THE NAA 47
U - 10 Mo FUEL
IRRADIATION PROGRAM
FOR
HNPF CORE I

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ABSTRACT

Atomics International conducted a fuel irradiation program to investigate and define the behavior of U - 10 Mo fuel at the temperatures and fission rates anticipated in Hallam Nuclear Power Facility (HNPF) Core I. Each of the eight irradiation tests contained nine 0.5-in. diameter, 2.0-in. long, cast U - 10 Mo fuel specimens, NaK bonded in a double-walled stainless steel capsule. Initially, the fuel specimens were either as cast, partially transformed ($\alpha + \gamma' + \gamma$), or homogenized (γ).

The experiments were irradiated in the Materials Testing Reactor, for periods up to 3 years, to achieve substantial burnup levels at the relatively low Core I burnup rates, and then were disassembled and examined.

Only the low-burnup specimens in Experiments NAA 47-1 and NAA 47-7, and the top specimens in certain other capsules, were relatively stable. Most of the higher-burnup fuel specimens swelled excessively at temperatures from 800 to 1300°F and fission rates from 0.3 to 1.4×10^{13} fissions/cc-sec. Imprints of thermocouple and spacer wires on the surfaces of the fuel indicated low in-pile strength and radiation-induced plasticity. Some axial surface cracking was observed. The NaK bond had extensively permeated the highly swelled fuel.

Post-irradiation metallographic studies were made to investigate the phase stability of U - 10 Mo irradiated at fission rates $< 1 \times 10^{13}$ fissions/cc-sec and temperatures $< 1100^\circ\text{F}$. At higher temperatures (900 to 1000°F), the transformation products were found to be in a spheroidal form. At intermediate temperatures ($\sim 850^\circ\text{F}$), the specimens had a lamellar transformed ($\alpha + \gamma'$) microstructure. At lower irradiation temperatures, the specimens displayed a γ -phase structure. Many of the specimens, irradiated at temperatures $> 875^\circ\text{F}$, showed intergranular cracking.

The results from these irradiations, at fission rates $< 1 \times 10^{13}$ fission/cc-sec and temperatures $< 1100^\circ\text{F}$, extended the findings of previous investigations, conducted at fission rates from 2 to 7×10^{13} fissions/cc-sec. Fuel instability is attributed primarily to the effects of continuous in-pile phase transformation, which occurs at fission rates insufficient to retain the metastable γ phase at temperatures $< 1100^\circ\text{F}$. This instability appears to be most severe at intermediate temperature levels, from 800 to 1000°F.

Extrapolation of test capsule data, to predict power reactor fuel element behavior, involves considerable uncertainty. The variation in irradiation conditions and the large number of severe thermal cycles, associated with as much as 3 years of irradiation, may have had a significant influence on the observed instability. As a result of the NAA 47 studies, the peak allowable burnup for HNPF Core I was substantially reduced from earlier estimates, to ~4000 Mwd/MTU, to insure that fuel rod integrity, as limited by internal fuel rod pressure, would be maintained throughout the core life.

I. INTRODUCTION

Under contract with the United States Atomic Energy Commission, Atomics International has designed and built the Hallam Nuclear Power Facility (HNPF), for operation by the Consumers Public Power District.¹ The reactor, located at the Sheldon Station, 21 miles from Lincoln, Nebraska, is a sodium-cooled, graphite-moderated system, designed to operate at 240 Mwt and 75 Mwe. For several years prior to and during construction of the plant, AI conducted a development program to establish and substantiate many design aspects of the nuclear plant. The HNPF Core I Uranium Alloy Fuel Evaluation Project was an integral part of this development program.

Originally, UO_2 was specified as the reference fuel for Hallam, but consideration of the ceramic fuel was dropped in November 1958. In its place, U - 10 wt % Mo was selected for Core I, as it was felt that this fuel would require the minimum development time for sodium reactor use, due to the considerable amount of irradiation data which were available. At the time, the data, which were generated primarily by Atomic Power Development Associates and Battelle Memorial Institute, in conjunction with the selection and development of fuel for Enrico Fermi Core I, indicated that U - 10 Mo possessed suitable irradiation stability.^{2,3,4} Consequently, the alloy was considered to be an excellent fuel for an initial, interim HNPF core, prior to the planned insertion of UC.

After selection of U - 10 Mo as the Core I fuel, the HNPF Uranium Alloy Evaluation Project was initiated, to provide specific information at the proposed design conditions. Irradiation experiments, designated as the NAA 47 series, were planned as material proof tests, to define the behavior of the alloy at the anticipated fission rates, temperatures, and burnup levels of HNPF. The fission rate and temperature ranges of Core I are shown in Figure 1, while the original peak design conditions for the NAA 47 experiments are presented in Table 1. Although the experiments did not simulate actual fuel element design, the individual fuel specimens, which were 0.5-in. diameter cast slugs, approximated the 0.590-in. diameter Hallam slugs. The design specifications of the subject Core I element are given in Table 2.

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Figure 1. Range of Temperature and Fission Rate Conditions for HNPf Core I U - 10 Mo Fuel

TABLE 1

NAA 47 ORIGINAL PEAK DESIGN CONDITIONS

Experiment	Fuel Temperature (°F)		Fuel ΔT (°F)	Fission Rate (f/cc-sec) ($\times 10^{-13}$)	Power (kw/ft)
	Surface	Center			
47-1, -2, -9	1073	1300	227	1.39	14.8
47-3, -4, -10	1152	1300	148	1.04	9.9
47-5, -6, -11	1193	1300	107	0.69	7.4
47-7, -8	1230	1300	70	0.46	5.0

NOTE: NAA 47-9, -10, and -11 were replacement capsules for NAA 47-1, -3, and -5 (see Section II-D-2)

TABLE 2

HALLAM NUCLEAR POWER FACILITY - CORE I DESIGN
SPECIFICATIONS

Nominal Full Power Rating	240 Mwt
Net Electrical Power	75 Mwe
Moderator	Graphite, Type 304 Stainless Steel Clad
Coolant	Sodium
Inlet Temperature	610°F
Outlet Temperature	945°F
Fuel Elements	151 eighteen-rod clusters
Cladding	Type 304 Stainless Steel
Length	180 in.
Inside Diameter	0.640 in.
Thickness	0.010 in.
Bond	Sodium
Thickness	0.025 in.
Fuel	3.6% enriched as-cast U - 10 Mo
Length	159 in. (3-1/2 in. /slug)
Diameter	0.590 in.
Enrichment	3.6%
Cover Gas	Helium (~ 1 atm pressure)

II. PROCEDURES AND RESULTS

A. EXPERIMENT DESIGN⁵

The NAA 47 irradiation experiments were considered as fuel material tests; consequently, the capsules did not simulate actual fuel rod design. As the schematic shown in Figure 2 indicates, each experiment contained nine 2.0-in. long U - 10 wt % Mo specimens, which were NaK bonded in a double-walled Type 304 stainless steel capsule. Since the inner 0.030 to 0.050-in. wall tube had an ID of 0.750 in., the specimens were centered in the tube by enclosing them in a perforated, 0.010-in. thick Type 304 stainless steel basket, which was axially slit to permit unrestricted diametral fuel growth. The basket was welded to concentric end hardware, and it contained 1/16-in. thermocouple and spacer wires which centered the fuel within the basket. Small-diameter wires held the slit basket together at three points along its length. A tubular extension, attached to lower hardware, was provided with a puncture diaphragm for post-irradiation sampling of fission gases contained in the inner tube.

The entire inner tube assembly was contained within an outer 0.050-in. thick tube filled with helium. The radial gap between the inner and outer tube varied with the thickness of the 0.030 to 0.050 in. inner tube wall. The four different radial gaps which were chosen permitted the capsules to operate at four design power generation or fission rates, while retaining the same design peak central temperature. The initial eight experiments in the series had spacer rings machined in the inner tube, which centered the inner assembly within the outer tube. However, these rings produced excessively high, localized stresses which contributed to the failure of three of the initial eight capsules;⁶ consequently, the rings were reduced to point contacts in the three replacement capsules: NAA 47-9, 47-10, and 47-11.

Each capsule was instrumented with eight, 1/16-in. diameter, stainless steel-sheathed, Chromel-Alumel thermocouples, six located in the NaK annulus between the fuel and the perforated basket, and two located on the axis of the fuel column. The locations of the six annulus thermocouples are shown in Figure 2. One of the in-fuel thermocouples ran down the side of the fuel column and extended 2.0 in. up through the bottom cored specimen, and the other central thermocouple extended downward through the top five cored specimens to a

Thermocouple	Distance from Base of Fuel Column (in.)
1	15.5
2	12.5
3	9.5
4 (in fuel)	2.0
5	2.0
6	9.5
7 (in fuel)	9.5
8	6.5

Figure 2. Schematic of NAA 47 Capsule Design

point 9.5 in. above the base of the fuel column. The sections of these thermocouples which extended into the fuel were wrapped with tantalum foil to prevent U-Fe eutectic reactions.

B. MATERIAL AND FABRICATION HISTORY

The 0.5-in. diameter fuel cylinders were fabricated by vacuum melting and casting in graphite molds. Specimens were obtained from three separate casting heats: Experiments NAA 47-1 through NAA 47-8 contained 6.06% enriched material from Heats 530 and 531; while NAA 47-9, -10 and -11 contained 6.42% enriched material from Heat 892. After casting, the fuel was machined to the proper 2.0-in. length, and a 0.093-in. diameter thermocouple hole was drilled through the required number of the specimens. Two groups of specimens were then subjected to vacuum heat treatment: one group was homogenized, or "gammatized," by holding the fuel at 1650°F for 24 hr and water quenching to room temperature; the second group was partially transformed by heating at 930°F for 100 hr and air cooling to room temperature. A third group of specimens was retained in the as-cast condition.

All structural components in the capsules, fabricated from Type 304 stainless steel, were cleaned, assembled, and welded; and the completed assemblies were subsequently helium leak checked. After fuel loading, the inner capsule was filled with NaK to a level approximately 1.0 in. above the top of the fuel column, leaving about 2 in. of He-filled expansion space at room temperature. Although NaK loading was performed in glove boxes under a helium atmosphere, no chemistry samples were taken; consequently, the impurity content of the NaK was not determined. After loading, the NaK fill tube was sealed, and the entire test assembly was heated to 1100°F to wet all internal components.

C. PRE-IRRADIATION TESTS AND ANALYSES

Over a period of 18 months, both before and during assembly and irradiation of the experiments, numerous measurements, tests, and analyses were performed to determine the dimensions, properties, and characteristics of the fuel and other capsule components. The procedures and results associated with these studies are described in the following sections.

1. Measurement and Determination of the Dimensions and Properties of the Fuel

Prior to capsule assembly, the dimensions, density, and electrical resistivity of each fuel specimen were obtained. Dimensional data consisted of several micrometer measurements of the length and diameter of each specimen. Two perpendicular diameter measurements were taken at each of three locations on the specimens. Densities were calculated after determining the specimen weight in air and the specimen weight immersed in octanol.

The electrical resistivity of each specimen was determined, at both room temperature (68°F) and at the boiling point of nitrogen (-321°F).^{7,8} Since the resistivity measurements were made over a period of time on different apparatus, the actual values were not too accurate. However, the results did provide an indication of the degree of transformation. As expected, the resistivity values of the as-cast specimens and the gammatized specimens were similar, while the resistivities of the partially transformed specimens were lower. The resistivities of the as-cast and gammatized specimens decreased with temperature, while the resistivities of the partially transformed fuel increased with temperature.

2. Chemical Analysis of the Fuel

Two analytical techniques were used to determine the distribution of molybdenum in the fuel alloy. The results of x-ray fluorescence, which are presented in Tables 3 and 4, indicated a range of individual values from 8.4 to 11.5 wt % Mo. Averages of the individual measurements of five 3/16-in. diameter areas on the surface of each specimen ranged from 9.8 to 10.5 wt % Mo.

In an effort to obtain more detailed information, six specimens were sent to Advanced Metals Research Corporation for an electron beam microprobe analysis.⁹ A probe, 3 μ in diameter, was utilized to make two perpendicular diametral scans across the face of each specimen. The individual analyses in each scan were taken at intervals of approximately 0.010-in.; and, at each point, the ratio of the intensities of the fluorescent lines MoK_{α} to UL_{α} was computed. Since no standard homogeneous U - 10 Mo alloys were available, the average net MoK_{α} intensity for all points from each specimen was taken to represent 10% Mo. The following ranges of Mo content were reported for the two specimens representing each heat treatment:

TABLE 3
MOLYBDENUM ANALYSIS X-RAY FLUORESCENCE

Specimen	wt % Mo					
	Position*					
	1	2	3	4	5	Average
<u>Heat 530</u>						
OA1	9.5	10.4	8.6	10.2	8.4	9.8
OT2	9.4	10.0	10.0	10.6	10.4	10.1
OG2	11.1	10.6	10.1	10.1	10.7	10.5
OT1	10.1	10.1	10.3	9.9	10.0	10.1
OT3	9.9	10.0	9.7	10.3	10.1	10.0
OG3	10.5	9.9	9.5	9.7	10.2	10.0
<u>Heat 531</u>						
IA2	9.9	11.5	9.9	8.8	10.2	10.1
IT3	10.4	10.4	9.4	9.5	9.1	9.8
IG2	10.8	10.9	9.7	9.3	11.0	10.3
IA1	10.4	10.0	10.1	10.5	10.6	10.3
IA3	10.0	10.1	9.9	10.1	9.9	10.0
IG1	10.2	10.2	10.4	10.1	10.3	10.2

*Position of 3/16-in. diameter spot on the U - 10 Mo specimen

a) As-cast specimens: 7.0 to 11.7 and 7.7 to 12.4%

b) Gammatized specimens: 8.5 to 11.6 and 8.2 to 11.1%

c) Transformed specimens: 9.0 to 10.8 and 8.4 to 12.0%

The mean probable error of an individual determination was estimated as $\pm 0.2\%$ Mo.

Conductometric carbon analyses were also performed on selected specimens. The results of these analyses, shown in Table 4, indicated that the material in Heats 530 and 531 contained 180 to 280 ppm carbon, while material from Heat 892 contained 600 to 840 ppm carbon.

TABLE 4
MOLYBDENUM AND CARBON ANALYSES
(NAA 47-1 through NAA 47-8 — 6.06% Enriched)

Specimen	Heat Number	Heat Treatment	wt % Mo*	wt % C†
OA1(T)§	530	AC**	9.8	--
OT2(M)	530	PT	10.1	--
OG2(M)	530	G	10.5	--
OT1(T)	530	PT	10.1	--
OT3(B)	530	PT	10.0	--
OG3(B)	530	G	10.0	--
OA2	530	AC	--	0.024
OA3	530	AC	--	0.022
OG1	530	G	--	0.018
IA2(M)	531	AC	10.1	--
IT3(B)	531	PT	9.8	--
IG2(M)	531	G	10.3	--
IA1(T)	531	AC	10.3	--
IA3(B)	531	AC	10.0	--
IG1(T)	531	G	10.2	--
IG3	531	G	--	0.019
IT1	531	PT	--	0.024
IT2	531	PT	--	0.028
AT	892	--	10.0	0.079, 0.076
AB	892	--	9.8	0.081, 0.081
BT	892	--	10.0	0.084, 0.082
BB	892	--	9.8	0.082, 0.084
CT	892	--	9.8	0.060, 0.063
CB	892	--	9.8	0.066, 0.065

*X-ray fluorescence analysis

†Conductometric

§T, M, B: top, middle, bottom of 2-in. long, 0.5-in. diameter specimen

**AC: As-Cast

PT: Partially Transformed — 100 hr at 500°C, air cooled

G: Gammatized — 24 hr at 900°C, water quenched

3. Pre-irradiation Metallography¹⁰

Specimens for metallographic analysis were selected from the same casting heats used in the NAA 47 capsules. An etchant, composed of four parts acetic acid and one part chromic acid, was used to delineate the structure. Figure 3 presents photomicrographs representative of the three structures: as-cast, gammatized, and partially transformed.

Examination of the photomicrographs indicated that the cooling rate of the as-cast material was sufficient to maintain the metastable γ phase at room temperature; however, the resultant structure showed some evidence of coring. The grain size ranged from 5 to 6 on the Timken ASTM scale. The gammatized structure appeared as a homogeneous single phase with a grain size of 0 on the Timken ASTM scale. The transformed structure, with a Timken ASTM grain size of 2 to 3, contained approximately 30% of the two-phase material concentrated primarily at the grain boundaries and adjacent to the precipitates. These precipitates, which were observed in all the material, were believed to be carbides.

Diamond pyramid hardness readings, taken at 1/16 to 1/8-in. intervals across the face of the specimens, indicated an average DPM of 280 to 290 for the as-cast and gammatized specimens, and 340 to 350 for the partially transformed specimens (Table 5).

4. U - 10 wt % Mo Oxidation Tests¹¹

As a part of the safeguards studies, oxidation tests were performed in air at 600 and 900°F, to determine the oxidation rates of as-cast and gamma-tized fuel material. U - 10 wt % Mo wafers, machined from cast specimens and electrolytically cleaned, were weighed, dimensioned, and heated in air for periods from 0.5 to 24.0 hr. At 600°F, a reasonably protective, dark, shiny film was formed, and the maximum rate of weight gain, $0.125 \text{ mg/cm}^2\text{-hr}$, observed during the first 0.5 hr, decreased steadily to a value of $0.015 \text{ mg/cm}^2\text{-hr}$ after 24 hr. At 900°F, the multilayered oxide film, which formed, fissured and spalled quite readily; and the weight gain was essentially linear with time, at a rate of about $1.8 \text{ mg/cm}^2\text{-hr}$.

a. As-Cast Structure
(Dark areas indicate
coring)

b. Gammatized Structure

c. Partially Transformed
Lamellar Structure

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Figure 3. Typical Pre-irradiation Structure of U - 10 wt % Mo Specimens

TABLE 5
DIAMOND PYRAMID HARDNESS TRAVERSES

Specimen	DPH									
	POSITION*									
	1	2	3	4	5	6	7	8	9	Average
OA1	278	233	282	309	309	309	292	278	278	286
OG2	292	278	288	278	304	304	297	297	292	292
OT2	278	376	355	383	355	321	348	309	315	338
IA2	267	282	226	254	282	297	315	288	282	277
IT2	376	288	348	385	348	321	340	394	321	347

* Traverse across the 0.5-in. diameter specimen; readings taken every 1/16 to 1/8 in.

5. Component Compatibility Studies¹²

Several tests were performed at representative NAA 47 fuel surface temperatures, to investigate the compatibility relationships between U - 10 wt % Mo and stainless steel in a Na or NaK environment. In these tests, wafers of cast U - 10 wt % Mo were Na or NaK bonded in four Type 304 stainless steel capsules. Two NaK-bonded capsules and two Na-bonded capsules were placed in a furnace at 1000°F, and one capsule of each type was removed after 30 days and after 90 days. Visual inspection and metallographic examination revealed no sign of interalloying, the only changes being the expected transformation of the U - 10 wt % Mo and the sensitization of the stainless steel. No significant weight or dimensional changes were noted.

D. IRRADIATION

1. Data Accumulation and Reduction Procedures

The in-pile capsule temperatures, which were continuously monitored and recorded, formed the basis for periodic estimates and final calculations of the irradiation histories of the experiments. As mentioned in Section II-A, each capsule was initially instrumented with a total of eight thermocouples. Although a number of these thermocouples failed at various times over the long irradiation periods, each capsule had at least two operational thermocouples when exposure was terminated. During irradiation, the peak fuel central

temperature for each capsule was determined on a weekly basis, and these temperatures were averaged to provide an estimate of the time-averaged fuel central temperature for each capsule. Based on the design burnup-temperature relationships, the accumulated in-pile time, and the assumed MTR power factor, weekly estimates of the time-averaged temperatures and the accumulated burnups were made for each capsule. These estimates provided the bases for decisions on removal dates for the experiments.

In addition to the weekly estimates, an extensive data reduction system was devised to provide final, comprehensive irradiation histories for the experiments. In the initial phase of this system, the recorder strip charts for each experiment were completely scanned, and temperature data points were recorded, periodically or whenever reactor power level or recorded temperatures varied significantly. The data served as input for Part I of the AIDAR Modified Data Reduction Code,¹³ which automatically calculated fuel surface and central temperatures for each thermocouple and presented these temperatures as a function of time. It also provided axial temperature profiles for any period specifically requested, and these profiles were studied to eliminate obviously faulty thermocouples. New inputs were subsequently prepared for Part II of the code, which then provided the following information:

- a) A listing of the time-averaged central and surface temperatures for each thermocouple for each reactor cycle.
- b) The peak fuel central temperature at each thermocouple location during each cycle.
- c) A listing of the burnup (Mwd/MTU) at each thermocouple location for each cycle, calculated from a comparison of actual and design conditions.

The final summary irradiation histories for Experiments NAA 47-2, -4, -6, and -7, which were prepared from this information, are described in References 14, 15, 16, and 17.

2. Irradiation History

The initial eight NAA 47 experiments were inserted in the MTR during July and August of 1959, and all eight remained in the reactor until November 1959, when three experiments, NAA 47-1, -3, and -5, were removed due to

confirmed or suspected failures. NAA 47-1 and -3 were removed when activity was detected in the MTR process water after these capsules had experienced temperature excursions, and NAA 47-5 was removed due to excessively high-temperature operation and the loss of five of its thermocouples. A subsequent post-irradiation examination of the failed capsules, described in Reference 6, revealed that:

- a) The outer tube of NAA 47-1, which had experienced fuel temperatures of 1700°F for approximately 5 hr, contained a longitudinal crack, about 1.0 in. long, at a point about 5.0 in. from the bottom of the fuel column. Although water had entered the He annulus, the inner capsule was intact.
- b) Both the inner and outer tubes of NAA 47-3 contained an axial split, 6.0 in. long, which had permitted water to enter the inner capsule, where it had reacted with the fuel and the NaK.
- c) Experiment NAA 47-5 was intact, but the fuel at the bottom of the column had reacted with the stainless steel spacer and thermocouple wires.

Evaluation indicated that failure in the outer tubes of Experiments NAA 47-1 and -3 was due to stress corrosion caused by a combination of the following factors: (a) improper heat treatment, which caused sensitization of the material; (b) excessively high, localized stresses adjacent to certain spacer rings; and (c) an environment favorable to stress corrosion. The secondary failure in NAA 47-3 was due to high-temperature corrosion of the inner tube, which allowed a slow, progressive reaction between the NaK and water. Since the observed failures had been gradual in nature, and since operating thermocouples were present to give adequate indication of any trouble, the decision was made to continue irradiation of the remaining five experiments.

Only one of the three discharged experiments, NAA 47-1, yielded any significant fuel data; consequently, three additional experiments, NAA 47-9, -10, and -11, were fabricated as replacement capsules for NAA 47-1, -3, and -5, respectively. These experiments were identical to the original capsules, with the exception that the circumferential centering rings were reduced to several point contacts, to lessen thermal stresses. In addition, the 1100°F proof test was deleted from the fabrication procedure, to prevent sensitization

TABLE 6
SUMMARY OF NAA 47 IRRADIATION CONDITIONS

Capsule	Peak Conditions			Date of		Time In pile (months)	Thermal Cycles*	Comments
	Time Average Central Temperature (°F)	Burnup (Mwd/MTU)	Fission Rate (fission/cc-sec) ($\times 10^{-13}$)	Insertion	Discharge			
NAA-47-1	1400	1800	1.4	8-10-59	11-9-59	3	54	Outer capsule leaked and water vapor in He annulus caused central temperature of fuel at bottom of column to rise as high as 1700°F for a short period.
NAA-47-2†	940	14,000	0.97	7-24-59	2-19-62	31	282	Peak recorded central temperature was 1190°F. Peak central temperature averaged ~1000°F for 2-1/2 months, dropped to ~850°F for 5-1/2 months, rose to 1130°F over the next 3 months, subsequently decreased to ~900°F.
NAA-47-4†	1060	12,000	0.73	7-24-59	6-25-62	35	312	Peak central temperature averaged ~1100°F for initial cycles, dropped to 880°F after 7 months, rose to 1380°F after 18 months, then averaged 1000 to 1175°F for remaining 17 months.
NAA-47-6†	1000	6600	0.50	7-24-59	11-6-61	27-1/2	262	Probably most consistent, most closely monitored experiment in series. No peak temperatures > 1205°F were recorded; experiment did experience extended temperature fluctuations (Figure 4).
NAA-47-7	1020	4600	0.35	8-15-59	12-18-61	28	268	Experienced temperature > 1400°F after insertion, lost several T/C's, was pulled up in X-basket, remained at elevated position with recorded temperatures < 1120°F until after MTR activity checks in November 1962, returned to lower position. At lower position, temperatures returned to initial high levels.
NAA-47-8	1320	7000	0.50	7-24-59	5-14-59	38-1/2	302	Ran at highest average temperatures in series. Peak burnup ~6000 to 7500 Mwd/MTU. Only 2 T/C's remained at time of discharge.
NAA-47-9	1290	12,000	1.38	5-30-60	2-19-62	20-1/2	169	T/C readings fluctuated. Comprehensive data reduction not possible; definitely experienced central temperatures as high as 1570°F for 6 hr prior to discharge, due to a flux increase when an adjacent loop test was removed.
NAA-47-10	1230	9900	0.90	5-13-60	8-26-62	27-1/2	210	Last experiment discharged. Not examined because of similarity of exposure history to -8 and -9.

*Thermal cycle: Whenever the MTR power dropped to levels below 5 Mw; full power is 40 Mw

†Irradiation data completely reduced by AIDAR Code.

of the stainless steel. The three replacement capsules were inserted in the MTR in May 1960.

All five original capsules and the three replacement capsules remained in pile until October 12, 1960, when Experiment NAA 47-11 was prematurely discharged by MTR personnel. This modified replacement capsule was removed after tripping the alarm circuit when fuel central temperatures rose to 1485°F. No fission product activity was detected, and observation of the capsule in the

MTR storage canal disclosed no exterior abnormalities. Although the temperature history of another capsule in the same "A" piece disclosed no similar high temperatures, examination of NAA 47-11 temperature data showed that all temperatures in the capsule had risen proportionately, and that the thermal resistance of the helium-filled annulus had not changed, strongly indicating that the capsule was intact and that the temperature rise had been due to a power increase. Due to its short irradiation exposure, the NAA 47-11 fuel had received a peak burnup of less than 1000 Mwd/MTU; consequently, the capsule was not examined.

The actual exposures and in-pile residence times of the seven remaining NAA 47 experiments were considerably in excess of those anticipated in the original design schedule. The original plan, derived from the initial Core I peak burnup goal of 7000 Mwd/MTU, specified a comparable peak burnup for the experiments, and a maximum residence time of approximately one year. However, when the target Core I peak burnup was subsequently increased to 15,000 Mwd/MTU, the scheduled irradiation periods were extended, in an effort to provide data at the higher burnup levels. As a consequence, the actual irradiation periods of the seven capsules ranged from 20 to 37 months, with resulting peak burnups which varied from 4600 to 14,000 Mwd/MTU. Data summarizing the irradiation histories of the experiments are presented in Table 6, while detailed data for the individual specimens are summarized in Tables 7, 8, and 9.

The tables and figures mentioned previously disclose some important characteristics of the irradiation histories:

a) The capsules experienced several hundred thermal cycles, due to MTR power fluctuations and shutdowns.

b) The experiments experienced a wide range of fuel temperatures and fission rates.

c) The fuel temperatures for individual experiments varied considerably during the long periods in pile. In many cases, temperatures, which were initially relatively high, dropped steadily to lower levels, and subsequently returned to the higher levels. An example of these extended variations, which were observed in the irradiation histories of NAA-47-2, -4, -6, and -7, is shown in Figure 4, which gives the MTR average cycle temperatures for NAA-47-6.

TABLE 7
DATA FROM EXPERIMENT NAA 47-1

Specimen Number	Burnup (Mwd/MTU)	Central Temperature (°F)	Fission Rate ($\frac{\text{fission}}{\text{cc-sec}}$) ($\times 10^{-13}$)	% Δ V	% Δ Dim.*
1-1	1150	970	0.90	-0.47	-0.16
1-2	1250	1120	0.97	0.06	0.02
1-3	1430	1230	1.11	0.48	0.16
1-4	1550	1290	1.21	0.48	0.16
1-5	1650	1350	1.28	0.42	0.14
1-6	1750	1400-1750	1.36	1.12	0.37
1-7	1800	1400-1750	1.4	1.54	0.51
1-8	1800	1380-1750	1.4	1.30	0.43
1-9	1750	1320-1750	1.36	1.77	0.59

*% Dimensional change assumed to be $\frac{\% \Delta V}{3}$

E. POST-IRRADIATION EXAMINATION

After removal from the MTR, the experiments were held for varying lengths of time in the MTR storage canal, and subsequently shipped to AI's Component Development Hot Cells at Santa Susanna, California, where the capsules were unloaded from the casks and moved into the cells for post-irradiation examination. Examinations of 6 of the experiments were initiated at intervals over a period of approximately 9 months, from April 1962 to January 1963. These examinations are described in detail in References 18, 19, 20, 21, 22, and 23. The examination of the seventh experiment, NAA 47-1, which was performed in 1959, is described in Reference 6. One experiment, NAA 47-10, was not examined, because of the similarity of its irradiation history to those of NAA 47-8 and -9. After being loaded into the cell, each experiment was visually examined, but no abnormalities or gross defects were noted in the outer capsules.

1. Gamma Scanning

The first operation performed on the irradiated experiments was to take a series of four gamma activity scans, at 90° rotations over the length of the

TABLE 8

DATA FROM EXPERIMENTS NAA 47-2, 47-4, 47-6, AND 47-9

Specimen	Burnup (Mwd/MTU)					Average Central Temperature (°F)	Fission Rate (fissions (cc-sec) (x 10 ⁻¹³)	%ΔD	%ΔL	$\frac{2 (\% \Delta D) + \% \Delta L}{3}$	Density Decrease (%)
	AIDAR	Radiochemical			Mass Spectrometer						
		Cs	Sr	Ce							
NAA 47-2-1	11,000	8,440	7,850	10,000	8,630	750	0.74	5.4	--	--	8.3
2-2	12,000	--	--	--	--	820	0.82	17.4	--	--	--
2-7	14,000	--	--	--	--	940	0.97	33.4	--	--	--
2-8	13,600	--	--	--	--	920	0.94	40.9	12.7	31.5	--
2-9	13,100	6,660	2,620	14,300	11,400	890	0.91	23.8	10.0	19.2	45.5
NAA 47-4-1	9,000	--	--	--	--	820	0.55	14.6	13.3	14.2	--
4-2	10,100	--	--	--	--	900	0.62	29.8	12.6	24.1	--
4-3	11,000	--	--	--	--	970	0.68	28.2	17.3	24.6	--
4-4	11,300	--	--	--	--	1020	0.71	26.1	--	--	--
4-5	11,800	5,770	4,260	10,400	10,600	1040	0.73	24.5	1.6	16.9	--
4-6	11,900	--	--	--	--	1050	0.73	26.3	--	--	--
4-7	11,700	--	--	--	--	1030	0.72	25.7	--	--	--
4-8	11,400	--	--	--	--	1000	0.69	21.9	0.3	14.7	--
4-9	10,900	--	--	--	--	960	0.66	32.1	19.4	27.9	--
NAA 47-6-1	4,000	--	--	--	--	670	0.34	0.5	0.4	0.5	1.7
6-2	4,900	--	--	--	--	770	0.39	5.4	4.4	5.1	10.7
6-3	5,600	4,750	3,850	4,930	4,960	860	0.44	8.1	9.2	8.5	21.1
6-4	6,100	5,540	4,460	6,510	5,640	930	0.48	9.7	10.1	9.8	24.4
6-5	6,500	--	--	--	--	980	0.50	10.6	9.9	10.4	--
6-6	6,600	--	--	--	--	1000	0.51	10.9	9.6	10.5	25.4
6-7	6,300	4,300	4,170	5,900	5,640	980	0.50	9.7	8.5	9.6	23.7
6-8	5,600	--	--	--	--	880	0.43	8.7	7.6	8.3	21.3
6-9	4,700	4,800	4,412	5,700	4,790	730	0.36	8.5	1.6	6.2	22.8
NAA 47-9-1	8,000*	--	--	--	--	Est. as 1100 to 1300	Est. as 9.0 1.1 to 1.4	9.7	4.5	8.0	17.8
9-2	10,000	--	--	--	--			6.0	4.9	5.6	13.6
9-3	11,500	11,000	10,200	8,260	11,500			9.0	4.2	7.4	16.5
9-4	12,200	7,680	10,200	14,700	12,200			16.6	5.1	12.8	27.6
9-5	12,400	--	--	--	--			17.9	4.5	13.4	27.8
9-6	12,200	--	--	--	--			19.9	3.9	14.6	30.9
9-7	11,800	6,340	9,540	11,900	11,800			16.8	3.6	12.4	28.2
9-8	11,500	--	--	--	--			12.7	2.8	9.4	21.9
9-9	11,000	--	--	--	--			11.7	5.4	9.6	24.6

*Burnups for NAA 47-9 specimens estimated from mass spectrometric data.

TABLE 9
DATA FROM EXPERIMENTS NAA 47-7 AND 47-8

Specimen	Estimated Burnup, Average Central Temperature, and Fission Rate Ranges	% Δ D	% Δ L	$\frac{2(\% \Delta D) + \% \Delta L}{3}$
NAA 47-7-1	47-7-1 through 47-7-4	0.0	0.0	0.0
7-2	2500-3800 Mwd/MTU	0.0	0.0	0.0
7-3	600-900°F	0.3	0.5	0.4
7-4	0.2-0.3x10 ¹³ fissions/cc-sec	0.9	0.9	0.9
7-5	47-7-5 through 47-7-9	4.0	3.6	3.6
7-6	4000-5000 Mwd/MTU	10.8	5.9	9.2
7-7	900-1100°F	16.4	7.9	13.6
7-8	0.3-0.4x10 ¹³ fissions/cc-sec	24.8	3.2	17.6
7-9		15.5	5.5	12.2
<hr style="border-top: 1px dashed black;"/>				
NAA 47-8-1		9.5	9.6	9.5
8-2	4000-7500 Mwd/MTU	11.9	10.2	11.3
8-3		12.1	5.6	9.9
8-4	900-1320°F	14.6	8.8	12.7
8-5		17.7	6.7	14.0
8-6	0.3-0.5x10 ¹³ fissions/cc-sec	19.7	6.8	15.4
8-7		16.3	4.4	12.3
8-8		9.1	5.4	7.9
8-9		15.7	3.2	11.5

fuel column. This provided a profile of the activity, or relative burnup, for each capsule, and also provided an estimate of the post-irradiation fuel column length. The ratios of the peak to the average activity for the experiments ranged from 1.1 to 1.3, for all of the experiments except NAA-47-7. The fuel column of this experiment, which was scanned both before and after removal of the NaK and inner cladding, exhibited peak-to-average ranges of 1.25 to 1.55 and 1.75 to 2.21.

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Figure 4. NAA 47-6 Average Fuel Central Temperatures for Each MTR Cycle at Four Thermocouple Locations

In general, the activity profiles for the fuel columns correlated with the profiles calculated from the temperature data. The indicated fuel column lengths ranged from about 18-3/4 to 20-1/4 in. The 20 in. length, noted in Experiments NAA-47-2, -4, -6, and -8, represented the maximum observed increase of 2 in. In the NAA-47-6 scan, pronounced dips in activity indicated separations between several of the fuel specimens.

On several scans, activity appreciably above background was noted in the region of the NaK reservoirs, above and below the fuel column, possibly indicating the presence of dissolved fission products in the NaK.

2. Fission Gas Sampling and Analysis

After the outer containment tube was removed, the inner assembly of each capsule was placed in the fission gas sampling apparatus, which was subsequently evacuated to equilibrium pressures, in the order of 10 to 20 μ . The fission gas diaphragm at the bottom of the inner tube was then punctured, releasing the fission gas, which was subsequently collected on activated charcoal in two liquid-nitrogen-cooled traps, connected in series. When evacuation was completed, the traps were sealed and sent to analytical chemistry for fission gas analysis, based on Kr-85. As Table 10 indicates, the measured fission gas released, as a percentage of the gas theoretically produced, varied considerably for the six capsules; the highest, 33% for Experiment NAA-47-4, and the lowest, <1% for Experiments NAA-47-7 and -8.

TABLE 10
FISSION GAS RELEASE DATA

Experiment	Assumed Average Burnup (Mwd/MTU)	Fission Gas Release (%)
NAA 47-2	13,000	6.6
NAA 47-4	12,000	33.0
NAA 47-6	5,500	5.7
NAA 47-7	3,800	0.5
NAA 47-8	6,600	0.2
NAA 47-9	10,000	11.0

3. Disassembly of the Inner Capsule and Removal of the Fuel

While numerous difficulties were encountered during disassembly operations, due to the highly swelled and damaged condition of the fuel, the most serious problems were encountered during disassembly of the initial experiment, NAA-47-2. The procedure for disassembly of this experiment specified that the ends of the inner capsule were to be removed, exposing the perforated support basket and contained fuel, which were to have been easily drawn from the inner tube. However, after removing the ends of the inner capsule, the support basket could not be moved, as the fuel had swelled diametrically to force the basket and the thermocouple and spacer wires against the inner cladding. When another circumferential cut was made through the cladding in the bottom third of the fuel column, the fuel column and cladding broke into two sections at the cut, which was near the top end of Specimen NAA 47-2-7, exposing the fuel cross sections shown in Figure 5. The bottom two fuel specimens and the major portion of Specimen 2-7 were finally removed by axially sectioning the cladding.

Further difficulties were encountered during storage of the NAA 47-2 fuel when a small in-cell fire destroyed the three specimens from the center of the fuel column and partially damaged the top three slugs. Apparently, the NaK, which had thoroughly permeated the swelled specimens, reacted with the moisture in the cell atmosphere.

Although NAA 47-4 was also disassembled by sectioning and slitting the cladding, disassembly of the other four experiments was relatively simple, as the fuel and cladding were easily separated. Since almost all of the fuel from these experiments had absorbed appreciable quantities of NaK, definite precautions were taken to handle and store all fuel under an inert atmosphere or under vacuum.

4. Examination and Photographs of the Fuel

The fuel specimens from all six capsules were extensively examined and photographed. The condition of the fuel showed various stages of damage, and the more important characteristics of fuel from each experiment are summarized in the following sections.

Figure 5. Cross Section of Capsule Hardware and Fuel Near Top of Specimen NAA 47-2-7
(Note slight deformation of 0.050-in. tubing wall in regions where thermocouple and spacer wires were compressed)

435B

Figure 6. Variation in Fuel Swelling Observed in Experiment NAA 47-2 (Specimen NAA 47-2-1 had experienced in-cell fire, prior to this photo)

501B

a. NAA 47-2

Three specimens from the center of the experiment were destroyed by an in-cell NaK fire. The top three specimens were also damaged and severely cracked. The top specimen, 2-1, was relatively stable, in comparison to the bottom three specimens, 2-7, 2-8 and 2-9, which swelled diametrically to fill the available annulus (Figures 5 and 6). There were deep indentations in the fuel surface from thermocouples, spacer wires, and the support basket. Swelling was restricted at the base of Specimen 2-9. The bottom two specimens were bound together.

b. NAA 47-4

The top specimen was tapered, cracked, and slightly warped. The specimens nearly filled the available annulus. There was an inconsistent variation in diametral swelling, over the lengths of individual slugs and over the length of the fuel column. The swelling at the base of Specimen 4-9 was restricted (Figure 7 and Figure 12). There were impressions in fuel surfaces from the capsule hardware. There were also severe damage zones near the surfaces of some specimens (Figure 8). Several specimens were bonded together, and were forcefully separated by breaking at or near slug junctions.

c. NAA 47-6

The surface condition of the fuel was excellent (Figure 9), having a typical as-cast surface. Swelling at the base of Specimen 6-9 was restricted. There was slight edge chipping on the upper specimens. Specimen 6-3 was slightly warped.

d. NAA 47-7

There was a continuous increase in the severity of damage and swelling from the top of the fuel column to the center of Specimen 7-8, in agreement with the gamma activity scan (Figures 10 and 14). The top four specimens were in excellent condition, with virtually no damage. The bottom four specimens bonded together. There were axial surface cracks.

e. NAA 47-8

There was an inconsistent variation in diametral swelling. The top specimens were warped. There was severe axial surface cracking, especially in the bottom specimens; this cracking was most severe in the gammatized specimens.

Figure 7. Specimen NAA 47-4-9, Illustrating Restricted Growth Area at Bottom of Specimen (Note inconsistent variation in diametral swelling over length of specimen)

718B

693B

Figure 8. Severe Damage Zones Near Surface of Specimen NAA 47-4-3 (Bottom)

601B, 602B

Figure 9. NAA 47-6 Fuel Specimens

827B, 828B, 829B, 835B, 836B

Figure 10. NAA 47-7 Fuel Specimens (Note increase in damage from top to bottom of fuel column)

519B

a. NAA-47-9-5, Bonded to Adjacent Specimens

584B

583B

1) Top

2) Bottom

b. End Surfaces of NAA-47-9-6, After Separation From Adjacent Fuel

Figure 11. Severe Surface Cracking and Bonding of NAA 47-9 Specimens

f. NAA 47-9

The surfaces of the top two specimens were in good condition. There were severe axial surface cracks in the bottom seven specimens, concentrated on one side of the fuel column. The gammatized specimens were the most severely cracked. The lower eight specimens were bonded into a continuous column, with the propagation of cracks extending across specimen interfaces. The specimens were separated by forcefully breaking them at these specimen interfaces (Figure 11).

5. Dimensional Measurements

Post-irradiation dimensional measurements, taken at the same locations as the pre-irradiation measurements, were performed to determine the percent dimensional increases for each specimen.

Two diameter measurements, at 90° rotations, were taken at each of three locations on all integral specimens, and the two measurements at each point were averaged. Measurements were taken at the center of each specimen and 0.25 in. in from each end. Although the precision of in-cell measurements was estimated as ± 0.001 in., the measured diameters on highly damaged specimens were somewhat uncertain, due to the cracked, grooved, or roughened fuel surface. Length measurements were taken on all specimens which could be cleanly separated at the slug junctions. The length measurements were somewhat inaccurate when the slugs were warped or when the end surfaces were not parallel. The percent diameter change at each point on the specimens and the percent length change for each specimen in Experiments NAA 47-4, -6, -7, -8, and -9 are illustrated in Figures 12 through 16. The average percent diameter change and the percent length change for each specimen in Experiments NAA 47-2, -4, -6, and -9 are given in Table 8.

6. Density Measurements

Density measurements were performed on certain fuel specimens from the first three experiments which were examined. Two specimens from NAA 47-2 were measured, while nine specimens from NAA 47-6 and nine from NAA 47-9 were checked. The percent density change during irradiation is presented in Table 8.

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Figure 12. NAA 47-4 Dimensional Changes During Irradiation

7508-51341

Figure 13. NAA 47-6 Dimensional Changes During Irradiation

7508-51342

Figure 14. NAA 47-7 Dimensional Changes During Irradiation

7508-51343

Figure 15. NAA 47-8 Dimensional Changes During Irradiation

7508-51344

Figure 16. NAA 47-9 Dimensional Changes During Irradiation

Prior to the density measurements, the fuel specimens were extensively cleaned. As a first step, the fuel was held at 300°F under vacuum for several hours, in an effort to remove the residual NaK which had permeated the fuel. However, even repeated distillation operations at higher temperatures failed to remove all of the NaK from the highly damaged fuel. After removal from the furnace, the specimens were ultrasonically cleaned in ethyl alcohol, and the cored slugs were further cleaned by brushing the internal surfaces of the thermocouple holes.

Post-irradiation density measurements were obtained in the following manner. Dry weights were obtained on individual specimens which were suspended from a wire sling inside of a balance. Four separate weights were obtained for each specimen, and then an average dry weight was computed. Next, the wet weights were determined by suspending the specimen in a Plexiglas container filled with octanol at a known temperature. Four wet weights were obtained: the initial weight was taken as soon as the balance reached equilibrium, and the next three determinations were made at subsequent 5-min intervals. An average of the four wet weights was recorded as the wet weight of the specimen, and the average wet weight and the average dry weight were then used to calculate the density of the specimen.

7. Burnup and Metallographic Specimens

After completion of the dimensional and density determinations, the fuel specimens were sectioned to obtain 1/4- to 1/2-in. thick samples for burnup analyses and metallography. Despite previous vacuum distillation operations, appreciable NaK was observed seeping from the exposed fuel surfaces. Although the material had a metallic lustre and appeared to be solid, it obviously contained extensive microscopic porosity. Consequently, the sample discs were given additional NaK distillation.

8. Burnup Analyses

Various methods were used to analyze and calculate fuel burnup for the different experiments (Table 8). The methods utilized were: (a) the previously mentioned AIDAR Code; (b) cesium, strontium, and cerium radiochemical analyses; and (c) mass spectrometric analysis of U^{235} -to- U^{236} ratios.

TABLE 11
POST-IRRADIATION METALLOGRAPHY

Specimen	Fission* Rate (fissions cc-sec (x 10 ⁻¹²))	Average Temperature* (°F)	Dia- meter Change (%)	Pre- irradiated Structure	Post- irradiated Structure	Inter- granular Cracking
NAA 47-2-1-S	7.9	625	1.0	AC	G	---
2-1-C	7.9	750	1.0	AC	G [§]	---
4-2-S	6.35	855	25.2	G	T	X
4-2-C	6.35	965	25.2	G	T&S	
4-5-S	7.1	960	23.1	G	T&S	X
4-5-C	7.1	1085	23.1	G	T&S	
4-9-S	5.9	805	35.3	PT	G	X
4-9-C	5.9	910	35.3	PT	T	
6-3-S	4.6	825	9.0	PT	T	---
6-3-C	4.6	905	9.0	PT	T	---
6-4-S	5.2	920	10.2	AC	T	X
6-4-C	5.2	990	10.2	AC	T	
6-7-S	5.3	890	10.6	AC	T&S	---
6-7-C	5.3	975	10.6	AC	T&S	---
6-9-S	4.2	790	12.3	PT	T	---
6-9-C	4.2	810	12.3	PT	T	---
7-3-C	2.5†	725†	0.8	PT	G	---
7-6A-C	3.3†	1025†	5.6	PT	T&S	---
7-6B-C	3.5†	1025†	14.8	PT	T&S	X

*The fission rate and average temperature values shown in the table are for the final stages of irradiation, since it was assumed that this period had the greatest effect on the observed microstructure. The fission rate at the center of the specimens is not corrected for absorption by the fuel.

†Average central temperature and fission rate for entire irradiation

§Some inhomogeneity noted after irradiation

S - Surface
C - Center
AC - As-Cast
X - Intergranular Cracking

PT - Partially Transformed
G - Gammatized
T - Transformed
T&S - Transformed and Spheroidized

All three types of analyses were utilized on specimens from NAA 47-2, -4, and -6. In these experiments, the Cs and Sr results were consistently lower than the temperature-based AIDAR values; however, the Ce and mass spectrometric results showed sufficient agreement to justify using the AIDAR levels for subsequent evaluation purposes.

NAA 47-7 burnup was analyzed by both AIDAR Code and radiochemical methods, but the results were not consistent, or were not in agreement with the indicated gamma scan activity. Consequently, the burnup for this experiment was given as a range, estimated from temperature and gamma scan data (Table 9).

Both radiochemical and mass spectrometric analyses were performed on NAA 47-9, and the burnup profile was estimated from the results (Table 8). No burnup determinations were performed for NAA 47-8; consequently, the burnup was estimated, using data from the weekly temperature reports.

9. Post-irradiation Metallography

Eleven NAA-47 specimens were metallographically examined, in an attempt to provide an explanation for the observed irradiation instability. The effects of phase transformation, and the nature and extent of microscopic damage, were studied. The results of the study are summarized in Table 11.

All of the specimens exhibited phase changes due to irradiation, and these changes varied with the temperature and fission rate conditions.

In general, the fuel irradiated below 750°F during the final stages of the irradiation period, exhibited a gammatized structure, as shown in Figure 17; however, one relatively higher fission rate specimen, NAA 47-4-9, exhibited a γ -phase structure near its surface at a temperature of about 805°F. At higher temperature, up to 900°F, the material transformed to a typical lamellar $\alpha + \gamma'$ structure, as shown in Figure 18; while at temperatures above 900°F, the transformed structure was spherodized, as shown in Figure 19.

During the examination, porosity was observed in specimens irradiated in the temperature range from 800 to 1085°F (Figure 19). This porosity consisted of intergranular cracks or cavities, which occurred predominantly at intermediate temperature zones on the fuel cross sections. There were two distinct types³¹ of cracks: elliptical at grain boundary junctions, and wedge-type along grain boundaries. In some specimens, the cracks were circumferentially oriented, as shown in Figure 20; while, in other specimens, the cracks extended in a radial direction. In all cases, the cavities were observed in specimens whose central temperatures exceeded the eutectoid temperature at some period during their irradiation.

10. Post-irradiation Heating Studies²⁵

In an effort to isolate and investigate the effects of temperature on the stability of the fuel, post-irradiation heating experiments were performed, using the specimens from NAA 47-6 and -9 which showed the least amount of swelling during irradiation. After being accurately dimensioned, the four

Figure 17. Gammatized
Microstructure at Edge of
Specimen
NAA 47-7-3

M1120

Figure 18. Transformed
Microstructure Near
Center of Specimen
NAA 47-6-3 (Note
lamellar structure)

M1032

Figure 19. Specimen NAA 47-7-6B, Illustrating Types of Void Structure Observed, and Their Relation to Grain Boundaries (Structure is transformed and spheroidized)

M1125

Figure 20. Circumferential Cracking in Specimen NAA 47-7-6B (This cracking occurred in an intermediate temperature zone)

M1112

specimens, NAA 47-6-1, -6-2, -9-1, and -9-2, were subjected to the various heat treatments described in Table 12. Following each annealing phase, the slugs were redimensioned and the percent volume increase was computed. The test results, presented in Table 12, revealed the following:

a) When Specimens 6-1 and 9-2 were heated in the transformation range at 900°F for about 90 hr, Specimen 6-1 showed an insignificant volume decrease, while Specimen 9-2 swelled about 3%. Specimen 9-2 expanded an additional 12% during a subsequent 3-hr anneal at 1500°F.

b) When Specimens 6-2 and 9-1 were heated for 3-hr periods at six temperature levels from 1110 to 1500°F, Specimen 6-2 showed no significant volume change, while Specimen 9-1 expanded about 4% at 1110°F but showed no further increase until the specimen expanded an additional 10% at 1500°F.

TABLE 12

POST-IRRADIATION ANNEALING DATA FOR NAA 47-6 AND NAA 47-9

Heat Treatment	Volume (in. ³)				Change in Volume From As Received (%)			
	47-6-2	47-9-1	47-6-1	47-9-2	47-6-2	47-9-1	47-6-1	47-9-2
As Received	0.4654	0.4955	0.4121	0.4685	--	--	--	--
2 hr 56 min at 1110°F	0.4789	0.5145	--	--	--	+ 3.8	--	--
2 hr 44 min at 1165°F	0.4753	0.5204	--	--	--	--	--	--
3 hr at 1211°F	0.4755	0.5102	--	--	--	--	--	--
3 hr at 1265°F	0.4783	0.5234	--	--	--	--	--	--
2 hr 30 min at 1355°F	0.4581	0.5122	--	--	--	--	--	--
3 hr at 1500°F	0.4598	0.5651	--	--	-1.2	+14.1	--	--
60 hr at 950°F	--	--	0.4105	0.4786	--	--	--	+ 2.2
91 hr at 950°F	--	--	0.4078	0.4848	--	--	-1.0	+ 3.5
2 hr 57 min at 1110°F	--	--	--	0.4839	--	--	--	+ 3.3
3 hr at 1500°F	--	--	--	0.5399	--	--	--	+15.2

III. DISCUSSION

A. IRRADIATION STABILITY AND IN-PILE PHASE TRANSFORMATION

The extreme irradiation instability of the U - 10 Mo at relatively moderate temperatures and low fission rates appears to be the direct result of continuous in-pile phase transformation. The U - Mo phase diagram (Figure 21) indicates that U - 10 Mo possesses a disordered, single-phase (γ) structure at temperatures above 1100°F, but transforms to a two-phase ($\alpha + \gamma'$) structure at temperatures below 1060°F. During irradiation, an additional mechanism, the fission spike, is believed to modify the equilibrium conditions by constantly disordering the ($\alpha + \gamma'$) phase, in opposition to the thermally-induced eutectoid decomposition reaction, $\gamma \rightarrow \alpha + \gamma'$. If the fission rate is sufficient, fission spike homogenization prevails, and the γ phase can be retained at lower temperatures; if fission rate is low, the disordered γ -phase material constantly reverts to ($\alpha + \gamma'$). This continuous transformation at low fission rates probably promotes fission gas bubble growth and/or reduces the in-pile strength of the fuel to such an extent that extreme irradiation instability results.

The gross instability of U - 10 Mo at low fission rates was conclusively demonstrated in an APDA experimental program which had been specifically designed to investigate the effects of temperature and fission rate on fuel stability.²⁶ This program was initiated soon after results from an earlier experiment, CP-5-1, had indicated that U - 10 Mo experienced instability at a fission rate of about 2.5×10^{13} fissions/cc-sec in a fuel temperature range around 985°F.²⁷ Since the observed swelling in CP-5-1 (~14% change in diameter at 11,000 Mwd/MTU) was believed due to dimensional instability of the ($\alpha + \gamma'$) phase at low fission rates, APDA initiated its recent program to investigate the effect of fission rate on transformation and stability of Enrico Fermi fuel pins. The results of this program, which were reported at Venice in the spring of 1962, substantiated the earlier CP-5-1 data, and led APDA to conclude that a fission rate $> 3.6 \times 10^{13}$ fissions/cc-sec is necessary to maintain the γ phase at irradiation temperatures of 850°F; while, at temperatures above 930°F, a level of 7×10^{13} fissions/cc-sec will produce or retain the γ phase.²⁶ In addition, APDA stated that diametral swelling of U - 10 Mo, irradiated at 975°F to about 12,000 Mwd/MTU, appeared to be inversely proportional to fission rate between 2.5 and 7.0×10^{13} fissions/cc-sec.

7508-51345

Figure 21. Uranium-Molybdenum Phase Diagram

B. IRRADIATION STABILITY AND TEMPERATURE

1. Average Operating Temperature

Since the stability of U - 10 Mo appears to be highly sensitive to in-pile phase transformation, temperature and temperature fluctuations must be considered as important variables influencing the stability of the fuel. Although it would be useful to isolate the effects of temperature and describe dimensional increase as a function of average fuel temperature, the wide variations in operating temperatures which the fuel experienced during irradiation reduce the significance of average temperature as an independent variable. In addition, it is difficult to isolate the effects of temperature and fission rate. However, in evaluating the data, some general trends were evident.

Below some temperatures, the fuel apparently behaves in a stable manner, retaining or transforming to the γ phase during irradiation. This was evident from the dimensional stability of fuel at the top of the fuel columns in the lower temperature experiments, such as NAA 47-2, -6, and -7, and from the metallographic studies of specimens from NAA 47-2 and -7. In general, these stable fuel specimens were irradiated at temperatures $< 800^{\circ}\text{F}$.

At fuel central temperatures from 800 to 1100°F and at fission rates $< 1 \times 10^{13}$ fissions/cc-sec, the fuel appears to be very unstable, and it transforms to a two-phase structure during irradiation. The data from NAA 47-2, -4, -6, and a portion of -7 showed severe fuel instability in this temperature range, and metallography on unstable specimens from these experiments disclosed the presence of the $(\alpha + \gamma')$ structure. At these temperatures, where the $\gamma \rightarrow \alpha + \gamma'$ reaction is most rapid, transformation and possible fuel instability would be expected, if the proposed relationship between irradiation conditions and fuel stability is correct.

At fuel central temperatures $> 1100^{\circ}\text{F}$, fuel stability may be slightly improved. Comparison of the data from NAA 47-2 and -4 with the data from NAA 47-9 shows that, at comparable burnups, the swelling of NAA 47-9 fuel was generally less than that of the two low-temperature experiments. Although this may be a valid observation, it is important to note two other considerations: (a) the fission rates of NAA 47-2 and -4 were slightly less than that of NAA 47-9, and (b) NAA 47-8 fuel, irradiated at very low fission rates, was very

unstable at an estimated average temperature in excess of 1300°F. Thus, fission rate, not temperature, may actually be the dominant variable in these experiments.

2. Temperature Variations

As mentioned previously, the NAA 47 experiments experienced wide variations in operating temperature during irradiation. These variations, which involved both abrupt and extended temperature fluctuations, may have had a pronounced effect on fuel stability.

a. Thermal Cycles

The NAA 47 fuel specimens were subjected to several hundred thermal cycles, due to MTR power fluctuations or scrams. These abrupt temperature changes, which were as great as 1000°F, probably have a deleterious effect on fuel stability. The rapid thermal stresses induced by the cycling may have promoted both the surface and internal cracking which contributed to the observed dimensional changes of the fuel.

b. Extended Temperature Variations

As the comments in Table 6 indicate, the NAA 47 experiments were also subjected to extended temperature fluctuations during these long in-pile exposures. The temperature history of NAA 47-6, shown in Figure 4, illustrates the variations which were also typical of NAA 47-2, -4, -7, and possibly 47-8; fuel temperatures, which were initially quite high, dropped steadily to low levels, and subsequently returned to the higher levels. These extended temperature changes, which resulted from reactor flux variations, undoubtedly produced gross phase changes which could have contributed to fuel instability.

3. Post-irradiation Annealing

The post-irradiation annealing study did provide some information on the specific effects of temperature in the absence of a neutron flux.²⁸ The test results indicated that both a stable specimen, NAA 47-6-1, and a somewhat unstable specimen, NAA 47-9-2, do not swell appreciably when held at optimum transformation temperatures out of pile. Since a metallographic examination was not performed on these specimens, it was not possible to determine whether transformation occurred during the heating period. However, NAA 47-6-1,

which had been irradiated at very low temperatures, should have retained the γ phase during irradiation, and should have partially transformed during subsequent exposure at 900°F.

Although post-irradiation heating for 3 hr at 1500°F produced significant, comparable volume increases in the NAA 47-9 specimens, NAA 47-6-2 showed a slight decrease in volume when exposed to this temperature, which greatly exceeded NAA 47-6 operating temperatures. Since the NAA 47-6 burnup was significant, swelling would be expected unless the factors which lead to in-pile swelling at lower temperatures are more effective than post-irradiation heating in promoting fuel volume changes.

C. IRRADIATION STABILITY AND FISSION RATE

As the previous discussion has indicated, fission rate is an extremely critical parameter influencing swelling. All fuel irradiated at fission rates $< 1 \times 10^{13}$ fissions/cc-sec and temperatures above $\sim 800^\circ\text{F}$ exhibited severe dimensional changes, and these changes were a factor of 2 to 3 greater than those noted in the previous APDA experiments. Even swelling of the high-temperature experiments appeared to be affected by fission rate. NAA 47-8 fuel, irradiated at 0.3 to 0.5×10^{13} fissions/cc-sec, swelled at a much faster rate than NAA 47-9 fuel, which was irradiated at a fission rate $> 1 \times 10^{13}$ fissions/cc-sec.

Although swelling at a fission rate of $< 1 \times 10^{13}$ fissions/cc-sec was much more severe than swelling above this fission rate, in the range from 0.2 to 1.0×10^{13} fissions/cc-sec, the swelling rate appeared to be fairly constant, regardless of fission rate. Fuel specimens from NAA 47-2, -4, and -6, which were irradiated at progressively lower fission rates, all exhibited about the same rate of dimensional increase as a function of burnup.

The metallographic studies, which were primarily oriented toward exploring the temperature-transformation relationship, did provide some marginal information regarding the fission rate-transformation relationships. The fuel from the surface of specimen NAA 47-6-3, irradiated at 825°F at a fission rate of 0.4×10^{13} fissions/cc-sec, was transformed after irradiation; while fuel from the surface of specimen NAA 47-4-9, irradiated at 805°F and 0.7×10^{13} fissions/cc-sec, did not transform, possibly indicating an increase in γ -phase stability at higher fission rates. In general, however, the metallography showed

Figure 22. Variation of Diameter Change with Burnup in NAA 47 Experiments

Figure 23. Variation of Dimensional Change with Burnup in NAA 47 Experiments

that fission rates $< 0.7 \times 10^{13}$ fissions/cc-sec were insufficient to retain the γ phase between 850 and 1050°F. At temperatures below 750°F, the γ phase was maintained or formed at fission rates $> 0.2 \times 10^{13}$ fissions/cc-sec. No metallography was performed on specimens with fission rates $> 0.8 \times 10^{13}$ fissions/cc-sec.

D. IRRADIATION STABILITY AND BURNUP

Although some irregularities were noted, in general, fuel swelling appeared to increase with burnup. This correlation was evident in the axial profiles of the diameter changes for Experiments NAA 47-7, -9, and particularly -6.

The relationship between burnup and dimensional change is illustrated in Figures 22 and 23. Figure 22 shows percent diameter change as a function of burnup; while Figure 23 gives average percent dimensional change, $[2(\% \Delta D) + \% \Delta L]/3$, as a function of burnup. Figure 23 attempts to correct for dimensional data from experiments which showed anisotropic dimensional increases. Fewer points are shown in Figure 23, because post-irradiation length measurements could not be obtained on some specimens. NAA 47-7 and -8 data are shown as a range, because of the uncertainty in the the burnup of specimens from these experiments.

The data points from NAA 47-6, -4, and -2, experiments with progressively higher burnups, show a general increase in diameter change with burnup (Figure 22). Since the NAA 47-6 experiment yielded the most consistent and complete irradiation and post-irradiation data, the points from this experiment were given primary emphasis in evaluating the dimensional change-burnup relationship (Figure 23). The curve in Figure 23 (which is reproduced on Figure 22) represents a line drawn through the NAA 47-6 data points and extended downward through the NAA 47-1 points. As Figure 23 indicates, the NAA 47-2 and -4 points cluster around an extension of this line; while the NAA 47-9 data fall below the line. The burnup for NAA 47-8, a high-temperature experiment, represents a very rough estimate, based on temperature data. Nevertheless, dimensional changes for the estimated burnup range are considerably above those of NAA 47-6, although they do approximate the changes observed in specimens from the bottom, high-temperature end of NAA 47-7. Points from the relatively cool top end of NAA 47-7 fall far below the estimated swelling curve.

As the previous discussion indicated, accurate determination of a dimensional change-burnup function was not possible, due to the recurring problem of isolating the interrelated effects of temperature, fission rate, and burnup. However, based on the general trends shown by specimens which experienced temperatures and fission rates in the HNPF Core I range, the following conclusions can be drawn:

- 1) At burnup levels above 3000 to 4000 Mwd/MTU, dimensional change increases rapidly with burnup, at a linear rate of about 2.5 to 3.0% dimensional increase per 1000 Mwd/MTU
- 2) Based on the limited data available, below 3000 to 4000 Mwd/MTU, swelling appears to be less severe. This second conclusion is drawn from both high- and low-temperature irradiation data — from the relatively high-temperature specimens from NAA 47-1, and from the lower-temperature specimens at the top end of NAA 47-6 and the top end of NAA 47-7.

E. CHARACTERISTICS OF THE IRRADIATION DAMAGE

1. Plasticity

The indentations in areas of the fuel which contacted thermocouple wires, spacer wires, the support basket, and other capsule components provided definite evidence of the in-pile plasticity which characterized the material.

2. Directionality of Swelling

In general, the relative length increases of severely damaged specimens were substantially less than the corresponding diameter increases, but this anisotropic behavior may also have been a result of fuel plasticity. After a certain amount of swelling, contact with the support basket and end hardware appeared to have restricted further axial movement of the fuel column, forcing the specimens to swell diametrically rather than axially. In some experiments, the axial force apparently produced a slight buckling of the relatively unsupported top specimens.

Specimens from NAA 47-6 and -7, which experienced a relatively small amount of swelling, expanded in the isotropic manner characteristic of cast material with randomly oriented, equiaxed grains.

3. Bonding at Specimen Interfaces

The axial forces due to restriction of swelling, in addition to the possible effects of temperature and neutron flux, apparently produced a reasonably strong bond at specimen interfaces. Examination of the end surfaces of some specimens also indicated that the presence of NaK may have contributed to the bonding between some specimens. The bonding occurred only in regions of the fuel columns which experienced severe damage and/or relatively high temperatures. Considerable force was necessary to separate the specimens at the junctions; and, in several instances, the break occurred within the specimen, rather than at the junction.

4. Axial Surface Cracking

Specimens which experienced extreme swelling, or specimens which experienced temperature excursions, exhibited irregular paraxial surface cracks or tears. In the case of highly swelled fuel, the tearing was probably due to extensive deformation and overstressing of the outer fibers. In experiments such as NAA 47-7 and -9, which experienced temperature excursions just prior to removal, rapid swelling and sudden thermal stresses undoubtedly accentuated the surface damage. As illustrated in Figure 11, the cracking appeared to be more severe in the specimens which were given a gamma heat treatment, prior to irradiation. The flux experienced by NAA 47-7 and -9 was undoubtedly skewed, since the cracking and damage was more severe on one side of the fuel column.

5. As-Cast Fuel Surface

Except for the axial cracking of the expanded fuel specimens, the fuel surface had a typical as-cast appearance. Although NaK had to be repeatedly cleaned from the fuel surface, no evidence of scaling, roughening, or inter-alloying was noted, indicating that fuel surface temperatures did not reach excessive levels. In addition, the results corroborated the findings of out-of-pile studies, which showed no compatibility problems in the U - 10 Mo-NaK-Type 304 stainless steel system at HNPF temperatures. The substitution of sodium for NaK as a bonding agent in the actual fuel rods should not alter this favorable compatibility relationship.

6. Effect of Pre-irradiation Heat Treatment

Pre-irradiation heat treatment appeared to have no appreciable effect on the irradiation stability of the fuel; this observation is in agreement with previous findings by APDA.²⁶ In general, the fuel swelled, regardless of heat treatment, the only difference being the more severe surface cracking of the gammatized specimens.

7. Internal Porosity

Metallographic examination of the fuel material revealed extensive microscopic porosity, which accounts for the propensity of the material to absorb considerable amounts of NaK. Much of the internal porosity consisted of intergranular cracks, which undoubtedly contributed to the observed volume changes of the fuel. The cracks appeared to be similar to cavities observed near fracture surfaces in high-temperature tensile specimens. Since the radial and circumferential cracks were mutually exclusive, different mechanisms may have led to their formation.

The inner capsule did not rupture, despite the fact that the increase in fuel volume in most experiments exceeded the available gas space, probably because of extensive NaK permeation of the porous fuel. While fuel swelling and fission gas release increase capsule pressure, NaK permeation of the fuel reduces this pressure. Undoubtedly, some fuel swelling and some increase in capsule pressure was necessary, before the NaK began to enter the fuel; however, it was not possible to determine the level of pressure and/or fuel damage necessary for NaK permeation.

8. Fission Gas Release

The results of the fission gas release measurements, shown in Table 10, appear somewhat contradictory. Data from four of the experiments showed an appreciable percentage of gas released; while results from two experiments, NAA 47-7 and -8, indicated a low level of release. Three of the four high-level results fell in a range from 5.7 to 11.0%; one result, 33% for NAA 47-4, appeared to be exceptionally high. In view of the extensive damage to part of the fuel in NAA 47-7 and all of the fuel in NAA 47-8, the measured percentage of gas released, which was in the range of that expected for recoil alone, appears to be anomalously low.

9. Swelling Irregularities

Although the diameter increase of the fuel specimens generally conformed to the indicated burnup profiles, several irregularities were observed.

a) The anomalous diametral bulges in specimens from NAA 47-4 and -8 are prominent exceptions, as indicated in the swelling profiles in Figures 12 and 15. One possible explanation is a variable temperature, which may have resulted from variation in the radial heat transfer, due to the presence of the annular spacer rings.

b) The upper end of the top specimen in all of the experiments except NAA 47-9 experienced appreciably less swelling than material farther down in the fuel column. Fuel at the lower end of the bottom specimen also showed considerably less diametral swelling. In the top specimen, this apparent stability was probably due to the cooling effects of the NaK. Although cooling due to NaK and end hardware probably contributed to the relatively small amount of swelling at the very bottom of the fuel column, the restrictive force of the attached support basket was undoubtedly an additional factor which reduced diametral expansion in this region.

F. HNPF CORE I FUEL

Since the primary objective of the NAA 47 series of experiments was to determine the irradiation behavior of U - 10 Mo under HNPF Core I conditions, one of the most important parts of the study was to evaluate the experimental results in view of Core I design limitations. As the U - 10 Mo elements had been fabricated, the main problem in the evaluation was to determine the anticipated life of the fuel rods, so that replacement elements would be available when the initial loading was removed. From a design standpoint, the life of the Core I rod was considered to be limited by the following factors:

1) Possible incompatibility between the fuel rod components at design temperatures. The results of the NAA 47 experiment corroborated the earlier data from out-of-pile tests and SRE Core I, which had shown that no incompatibilities existed. Thus, loss of fuel rod integrity due to this mechanism was not given further consideration.

2) Diametral growth of the fuel beyond the available annulus, which could eventually result in excessive cladding strains. The design limit or

failure, due to this mechanism, was conservatively assumed to occur when the fuel contacted the cladding. Based on this limitation, the nominal 0.025-in. annulus surrounding the 0.590-in. diameter fuel allowed for 8-2/3% diametral expansion.

3) Internal pressure buildup within the rod, due to fission gas release and fuel expansion. As fuel volume increases, internal pressure rises at an increasing rate. If fission gas release is assumed to be 6% and swelling is isotropic, this rate increases very rapidly, after an average of 6% volume (or 2% dimensional) change at a pressure of 250 psi. This pressure corresponds to a nominal cladding hoop stress of 8500 psi. Although this stress is appreciably less than the yield strength at operating temperatures, the 6% volume change was chosen as a conservative swelling limit.

In performing the evaluation and analysis of the Core I rod, the NAA 47-6 data were given primary consideration. Thus, fission gas release was assumed to be 6%, and fuel swelling was assumed to be isotropic and in conformance with the curve in Figure 23. This primary reliance on NAA 47-6 data was necessary for several reasons. The data from this experiment were comprehensive and complete, and the irradiation history adequately represented average conditions in Core I. In addition, the irradiation temperatures and the burnup did not reach extreme levels; and, consequently, fuel damage had not been greatly in excess of HNPF limitations.

Although data from the other experiments were considered to be useful as supplementary information, each of these experiments had shortcomings. In some experiments, such as NAA 47-1, -7, -8, and -9, temperatures had reached excessive levels; while, in other lower-temperature experiments, such as NAA 47-2 and -4, burnup had reached a point where fuel swelling was far in excess of Core I design limits. The peak fuel burnup conforming to the 2% dimensional change was calculated in the following way.

Using the curve in Figure 23 and an estimated axial burnup profile for the Hallam rod, and assuming various peak burnup levels, the diameter change at each 1-ft section of fuel was determined from the curve. For each assumed peak burnup, the diameter changes at each section were summed, and the average diameter change was computed. This method indicated that the limiting average dimensional increase of 2% would occur at a peak burnup of about 4000 Mwd/MTU.

Since low burnup data defining the Figure 23 curve were somewhat uncertain, a second calculation was made, assuming that NAA 47-6 dimensional change was proportional to burnup over the range from 0 to 6600 Mwd/MTU. This calculation yielded a limiting peak burnup of 2000 Mwd/MTU.

Since conservative assumptions were used in calculating the 4000 and 2000 Mwd/MTU limits, it was qualitatively concluded that the limiting peak burnup could be as high as 6000 Mwd/MTU, which is just beyond the burnup on the Figure 23 curve where the fuel would contact and possibly stress the cladding, assuming isotropic growth. At this peak burnup level, the swelling calculations indicated an average fuel column dimensional change of 4.8%. Although internal pressure and cladding stress would theoretically be excessive if fuel completely displaced the sodium, 4.8% dimensional change could be tolerated if sodium extensively permeated the expanded fuel. If sodium does not enter the fuel, the Hallam fuel will have to be appreciably more stable than NAA-47 data indicate, if Core I is to achieve a peak burnup of 6000 Mwd/MTU.

G. UNCERTAINTIES

Since the peak burnup level of 4000 Mwd/MTU was chosen as the most probable limit for the core, it is important to discuss some of the uncertainties and assumptions involved in the evaluation and analysis.

1) Although it would have been desirable to consider both temperature and fission rate as separate variables influencing fuel stability and fuel rod life, variations anticipated in Hallam, in addition to those observed in the NAA 47 experiments, make such an analysis impossible. Thus, all of the uncertainties involving the quantitative relationships between swelling and variables such as temperature, temperature fluctuations, and fission rate represented possible sources of error in predicting Core I behavior. Although fuel in HNPF Core I will not experience severe thermal cycles, it will be exposed to an appreciable

range in temperatures and large fluctuations in fission rates, due to the reactor power changes; consequently, a given temperature or fission rate cannot be associated with any specific region of the fuel rod.

2) The relative emphasis on NAA 47-6 results may also introduce some error. Since NAA 47-6 conditions approximated those of an average Core I rod at full power, the peak power rods will experience both higher temperatures and higher fission rates; and, consequently, the fuel in these rods may exhibit less instability than NAA 47-6 data indicate.

3) Although a fission gas release level of 6% was assumed in this analysis, the variability of the results from NAA 47 fission gas analyses make this figure somewhat uncertain.

4) It was difficult to analyze the probable behavior of the cladding and the maximum deformation which could be tolerated, prior to cladding failure. In determining the 2000 and 4000 Mwd/MTU limits, it was conservatively assumed that stresses would be held considerably below the yield stress. If the limiting stress was assumed to be the yield stress, about 15,000 psi at 1000°F, approximately 2.5% fuel dimensional increase, which corresponds to about 4500 Mwd/MTU, could be tolerated. If cladding deformation is assumed, slightly more dimensional change could be tolerated. Without cladding deformation, the internal pressure of the fuel rod increases very rapidly after about 2.5% dimensional change.

IV. CONCLUSIONS

The NAA 47 Core I Evaluation Experiments have shown that U - 10 wt % Mo is susceptible to severe damage, when irradiated at temperatures and fission rates characteristic of Hallam Core I. At fission rates $< 1 \times 10^{13}$ fissions/cc-sec, extreme fuel dimensional increases and corresponding density decreases can occur at fuel temperatures in excess of 800°F. In most instances, the fuel instability is characterized by a high level of fission gas release. The rate of fuel dimensional change may not be excessive at burnups less than 3000 to 4000 Mwd/MTU; however, at higher exposure levels, swelling increases rapidly with burnup, at a rate of 2.5 to 3.0% dimensional change per 1000 Mwd/MTU.

Although many factors may have contributed to the observed damage, it is attributed primarily to the effects of in-pile phase transformation at fission rates insufficient to retain the metastable γ phase at fuel temperatures between 800 and 1100°F. Under these temperature conditions at fission rates $< 1 \times 10^{13}$ fissions/cc-sec, the U - 10 Mo transforms to or retains the $(\alpha + \gamma')$ phase, regardless of pre-irradiation heat treatment. Although much of the observed swelling was probably due to the nucleation and growth of fission gas bubbles, a significant amount of the observed volume change resulted from internal micro-cracking of the fuel material. In addition to the effects of irradiation phase transformation, the severe temperature cycles and fluctuations undoubtedly contributed to fuel damage.

Uncertainties and assumptions are involved in predicting fuel rod behavior by extrapolation of test capsule results. The NAA 47 data show that Core I fuel may be limited to a design peak burnup of about 4000 Mwd/MTU, due to the build-up of pressure within the fuel rod. If the sodium bond permeates the expanding fuel and internal pressure is not limiting, the peak burnup may be reached when the fuel contacts the cladding, at an indicated peak burnup of 6000 Mwd/MTU. More favorable fuel behavior under the more stable power reactor conditions, and any ductile cladding behavior, will tend to increase fuel life; while rapid swelling at low-burnup levels could reduce the design peak to 2000 Mwd/MTU.

Despite some conservatism, the 4000 Mwd/MTU level should be accepted as a most probable design peak burnup.

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