only slightly affected by Q_f, it follows that a shift in Q_f will produce a corresponding shift in k_{eff}.

(3) Where k_{eff}'s are calculated as indicated above, the specimen center-line temperatures (T_c) are affected only by the small error which exists in T_s as a result of a variance in Q_f.

Specimen-Surface Temperature

Nominal specimen-surface temperatures are listed in Tables 4, 6, 8, 10, and 11. Figure 8 presents the nominal T_2 - T_s relation with an estimated uncertainty band derived on the basis of uncertainties in Q_f shown in Figure 5. Note that an error in the neighborhood of ±100 F is applicable over the temperature range of interest.

Specimen Effective Thermal Conductivity

An important aspect of this program is the evaluation of the thermal-conductivity behavior of UC during irradiation. For this, use was made of the basic heat-flow equation relating thermal conductivity, difference between the temperature measured by the central-core thermocouple and the reference thermocouple in each capsule, and rate of fission-heat generation:

$$\underbrace{T_1 - T_2}_{\text{(Term I)}} = \underbrace{Q_f \frac{\ln(r_2/r_s)}{2\pi k_{NaK}}}_{\text{(Term II)}} + \underbrace{Q_f \left[\frac{1}{2\pi k_{eff}} \left(\frac{1}{2} - \frac{r_1^2}{r_s^2 - r_1^2} \ln \frac{r_s}{r_1} \right) \right]}_{\text{(Term III)}}$$

where

Term I = the net temperature difference mentioned above

Term II = the difference between the temperature at the reference couple and at the specimen surface

Term III = the temperature drop across the specimen. The deviation in this term from the standard expression, $\Delta T = \frac{Q}{4\pi k}$ stems from the fact that a central hole is present to accommodate the thermocouple.

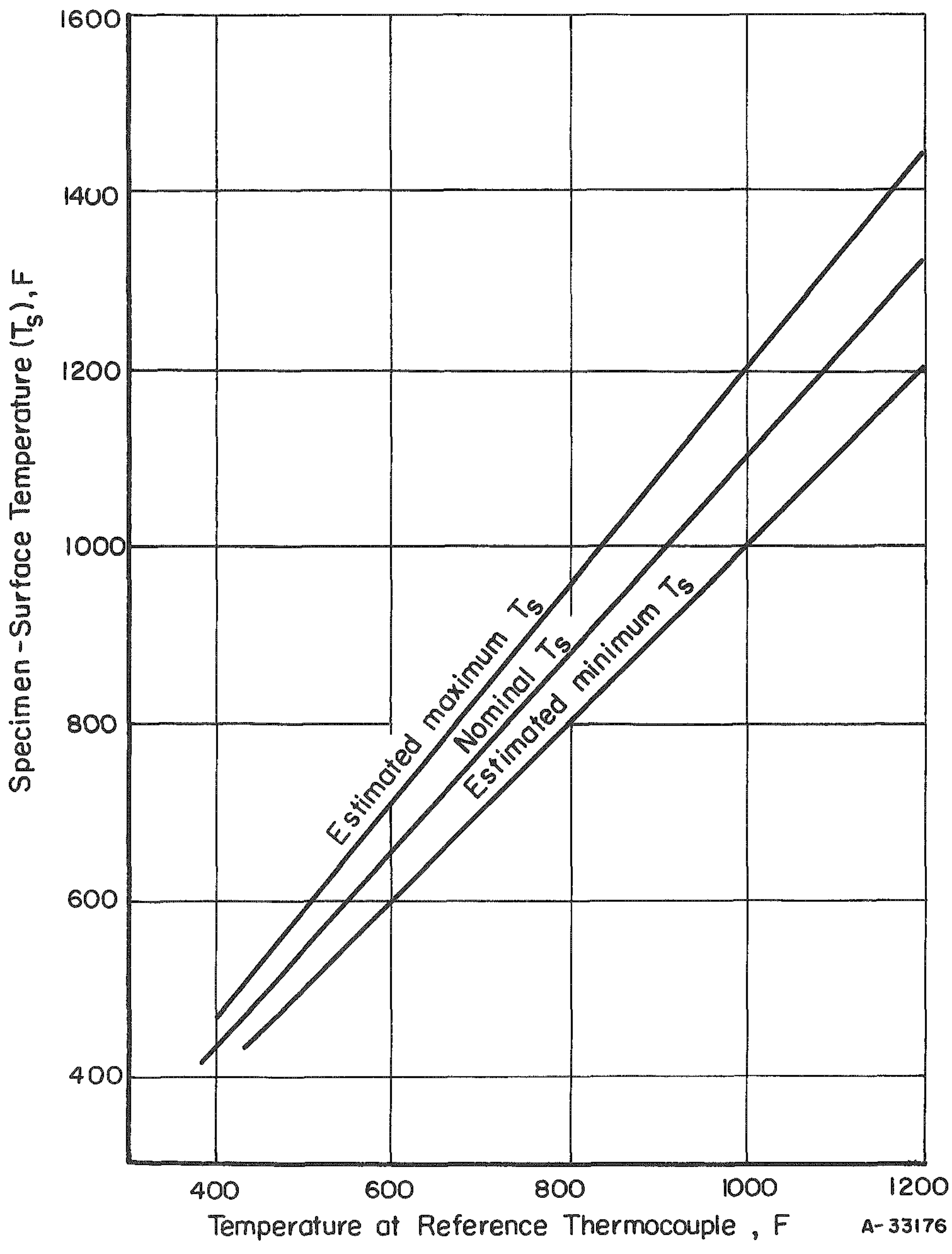

FIGURE 8. SPECIMEN-SURFACE TEMPERATURE VERSUS REFERENCE THERMOCOUPLE READING WITH ESTIMATED UNCERTAINTY BAND

Solving this equation for k_{eff} and making certain substitutions yields

$$k_{eff} = \frac{0.50}{\left(\frac{T_1 - T_2}{T_2 - 120}\right) - 0.12} .$$

The substitutions are:

(1) For Q_f, the linear relation (from Figure 4), 7.27 (T_2 - 120) where 120 F is an assumed reactor water temperature

(2) For $Q_f \left[\frac{\ln r_2/r_s}{2\pi k_{NaK}}\right]$, the temperature drop $T_s - T_2$

(3) For $T_s - T_2$, the linear relation (from Figure 4), 0.12 (T_2 - 120).

Table 16 gives estimated nominal effective thermal conductivities of the top specimens in Capsules BMI-23-1 through BMI-23-6 for each irradiation cycle. These values were calculated by substituting into the above equation for k_{eff} the averaged in-pile temperature data presented in Tables 4, 6, 8, 10, and 11. It may be noted from the estimated conductivities listed in Table 16 that variations during an irradiation do not appear to be significant. The conductivity at the start of an irradiation did not differ appreciably from that during the irradiation. As indicated previously, the fact that the calculated in-pile values are consistently lower than the measured out-of-pile value may stem from the fact that the calculations are based on Method I heat generation rates, which are suspected of being somewhat low.

To complete this analysis, the uncertainties associated with the nominal values were appraised by using the statistical approach previously outlined. The net uncertainty stems from the estimated uncertainty in fission-heat generation rate (Figure 5) and the resulting uncertainty in specimen-surface temperature (Figure 8).

Over the range of interest Figure 9 shows (a) the nominal relation between k_{eff} and the parameter involving thermocouple readings T_1 and T_2 and (b) the estimated uncertainty band. It might be noted that an uncertainty in the neighborhood of ±25 per cent exists here, even presuming that the input nominal Q_f data are entirely correct.

Specimen Center-Line Temperature

To round out the study of irradiation data, new center-line temperature-reference temperature curves were calculated on the basis of conductivity values in the range indicated in Table 16. Figure 10 shows the new nominal relations for the two limiting calculated values of k_{eff}. Included in the plots are error bands estimated from the uncertainties associated with fission-heat generation, reference thermocouple-to-specimen surface ΔT, and specimen conductivity (Figures 5, 8, and 9, respectively).

These curves can be used to estimate uncertainty limits for the nominal center-line temperatures listed in Tables 4, 6, 8, 10, and 11. As a whole, the error in T_c over the range of interest appears to be ±100 to ±200 F.

TABLE 16. EFFECTIVE IN-PILE THERMAL CONDUCTIVITIES OF IRRADIATED UC SPECIMENS CALCULATED FROM THERMAL DATA

Values Pertain to Top Specimens in Capsules

MTR Cycle	Carbon, w/o	Nominal k_{eff}, Btu/(hr)(ft)(F)[a]
	Capsule BMI-23-1	
107	5.0	6.2
	Capsule BMI-23-2	
110	5.0	7.4
111	5.0	8.4
112	5.0	8.7
113	5.0	8.7
114	5.0	8.0
115	5.0	8.2
	Capsule BMI-23-3	
116	5.0	6.9
117	5.0	6.5
118	5.0	8.0
119	5.0	8.1
120	5.0	7.6
121	5.0	8.1
122	5.0	8.2
123	5.0	8.4
124	5.0	7.8
125[b]	5.0	--
126	5.0	--
127	5.0	--
	Capsule BMI-23-4	
115	5.0	6.9
116	5.0	6.9
117	5.0	6.6
118	5.0	6.5
119	5.0	6.5
120	5.0	6.5
121	5.0	6.5
122	5.0	6.8
123	5.0	7.0
124	5.0	7.1
125	5.0	7.2
126	5.0	7.3
127	5.0	7.0

TABLE 16. (Continued)

MTR Cycle	Carbon, w/o	Nominal k_{eff}, Btu/(hr)(ft)(F)[a]
	Capsule BMI-23-5	
122	4.6	8.7
123	4.6	10.2
124	4.6	8.9
125	4.6	8.5
126	4.6	8.5
127	4.6	9.7
	Capsule BMI-23-6	
126[c]	4.8	--
127	4.8	--

(a) The measured out-of-pile conductivity in the 1000 to 1500 F range is approximately 14 Btu/(hr)(ft)(F).

(b) The thermocouple adjacent to the bottom specimen failed.

(c) The thermocouple located in the hollow of the top specimen failed at startup.

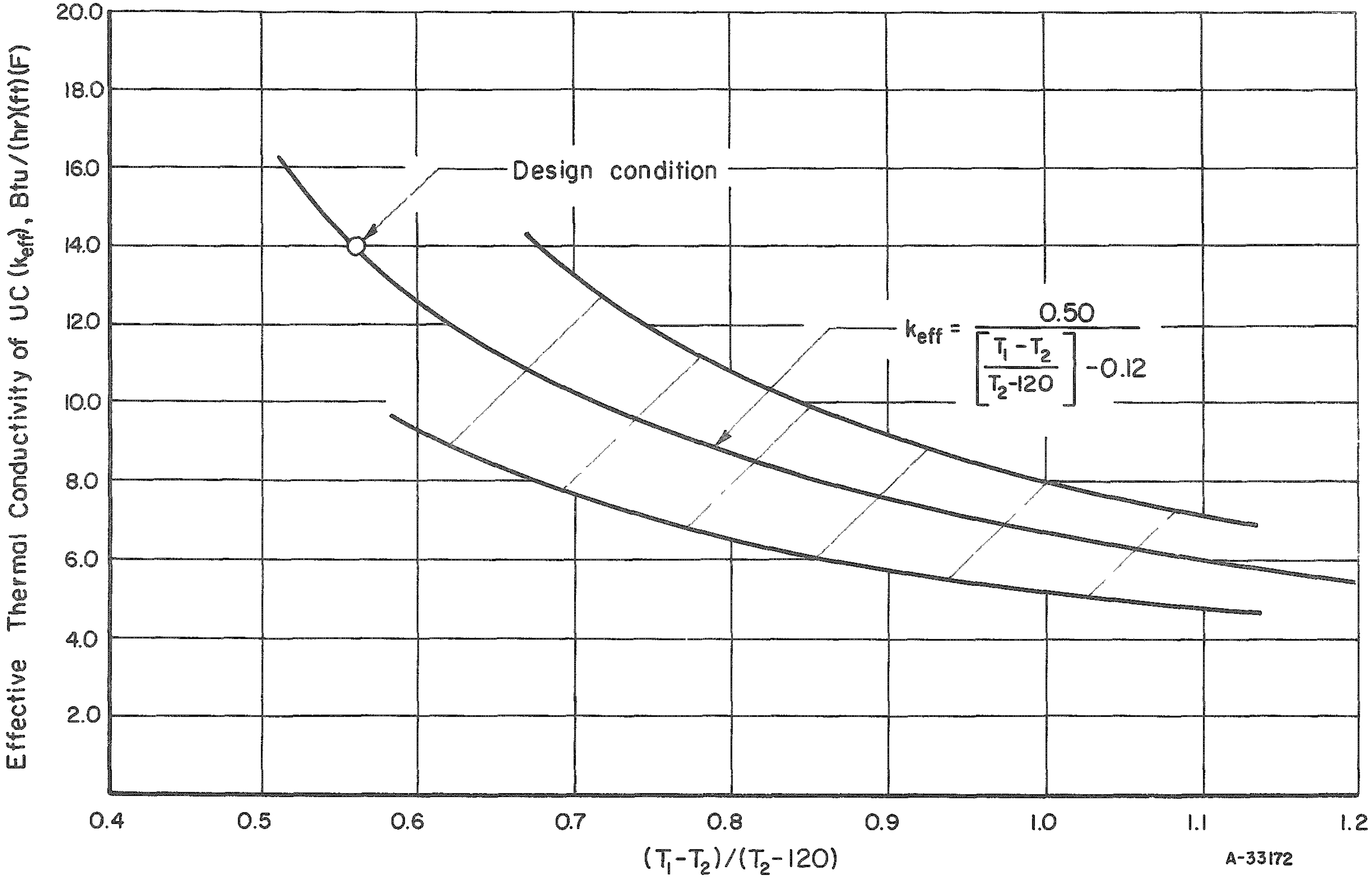

FIGURE 9. EFFECTIVE THERMAL CONDUCTIVITY OF UC VERSUS THE $(T_1-T_2)/(T_2-120)$ RATIO WITH UNCERTAINTY BAND

Because T_1 is a parameter, the plot is strictly applicable only to top specimens in the capsule.

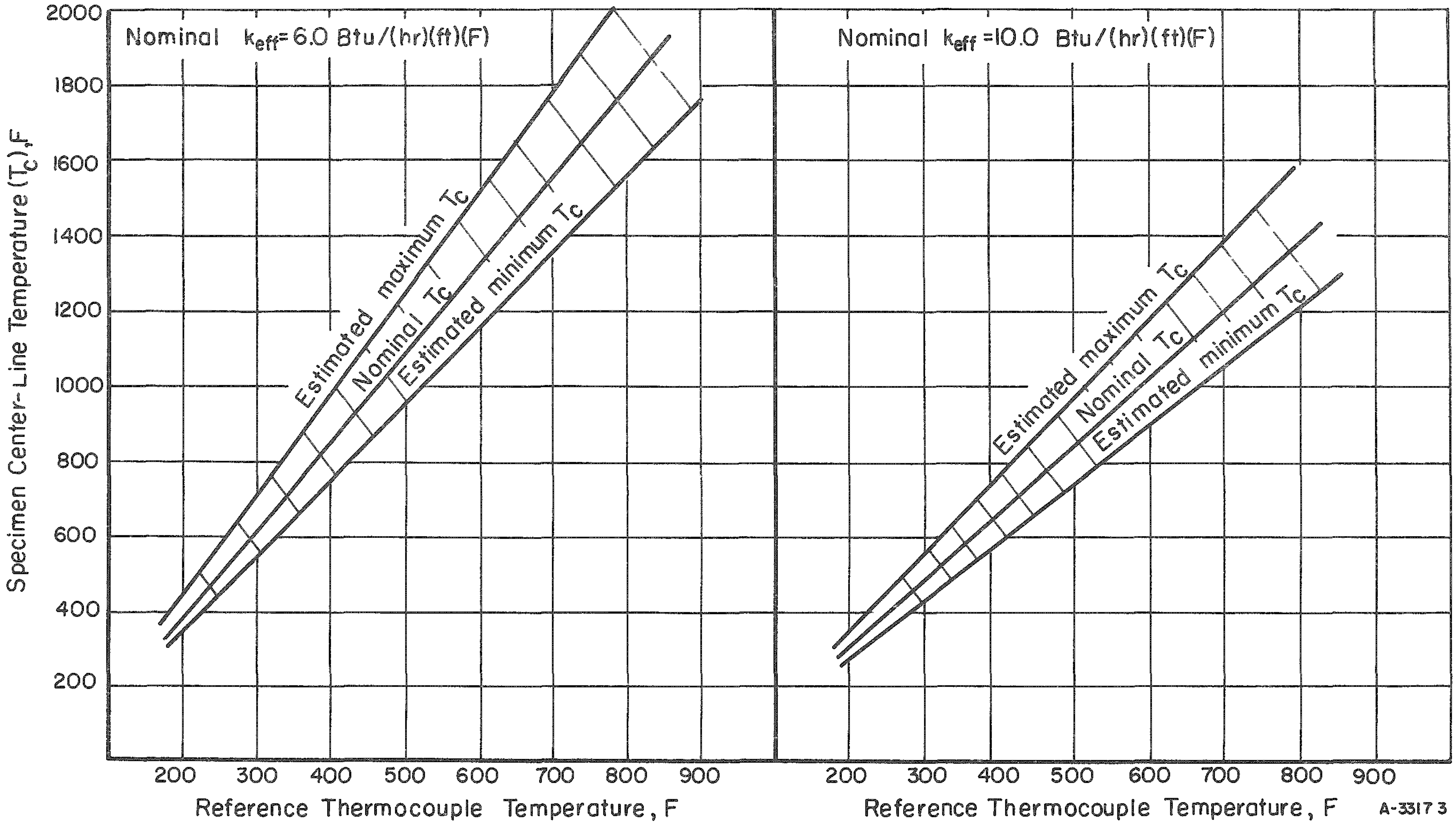

FIGURE 10. SPECIMEN CENTER-LINE TEMPERATURE VERSUS TEMPERATURE AT REFERENCE-THERMOCOUPLE WITH k_{eff} OF 6.0 AND 10.0 BTU/(HR)(FT)(F)

CONCLUSIONS

Capsules were designed and constructed for the irradiation at the MTR of small cylindrical specimens of enriched UC. The capsule design conditions included a specimen-surface temperature of 1100 F and a specimen center-line temperature of 1500 F. Temperature data obtained from thermocouples incorporated in the capsule systems indicated that of the six capsules involved in the program, five operated at temperatures close to the design conditions. The temperatures in the sixth capsule, BMI-23-4, were somewhat lower than anticipated, presumably because the irradiation was conducted where the neutron flux was lower than the design value.

In order to obtain an evaluation of the radiation stability of UC, a wide range of fissionable-material burnups (1,000 to 20,000 MWD/T of uranium) was specified for these irradiations. To provide this range, the various capsules were irradiated for appreciably different periods of time. While a fairly detailed evaluation of uncertainties involved in predicting burnup levels has been used to demonstrate the potential for sizable errors, there is good evidence that the target range of burnup desired was achieved.

The irradiations provided information that could be employed in the prediction of effective in-pile specimen thermal conductivity. Findings along this line indicated that the conductivity did not vary appreciably during the various irradiation exposures.

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