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ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM

In-Pile Test of Prototype
ML-1 Fuel Elements

June 1964

AEROJET-GENERAL NUCLEONICS
SAN RAMON, CALIFORNIA

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ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM

IN-PILE TEST OF PROTOTYPE
ML-1 FUEL ELEMENTS

by

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ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM

IN-PILE TEST OF PROTOTYPE
ML-1 FUEL ELEMENTS*

ABSTRACT

Prototype fuel elements of the design used in the first core loading of the ML-1 (mobile nuclear power plant) reactor were tested in-pile under Contract AT(10-1)-880 between the U.S. Atomic Energy Commission and Aerojet.

The IB-8T-1 (instrumented) test element operated for 6415 hr in 99.5 wt% nitrogen with 0.5 wt% oxygen under ML-1 operating temperature conditions. Test data indicated that the fuel pin cladding temperatures were in excellent agreement with design calculations and that the element would operate at full power (55 kw) without exceeding the specified cladding hot spot temperature (1750°F). No change in pressure drop across the test element was observed during the test, indicating that the configuration of the assembly was not altered by the exposure. No evidence of cladding failure or other malfunction was observed.

Post-irradiation examination of the IB-8T-1 test element revealed that the fuel pins were deformed by spiral bending; analysis indicated that this condition was the result of constraint imposed by the spacer wires during thermal growth of the fuel pins. Metallographic evaluation revealed a massive second phase in the cladding of certain fuel pins. After comprehensive investigation, this formation was attributed to exposure of the Hastelloy X cladding to pure nitrogen; the second phase was identified as a variant of the pi-phase with a lattice parameter of 6.35 to 6.40 Å. Examination indicated that the BeO-UO₂ and UO₂ fuel in the test element had performed satisfactorily.

The IB-8T-2 (uninstrumented) test element operated for 6415 hr in oxygenated nitrogen plus 3736 hr in air for a total of 10,151 hr. Performance of the element during the test was completely satisfactory. Post-irradiation examination revealed excellent resistance of the Hastelloy X fuel pin cladding to oxidation by the gas, the maximum oxidation penetration observed was 0.0044 in. No evidence of the pi-phase was observed in the cladding; the structure and physical properties of the irradiated material closely resembled that of laboratory specimens exposed out-of-pile for similar periods at the same temperature. The IB-8T-2 test element, after removal from the test loop, appeared capable of operating for several thousand additional hours.

*Published by Aerojet-General Nucleonics, San Ramon, California

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ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM

IN-PILE TEST OF PROTOTYPE
ML-1 FUEL ELEMENTS

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I. INTRODUCTION

1.0 BACKGROUND

The goal of the Army Gas-Cooled Reactor Systems Program (AGCRSP) is the development of a mobile, nuclear power plant for military field use. The most significant single accomplishment under the program is the design, fabrication and test operation of a demonstration model, the ML-1. The ML-1 is a closed-cycle, gas-cooled nuclear power plant designed to produce 300 to 500 kw(e) from a nuclear reactor operating at about 3 Mw(t).

The core of the ML-1 reactor consists of 61 fuel elements arranged in a hexagonal lattice in the reactor calandria*. Each fuel element consists of 19 pins (12 in the outside ring, 6 in the next ring, and a single pin in the center) in a cylindrical tube. The 12 outer ring pins are fueled with $\text{UO}_2\text{-BeO}$ and the 6 inner ring pins are fueled with UO_2 ; the single center pin is unfueled. The calandria is immersed in the moderator water. The reactor coolant gas (the working fluid for the power conversion system) flows downward through passages provided in the fuel elements. (Additional information on the AGCRSP and the ML-1 is provided in Appendix A.)

The development of the fuel elements for the ML-1 to the highly restrictive specifications (Appendix B - ML-1 Military Characteristics) established for the power plant by the USAEC began in mid-1959. The development program for the first core fuel elements (ML-1-I) was supported by a number of in-pile tests. Four developmental models (the IB-1 α T, IB-1 β T, and IB-2T-1 and

*The calandria consists of 61 pressure tubes expanded and seal-welded into tube sheets (which are actually integral with the plenum chambers) on each end.

IB-2T-2) were tested to evaluate the heat transfer and fluid flow characteristics of the proposed design (Refs. 1-1 and 1-2). The rate of fission product leakage through failed fuel cladding was evaluated with the IB-7T-1, -2 and -3 test elements (Ref. 1-3). A full core loading of 71 IB-2L fuel elements was irradiated in the GCRE-I test reactor (Ref. 1-4).

The IB-8T tests, the subject of this report, were conducted to prove the final design of the ML-1-I fuel element. Two test elements were irradiated in an in-pile test loop; one was a prototype of the ML-1-I fuel with special instrumentation to sense fuel pin cladding temperatures and the second was a true prototype element.

Ref. 1-1: Army Gas-Cooled Reactor Systems Program, Interim Report In-Pile Tests of GCRE-IB Prototype Fuel Elements, IDO-28552, December 1960, Aerojet-General Nucleonics, San Ramon, California

Ref. 1-2: Internal communications:

W.D. Wilson/L.W. Weisbecker, Preliminary Data on the IB-2T-1 In-Pile Test, AN-FDD-110, 17 October 1960

L.W. Weisbecker, IB-2T-1 Post-Irradiation Examination, AN-FDD-121, 22 November 1960

H.S. Dreyer, Trip Report: IB-2T-1 Hot Cell Examination Nov 22 to Dec 9, 1960, AN-FDD-128, 19 December 1960

H.T. Watanabe, Trip Report to GE-ANPD and Area Hot Cells, NRTS, Idaho, December 18 Through 22, 1960, AN-FDD-133, 30 December 1960

L.W. Weisbecker, IB-2T-1 Post-Irradiation Examination Leak Check, AN-FDD-151, 27 February 1961

L.W. Weisbecker, IB-2T-1 Pin Puncturing-Fission Gas Release Measurements, AN-FDD-163, 31 March 1961

A.B. Burgess, Data From IB-2T Puncture Test, AN-FDD-170, 5 June 1961

L.W. Weisbecker, Results of Post-Irradiation Gamma Scanning of IB-2T-1 In-Pile Test Element, AN-FDD-218, 12 September 1961

Ref. 1-3: Army Gas-Cooled Reactor Systems Program, Defective Fuel Element Testing For the ML-1, AGN-TM-404, September 1963, Aerojet-General Nucleonics, San Ramon, California

Ref. 1-4: Army Gas-Cooled Reactor Systems Program, Final Summary Report of the Gas Cooled Reactor Experiment I, IDO-28598, October 1963, Aerojet-General Nucleonics, San Ramon, California

2.0 REPORT ORGANIZATION

This report describes the design of the test elements and the in-pile test loop; the operation of, and results from the test program; and results of the post-irradiation hot cell and of the metallurgical evaluations of the elements. This information is presented under five major headings: Introduction, Summary, In-Pile Test, Test Evaluation, and Metallurgy. Specific phases are identified by consecutive numbers (i.e., 1.0, 2.0, 3.0, etc.) throughout the report for convenience in cross referencing information. These sections are, in turn, divided into decimal subsections (i.e., 1.1, 1.2, and 1.3). Figures and tables are inserted close to the relevant portion of the text and are numbered according to the sections of the text in which they appear (e.g., Fig. 5-1, Table 6-2). References are listed at the end of each section of text.

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II. SUMMARY

3.0 SUMMARY OF TEST

The in-pile tests of the ML-1-I prototype fuel element were conducted in a loop in the General Electric Test Reactor at Vallecitos, California. The loop had been originally designed for a test program for the Oak Ridge National Laboratory and was modified for use in the AGCRSP. The six test elements included two dummy elements; a flux measuring element; two prototype elements instrumented to provide information on coolant, cladding and central fuel temperatures; and one uninstrumented prototype element.

After modification of the loop, tests were conducted with the dummy element to check out instrumentation, to calibrate the loop heaters, to establish the flow characteristics of the loop, and to verify the integrity of the loop and of the mechanical procedures for handling the test elements. At the conclusion of this work, the flux at the test section was measured with the flux measuring element. The in-pile irradiation of one instrumented and one uninstrumented prototype test element was begun on 2 December 1960. The coolant in the loop was oxygenated nitrogen (99.5 wt% N_2 + 0.5 wt% O_2). The instrumented (IB-8T-1) test element was removed from the loop on 1 January 1962 after 6415 hr of operation. Air was substituted for oxygenated nitrogen as the loop coolant and test irradiation of the uninstrumented (IB-8T-2) test element continued until 7 October 1962, when this element was removed after 10,131 hr of operation.

4.0 SUMMARY OF TEST RESULTS

During the early phases of the test, the temperatures observed in the IB-8T-1 test element agreed (within the limits of experimental error) with those predicted analytically, and the pressure drop across the element agreed quite closely with that predicted by laboratory flow experiments.

No fission product activity was observed in the loop coolant at any time during the test, indicating that the cladding operated without failure throughout the test. The Ar-41 activity in the loop increased by a factor of approximately 100 at the time the loop coolant was changed from oxygenated nitrogen to air; this activity increase did not cause any problems in the operation of the equipment.

Examination of the test elements after irradiation revealed the following:

- The fuel pins in both elements were twisted and spiralled, particularly in the lower section.
- The corrosion resistance of the Hastelloy X fuel pin cladding was very satisfactory; the maximum depth of oxidation penetration was 0.0044 in.
- The cladding of certain fuel pins in the IB-8T-1 elements was severely embrittled, a condition subsequently attributed to the formation of a second phase in the material induced by the presence of nitrogen in the interior of the fuel pin.
- The fuel pin cladding in the IB-8T-2 was only moderately altered by the exposure; the condition of the material was such that the element was considered serviceable after the 10,000-hr irradiation.
- The UO_2 fuel was severely cracked, but only minor cracking was observed in the BeO-UO_2 fuel. Both fuels were considered satisfactory for continued operation.
- The fission gas release in the IB-8T-2 (10,000 hr) test element was less than 1% in the BeO-UO_2 fuel (9.5% U-235 burnup) and 5.7% in the UO_2 fuel (2.9% U-235 burnup).

Post-irradiation evaluation of the IB-8T-1 element extended over approximately 18 months because of the need to identify and determine the cause of the embrittling second phase in the fuel pin cladding. The IB-8T-2 was evaluated in much less time since no complicating changes were observed.

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III. IN-PILE TEST

5.0 PURPOSE AND OBJECTIVES

The purpose of the IB-8T in-pile test was to verify the adequacy of the ML-1-I fuel element design.

The design of the ML-1 reactor requires that the 61 fuel elements furnish up to 3 Mw of thermal energy to the working fluid at a maximum reactor outlet temperature of 1200°F. The performance specifications of the power plant require that the fuel elements be capable of at least 3,000 hr of continuous full power operation; a design lifetime of 10,000 hr was established as being highly desirable.

The design of the ML-1-I (first core loading for the ML-1) fuel elements was begun in mid-1959. The first objective of this effort was the design of pin-type fuel elements (designated IB-2L) for test operation in the Gas-Cooled Reactor Experiment (GCRE-I) at NRTS. This core loading was completed in mid-1960 and operated in the GCRE-I reactor until April 1961 when the GCRE program was deactivated because of problems with the reactor calandria. During this period the IB-2L core accumulated 750 Mwh of exposure which was not an adequate demonstration of the ML-1 fuel element design concept.

Extensive neutronic, hydraulic and thermal experiments were conducted during the design of the ML-1-I fuel elements. In addition, seven developmental test models were exposed in in-pile tests for periods up to 2300 hr to evaluate various aspects of the design.

5.1 Cladding Surface Temperature

The configuration of the ML-1-I pin-type fuel element (a bundle of highly absorbing fuel pins) and the ML-1 core (non-uniform fuel element spacing to reduce core radial power peaking) as well as the fission spectrum of the ML-1 reactor (intermediate as a result of undermoderation) combine to make the analytical determination of fuel cladding temperature extremely difficult. In the case of the ML-1-I design, the pin-to-pin power ratio, the power skewing across the fuel elements and the reactor axial power distributions were measured in critical assembly experiments. The neutronic measurements were related to

the ML-1-I fuel element thermal design by the HECTIC (Ref. 5-1) computer code. HECTIC is a heat transfer analysis code capable of considering many different fuel element geometries with parallel single-pass flow channels enclosed by heated or unheated surfaces. For given parameters (geometry, coolant fluid, power distribution, coolant inlet distribution, coolant flow rate, fuel element power and heat transfer correlation), HECTIC calculates the channel surface and fluid temperature distributions. Conduction between surfaces, convection to the fluid, radiation between surfaces and turbulent interchange between fluid channels are also considered in the calculations. The HECTIC code was used to iterate between pin bundle power distributions and pin surface temperature distributions until a fuel element design was achieved which satisfied the thermal hot spot criterion (established by the material considerations discussed below).

Thermal capacity laboratory tests demonstrated that the average heat transfer coefficients in the elements were in good agreement with those predicted by the analysis. Mass transfer tests showed less than 10% local variation in heat transfer coefficients around the periphery of the fuel pins.

A primary objective of the IB-8T test was to verify the laboratory experiments and design calculations that indicated the element would operate at full power without exceeding the fuel pin cladding thermal hot spot limit. For this purpose, one test element (IB-8T-1) was equipped with thermocouples to measure the fuel pin cladding temperature.

5.2 Cladding Material Selection

Literature surveys and out-of-pile screening tests had led to the conclusion that Hastelloy X was the best material for the fuel pin cladding in the ML-1-I fuel element design. However, the limited long term, high temperature creep data on Hastelloy X indicated that the cladding would probably collapse onto the fuel pellets during operation.

An objective of the IB-8T test was to establish that Hastelloy X had sufficient strength and resistance to corrosion by the coolant gas to operate satisfactorily for the design lifetime. In addition, the test was designed to determine whether the cladding collapsed onto the fuel pellets and to assess the effect of such collapse, and of the subsequent changes in the fuel element configuration, on the temperature distribution in the fuel and cladding.

5.3 Fission Gas Release

Although fission gas release from UO_2 fuel had been measured by many investigators (Ref. 5-2) and the data had been successfully correlated to a temperature-dependent diffusion model, most data were based on assumed reactor power density and fuel temperature factors considerably higher than

those associated with the ML-1-I design. It appeared quite likely that the mechanism for fission product release at ML-1-I conditions would prove to be the recoil mechanism rather than the diffusion mechanism. In addition, very little information existed concerning fission gas release from BeO-UO₂ fuel.

The release of fission gases from the fuel pellets in the ML-1-I design dilutes the helium gas with which the pin was filled at the time of fabrication and consequently decreases the thermal conductivity of the gas mixture in the gap between the fuel and cladding. As the gas thermal conductivity is decreased, the fuel temperature is increased, thus further increasing the amount of fission gases released from the fuel.

Because of the variation in gas release data found in the literature and the absence of a basic understanding of the fission gas release mechanism in the ML-1-I element, the design of the upper end of each fuel pin incorporated a gas expansion chamber. It was planned to puncture each fuel pin and remove the gas from the pin for analysis. Analysis of the gas would provide data on the assumptions concerning fission gas release and the effect of such release on temperatures in the ML-1-I design.

5.4 Pressure Drop

The closed Brayton cycle used in the ML-1 power plant is extremely sensitive to pressure losses; the net thermodynamic output of the cycle is reduced 6 kw for each increase of 1 psi in system pressure drop. Hydraulic experiments were conducted to establish the experimental friction factor correlation for the ML-1-I fuel element design. An objective of the IB-8T experiment was to confirm this correlation in the reactor environment and to monitor the pressure drop over the test lifetime to evaluate the influence of corrosion product formation or changes in the configuration of the fuel element on pressure drop.

5.5 Lifetime

An objective of the IB-8T experiment was to assess the ability of the ML-1-I fuel elements to operate satisfactorily for the 3,000 hr required lifetime, and to assess the capability of the design for longer periods of operation. Such a demonstration can be conducted only in an in-pile test where the interaction of all the variables affecting the design is summarized in a prototype situation and the influence of transients associated with reactor startup, shutdown and scram can be evaluated.

Ref. 5-1: HECTIC-II, An IBM 7090 Fortran Computer Program for Heat Transfer Analysis of Gas or Liquid-Cooled Reactor Passages, IDO-28595, December 1962, Aerojet-General Nucleonics, San Ramon, California

Ref. 5-2: Belle, J.A., Editor, "Uranium Dioxide, Properties and Nuclear Applications," Chapter 9, pp. 477-526, U. S. Government Printing Office, Washington, D. C., July 1961

6.0 IN-PILE TEST LOOP DESCRIPTION

6.1 The GETR Facility

The IB-8T test elements were tested in the AGN-GETR loop in the General Electric Test Reactor (GETR) at the Vallecitos Atomic Laboratory. This loop was originally used by the Oak Ridge National Laboratory to test prototype elements for a helium-cooled reactor and was modified for the IB-8T tests.

The GETR is a versatile facility, capable of a wide variety of irradiations under varying environmental conditions. The facility consists of a pressurized water-cooled and water-moderated reactor and supporting auxiliaries (Fig. 6-1). The reactor normally operates at a power level of 30 Mw(t). The reactor core, contained within a 24-in. diameter aluminum pressure vessel, is submerged in a water pool 9 ft in diameter. The core is a 24- by 24-in. matrix with an active length of 36 in. Beryllium and aluminum reflector pieces are provided to fill the annular spaces between the square core and the round pressure vessel. The reactor is controlled by six bottom-mounted, top-entry control rods each of which consists of a poison section and a fuel section.

The reactor core is forced-convection-cooled by demineralized water circulated in the primary system. The water enters near the top of the pressure vessel, flows downward through the core, and discharges near the bottom of the vessel. The water then flows through the primary heat exchanger where it is cooled by the secondary water system. The secondary water system discharges heat to the atmosphere through a cooling tower.

The ORNL loop was a multiple test section helium loop that could irradiate 15 elements simultaneously. The loop was declared surplus in December 1959 at which time it was transferred to the AGCRSP and modified for long term testing at ML-1 conditions. The AGN-GETR loop was operated for AGN by the General Electric Co. under Project Agreement No. 18 of Contract AT(04-3)-189 and IPSO No. 1 of Contract AT(10-1)-1034.

6.2 AGN-GETR Loop

The operating characteristics of the loop are summarized in Table 6-1 on page 14.

FIGURE 6-1. GENERAL ELECTRIC TEST REACTOR EXPERIMENTAL FACILITIES

TABLE 6-1 - TYPICAL OPERATING CHARACTERISTICS OF AGN-GETR GAS LOOP

Element power	55 kw
Coolant temperatures	
Inlet	800°F
Outlet (mixed mean)	1200°F
Coolant mass flow rate	1560 lb/hr
Loop operating pressure	315 psia

The loop consists of three sub-loops (hereafter termed facility tubes) with common systems for gas circulation, heating and supply (Fig. 6-2). Only two of these facility tubes were used during the IB-8T test. The facility tubes were located in the reflector section of the GETR reactor (Fig. 6-3). Each U-shaped facility tube consists of a fuel element test section in the leg nearest the reactor and a heat exchanger and effluent filter in the leg farthest from the reactor (Fig. 6-4). A positioning mechanism is provided at the bottom of the facility tube to provide for adjustment of the test section to control the power generated in the test element (Fig. 6-5).

A flux controlling assembly was placed around the test section to minimize the steep flux gradient through the reflector region of the GETR. A stainless steel shutter was provided to absorb thermal neutrons on the high flux side of the test position and an aluminum "window" was provided to minimize thermalization by water on that side. This arrangement permitted maximum penetration of fast neutron flux into the test element, thus increasing the power on the low flux side of the test section. A significant flux gradient existed across the test section during the experiment despite this arrangement. The result was that the part of the test element nearest the reactor received about twice the exposure of similar parts on the side away from the reactor.

Four axial-flow, rotary, spiral compressors were provided to circulate the coolant gas in the loop (Table 6-2). Two spare units were installed in parallel to preclude emergency loop shutdown if the primary units failed. The four compressor and motor assemblies were housed in containment vessels vented to the GETR control vent system.

The compressor delivered coolant to the high pressure (low temperature) pass of the regenerative heat exchanger (Fig. 6-2). This shell and tube unit was 8-in. OD by 32 ft long. The high pressure pass consisted of 60 tubes with a total heat transfer area of 290 sq ft. The gas entered the high pressure pass at 160°F and exited at 680°F. The gas entered the shell side at 820°F and exited at 290°F. The operating pressure on the tube side was 318 psia, and on the shell side was 295 psia. The exchanger transferred about 6.25×10^5 Btu/hr under these conditions.

A bulk gas heater was provided to increase the temperature of the coolant gas entering the facility tube from about 665°F to about 835°F. The maximum output of the bulk gas heater was about 250 kw, generated in finned Nichrome electrical heating elements. An electrical trimming heater (15 kw) was installed in the inlet leg of each facility tube to provide final adjustment of the temperature before the coolant gas entered the facility tube.

FIGURE 6-3. LOCATIONS OF GETR EXPERIMENTAL FACILITIES

FIGURE 6-4. AGN-GETR LOOP SHOWING ONE FACILITY TUBE IN PLACE ADJACENT TO THE GETR REACTOR

FIGURE 6-5. TYPICAL TEST SECTION OF AGN-GETR GAS-COOLED LOOP

TABLE 6-2 - PERFORMANCE SPECIFICATIONS FOR
AGN-GETR GAS LOOP CIRCULATING COMPRESSORS

	<u>Stage I</u>	<u>Stage II</u>
Speed, rpm	1180	1180
Suction pressure, psia	288	308
Compression ratio	1.070	1.062
Discharge pressure, psia	308	327
Suction temperature, °F	120	143
Discharge temperature, °F	143	162
Brake horsepower	19.7	18.6
Suction flow, cfm	126	128
Discharge flow, cfm	118	120
Blow back flow, cfm	---	10
Flow, lb/hr	9700	9700

The gas from the test section of each facility tube passed through the heat dissipation leg, the Fiberfrax filter and into the regenerative heat exchanger. (These filters removed micron-size particles from the gas stream.) In addition, a portion of the coolant was bypassed at the compressor discharge through a silica gel purifier and returned to the loop at the compressor suction. A high level of coolant purity was thus maintained in the cooling system.

The gas from the regenerative heat exchanger passed through a precooler. The precooler was a gas-to-water heat exchanger provided to reduce the temperature of the gas at the suction side of the compressor to 120°F. The precooler had a heat transfer surface of 182 sq ft and a capacity of 850,000 Btu/hr. The normal temperature of the inlet water was about 85°F, and the inlet gas temperature was about 190°F. Water flow through the precooler was regulated to maintain a constant temperature in the outlet gas.

Coolant gas was supplied to the loop through a pressure system from a bank of ten cylinders (300 scf each). In addition, an independent system, consisting of ten 300 scf cylinders connected to a separate manifold, was available for emergency cooling.

The coolant gas in the loop could be transferred into three shielded 1800-cu-ft dump tanks. Radioactive gas could be held in these tanks until the radiation level permitted gradual venting to the GETR containment vessel stack. A 2 hp roughing vacuum pump was provided in this system for evacuation of air from the loop during charging, and for the transfer of contaminated loop gas into the dump tanks.

The instrumentation required for loop operation was located on the loop control panel in the experimental area (Table 6-3). The parameters critical to reactor operation were also monitored in the GETR control room.

TABLE 6-3 - AGN-GETR LOOP INSTRUMENTS AND CONTROLS

Parameter	Recorder Location*	Alarm & Scram Setting**		
		Alarm		Scram
	Reactor Control Rm	High	Low	
Compressor discharge pressure, psig	X	335	280	250
Facility tube flow, lb/hr	X	---	1320	1090
Facility tube exit gas temperature, °F	X	875	---	900
Primary coolant activity, mr/hr	X	500	---	1000
Bypass Δ P, psi	X	40	25	50
Supply header pressure, psig	-	---	700	---
Compressor suction gas temperature, °F	-	140	---	---
Facility tube inlet gas temperature, °F	-	840	---	---
Precooler water outlet temperature, °F	-	120	---	---
Precooler water outlet flow, %	-	---	50	---
Main loop flow, %	-	---	80	---
Dump tank radiation, mr/hr	-	10	---	---

*Note: Recorders for these parameters were all located in the loop console in the experimental area. Those indicated "X" were duplicated in the reactor control room.

**For typical test run.

The loop operated automatically under steady state conditions. Manual control was used for startup, shutdown, correction of malfunctions and elimination of transients. The mass flow of coolant gas to each facility tube was controlled by a valve in the discharge leg through a recorder-controller which sensed a signal from one of the four thermocouples located at the outlet end of the test section. These four thermocouples were operated in pairs, one pair for sensing to control the mass flow of coolant gas, and the other for sensing to protect against high temperatures. Failure of the on-stream mass flow control thermocouple would drive the recorder-controller up-scale, opening the flow control valve. Manual switching to the other thermocouple returned the loop to automatic control. The alternate thermocouples were used for reactor scram on high exit gas temperatures. The temperature of the gas entering the test section was controlled by the trimming heater on the basis of the temperatures sensed by one of two thermocouples at the inlet to the test section.

All potentially radioactive portions of the loop were shielded with concrete to permit normal operations in the vicinity of the loop equipment. The loop was designed for, and operated with, the reference ML-1 gas coolant (99.5 vol% N₂ + 0.5 vol% O₂) until the end of 1962 when air coolant was substituted for the balance of the IB-8T-2 irradiation program.

7.0 TEST ELEMENT DESCRIPTIONS

Three IB-8T test elements were fabricated for in-pile testing. In addition, two dummy elements and a flux mapping element were fabricated and used in preliminary investigations. The various elements were designated as follows:

- IB-8T-1: Instrumented to provide information on coolant, cladding and central fuel temperatures
- IB-8T-2: An "uninstrumented" element*
- IB-8T-3: Duplicate of the IB-8T-1 (provided as a spare and not tested)
- Dummy elements: Used in pre-startup tests to determine heat loss and pressure drop in the facility tubes
- Flux mapping element: Used to determine the power distribution in the test section

7.1 In-Pile Test Elements

The IB-8T test elements and those fabricated for the ML-1 first core loading were identical except for minor deviations needed to adapt the element to the test section. The assembly was basically an insulated cylindrical tube containing 19 fuel pins (Fig. 7-1). The elements were 32.6 in. long, 1.720-in. OD and weighed about 10 lb. The fuel pin tubing was low cobalt (about 0.7%) Hastelloy X (Table 7-1); the outside diameter of the tubing was 0.241 in. and the wall thickness was 0.030 in. The pins were arranged in two rings about the central pin; the inner ring contained 6 pins and the outer ring 12 pins. All pins were welded to an upper spider and were positioned by, and free to move within a lower spider. A Hastelloy X spacer wire (0.040-in. OD) was wrapped around each fuel pin (except the center one) on a 6.75 in. pitch to maintain pin spacing. The fuel pin bundle was surrounded by a double liner assembly. The annulus between the liners was filled with thermal insulation to prevent heat loss to the calandria tube and moderator water. A flexible stainless steel aircraft cable, with swaged fittings, was provided at the inlet end for handling.

*The IB-8T-2 element was instrumented to measure only inlet and outlet gas temperatures.

CLAD TEMPERATURE
THERMOCOUPLE DETAIL

FIGURE 7-1. IB-8T-1 TEST ELEMENT

TABLE 7-1 - CERTIFIED CHEMICAL ANALYSIS OF HASTELLOY X TUBING
USED AS FUEL PIN CLADDING IN IB-8T ELEMENTS

<u>Element</u>	<u>Abundance, wt%</u>	<u>Element</u>	<u>Abundance, wt%</u>
Nickel	48.51	Cobalt	0.07
Chromium	22.37	Zirconium	0.064
Iron	18.41	Copper	0.03
Molybdenum	8.73	Calcium	0.01
Manganese	0.75	Phosphorous	0.01
Silicon	0.61	Sulfur	0.005
Tungsten	0.31	Lead	0.001
Carbon	0.12		

The active fuel material in the IB-8T elements was fully-enriched UO_2 . The outer ring of pins was loaded with pellets containing 60 wt% UO_2 with a BeO diluent; the inner ring was loaded with fully-enriched UO_2 pellets. The central pin was not fueled. All pellets were cold-pressed and sintered to 95% theoretical density. The total loading in the IB-8T-1 element was 706.3 grams of U-235; the total U-235 loading in the IB-8T-2 element was 728.6 grams. This difference resulted from the use of cored pellets in seven of the fueled IB-8T-1 pins to permit thermocouple installation.

The burnable poison assembly consisted of two concentric cylinders of metal foil placed outside of the inner liner at the lower end of the element. The poison foil inner cylinder was formed from 0.94 wt% cadmium in copper, coated on both sides with Kanigen nickel plating 0.0005 in. thick. The poison foil was formed into a cylinder 9 in. high by 4.58 in. diameter. A cover foil of AISI Type 316 stainless steel was formed into a cylinder 12.5 in. high and placed over the poison foil. The foil assembly was positioned so that the top of the burnable poison was at the mid-plane of the fueled region of the test element thus depressing the flux in the lower half of the element. The assembly was attached to the inner liner by spot welds at the ends of the cover foil. This configuration differed from that of the ML-1 fuel element where the poison foil is located on the inner surface of the outer liner. The change was made to facilitate assembly of the IB-8T element in which the outlet gas thermocouple was routed between the inner and outer liners. The IB-8T arrangement, however, located the foil assembly in a higher temperature region than is the case in the ML-1 design.

In the IB-8T-1 and -3 test elements, two thermocouples sensed the temperature of the gas at the inlet end of the element, and four sensed the outlet gas temperatures (see Table 7-2 for thermocouple locations in the IB-8T elements). One thermocouple sensed fuel pellet central temperature, and seven sensed axial and radial cladding temperatures (Fig. 7-2). The coolant sensing thermocouples were fabricated from chromel/alumel wires (0.010 in. diameter) sheathed in 0.060-in. diameter Inconel tubing. The cladding and fuel temperature sensing thermocouples were fabricated from 0.005-in. diameter chromel/alumel wires sheathed in 0.040-in. diameter Inconel tubing. The cladding

TABLE 7-2 - IB-8T TEMPERATURE INSTRUMENTATION

<u>Test Element</u>	<u>Function</u>	<u>THERMOCOUPLE</u>		<u>Sheath Diameter*, in.</u>
		<u>No.</u>	<u>and Position</u>	
IB-8T-1 & -3	Gas Coolant	2	Inlet	0.060
		4	Outlet	0.060
	Fuel Central Pellet	1	Pellet Core	0.040
	Cladding	7	Axial & Radial	0.040
IB-8T-2	Gas Coolant	2	Inlet	0.060
		4	Outlet	0.060

*Inconel tube

thermocouples terminated in stainless steel discs which provided a heat path from the interior of the cladding to the thermocouple junction. The discs were insulated from the fuel pellet on either side with zirconia wafers. The fuel pellets in this element were cored (0.050-in. dia) to accommodate the thermocouples for cladding and fuel temperature sensing. The thermocouple sheaths were brazed to the Hastelloy X upper plug, and the leads were spliced to Inconel-sheathed chromel/alumel thermocouple extensions immediately above the element. Figure 7-3 shows the arrangement of the cladding and fuel thermocouples in the fuel pins.

The IB-8T-2 (uninstrumented) element was not equipped with thermocouples in the fueled region but was equipped with four thermocouples to sense outlet gas temperatures and two to sense inlet gas temperatures.

7.2 Dummy Elements

Two dummy elements were fabricated for use in non-nuclear tests of the loop. The elements, identical except that one was instrumented, included insulated liner assemblies identical with those described for the IB-8T test elements. The fuel pin bundle was omitted from the dummy elements and an orifice (0.650-in. ID) was installed to approximate the pressure drop of the pin bundle at reference flow conditions. One dummy element was equipped with thermocouples to sense coolant gas temperature at the inlet and outlet of the element and the outer liner temperature.

7.3 Flux Measuring Element

The flux measuring element was a 19-pin bundle equipped with a 6-in. long fuel stack, and inserted into the liner assembly so that the center plane of the flux element coincided with the center plane of the GETR. A spring spacer held the fuel pin bundle in the flux element and facilitated disassembly for counting after irradiation (Fig. 7-4). The probe and bracket on the lower end of the assembly anchored the flux element in the GETR pool, duplicating the location of the test elements. Cored fuel pellets, identical with those used

FIGURE 7-2. THERMOCOUPLE LOCATIONS IN THE IB-8T-1 AND -3

FIGURE 7-3a. ARRANGEMENT OF IB-8T-1 FUEL CLADDING THERMOCOUPLE

FIGURE 7-3b. ARRANGEMENT OF IB-8T-1 CENTRAL FUEL THERMOCOUPLE

FIGURE 7-4. IB-8T FLUX MEASURING ELEMENT

in the IB-8T-1 test element, were used in the flux element. Cadmium discs (0.020-in. thick) and cadmium cylinders were used to shield the ends of the flux assembly from axial neutron streaming. Cadmium-covered gold foils, bare gold foils and axially placed gold wires were installed in the flux element (Fig. 7-5). Aluminum foils were placed between the pellets of pins 3, 6, 9, 12, 13, 15 and 17 at the vertical center of the fuel stack. These detectors were counted after irradiation to provide data from which the flux distribution was calculated.

FIGURE 7-5. IB-8T FLUX MEASURING ELEMENT - FLUX MONITOR LOCATIONS

8.0 PRELIMINARY TESTS

The following preliminary tests were performed before the IB-8T test elements were inserted into the loop for irradiation:

- The characteristics of the compressors were established and the heaters were calibrated during checkout of the circulation system components.
- The flux mapping element was installed in the test position and the reactor was operated for 10 minutes to obtain the data needed for flux mapping.
- After the loop had been modified by installation of the facility tubes, the accuracy and speed of response of the loop instrumentation were measured.
- Pressure drop measurements were taken across dummy elements installed in two of the facility tubes (the third was valved off).
- A test was conducted with the instrumented dummy element to measure heat loss and mass flow rates to be expected during testing.

8.1 Circulation System Component Checkout

Temporary piping was installed to connect the discharge and suction headers of the circulation system and tests were performed to establish the characteristics of the compressors and to calibrate the heaters. The circulation system was operated for about 500 hr. Each compressor was operated with the containment vessel open and closed under no load and full load. In addition, each bank of compressors was operated to verify series performance. The electrical heater controllers were calibrated against measured gas temperatures at various flow rates. The main compressors operated about 300 hr, and the standby units operated about 200 hr during these tests. Design temperatures had been maintained in the loop for about 230 hr of the 500-hr total.

8.2 Flux Mapping Experiment

The flux mapping experiment was conducted prior to the installation of the facility tubes by installing the flux mapping element in the test position and operating the GETR reactor at 13.7 kw for 10 minutes. After 4 hr of cooling, the element was disassembled and the flux monitors removed for counting. The gold wires and fuel pins without foils were scanned to determine distribution of gamma activity (Fig. 8-1). The aluminum foils and the central fuel pellets from the disassembled pins were counted to obtain data to establish the pin-to-pin radial power. Table 8-1 compares experimental results with calculated values. The effective neutron flux was calculated from the observed activation of the gold foils. The activity of the bare gold foils was extrapolated to a position corresponding to the location of the cadmium-plated gold foil by use of the curves in Fig. 8-1. The cadmium-ratio at this point was then determined and assumed to be constant along the length of the fuel pin. These data were used to determine the bare gold foil activity and the thermal and epicadmium saturated activity at the mid-point of the fuel pin.

TABLE 8-1 - IB-8T FLUX ELEMENT RADIAL POWER DISTRIBUTION
(Relative Count Rate)

<u>Pin No.</u>	<u>Aluminum Foil Activity</u>	<u>Fuel Pellet Activity</u>	<u>Calculated Values</u>
1	1.36	1.36	1.37
2	1.23	1.23	1.30
3	1.02	1.02	1.26
4	0.76	0.77	1.02
5	0.62	0.64	0.92
6	0.64	0.63	0.80
7	0.71	0.71	0.83
8	0.82	0.84	0.88
9	1.05	1.08	1.30
10	1.25	1.25	1.42
11	1.53	1.45	1.47
12	1.50	1.48	1.44
13	1.10	1.10	0.73
14	0.83	0.83	0.70
15	0.66	0.66	0.61
16	0.78	0.78	0.56
17	0.98	0.98	0.65
18	1.16	1.16	0.74

FIGURE 8-1. IB-8T FLUX ELEMENT AXIAL FLUX DISTRIBUTION

The results of these calculations as well as the calculated thermal flux values for the normal GETR operating power of 30 Mw are given in Table 8-2. The sharp difference between the measured and calculated thermal flux for pins 13, 15 and 17 is the result of neutron streaming between the pins in the outer row; the mathematical model used in the calculation did not provide for this streaming.

TABLE 8-2 - IB-8T NEUTRON FLUXES (n/cm²-sec)

<u>Pin No.</u>	<u>Thermal Activation</u>	<u>Epicadmium Activation</u>	<u>Thermal Flux Extrapolated to 30 Mw</u>	<u>Calculated Thermal Flux at 30 Mw</u>
3	0.501×10^8	0.781×10^8	3.55×10^{12}	3.84×10^{12}
6	0.301×10^8	0.519×10^8	2.13×10^{12}	2.45×10^{12}
9	0.552×10^8	0.648×10^8	3.91×10^{12}	3.98×10^{12}
12	0.621×10^8	1.00×10^8	4.40×10^{12}	4.37×10^{12}
13	0.346×10^8	0.644×10^8	2.45×10^{12}	0.588×10^{12}
15	0.151×10^8	0.489×10^8	1.07×10^{12}	0.491×10^{12}
17	0.084×10^8	0.604×10^8	0.595×10^{12}	0.522×10^{12}

The data from the flux mapping test were used to calculate the thermal power which would be generated in the IB-8T elements. The calculations indicated that, at the position in which the flux mapping element was irradiated, the IB-8T elements would generate 89 kw. Since the desired operating power for the test elements was 55 kw, provision was made to position the facility tube further away from the reactor core before initiating the nuclear tests.

8.3 Instrumentation Checkout

After the modification of the loop had been completed by installation of the facility tubes, the accuracy and speed of response of the loop instrumentation were measured. The scram points on the various instruments were set and checked and the functioning of the instrumentation system was demonstrated.

8.4 Flow Calibration

The dummy elements were installed in two of the facility tubes and the third tube was valved off. Pressure drop measurements across the dummy elements were taken at various flow rates at ambient temperature and at design inlet temperature, and excellent agreement was obtained between the measured and calculated pressure drops (Table 8-3).

8.5 Heat Loss Experiments

Calculations were performed which indicated that a difference of as much as 300°F could exist between the local gas temperatures in the IB-8T elements and the temperatures indicated by the gas outlet thermocouples installed in the elements. This difference resulted from the power skewing in the element

TABLE 8-3 - AGN-GETR GAS LOOP PRESSURE DROP CALIBRATION

	Compressor Discharge Pressure, psig	Orifice Temp, °F	Flow Rate lb/hr	Pressure Drop, psi	
				Measured	Calculated
<u>COLD RUN</u>					
<u>Facility Tube 2</u>					
Run A	110	85	1125	7.0	7.7
Run B	116	88	1040	4.7	6.2
Run C	119	90	729	3.6	2.9
<u>Facility Tube 3</u>					
Run A	110	92	1150	6.3	8.1
Run B	116	90	1040	4.1	6.2
Run C	119	95	729	1.7	2.9
<u>HOT RUN</u>					
<u>Facility Tube 3</u>					
Run A	315	800	1285	8.3	8.0
Run B	317	800	1556	12.0	11.7
Run C	317	800	1700	14.4	14.0
Run D	316	800	1910	17.8	17.7

and the relatively short mixing length between the end of the fuel pin and the thermocouple location. A test was conducted with the instrumented dummy element to measure the heat loss from the facility tube over the range of temperatures and mass flow rates anticipated during testing. These data were analyzed to develop a relationship between the indicated and true mixed mean outlet gas temperature. This relationship is shown in Figure 8-2.

FIGURE 8-2. FACILITY TUBE HEAT LOSS CORRELATION FOR THE AGN-GETR LOOP
(Based on 70°F pool temperature)

9.0 TEST ELEMENT IRRADIATION9.1 Startup

Prior to the startup of the GETR reactor and the initial irradiation of the IB-8T-1 and -2 test elements, flow tests were made to determine the pressure drop in the test elements (Table 9-1).

TABLE 9-1 - STARTUP FLOW TEST DATA

IB-8T-1 Fuel Element
(Facility Tube 3)

<u>Flow (cfm)</u>	<u>Temperature (°F)</u>		<u>Pressure Drop (psi)</u>
	<u>Inlet</u>	<u>Outlet</u>	
30.0	760	750	5.4
36.5	780	770	7.5
39.9	800	795	9.0
45.0	800	790	11.0

IB-8T-2 Fuel Element
(Facility Tube 2)

30.0	765	755	5.5
36.5	790	783	7.7
39.9	805	800	9.0
45.0	805	800	11.1

The loop was placed in operation to maintain an inlet temperature of 500°F to the test sections. The GETR reactor was then started up and irradiation of the test elements began on 2 December 1960. When the GETR reached 50% of full power, thermodynamic balances indicated that the test elements were each generating about 52 kw. This value, when extrapolated to full GETR power, would have been significantly in excess of the desired 55 kw operating level. The GETR reactor was shut down and the facility tubes were moved away from the core. The GETR reactor was restarted and at full power the test elements

were generating from 55 to 65 kw. Attempts to increase the coolant gas temperature at the inlet to the test elements were discontinued when the inlet temperature reached approximately 700°F, at which time the indicated cladding temperature reached the design limit (1650°F). The test continued at these conditions for the balance of the the GETR cycle.

At the end of the first GETR cycle, shutters fabricated from 1/2-in. thick stainless steel were installed on the reactor side of the facility tubes to reduce the power skewing in the element which had prevented the attainment of reference inlet temperatures in the first cycle. This modification reduced the temperature skewing in the elements from approximately 430° to approximately 200°F. No further modification of the test section configuration was necessary for the remainder of the tests. However, the facility tube was moved with respect to the GETR core several times during the test to compensate for shifts in the spatial distribution of the GETR flux.

9.2 Normal Operation

The IB-8T-1 operated for 11 GETR cycles and the IB-8T-2 for 18 GETR cycles. Each cycle was about six weeks long and averaged about 600 operating hours at full power. A schematic representation of the operating history of the elements and selected performance data are presented in Figure 9-1. The operating data for a typical cycle is summarized in Table 9-2.

TABLE 9-2 - SUMMARY OF IB-8T OPERATING DATA*

	<u>IB-8T-1</u>	<u>IB-8T-2</u>
Element energy output, kwh	34,608	34,008
Power, kw		
Maximum	65.0	64.0
Minimum	51.2	52.0
Average	56.8	52.7
Operating time, hr	610	610
<u>GETR energy output, Mwd</u>	750	750

*Taken from cycle 28, 30 October to 26 November 1961

The IB-8T experiment is summarized in Table 9-3.

TABLE 9-3 - CUMULATIVE POWER AND ENERGY FOR IB-8T TEST

	<u>IB-8T-1</u>	<u>IB-8T-2</u>
Element energy output, kwh	348,288	539,018
Average power, kw	54.3	53.1
Operating time at or above GETR critical, hr	6,415	10,151

FIGURE 9-1. IB-8T TEST PROGRAM HISTORY

Several significant incidents that occurred during the test are summarized below:

- A loss of coolant flow occurred during a GETR shutdown in February 1961. The indicated fuel element temperatures did not exceed nominal values.
- Cladding temperatures in excess of 1850°F were measured during the GETR startup in April 1961. Operation at temperatures in excess of the specified limit continued for less than 60 seconds. The high temperatures were caused by higher than nominal power in the GETR reactor and aggravated by surface deposits on the fuel. The surface deposits resulted from contamination of the gas coolant by material from the molecular sieve. The inlet gas temperature was reduced and the foreign material removed from the coolant gas by use of the in-line filters. Normal operation was resumed within four days of this incident.
- The pressure shell of the loop bulk heater ruptured during the shutdown in April 1962. The rupture occurred because the electrical heating elements were not turned off at the time the coolant flow in the loop was discontinued. A rapid depressurization of the loop resulted from the rupture; the IB-8T-2 test element was blown upward in the facility tube about 24 in. but was re-seated without difficulty. In the following cycle, the inlet temperature to the test element was significantly lower than the nominal 800°F temperature and the power generated in the test element was held at 41 kw.
- In May 1962, the inlet temperature to the test element was maintained at 500°F to avoid exceeding the temperature limits in the IB-14R experiment in the adjacent facility tube. This condition was corrected by repositioning of the IB-14R experiment and normal operating conditions were attained during the following cycle.

A computer code, DART-8 (Ref. 9-1), was used to reduce the operating data generated by the test elements. Typical computer outputs are shown in Figures 9-2 and 9-3. The power was calculated from a thermodynamic balance across the test section. The inlet temperature data was read directly from the thermocouples at the inlet end of the test element but the computer applied correction factors to the output from the thermocouples located at the outlet of the test element to correct the power skewing and these corrected outlet temperatures are shown in the figures. The flow rate was calculated from the pressure drop correlations established during the non-nuclear tests prior to each cycle. The data identified as "coefficient A" in the figures is the result of a computer calculation of the value of the coefficient in the following expression:

$$A = f' Re^{0.2}$$

where f' is the Fanning friction factor and
 Re is the Reynolds number.

The value of the coefficient was monitored to detect changes in the configuration of the element since this value should vary with pressure loss but not with the rate of coolant flow.

The cladding temperature thermocouples in the IB-8T-1 element performed relatively satisfactorily for most of the test period. Cladding temperatures varied significantly during GETR cycles (as shown in Figure 9-1) because of the change in the GETR flux and the corresponding change in test element power. In addition, evidence of physical changes in the thermocouples became apparent after about 4500 hr of operation, at which time reproducible calibration was not possible. In the fall of 1961, it was observed that the cladding temperature thermocouples indicated temperatures below the coolant gas temperatures indicated during non-nuclear test runs. The performance of the thermocouples in the IB-8T-1 elements is summarized in Table 9-4.

TABLE 9-4 - PERFORMANCE OF IB-8T-1 FUEL PIN THERMOCOUPLES

<u>Type</u>	<u>Pin No.</u>	<u>Distance Along Fuel, in.</u>	<u>Performance Summary</u>
Cladding Temperature	1	19.8	Reading drifted downward after 8 cycles
Cladding Temperature	6	16.5	Reading drifted downward after 8 cycles
Cladding Temperature	10	6.5	Reading drifted downward after 8 cycles
Cladding Temperature	12	16.5	Reading drifted downward after 8 cycles
Cladding Temperature	13	16.5	Failed in fourth cycle
Cladding Temperature	17	6.5	Failed in second cycle
Central Fuel Temperature	18	16.5	Failed during initial startup
Outlet Gas Temperature	19	22.7	Failed in second cycle

The coolant gas in the loop was sampled periodically and analyzed by chemical and radiochemical techniques at G. E. Vallecitos. Typical results of chemical analysis for water vapor, oxygen, carbon dioxide and combustible carbon are shown in Table 9-5, on page 43.

Typical results of radiochemical analysis of the coolant gas for Argon-41 are shown in Table 9-6, page 43.

As shown in Figure 9-1, argon activity increased significantly when the coolant gas was changed from oxygenated nitrogen to air early in 1962.

FIGURE 9-2. IB-8T-1 TEST ELEMENT OPERATING DATA

FIGURE 9-3. IB-8T-2 TEST ELEMENT OPERATING DATA

Loop gas samples were also analyzed periodically for fission product activity. No activity was detected in any of these samples using analysis equipment capable of detecting fission product activity equal to or greater than 20% of the Ar-41 activity in any sample.

TABLE 9-5 - CHEMICAL ANALYSES OF TYPICAL LOOP GAS SAMPLES*

<u>Date</u>	<u>H₂O</u> <u>ppm by volume</u>	<u>O₂</u> <u>volume percent</u>	<u>CO₂</u> <u>ppm by volume</u>	<u>Combustible</u> <u>Carbon</u> <u>ppm by volume</u>
26 Sept 61	210	0.7	52	302
3 Oct 61	180	0.7	60	261
10 Oct 61	160	0.7	58	248
31 Oct 61	180	0.9	47	251
7 Nov 61	210	0.8	39	267
14 Nov 61	118	0.7	29	254
21 Nov 61	126	0.7	67	99
6 Dec 61	220	---	--	---
13 Dec 61	150	0.7	--	235**

*Cycles 27, 28 and 29

**40 ppm of the combustible carbon was CH₄

TABLE 9-6 - RADIOCHEMICAL ANALYSES OF TYPICAL LOOP GAS SAMPLES*

<u>Date</u>	<u>Argon-41</u> <u>(dpm/ml)</u>
26 Sept 61	2.4×10^2
3 Oct 61	3.5×10^2
10 Oct 61	2.9×10^2
31 Oct 61	3.2×10^2
7 Nov 61	3.0×10^2
15 Nov 61	7.9×10^2
21 Nov 61	5.5×10^2
6 Dec 61	3.2×10^2
13 Dec 61	4.2×10^2

*Cycle 24, 27, 28 and 29

Ref. 9-1: Internal communication: R. E. Hoogstoel, Modification of DART-8 Code, AN-AGCR-2, 8 March 1962

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IV. TEST EVALUATION

10.0 ANALYSIS OF TEST RESULTS

10.1 Analysis of Temperature Data

One of the primary objectives of the IB-8T experimental irradiation was to verify the thermal analysis used in the design of the ML-1-I element to demonstrate that the element would operate at full power without exceeding the cladding hot spot temperature limit. As indicated earlier, the IB-8T-1 test element was equipped with several thermocouples to provide data for this demonstration. The design of the thermocouples has been described earlier in this report (Section 7.1). The thermocouple configuration was based on calculations of the conductive heat flow paths needed to make the temperature sensed by the thermocouple equal to the unperturbed cladding surface temperatures. Uncertainties in this calculation and minor variations in component dimensions or assembly procedures altered the conductive heat flow paths and, as a consequence, introduced certain errors between the observed and actual temperatures. Assessment of these errors led to the conclusion that the IB-8T-1 thermocouples indicated correct cladding surface temperatures within 25°-50° F in the normal operating temperature range.

The thermal design of the ML-1-I fuel element was verified by comparing temperatures measured in the IB-8T-1 with corresponding temperatures predicted by calculations performed with the HECTIC code (Ref. 10-1). Normally, the power distribution data from the flux measuring experiment (IB-8Ø) could be used to predict the temperatures in the test element. However, in the IB-8T irradiations, changes made temperature predictions based on IB-8Ø flux data extremely difficult. These changes are summarized below:

- The IB-8Ø run was made with the test element in the position which corresponded to the nearest approach of the facility tube to the reactor core. The IB-8T-1 element did not operate in this position for any significant part of the irradiation.
- A stainless steel shutter was installed in front of the test section after the IB-8Ø run was completed to minimize the flux skewing across the test element.

- During the course of any GETR cycle, the control rod bank moved upward in the core, inserting more fuel. Such a movement tends to shift the axial power peak toward the inlet end of the test element (see Fig. 10-1). This flux shift tended to lower the hottest fuel pin cladding surface temperature.
- Partial refueling of the GETR core, or the introduction of highly absorbing materials in a nearby test position, influenced power distributions in the IB-8T element.
- It was necessary to reposition the facility tubes several times during the course of the irradiation to control the power generated in the test element. This repositioning altered the amount of moderation between the reactor core and the test element and consequently altered the flux and power distribution in the element.

The above variations made it very difficult to predict temperatures in the test element and to make meaningful comparisons between measured and predicted fuel pin surface temperatures. For this reason, two alternative methods of predicting temperatures were utilized:

- 1) During the post-irradiation examination, the axial power distribution in each of the fuel pins was determined by gamma scans. (Typical gamma scans are shown in Figure 10-2). The axial power distribution thus derived was then used as the basis for predicting cladding surface temperatures at the points of interest in the test element.
- 2) During the post-irradiation examination, each fuel pin was punctured and the gas extracted from the interior of the pins was analyzed in a mass spectrometer. The Xe-136 fraction of the gas provided the basis for calculating the effective thermal flux in each pin and the ratio of fluxes between the various pins. Calculations of fuel pin cladding surface temperatures were then based on these data.

The pin-to-pin power ratios calculated by the three methods described above are shown in Table 10-1.

The ratios from the gamma scan and Xe-136 data are almost identical for pins in similar positions on either side of a plane of symmetry placed through the center of the test element (an extension of a radius of the GETR core through the 6 o'clock and 12 o'clock positions of the test element). This symmetry is not apparent in the IB-80 data; it is felt that the smaller amount of water moderation and the absence of the stainless steel shutters resulted in the flux asymmetry reflected in the data.

The data presented in Table 10-2 show typical comparisons between the measured and predicted cladding surface temperature values. The cycle 18 data are based on IB-80 power distributions. The indicated agreement between the measured and predicted temperatures shown in the table and in Figure 10-3 is considered quite satisfactory. The data for cycles 20 through 24 is compared with the predictions based on gamma scan power data. The agreement between the measured and predicted temperatures is generally within 100°F and is considered quite satisfactory in view of the complications involved in the prediction calculations (these data are presented graphically in Fig. 10-4).

FIGURE 10-1. GETR LOOP AXIAL POWER DISTRIBUTION FOR VARIOUS RCD BANK POSITIONS

FIGURE 10-2. TYPICAL IB-8T-1 GAMMA SCAN AXIAL DISTRIBUTION

FIGURE 10-3. PREDICTED AND MEASURED IB-8T-1 CLADDING TEMPERATURES
(Cycle 18)

FIGURE 10-4. PREDICTED AND MEASURED IB-8T-1 CLADDING TEMPERATURES
(Average cycles 20 through 24)

TABLE 10-1 - PIN-TO-PIN POWER RATIOS

Pin No.	Facility Tube 3			Facility Tube 2		
	IB-8T-1			IB-8T-2		
	IB-8Ø Assembly 2 ^(a)	Gamma Scan	Xe-136 Data	IB-8Ø Assembly 4 ^(b)	Gamma Scan	Xe-136 Data
1	1.36 ^(c)	1.21	1.44 ^(d)	1.63	1.21	1.30
2	1.23	1.10	1.41	1.49	1.19	1.27
3	1.02	1.01	1.03	1.34	1.07	1.09
4	0.77	0.90	0.79	1.10	1.08	0.98
5	0.64	0.81	0.68	0.86	0.83	0.78
6	0.63	0.79	0.77	0.73	0.72	0.72
7	0.71	0.72	0.67	0.65	0.72	0.66
8	0.84	0.79	0.70	0.68	0.84	0.72
9	1.08	0.86	0.77	0.86	0.82	0.96
10	1.25	1.04	0.95	1.11	1.08	1.03
11	1.48	1.15	1.20	1.40	1.19 ^(d)	1.14
12	1.48	1.19	1.44 ^(d)	1.51	1.21 ^(d)	1.33
13	1.10	1.19	1.06	0.98	1.09	1.01
14	0.83	0.98	1.06	0.81	1.01	1.11
15	0.66	0.99	0.93 ^(d)	0.62	0.88	0.94
16	0.78	0.92	0.93	0.58	0.82	0.81
17	0.98	1.11	1.17	0.70	1.07	1.04
18	1.16	1.21	1.00	0.93	1.18	1.11
19	0	0	0	0	0	0

(a) 12 outer ring pins are 60 wt% UO₂ - 40 wt% BeO

(b) 12 outer ring pins are 70 wt% UO₂ - 30 wt% BeO

(c) Average pin power = 1.00

(d) Assumed value, data not available

It is concluded that the ML-1-I design will, in view of the experimental performance of the IB-8T-1 test element, operate satisfactorily at full power without exceeding the cladding hot spot temperature limit (1750°F). In addition, the good agreement between the measured and predicted temperatures permits assigning a significant level of confidence to the calculational techniques employed in the design.

TABLE 10-2 - COMPARISON OF MEASURED AND PREDICTED IB-8T-1 CLADDING SURFACE TEMPERATURES

<u>Cycle No.</u>		18 ^(a)	--	20 ^(c)	21	22	23	24	--
<u>Pin No.</u>	<u>X/L</u>	<u>Measured Temp, °F</u>	<u>Predicted Temp, °F^(b)</u>	<u>Measured Temperature, °F</u>					<u>Predicted Temp, °F^(d)</u>
1	0.9	1400	1460	1500	1550	1610	1540	1600	1530
6	0.75	1060	1050	1315	1300	1300	1300	1270	1240
10	0.3	1025	1100	1165	1110	1150	1110	1100	1180
12	0.75	1500	1530	1475	1400	1430	1410	1390	1425
13	0.75	1400	1450	1560	1550	1520	1560	1500	1425

(a) Operating data: Power, 59.4 kw; flow rate, 1540 lb/hr; inlet gas temperature, 500°F; outlet gas temperature, 1000°F

52 (b) Prediction based on IB-80 data

(c) Operating data:

<u>Cycle</u>	<u>Power, kw</u>	<u>Flow, lb/hr</u>	<u>Inlet Gas Temp, °F</u>	<u>Outlet Gas Temp, °F</u>
20	58	1810	800	1200
21	58	1810	800	1200
22	49	1660	740	1160
23	54	1690	800	1200
24	55	1720	800	1190

(d) Prediction based on gamma scan data

10.2 Pressure Drop Data Analysis

As discussed earlier, the experimentally determined friction factor correlation was monitored throughout the test irradiation to detect changes in configuration which might reflect incipient failure or abnormal operation of the test element. In addition, the pressure drop was measured periodically to monitor the performance of the elements.

No significant change in the friction factor correlation was noted during the experiment and the variation in pressure drop in the test elements was negligible (see Table 10-3).

TABLE 10-3 - IB-8T PRESSURE LOSS DATA

Test Conditions: 800°F inlet gas temperature
1200°F outlet gas temperature
55 kw test element power

<u>IB-8T-1</u>			<u>IB-8T-2</u>		
<u>Date</u>	<u>Δ P, psi</u>	<u>Flow, lb/hr</u>	<u>Date</u>	<u>Δ P, psi</u>	<u>Flow, lb/hr</u>
4 Feb 1961	11.2	1653	12 Jan 1961	11.7	1770
18 June 1961	11.8	1775	4 Feb 1961	12.0	1688
20 June 1961	11.4	1765	12 Apr 1961	13.5	1770
14 Sept 1961	12.2	1770	13 Aug 1961	11.8	1770
16 Oct 1961	11.7	1770	10 Oct 1961	11.6	1765
23 Nov 1961	12.0	1783	11 Nov 1961	11.6	1760
5 Dec 1961	11.8	1775	15 Dec 1961	11.9	1690
11 Dec 1961	10.5	1657	30 Dec 1961	12.3	1765
18 Dec 1961	12.0	1770	4 July 1962	11.7	1770
30 Dec 1961	12.3	1775	16 Sept 1962	11.9	1762
			3 Oct 1962	11.4	1690

Ref. 10-1: HECTIC-II, An IBM /090 Fortran Computer Program for Heat Transfer Analysis of Gas or Liquid-Cooled Reactor Passages, IDO-28595, December 1962, Aerojet-General Nucleonics, San Ramon, California

11.0 HOT CELL EXAMINATION

11.1 IB-8T-1

a. Transfer to Hot Cell: After removal from the in-pile loop, the IB-8T-1 test element was allowed to cool in the GETR canal for about a month. The element was then loaded into an 8-ton horizontal cask originally designed for use in the BMI-16 experiment*. The cask was placed on wooden pallets and the assembly was loaded on a trailer truck for transport to the NRTS.

Despite the lack of special shock absorbing material and transportation of the element in the horizontal position, no evidence of damage was observed when the element was removed from the cask in the MTR canal.

b. Physical Examination: The IB-8T-1 element was transferred from the MTR canal to the Army Reactor Area (ARA) hot cells. A photograph of the element before disassembly is shown in Figure 11-1. Examination revealed that the element was structurally sound and that the cross-sectional areas of the coolant channel were unchanged. The exterior surface of the outer liner was slightly scaled near the high-temperature end but the dimensions had not altered.

The fuel pin cluster was removed by splitting the insulating sleeve into halves. A white powder was observed on the inner surface of the inner liner and on the outer surfaces of the fuel pins (Fig. 11-2). This deposit was too thin to measure and attempts to remove enough of the powder to permit analysis were unsuccessful. The appearance of the powder was similar to the molecular sieve material used as a desiccant in the GETR loop in the first five cycles of test operation. It was observed that the corrosion of the fuel pins and inner liner was greatest in the areas of highest temperature; i.e., the lower portions of the liner and pins were more heavily corroded than the upper portions, and the pins on the GETR side of the pin bundle were more heavily corroded than those on the side away from the reactor.

A spiral twisting of the fuel pins was apparent (Fig. 11-3). This distortion was so pronounced at a point 10 in. from the bottom of the assembly that the spacing between pins 1 and 12 had been reduced to less than the diameter of the spacer wire. However, there was no evidence that the fuel pins had touched nor that any over-temperature had occurred as a result of the reduced spacing.

*An in-pile test of a GCRE-I plate-type test element in the ETR.

FIGURE 11-1. EXTERIOR OF IB-8T-1 TEST ELEMENT (INLET END AT TOP)

FIGURE 11-2. IB-8T-1 TEST ELEMENT SHOWING HALVES OF TUBE BUNDLE
INNER LINER (INLET END IS AT TOP)

11.2-64-1371

FIGURE 11-3. IB-8T-1 FUEL PIN BUNDLE SHOWING SPIRAL BENDING OF FUEL PINS

When the fuel pins were removed from the cluster, pin 12 cracked about 4 in. from the bottom (Fig. 11-4). When the spacer wire was removed from this pin, the cladding fractured through the initial crack. The two pieces of the pin are shown in Figure 11-5 with the spacer wire still attached to the upstream portion. The spiral bending of the pin is apparent in this photograph; subsequent observation revealed that this bending occurred on a pitch of 6.5 in. which corresponds to the pitch of the spacer wires.

Each pin was placed in a straightening jig in preparation for a gamma scan operation. Pin 1 broke during this straightening process.

c. Fuel Pin Diameter Measurements: The diameter of the fueled pins was measured at seven locations along the length of each pin after the pins had been removed from the tube bundle. The measurements were made using a knife-edged caliper which was read through the hot cell periscope. Table 11-1 summarizes the results of the measurements; the values shown represent the difference between the average of three post-irradiation measurements at each location and pre-irradiation measurements at the same locations. As indicated by the data in the table, it is apparent that the fuel pin cladding collapsed over the fuel pellets during irradiation.

d. Gamma Scan: Each fuel pin was gamma scanned over its full length to establish the power generation profiles. Typical gamma scan profiles are shown in Figure 11-6. The induced activity in the fuel pin plugs is apparent at either end of the scan as is the unfueled void at the top of the pin. The higher operating level of pin 1 (which was closer to the GETR core) is apparent; the distortion of the power profile by the burnable poison can also be observed.

e. Fission Gas Analysis: Sixteen of the 19 fuel pins in the IB-8T-1 test element were punctured and the internal gas removed and analyzed (pins 1 and 12, broken during handling, and pin 19, unfueled, were not punctured). The analysis of the extracted gas revealed that four of the pins contained large quantities of nitrogen, two contained air and 11 contained fission product gases and helium (with which the pins were originally filled). The results of the gas analysis are shown in Table 11-2.

11.2 IB-8T-2

a. Transfer to Hot Cell: After discharge from the GETR loop and cooling in the canal at that facility, the IB-8T-2 test element was loaded into a 4-ton G.E. Series 200 cask. The cask was secured on a steel pallet so that the fuel element was in a vertical position. The pallet was transported to NRTS on a truck without any special equipment for shock absorption. During handling in the MTR canal at NRTS, the test element was accidentally dropped through 50 in. of water into the shipping cask. The element was subsequently transferred to the ARA hot cell for examination.

b. Physical Examination: The most obvious characteristics noted in the examination of the element were the results of the handling accident in the MTR canal: The outer liner was buckled by the impact; the fuel pins were undamaged but the upper spider had deflected downward; the inner row of pins moved downward 0.065 in. and the outer row moved downward 0.050 in.

FIGURE 11-4. CRACK IN IB-8T-1 PIN 12

FIGURE 11-5. IB-8T-1 PIN 12 AFTER COMPLETE FRACTURE

FIGURE 11-6. TYPICAL IB-8T-1 FUEL PIN GAMMA SCAN TRACES

TABLE 11-1 - DEVIATION OF IB-8T-1 FUEL PIN
DIAMETERS FROM PRE-IRRADIATION VALUES

<u>Pin No.</u>	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>	<u>6</u>	<u>7</u>	<u>8</u>	<u>9</u>	<u>10</u>	<u>11</u>	<u>12</u>	<u>13</u>	<u>14</u>	<u>15</u>	<u>16</u>	<u>17</u>	<u>18</u>
<u>Measurement Location⁽¹⁾</u>																		
4 ⁽²⁾	+1 ⁽³⁾	-1	-1	0	0	+1	+2	0	0	0	0	0	0	+1	+2	-1	0	+1
8	+1	-1	-1	-1	-1	-1	0	-1	0	0	-1	0	0	0	0	-1	-1	0
12	+2	-1	-1	-1	0	-1	0	-1	0	0	-1	0	0	0	-1	-1	-1	0
16	0	-1	0	0	0	-1	0	-2	-1	-2	0	-2	-1	-1	-1	-1	-2	0
20	+1	0	-1	-1	-1	0	-1	-1	-1	-1	-3	-2	0	-1	-1	0	0	+3
24	-1	-5	-2	-1	-1	-1	-1	-1	0	-2	-3	-1	0	-2	-2	-2	-2	+1
28	0	-1	-1	-1	0	0	0	-1	+1	0	0	-1	0	0	0	-1	0	+1

(1) Measurement locations are specified in inches from the top of the fuel pin.

(2) Unfueled regions

(3) Deviations in inches $\times 10^{-3}$; accuracy of caliper was 0.0015 in.

TABLE 11-2 - IB-8T-1 FISSION GAS ANALYSIS

Pin No.	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19
Volume, (cm ³ at stp)	(1)	4.15	4.02	3.75	3.87	1.64	3.71	2.98	3.96	2.60	4.14	(2)	1.41	1.47	(3)	1.92	1.95	1.42	(4)
Element (mol %)																			
H ₂		---	---	---	---	---	---	---	---	---	---		---	---		---	---	---	
He		95.1	98.8	>98	99	1.7	99	97.7	98.3	96.3	98		---	90		94.4	89.4	---	
N ₂		0.05	0.02	---	---	95.6	---	0.20	0.03	0.07	---		90	---		0.1	0.09	82	
O ₂		0.02	0.004	---	---	2.05	---	0.005	0.01	0.02	---		---	---		0.05	0.01	---	
Ar		---	---	---	---	0.40	---	0.02	---	0.05	---		1.1	---		0.11	0.13	---	
Kr		0.66	0.17	0.23	0.215	0.03	0.11	0.03	0.23	0.50	0.25		1.04	1.47		0.76	1.51	1.2	
Xe		4.18	1.05	1.38	1.21	0.20	0.58	0.20	1.41	3.06	1.49		8.0	8.5		4.33	8.83	2.2	
Isotope (atom %)																			
Kr-83		14.1	14.1	14.2	14.1	13.1	15	14.1	14.2	14.1	14.0		14.1	14.3		14.1	14.2	14.3	
Kr-84		27.1	27.2	27.0	26.7	29.6	27	27.2	27.1	27.0	27.3		26.8	26.9		26.9	26.9	27.0	
Kr-85		7.3	7.4	7.3	7.8	7.6	8	7.4	7.3	7.4	7.3		7.3	7.5		7.5	7.0	6.9	
Kr-86		51.5	51.4	57.5	57.5	49.7	50	51.2	51.4	51.5	51.4		51.8	51.3		51.5	51.9	51.8	
Xe-131		11.7	11.7	12.3	12.4	12.3	12.2	12.2	12.2	12.0	11.5		12.9	12.4		12.8	12.7	12.7	
Xe-132		17.1	18.3	18.5	18.6	18.3	18.7	18.6	18.6	18.2	18.1		19.5	19.5		19.7	19.8	19.6	
Xe-134		32.5	32.5	32.7	33.0	33.0	33.3	33.2	32.9	32.5	32.3		34.4	34.8		34.7	34.7	34.6	
Xe-136		38.7	37.5	36.5	35.9	36.4	35.8	36.0	36.4	37.2	38.1		33.2	33.2		32.8	32.8	33.0	

(1) Pin broke during handling prior to puncture.

(2) Pin broke during handling prior to puncture.

(3) Extraction equipment leaked.

(4) Unfueled pin - not punctured.

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Examination of the exterior of the IB-8T-2 test element indicated the presence of a heavier scale than had been observed on the IB-8T-1 but no damage was detected other than the buckling of the outer liner. When the outer liner was split and removed, it was observed that the Thermoflex insulation was intact and apparently functioning as designed. The insulation stripped away from the inner liner when the outer liner was removed (Fig. 11-7). It was observed that the fuel pin tips were all in place in the lower spider and moved freely in the spider. The expansion joint was not deformed and moved freely without evidence of binding.

Large, dark brown particles were observed in the cooling passages in the upper portions of the fuel pin bundle when the inner liner was removed from the fuel pin bundle. (Typical particles are shown in Fig. 11-8.) These particles were subsequently identified as Fiberfrax insulation from the filter in the in-pile loop. The pin bundle was uniformly coated with a dark brown deposit; no evidence of the patchy white surface deposit observed in the IB-8T-1 could be found. The brown deposit was heaviest on the heat generating surfaces of the fuel pins and a clear line of demarcation was evident at the top of the fuel pin stack. The spiral bending of the pins observed was similar to , but more severe than that seen in the IB-8T-1. In addition, it was apparent that the entire tube bundle had expanded and surfaces of the outer tubes had come in contact with the inner liner (Fig. 11-9). Figure 11-10 shows the general bowing as well as the spiral deformation of pin 1.

c. Fuel Pin Diameter Measurements: The diameter of the fuel pins was measured at seven locations along the length of each pin after removal of the pins from the tube bundle. The measurements were made using a knife-edged caliper which was read through the hot cell periscope. Table 11-3 summarizes the results of the measurements; the values shown represent the difference between the average of three post-irradiation measurements at each location and pre-irradiation measurements at the same locations. As indicated in the table, it is obvious that the fuel pin cladding collapsed over the fuel pellets during irradiation.

d. Gamma Scanning: Each fuel pin was straightened in a jig and gamma scanned to determine the axial power profile. Typical gamma scan traces are shown in Figure 11-11. The highly activated end plugs in the fuel pins are apparent at the end of the trace. The relative difference in the operating power between pin 1 (nearest the GETR) and pin 6 (farthest from the GETR) is quite apparent.

e. Fission Gas Analysis: Eighteen of the 19 fuel pins in the IB-8T-2 test element were punctured and the internal gases extracted and analyzed (the unfueled center pin was not punctured). The results of the analysis of the extracted gas are shown in Table 11-4. The puncturing and sampling equipment was operated in an argon atmosphere to permit detection of in-leakage during the extraction procedure.

FIGURE 11-7. IB-8T-2 AFTER REMOVAL OF OUTER LINER

FIGURE 11-8. FOREIGN MATERIAL IN IB-8T-2 COOLING PASSAGES
(View taken through split and partially separated inner liner
exposing portion of tube bundle)

FIGURE 11-9. IB-8T-2 FUEL PIN BUNDLE AND INNER LINER HALVES
SHOWING POINTS OF CONTACT BETWEEN FUEL PINS
AND LINER

FIGURE 11-10. IB-8T-2 FUEL PIN 1 SHOWING GENERALIZED BOWING AND SPIRAL BENDING

11.2-64-1379

FIGURE 11-11. TYPICAL GAMMA SCAN TRACES FROM 1B-8T-2 FUEL PINS

TABLE 11-3 - DEVIATION OF IB-8T-2 FUEL PIN

DIAMETERS FROM PRE-IRRADIATION VALUES

Pin No.	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>	<u>6</u>	<u>7</u>	<u>8</u>	<u>9</u>	<u>10</u>	<u>11</u>	<u>12</u>	<u>13</u>	<u>14</u>	<u>15</u>	<u>16</u>	<u>17</u>	<u>18</u>	<u>19</u>
Measurement Location ⁽¹⁾																			
4 ⁽²⁾	0 ⁽³⁾	0	0	-2	-1	-1	0	0	+1	-1	-1	-1	-1	-1	0	0	0	0	-1
8	0	0	0	-1	-1	-1	0	0	+1	0	-1	-1	0	-1	+1	-1	-1	0	-1
12	0	0	0	-1	-1	-1	0	0	0	-1	-2	-1	0	0	0	-1	-1	-1	-1
16	0	0	-1	-2	-1	-1	0	0	0	-1	-1	-1	-1	-1	0	-1	-1	-1	-1
20	-1	0	-1	-1	-1	-1	-1	0	0	-1	-1	-1	-1	-1	-1	(4)	(4)	(4)	-1
24	-1	0	-1	-2	-1	0	0	0	0	0	-2	+1	-1	-1	0	-1	0	-1	-1
28	-1	-1	0	-1	-1	-1	0	-1	0	-1	-3	0	0	0	-1	-1	-2	0	-1

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(1) Measurement locations are specified in inches from the top of the fuel pin.

(2) Unfueled region

(3) Deviations in inches $\times 10^{-3}$; accuracy of caliper was 0.0015 in.

(4) Data missing

TABLE 11-4 - IB-8T-2 FISSION GAS ANALYSIS

Pin No.	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19
Volume, (cm ³ at stp)	3.43	3.26	0.78	2.69	2.74	2.31	2.82	2.84	2.60	1.20	3.14	2.14	3.70	3.66	2.5	2.78	2.4	4.45	(1)
Element (mol %)																			
H ₂	0.03	0.03	0.02	0.03	0.03	0.03	0.02	0.02	0.02	0.02	0.03	0.41	---	0.06	0.04	0.01	0.03	0.05	
He	89.1	90.6	87.0	95.2	97.1	96.2	98.0	96.3	95.1	81.2	93.1	88.0	44.8	71.7	91.6	94.4	85.5	56.3	
N ₂	0.05	0.05	5.3	0.06	0.05	0.09	0.04	0.07	0.13	0.06	0.69	0.42	0.25	3.4	0.75	0.06	0.10	0.23	
O ₂	0.01	0.01	1.2	0.01	0.01	0.02	0.01	0.01	0.02	0.01	0.17	0.04	0.08	0.54	0.02	0.02	0.04	0.03	
Ar	0.01	0.02	0.32	0.15	0.04	0.14	0.04	0.10	0.06	0.95	0.05	0.11	0.11	0.13	0.10	0.15	0.18	0.04	
Kr	1.5	1.33	0.89	0.67	0.41	0.52	0.29	0.52	0.65	2.51	0.86	1.5	8.7	3.75	1.14	0.83	2.15	6.7	
Xe	9.3	7.9	5.2	3.9	2.4	3.0	1.6	3.0	4.05	15.3	5.1	9.2	46.0	20.3	0.3	4.6	12.0	36.5	
Isotope (atom %)																			
Kr-83	14.0	13.9	14.0	14.2	14.2	14.2	14.3	14.2	14.1	13.9	14.0	14.0	14.3	14.3	14.3	14.4	14.2	14.3	
Kr-84	27.1	27.1	27.2	27.0	27.0	26.9	26.8	27.0	26.8	27.3	26.9	27.1	27.1	27.0	27.1	27.0	26.9	27.0	
Kr-85	7.2	7.2	7.3	7.3	7.4	7.4	7.7	7.4	7.5	7.0	7.6	7.1	7.0	7.1	7.0	7.1	7.0	6.9	
Kr-86	51.7	51.8	51.5	51.5	51.4	51.5	51.2	51.4	51.5	51.9	51.4	51.8	51.5	51.6	51.6	51.5	51.9	51.8	
Xe-131	11.4	11.5	11.6	11.8	12.0	12.2	12.2	12.2	11.6	11.7	11.6	11.4	12.9	12.7	12.9	12.8	12.7	12.7	
Xe-132	18.1	18.2	18.4	11.3	18.5	18.6	18.9	18.6	18.3	18.2	18.2	17.8	19.7	19.7	19.7	18.9	19.8	19.6	
Xe-134	32.5	32.4	32.7	33.0	33.5	33.5	33.5	35.5	33.3	33.0	32.7	32.6	34.7	34.6	34.9	35.2	34.7	34.6	
Xe-136	38.0	37.9	37.3	36.9	35.0	35.7	35.4	35.7	36.8	37.1	37.5	38.1	32.7	33.0	32.5	32.1	32.8	33.0	

(1) Unfueled pin; not punctured.

12.0 ANALYSIS OF HOT CELL EXAMINATION DATA

12.1 Gamma Scan Data

A review of the gamma scan data from the pins from IB-8T test elements reveals that the power skew was significantly larger than that anticipated in the ML-1. The power generated in individual pins varied from 60 to 150% of the average pin power; the anticipated range of similar values in the ML-1 is $\pm 5\%$ for typical elements and $\pm 35\%$ for the peripheral elements.

The application of the gamma scan data in an alternative method of calculating fuel pin surface temperature is described in Section 10.1.

12.2 Fission Gas Analysis

Review of the analysis of the gaseous fission products extracted from the fuel pins of the IB-8T test elements indicated that the fission gas release was, as observed in other ML-1 type elements, significantly lower than the design values. In the IB-8T-1, the gaseous fission product release constituted less than 1% of the calculated total fission product generation. The ratio of xenon to krypton was in excellent agreement with the fission yield data. In the IB-8T-2, the BeO-UO₂ fuel released less than 1% of the calculated total gaseous fission products. In the UO₂ fuel, the release varied from approximately 0.5% on the cold side of the element to approximately 5.0% on the hot side of the element. The higher release is considered to have been associated with the longer exposure time and the significant amount of cracking observed in the UO₂ fuel pellets. The maximum release observed was significantly lower than the ML-1 design value of 30%.

The measured values of fission gas release for the IB-8T test elements and the ratio of pin power to the average pin power are presented in Table 12-1.

TABLE 12-1 - IB-8T FUEL GASEOUS FISSION PRODUCT RELEASE

<u>Fuel Composition</u>	<u>Pin No.</u>	<u>P_P/P_A *</u>	<u>Xe & Kr, Percent Release</u>	
			<u>IB-8T-1</u>	<u>IB-8T-2</u>
60 wt% UO ₂ -BeO	1	1.50	----	0.75
	2	1.36	0.73	0.67
	3	1.23	0.20	0.12
	4	1.02	0.29	0.37
	5	0.76	0.36	0.31
	6	0.62	----	0.40
	7	0.64	0.20	0.25
	8	0.71	0.45	0.43
	9	0.82	0.40	0.45
	10	1.05	0.44	0.62
	11	1.25	0.30	0.46
	12	1.53	----	0.46
UO ₂	13	1.10	0.58	5.61
	14	0.83	0.88	3.24
	15	0.66	----	0.87
	16	0.78	0.62	0.59
	17	0.98	1.02	1.06
	18	1.16	0.15	5.05

*Pin Power/Average Pin Power

12.3 Chemical Analysis

Chemical analysis of the brownish deposit observed in the coolant passages of the IB-8T-2 test element indicated that the material consisted of iron, chromium, nickel, cobalt and a ceramic whose composition was identical with the Fiberfrax used in the filters. The corrosion of stainless steel components in the loop was the source of the metallic elements; the ceramic material obviously came from the filters. Traces of antimony were also found; this material was used in the seals in the loop compressors.

12.4 Burnup Analysis

Samples of fuel from pins in the IB-8T elements were dissolved and analyzed for U-235 burnup. Mass spectrometric and radiocesium determinations were both made. The maximum measured burnup of U-235 atoms occurred in pin 12 of the IB-8T-2 test element. The measured burnup was 9.5% of the U-235 atoms, which corresponds to 5.7×10^{20} fissions/cm³ of the 60 wt% UO₂-40 wt% BeO fuel. The maximum burnup measured in the UO₂ fuel was 2.9% of the U-235 atoms. The burnup analysis data are summarized in Table 12-2.

TABLE 12-2 - IB-8T BURNUP ANALYSIS

<u>Pin No.</u>	<u>Sample Location*</u>	<u>IB-8T-1</u>		<u>IB-8T-2</u>	
		<u>Mass Spec.</u>	<u>Radiocesium</u>	<u>Mass Spec.</u>	<u>Radiocesium</u>
1	13.5	5.78	7.01	----	----
1	21.1	----	----	8.3	8.5
6	4.9	2.69	2.38	----	----
6	17.8	3.35	4.38	----	----
10	10.1	4.73	2.88	7.4	7.8
12	21.1	5.39	3.83	9.5	9.8
13	13.5	1.55	1.46	----	----
13	21.1	1.61	1.70	2.93	2.8
17	10.1	1.78	1.61	0.46	2.6

*Distance from top of fuel stack

12.5 Fuel Pin Cladding Embrittlement

The fracture of two of the IB-8T-1 fuel pins during handling in the hot cell indicated the possibility of metallurgical changes in the cladding during irradiation. However, the absence of any such occurrence during handling of the IB-8T-2 strongly suggested that some phenomena unique to the IB-8T-1 test element were responsible for the fracture. This subject is discussed fully in Section V of this report (Metallurgical Evaluation).

12.6 Pin Bending

Thermal analysis of the fuel pin bundle configuration revealed that the spiral bending of the fuel pins observed in the IB-8T test elements was the result of unequal expansions between the fuel pin and the spacer wire. The analysis indicated that a significant temperature difference (approximately 300°F) existed between these two components during normal operation. Under these conditions, the fuel pin would "grow" more than the spacer wire and be deformed in the pattern observed by the constraint imposed by the spacer wire.

V. METALLURGICAL EVALUATION

13.0 IB-8T-1

A comprehensive metallurgical evaluation was conducted to evaluate the condition of the IB-8T-1 test element after irradiation in the GETR loop. The original goals of this evaluation program were to determine the physical properties of the cladding and fuel, and to compare the results of these determinations with similar data derived from laboratory control tests. However, the severe embrittlement of the cladding material resulted in a significant expansion of the program and the conduct and evaluation of many more comprehensive examinations than had originally been intended.

Specimens of the cladding of selected fuel pins were obtained from a wide variety of locations (Fig. 13-1). In addition to the normal metallographic examination of these specimens, and the tensile testing and microhardness determinations to evaluate the physical properties of the cladding, extensive chemical analysis, electron microprobe analysis and X-ray diffraction analyses were performed to determine the cause of the observed embrittlement. This program is discussed in detail in Section 13.1 below.

Metallographic examination of specimens of fuel from the IB-8T-1 test element was conducted to evaluate the mechanical performance of the fuel and to compare the appearance of the material with that exposed in static capsules earlier in the program. This work is discussed in Section 13.2 below.

13.1 Cladding Evaluation

a. Chemical Analysis: Chemical analyses were performed on specimens of the IB-8T-1 fuel pin cladding to determine the amount of nitrogen, oxygen and carbon, and to establish whether longitudinal migration of the alloying constituents had occurred in the Hastelloy X. The nitrogen analysis was performed using the micro-Kjeldahl technique. This procedure requires that the metal sample be placed in solution, normally by dissolution in aqua regia ($\text{HNO}_3\text{-HCl}$). In this case, this reagent could not be used because of its nitrogen content and other possible acid combinations (HCl , $\text{HCl-H}_2\text{O}_2$, HCl-HF , H_2SO_4 , and $\text{H}_2\text{SO}_4\text{-H}_3\text{PO}_4$) were found to be unsatisfactory. Incomplete dissolution was obtained by employing anodic techniques with H_2SO_4 as an electrolyte. Subsequent analysis of the data indicated that the molybdenum-rich constituent comprised the main part of the undissolved material. Following dissolution of the 0.5 to 1.0-gram sample, standard Kjeldahl procedures were used to obtain the nitrogen content.

KEY - NUMBERS DENOTE DISTANCE FROM BOTTOM OF PIN IN INCHES

TENSILE TEST SAMPLES

CHEMISTRY ANALYSIS SAMPLES

INDICATES CUTS

V DENOTES FACE EXAMINED METALLOGRAPHICALLY

FIGURE 13-1. DIAGRAM OF IB-8T-1 CLADDING

Vacuum fusion analyses were performed on the fuel pin cladding to verify the nitrogen analysis obtained by the Kjehldahl procedure and to obtain data on the oxygen content. Standard laboratory vacuum fusion techniques were employed to analyze 0.25 to 0.30-gram samples.

Carbon analyses were determined by a combustion method using a Leco apparatus.

The results of the analysis performed on the IB-8T cladding specimens and on unirradiated control specimens are presented in Table 13-1.

TABLE 13-1 - CHEMICAL ANALYSIS OF HASTELLOY X TUBING

Sample Identification	Nitrogen, wt%		Oxygen, wt%	Carbon, wt%
	Kjehldahl	Vacuum Fusion		
IB-8T-1, Pin 7	0.036	----	----	0.11
	0.062	----	----	0.10
IB-8T-1, Pin 18	0.007	1.32	0.66	0.09
	0.002	0.009	0.12	0.08
			1.52	0.20
			1.63	0.33
IB-8T-1, Pin 1	0.13	----	----	0.07
	0.11	----	----	0.09
Hastelloy X Sheet (as received)	0.037	0.014	0.006	0.10
Hastelloy X Tubing (as received)	----	----	----	0.116
				0.110
				0.108
				0.093
Hastelloy X Tubing (exposed 5000 hr to ref. gas at 1750°F)	----	0.412	0.017	0.10
Hastelloy X Tubing (exposed 2500 hr to nitrogen at 1750°F)	0.38	----	----	0.29

As indicated in Table 13-1, the original Kjehldahl analysis of the cladding from pins 1, 7 and 18 yielded very small values of nitrogen content. These results were not consistent with the microscopic appearance of the material and, as a consequence, a control sample exposed to high purity nitrogen for 2500 hr at 1750°F was analyzed by the same techniques. This analysis yielded results significantly higher than those observed in the analysis of the cladding. In an effort to check the adequacy of the Kjehldahl technique, vacuum fusion analyses were run on samples from pin 18 and on laboratory control samples exposed to reference gas (nitrogen with 0.5% oxygen) for 5000 hr

at 1750°F. As indicated in the table, the nitrogen content determined by this technique was significantly higher than that established in the fuel pin cladding by the Kjehldahl technique and in reasonable agreement with that determined by the Kjehldahl method in the nitrogen-annealed control samples. It is felt that this situation is explained by the fact that Kjehldahl analysis will detect only combined nitrogen and will not indicate nitrogen held interstitially or in solid solution. It does not seem unlikely that the method of binding in the alloy changed as the result of internal structural rearrangement caused by the irradiation and that vacuum fusion was required to break down the resultant compounds.

Although the data in Table 13-1 indicate a significant increase in oxygen content between the as-received and both the laboratory-annealed and IB-8T-1 irradiated samples, the data is so scattered that the indicated values may not be significant. Based on out-of-pile experiments, the oxygen would not be expected to have any direct effect on the formation of precipitates in the alloy and consequently it seems likely that the data presented in the table reflect the surface oxidation of the samples.

It is considered that the carbon analysis presented in the table does not reflect any significant change between the as-received, laboratory-annealed, and irradiated cladding specimens.

Chemical analyses were also performed to determine whether longitudinal migration of the alloying constituents in the Hastelloy X occurred during irradiation. Samples were taken at the positions identified as C-1 through C-5 in Figure 13-1. The amount of the major alloying constituents (Cr, Mo, Ni, Fe) was determined by wet chemical techniques and the minor metallic constituents were analyzed by spectrochemical techniques at the Idaho Chemical Processing Plant at the National Reactor Test Station. An unirradiated tubing sample was also analyzed for control purposes.

The results of these analyses are shown in Table 13-2.

TABLE 13-2 - CHEMICAL ANALYSIS OF IB-8T-1 CLADDING SPECIMENS

Sample No.	Pin No.	Distance From Bottom, in.	Amount, wt%									
			Cr	Ni	Mo	Fe	Al	Co	Mg	Mn	Si	Zr
C-1	12	24.2-25.2	19.8	42.6	10.7	18.4	0.23	0.60	0.15	0.93	1.30	0.19
C-2	12	5.4- 6.9	21.5	44.4	6.4	19.0	0.83	0.08	0.14	0.91	0.79	0.13
C-3	12	2.0- 3.0	19.0	44.7	7.6	18.2	0.25	0.35	0.03	1.05	0.44	0.19
C-4	18	24.2-25.2	21.4	41.6	12.6	15.8	0.33	0.26	0.05	1.00	0.76	0.07
C-5	18	2.0- 3.0	22.1	44.2	7.3	16.8	0.77	0.56	0.89	0.96	1.90	0.09
C-6	Reference		--	--	--	--	0.16	<.05	0.05	1.00	0.78	0.23

The difficulties in dissolving the samples discussed earlier were also encountered in this work and consequently the data probably are not completely valid. However, it appears that generally more chromium and less molybdenum is present in the hottest sections (samples C-2 and C-5) than in cooler sections (C-1, C-3 and C-4) of the cladding.

An attempt was made to determine variations in the alloy content from the inner to the outer surfaces of the IB-8T cladding by selective dissolution and subsequent analysis of cladding samples. The sample was mounted on a tantalum anode and the inside surface was sealed with an epoxy potting compound. It was planned to remove successive layers 0.005- to 0.010- in. thick from the outside surface of the sample by anodic dissolution and, subsequently, to analyze the solutions spectrographically. However, no suitable reagent could be found which would completely dissolve the material, and consequently, this technique was discontinued.

b. Tensile Testing: A number of irradiated tubing specimens were prepared from samples obtained at the locations shown in Figure 13-1. Tensile tests of these specimens were performed at room temperature and at 1200° and 1400°F. Samples of tubing in the as-received condition and samples of tubing exposed for 5000 hr to reference gas at 1750°F were tested also. It was necessary to remove the fuel from the fuel pins prior to testing the irradiated cladding samples. It was found that vibratory techniques would not remove the fuel, consequently, chemical dissolution techniques were used. Boiling nitric acid removed the UO₂ fuel quite easily but did not remove the BeO-UO₂ fuel. A refluxing solution composed of 87 cc of nitric acid, 0.68 g of sodium fluoride, 1.044 g of aluminum nitrate, and 13 cc of water was used to remove the BeO-UO₂ fuel. Samples of irradiated and unirradiated tubing were tested in both the boiling nitric acid and refluxing solution and, based on visual examination and evaluation of weight change data, it was concluded that the cladding was not attacked by either of the solutions.

The tensile test techniques employed conformed to the ASTM Method No. E8-57T for the testing of small diameter tubing. This specification prohibits the use of reduced section test specimens. The unirradiated specimens were 6 in. long and contained a 1-1/2-in. long plug in each end to prevent tube collapse under the loads imposed by the grips. Conventional Templin grips were used for these tests. However, these techniques were not suitable for testing the irradiated specimens which were found to be so hard that the grips invariably slipped upon application of load. Special self-aligning test fixtures were welded to the irradiated tubing specimens to permit the application of the required test load and to compensate for specimen bending. Figure 13-2 shows the three steps in the preparation of the specimen. Care was exercised during the welding operation to avoid weakening the tubing by the introduction of a heat-affected zone; post-test metallography of the welded areas showed a significant microstructural change in only one specimen.

All tensile testing was performed in a remotely operated Rhiele testing machine with a total capacity of 30,000 lb. The cross head speeds were uniformly controlled at 0.005 in./min and cross head travel was measured with a deflectometer (rather than with a strain gauge or extensometer) because of the elevated temperatures. No gauge marks were made in the irradiated tubing and elongation was determined by measuring the distance between the welded holders before and after testing. Figure 13-3 shows a typical irradiated sample in the tensile machine and Figure 13-4 shows the wire-wound resistance heat furnace used to control the temperatures of the test.

Showing gripping fixtures, end plugs and test specimen before assembly

Showing test specimen assembled in gripping fixtures before welding

Showing completed welded assembly

FIGURE 13-2. IRRADIATED TUBING SPECIMENS FOR TENSILE TESTING

FIGURE 13-3. IB-8T-1 TENSILE TEST SPECIMEN IN TESTING GRIPS

FIGURE 13-4. ASSEMBLY FOR ELEVATED TEMPERATURE TENSILE TESTING OF IB-8T-1 TUBING

Table 13-3 presents the results of the tensile testing of IB-8T-1 tubing and of laboratory control samples. The embrittlement of the irradiated material is clearly indicated by the significant reduction in ultimate tensile strength. A portion of this reduction can be associated with the aging of the material at temperatures, as indicated by the strength of the several unirradiated samples. The elongation data is difficult to interpret; direct comparison of elongation values for the sheet and tubing specimens is not valid because similar microstructural changes result in more drastic ductility changes in tubing than in sheet because of geometric considerations. It is apparent, however, that exposure of the material at temperature results in a significant loss of ductility; it is not apparent that irradiation of the material results in any additional loss of ductility.

Figure 13-5 shows selected fracture surfaces of the IB-8T-1 tubing after tensile testing. The ragged appearance of the surfaces is typical of brittle intergranular failures.

After tensile testing, a comprehensive program of metallographic evaluation was conducted on the IB-8T-1 fuel cladding tensile specimens at Battelle Memorial Institute. The most significant aspect of the evaluation is the marked difference between the structure of the cladding from fuel pins found to contain nitrogen after irradiation (pins 1, 6, 12 and 18) and those which were found to contain primarily helium after irradiation (pins 2, 7 and 11).

Figures 13-6 through 13-11 show the structure observed in a longitudinal section taken through the fracture of the IB-8T-1 pin 1 cladding. The tensile specimen was taken from a point 5.1 to 10.3 in. from the fuel pin bottom. Figure 13-6 shows the typical intergranular-type fracture and the characteristic second phase (observed in all pins which contained nitrogen after irradiation) extending entirely through the cladding thickness. Figure 13-7 shows the structure of the material at the opposite end of the metallographic mount where the specimen was parted from the tubing. The presence of the second phase extending completely through the cladding is also apparent in this photomicrograph. In Figures 13-8 and 13-9 the fracture can be seen to have followed the grain boundaries of the second phase formation and separations can be seen in second phase sites near the fracture. Figure 13-10 shows the inside of the longitudinal section of the test specimen; the presence of the second phase is again obvious in this photomicrograph. Figure 13-11, the outside of the test specimen, shows the dark surface oxide film and characteristic second phase structure.

Figures 13-12 and 13-13 show the microstructure of a transverse section of the IB-8T-1 pin 12 cladding tensile specimen. This specimen was taken 8.5 to 13.2 in. from the pin bottom. In Figure 13-12 the massive gray second phase is seen to extend from the inside diameter very nearly to the outside diameter. As shown in Figure 13-13 the second phase has formed a continuous layer on the surface and extends intergranularly through the matrix.

Figures 13-14 through 13-17 show the structure in a longitudinal section of the tensile specimen taken from the cladding of IB-8T-1 pin 18 at 7.7 to 13.7 in. from the bottom. Figure 13-14 clearly shows the intergranular-type tensile fracture and, once again, the characteristic second phase extending completely through the cladding (the matrix near the inside diameter has been

TABLE 13-3 - TENSILE PROPERTIES OF IB-8T-1 CLADDING
AND CONTROL SPECIMENS OF HASTELLOY X

Specimen Designation	History	Material	Exposure Temp, °F	Tensile Test Temp, °F	Ultimate Tensile Strength, psi	0.2% Offset Yield Strength, psi	Percent Elongation
B-130-A	As-received	Tubing	----	70	124,000	69,000	39.6
B-130-B	As-received	Tubing	----	70	128,000	66,000	40.4
B-204-A	As-received	Tubing	----	70	119,000	71,000	19.3*
B-204-B	As-received	Tubing	----	70	119,000	70,000	18.3*
B-204-X	As-received	Tubing	----	70	123,000	71,000	35.5
A-221	As-received	Sheet	----	70	116,400	70,400	41
A-222	As-received	Sheet	----	70	116,500	64,000	38
A-223	As-received	Sheet	----	70	116,000	68,800	39
A-224	As-received	Sheet	----	70	116,000	66,400	42
A-225	As-received	Sheet	----	70	116,000	67,500	41
13A	5000 hr - Ref. Gas**	Tubing	1750	70	70,000	62,000	0.5
13B	5000 hr - Ref. Gas	Tubing	1750	70	90,000	53,000	8.0
27A	5000 hr - Ref. Gas	Tubing	1750	70	72,000	44,000	1.5*
27B	5000 hr - Ref. Gas	Tubing	1750	70	99,000	33,000	3.7*
17A	5000 hr - Ref. Gas	Tubing	1750	1400	49,100	37,500	5.8
34A	5000 hr - Ref. Gas	Tubing	1750	1400	49,300	40,600	18.6
A-200	100 hr - Ar + Oil Vapor	Sheet	1750	70	96,800	53,400	8.5
A-213	100 hr - Ar + Oil Vapor	Sheet	1750	70	91,600	50,800	3.5
IX-3	1000 hr - Nitrogen	Sheet	1750	70	66,000	29,200	11.0
IX-2	2500 hr - Nitrogen	Sheet	1750	70	56,700	48,700	1.0
NH-1	1000 hr - Ref. Gas	Sheet	1750	70	112,800	54,500	32
NH-2	1000 hr - Ref. Gas	Sheet	1750	70	115,900	54,200	27
IX-AA	2500 hr - Ref. Gas	Sheet	1750	70	92,400	---	19
IX-CC	5000 hr - Ref. Gas	Sheet	1750	70	110,500	51,600	22
7-A	Pin 7 - 3.6 to 10.1 in.***	Tubing	1240	70	101,000	64,800	9.2*
7-B	Pin 7 - 20.7 to 27.2 in.	Tubing	970	70	113,000	56,200	12.1
T-8	Pin 18 - 7.7 to 13.7 in.	Tubing	1330	1400	45,000	36,900	7.6*
T-1	Pin 1 - 5.1 to 10.3 in.	Tubing	1440	1400	37,800	34,400	0*
T-9	Pin 6 - 8.5 to 13.2 in.	Tubing	1210	1200	58,100	38,900	4.9
T-2	Pin 12 - 8.5 to 13.2 in.	Tubing	1390	1200	33,100	28,200	0*
2-A	Pin 2 - 3.2 to 9.7 in.	Tubing	1430	1400	39,700	36,900	4.3
2-B	Pin 2 - 21.4 to 27.9 in.	Tubing	1000	70	88,500	46,100	13.6
11-A	Pin 11 - 3.5 to 10.0 in.	Tubing	1430	1400	39,300	33,800	3.2
11-B	Pin 11 - 20.8 to 27.3 in.	Tubing	1000	70	93,400	48,600	11.8

* Broke at gage mark or weld zone

** Reference Gas is 99.5 vol% N₂ + 0.5 vol% O₂

***IB-8T-1 Distance from bottom

Pin 1, 5.2 - 10.2 in.
from Bottom

Pin 11, 20.8 - 27.3 in.
from Bottom

Pin 12, 8.6 - 13.2 in.
from Bottom

Pin 18, 7.7 - 13.7 in.
from Bottom

FIGURE 13-5. FRACTURE SURFACES TYPICAL OF IB-8T-1 CLADDING SAMPLES
AFTER TENSILE TESTING

FIGURE 13-6. LONGITUDINAL MICROSTRUCTURE OF IB-8T-1, PIN 1, TENSILE SPECIMEN AT FRACTURE

FIGURE 13-7. LONGITUDINAL MICROSTRUCTURE OF IB-8T-1, PIN 1, TENSILE SPECIMEN AT POINT REMOVED FROM FRACTURE

11.2-64-1391

FIGURE 13-8. FRACTURE SURFACE OF IB-8T-1, PIN 1, TENSILE SPECIMEN

FIGURE 13-9. FRACTURE SURFACE OF IB-8T-1, PIN 1, TENSILE SPECIMEN

INSIDE SURFACE

250X

FIGURE 13-10. INSIDE EDGE OF IB-8T-1, PIN 1, TENSILE SPECIMEN
5.1 to 10.3 in. from Bottom

OUTSIDE SURFACE

250X

FIGURE 13-11. OUTSIDE EDGE OF IB-8T-1, PIN 1 TENSILE SPECIMEN
5.1 to 10.3 in. from Bottom

11.2-64-1393

FIGURE 13-12. TRANSVERSE SECTION OF IB-8T-1, PIN 12, TENSILE SPECIMEN

FIGURE 13-13. INSIDE PORTION OF TRANSVERSE SECTION OF IB-8T-1, PIN 12, TENSILE SPECIMEN

FIGURE 13-14. LONGITUDINAL MICROSTRUCTURE OF IB-8T-1, PIN 18, TENSILE SPECIMEN AT FRACTURE

FIGURE 13-15. FRACTURE SURFACE OF IB-8T-1, PIN 18, TENSILE SPECIMEN

heavily attacked by the etchant). The appearance of the fracture at higher magnification is shown in Figure 13-15; the similarity between the appearance of the fracture with that shown in Figure 13-8 is striking. Figures 13-16 and 13-17 show the character of the second phase formed on the inside and near the center of the cladding specimen.

Figures 13-18 and 13-19 show the inside and outside surfaces of a transverse section from the IB-8T-1 pin 6 taken 1/2 in. above the tensile specimen fracture. (The tensile sample was taken 8.5 to 13.2 in. from the bottom.) The initiation of second phase formation is apparent on the inner surface of Figure 13-18; no second phase is observed in Figure 13-19. The very fine precipitate observed in these figures is carbide particles.

Figures 13-20 and 13-21 show the structures in a longitudinal section of the tensile specimen taken from IB-8T-1 pin 2 at 3.2 to 9.7 in. from the bottom. Pin 2 was located in the outer ring of fuel pins and was found after irradiation to contain principally helium. The structures shown resemble those observed in Hastelloy X tubing and sheet aged out-of-pile at temperatures in the range experienced by the sample during irradiation. A very fine precipitate had formed within the grain boundaries; the characteristic oxide scale is apparent on the outer surface. The intergranular separations shown in Figure 13-20 were the result of the tensile testing; the specimen would ultimately have failed at this point under test.

Figure 13-22 shows the transverse section of the tensile specimen taken from pin 2 at 21.4 to 27.9 in. from the bottom. This sample, taken from a relatively cool section of the fuel pin, exhibits a structure which closely resembles that of the as-received tubing, consisting of primarily M_6C -type carbides uniformly dispersed across the section. Figure 13-23 is a view of the same specimen at higher magnification.

Figure 13-24 shows the tensile fracture which occurred in the weld zone of the specimen from pin 7 taken 3.6 to 10.1 in. from the bottom. Figure 13-25 is another view of this specimen at high magnification. The formation of a second phase is not evident in either figure; the fine precipitate of $M_{23}C_6$ and M_6C -type particles closely resembles that in Hastelloy X aged out-of-pile under similar temperature conditions. Figure 13-26 shows the structure at the center of the tensile specimen taken from pin 7 at 20.7 to 27.2 in. from the bottom. This pin contained principally helium after irradiation and the structure closely resembles that of Hastelloy X aged out-of-pile.

A sample was taken from the cladding at the center of pin 11 (14.5 to 16 in. from the bottom) for metallographic examination and direct comparison with the cladding of other IB-8T fuel pins which contained nitrogen after irradiation (pin 11 contained principally helium after irradiation). Figure 13-27 shows the structure of the inside surface of the tensile specimen taken from pin 11 at 3.5 to 10.0 in. from the bottom. No evidence of second phase formation is visible. Figure 13-28 shows the relatively ductile nature of the fracture of the specimen taken from pin 11 at 20.8 to 27.3 in. from the bottom. Only primarily M_6C -type carbides can be observed in the structure of this sample (Figure 13-29). Figures 13-30 and 13-31 show the metallurgical structure of a sample of the cladding from pin 11 near the center; no evidence of second phase formation can be observed.

FIGURE 13-16. STRUCTURE OF INSIDE OF IB-8T-1, PIN 18, TENSILE SPECIMEN

FIGURE 13-17. CENTER STRUCTURE OF IB-8T-1, PIN 18, TENSILE SPECIMEN

250X

FIGURE 13-18. INSIDE EDGE OF IB-8T-1, PIN 6, TENSILE SPECIMEN

250X

FIGURE 13-19. OUTSIDE EDGE OF IB-8T-1, PIN 6, TENSILE SPECIMEN

64-1397-2 64-1397-2 64-1397-2

FIGURE 13-20. LONGITUDINAL SECTION THROUGH TENSILE SPECIMEN FROM IB-8T-1, PIN 2, 3.2 TO 9.7 in. FROM BOTTOM

FIGURE 13-21. OUTSIDE EDGE OF LONGITUDINAL SECTION THROUGH TENSILE SPECIMEN FROM IB-8T-1, PIN 2, 3.2 TO 9.7 in. FROM BOTTOM

FIGURE 13-22. TRANSVERSE SECTION OF IB-8T-1, PIN 2, TENSILE SPECIMEN, 21.4 TO 27.9 in. FROM BOTTOM

FIGURE 13-23. TRANSVERSE SECTION FROM INSIDE OF IB-8T-1, PIN 2, TENSILE SPECIMEN, 21.4 TO 27.9 in. FROM BOTTOM

FIGURE 13-24. IB-8T-1, PIN 7, TENSILE SPECIMEN 3.6 TO 10.1 in. FROM BOTTOM SHOWING FRACTURE THROUGH WELD ZONE

FIGURE 13-25. CENTER MICROSTRUCTURE OF IB-8T-1, PIN 7, TENSILE SPECIMEN 3.6 TO 10.1 in. FROM BOTTOM

11.2-64-14-00

500X

FIGURE 13-26. CENTER MICROSTRUCTURE OF IB-8T-1, PIN 7, TENSILE SPECIMEN 20.7 TO 27.2 in. FROM BOTTOM

500X

FIGURE 13-27. MICROSTRUCTURE OF INSIDE EDGE OF IB-8T-1, PIN 11, TENSILE SPECIMEN 3.5 TO 10.0 in. FROM BOTTOM

FIGURE 13-28. FRACTURE SURFACE OF IB-8T-1, PIN 11, TENSILE SPECIMEN 20.8 TO 27.3 in. FROM BOTTOM

FIGURE 13-29. INSIDE EDGE OF IB-8T-1, PIN 11, TENSILE SPECIMEN 20.8 TO 27.3 in. FROM BOTTOM

11.2-64-1402

FIGURE 13-30. INSIDE EDGE OF IB-8T-1, PIN 11, 14 1/2 TO 16 in. FROM BOTTOM

FIGURE 13-31. CENTER OF IB-8T-1, PIN 11, 14 1/2 TO 16 in. FROM BOTTOM

c. Electron Microprobe Analyses: Electron microprobe analyses were performed on a specimen of the cladding from the bottom portion of the IB-8T-1 pin 18 to establish the identity and relative quantities of the elements present in the matrix and in the second phase. Similar analyses were performed on a control specimen after exposure to nitrogen for 2500 hr at 1750°F.

The analysis equipment available was not shielded and, as a consequence, it was necessary to absolutely minimize the activity of the irradiated specimen. This was accomplished by using 1/8 by 0.030 in. segments of the cladding. The activity level measured from the specimen, after mounting in a Hysol metal-lurgical mount and further grinding to reduce the thickness, was 40 mr/hr at the surface. The unirradiated specimens were mounted and prepared in the same fashion.

The results of the analysis are given in Table 13-4.

TABLE 13-4 - ELECTRON PROBE ANALYSES OF HASTELLOY X
IB-8T-1 CLADDING AND CONTROL SAMPLES, wt%

	<u>Cr</u>	<u>Fe</u>	<u>Ni</u>	<u>Mo</u>
<u>Sheet As-Received</u>				
Matrix - Center	21.5	18.2	49.0	9.0
<u>Sheet - 2500 hr, 1750°F, Nitrogen</u>				
Matrix - 0.001 - 2 in. from edge	22	20	47	11
Matrix - Center	23	22	46	7
Second Phase - 0.001 - 2 in. from edge	19	18	37	23
Second Phase - Center	45	5	28	22
M ₆ C - Intragranular	33	13	33	18
M ₆ C - Located within second phase	24	5	23	45
<u>IB-8T-1 Tubing - Pin 18</u>				
Matrix - 0.005 in. from inside	9	27	59	4.2
Second Phase - 0.005 in. from inside	53	11	35	3
Matrix - 0.010 in. from inside	13	18	63	7
Second Phase - 0.010 in. from inside	48	13	34	5
Matrix - 0.020 in. from inside	10	24	57	9
Second Phase - 0.020 in. from inside	22	20	41	16.7
Matrix - 0.026 in. from inside	9	25	58	7
Second Phase - 0.026 in. from inside	24	20	36	21
Matrix - 0.031 in. from inside (outside)	14	23	56	7
Second Phase - 0.031 in. from inside (outside)	24	20	36	20

The accuracy of the results obtained on the irradiated samples was somewhat less than that obtained on the unirradiated samples because the background level in the equipment was increased by the specimen radiation levels. However, this difference is not considered significant in this study since the purpose was to determine relative rather than quantitative results.

The microprobe analysis of the center of the matrix of the as-received sheet agreed very well with the chemical analysis supplied by the vendor of the material and consequently these values were used as the reference for comparison with the other findings. A graphic presentation of the results of the microprobe analysis of the control sample is shown on Figure 13-32. The dotted lines are the reference values and the solid lines are the results of the analysis. It can be observed from this figure that the nickel concentration in the matrix was apparently less and the iron and chromium concentrations were somewhat higher than the reference values. The molybdenum content in the matrix was higher at the edges of the sample but lower in the center than the reference values. The composition of the second phase differs sharply from the reference analyses; the chromium content is significantly higher at the center and the nickel and iron contents are significantly lower at the center. The molybdenum content is relatively uniform but somewhat higher than the reference value. The microstructure of the specimen is shown in Figure 13-33; the characteristic M_6C precipitates and the massive second phase are evident in this photo.

The results of the microprobe analysis of the specimen from IB-8T-1 pin 18 are shown graphically in Figure 13-34. The concentrations of nickel and iron in the matrix are somewhat higher than the reference values but those of the chromium and molybdenum are somewhat lower. Analyses of the second phase material reveal sharp deviations from the reference values and from data developed in analysis of the control sample second phase. The chromium content at the fuel side of the sample is much higher than the reference values but decreases to approximately normal concentration at about the center of the tubing. The nickel content is less than the reference values across the entire thickness of the sample. The iron content is somewhat less than reference at the fuel side of the sample but increases to near the reference value at about the mid-point. The molybdenum content is lower than reference at the fuel side but increases significantly to somewhat higher than reference values at the opposite side of the sample. The microstructure of the sample is shown in Figure 13-35.

In general, it was concluded that both chromium and molybdenum had migrated from the matrix into the second phase, largely at the expense of nickel which, in turn, had migrated into the matrix.

d. X-Ray Diffraction Analyses: X-ray diffraction analyses were performed on samples of the IB-8T test element cladding and on control samples to identify the various precipitates observed in the alloy. The preparation and mounting of cladding specimens was similar to that described for the electron microprobe analysis work; the reduction in the activity of irradiated samples was also an important consideration for X-ray diffraction work. The identity of the precipitates was established after extraction of the precipitate bodies from the matrix by the use of an etchant ($10\text{ HCl}:2\text{HNO}_3$). The extracted particles were placed on a specially designed powder mount for the analysis. The results of the X-ray diffraction analyses performed on the control samples are summarized in Table 13-5, which tabulates the lattice parameters for all identified precipitates as well as for the austenitic base metal.

FIGURE 13-32. MICROPROBE ANALYSES VERSUS DISTANCE FROM EDGE FOR SHEET EXPOSED TO HIGH PURITY NITROGEN FOR 2500 hr AT 1750°F

Second Phase

 M_6C Within
Second PhaseIntra-granular
 M_6C

INCHES
.001
.002
.003
.004
.005
500X

FIGURE 13-33. MICROSTRUCTURE OF HASTELLOY X SHEET EXPOSED 2500 hr TO HIGH PURITY NITROGEN AT 1750° F ANALYZED BY ELECTRON MICROPROBE

500X

FIGURE 13-34. MICROPROBE ANALYSES VERSUS DISTANCE FROM FUEL FOR IB-8T-1, PIN 18, HASTELLOY X TUBING

FIGURE 13-35. MICROSTRUCTURE OF HASTELLOY X 1B-8T-1, PIN 18, CLADDING
SAMPLE ANALYZED BY ELECTRON MICROPROBE

11.2-64-1407

TABLE 13-5 - LATTICE PARAMETERS OF MATRIX AND PRECIPITATES IDENTIFIED
IN HASTELLOY X CONTROL SAMPLES AS DETERMINED BY X-RAY DIFFRACTION

Sample Designation	Material History	Material Form	Lattice Parameter, A_0				Second Phase
			Base Metal	M_6C	$M_{23}C_6$	Cr_7C_3	
4A	As-Received	Sheet	3.49	10.91			
1A	Annealed 1000 hr at 1750°F in ref. gas ^(a)	Sheet	3.58	10.83			
2A	Annealed 2500 hr at 1750°F in ref. gas	Sheet	3.59	10.83			
3A	Annealed 5000 hr at 1750°F in ref. gas	Sheet	3.58	10.93 ^(b)			
7A	Annealed 2500 hr at 1750°F in pure argon	Sheet	3.58	10.95			
11A	Annealed 5000 hr at 1750°F in ref. gas	Tube	3.60	10.95			
25A	Annealed 5000 hr at 1750°F in ref. gas	Tube	3.58	10.80			6.4
10A	Annealed 5000 hr at 1750°F in ref. gas	Tube	3.58	10.94 10.84			
6A	Annealed 2500 hr at 1750°F in high purity nitrogen	Sheet	3.58	10.83			6.4
16A	Annealed 100 hr at 1750°F in nitrogen plus oil vapor	Sheet	3.58	10.8			
15A	Annealed 100 hr at 1750°F in nitrogen plus oxygen plus oil vapor	Sheet	3.58	10.94			
22A	Annealed 100 hr at 1750°F in argon plus oil vapor	Sheet	3.58	10.95			(c)
18A	Annealed 204 hr at 1500°F in NH_3	Tube	3.56 ^(d) 3.58	10.9			6.4
20A	Annealed 218 hr at 1500°F in CH_4	Tube	3.58	10.92	10.6		

(a) Reference gas is 99.5 vol% N_2 + 0.5 vol% O_2

(b) Sample also contained some Fe_2Mo Laves-type phase

(c) Phase was present but lattice parameter was not measured

(d) Sample also contained CrN

As indicated in Table 13-5, Hastelloy X sheet exposed for periods up to 5000 hr in reference gas at 1750°F was found to contain primarily M_6C although some evidence of a Fe_2Mo -type Laves phase was observed at 5000 hr. Hastelloy X sheet samples exposed in argon were found to have the same phases present. Hastelloy X tubing annealed for 5000 hr at 1750°F in reference gas was also found to contain the M_6C precipitate although examination of one sample revealed a second phase with a pattern closely resembling that seen in irradiated IB-8T-1 cladding samples.

Sheet samples exposed to high purity nitrogen for 2500 hr at 1750°F and to nitrogen saturated with oil vapor for 100 hr at 1750°F produced an X-ray diffraction pattern very closely approximating that of the second phase, together with indications of M_6C and an austenite matrix. The presence of the massive second phase precipitate in Figure 13-36 and the evidence of the initiation of formation of the same precipitate shown in Figure 13-37 support the conclusion that this phase is produced in out-of-pile exposure of Hastelloy X to high purity nitrogen atmospheres. Evidence that the second phase is not produced by oil vapor alone is shown by the X-ray diffraction pattern of Hastelloy X sheet exposed to reference gas and oil vapor (sample 15A of Table 13-5) which did not reveal evidence of the second phase. This conclusion is also supported by the appearance of the sample (Fig. 13-38); only the austenitic base metal and an M_6C precipitate is apparent at the matrix although a Cr_7C_3 surface layer is evident. Further evidence that the second phase is not produced by a carburizing atmosphere is shown by the X-ray diffraction results on Hastelloy X sheet exposed to methane (sample 20A of Table 13-5).

A Hastelloy X sheet sample exposed to ammonia for 204 hr at 1500°F did contain the second phase beneath a surface layer of CrN. This structure is shown in Figure 13-39.

The results of X-ray diffraction analysis performed on samples of the cladding from IB-8T-1 pin 18 are presented in Table 13-6. The data in Table 13-6 indicate that, in addition to the austenitic matrix and the M_6C carbides, a second phase was present in the cladding of pin 18. In the center of the cladding, this phase has the same lattice parameter as the massive precipitate noted in the control specimens exposed to high purity nitrogen. At the interface between the cladding and the fuel, the second phase has a slightly different lattice parameter corresponding to the increase in chromium content noted in the microprobe analysis (see Section 13.2)

TABLE 13-6 - LATTICE PARAMETERS OF MATRIX AND PRECIPITATES
IDENTIFIED IN IB-8T-1 PIN 18 CLADDING

<u>Phases</u> <u>Observed</u>	<u>Lattice</u> <u>Parameter (Å)</u>	<u>Phases</u> <u>Observed</u>	<u>Lattice</u> <u>Parameter (Å)</u>
γ	3.57	γ	3.58
M_6C	10.8	M_6C	10.8
"Second"	6.40	"Second"	6.40
"Second"	6.35	"Second"

FIGURE 13-36. HASTELLOY X SHEET EXPOSED TO HIGH PURITY NITROGEN FOR 2500 hr AT 1750°F

FIGURE 13-37. HASTELLOY X SHEET EXPOSED TO NITROGEN AND OIL VAPOR FOR 100 hr AT 1750°F

FIGURE 13-38. HASTELLOY X SHEET EXPOSED TO REFERENCE GAS AND OIL VAPOR FOR 100 hr AT 1750°F

FIGURE 13-39. HASTELLOY X SHEET EXPOSED TO NH₄ FOR 204 hr AT 1500°F

An attempt was made to define the composition of the gray material formed at the fuel cladding interface of IB-8T-1 pin 18 but no pattern was obtained. This may indicate that the substance was amorphous.

e. Microhardness Measurements: Microhardness measurements were made on selected specimens of the IB-8T-1 cladding with a Wilson-Tukon Microhardness Tester installed at the MEATR hot cells at the NRTS. A diamond indenter with a 200 gram load was utilized in this testing. Five indentations at about 0.006 in. intervals across the thickness of the cladding were used to detect hardness variations.

The results of the microhardness measurements, presented in Table 13-7, indicate that the claddings were harder in the lower (hotter) portions of the pins than at the upper (cooler) portions. This gradient in hardness generally corresponds with the amount of second phase precipitate observed in various sections of the fuel pin cladding. The gray precipitate on the inside surface of the cladding proved to be very hard.

TABLE 13-7 - MICROHARDNESS OF IB-8T-1 FUEL PIN CLADDINGS, DPH

Pin No.	Distance From Bottom, in.	1 (inside)	2	3	4	5 (outside)	Average
12	7.7	353	349	353	411	234	340
12	13.2	541*	294	290	268	286	284
12	17.6	560*	298	283	306	298	296
12	24.2	302	279	249	243	261	261
1	4.4	392	411	398	374	302	375
1	17.6	771*	319	283	294	348	311
6	4.4	279	261	249	255	272	263
6	24.2	252	275	265	265	255	262
13	4.4	1229*	353	426	405	258	360
18	7.7	418	240	363	286	316	323
10	7.7	249	234	252	249	348	266
17	17.6	311	268	246	275	265	273

*Gray deposit on inside

Microhardness measurements were also made on some of the tubing sections that had been tensile tested at BMI (Table 13-8). These data indicate that the cladding of IB-8T-1 pins 12 and 18 was considerably harder than unirradiated tubing samples. The values for the irradiated samples agree well with those obtained at the MEATR hot cells.

TABLE 13-8 - MICROHARDNESS OF TENSILE SPECIMENS FROM
HASTELLOY X TUBING SAMPLES, KHN

<u>Specimen Designation</u>	<u>Identification</u>	<u>Microhardness (Range)</u>
B-130-A	As-Received	155
13-A	5000 hr, Ref. gas, 1750°F	144
T-8	Pin 18, 7.7 - 13.7 in. from bottom	285 - 1102
T-2	Pin 12, 8.5 - 13.2 in. from bottom	264 - 1407

b. Metallographic Evaluation:

1) Laboratory Control Metallography: Specimens of Hastelloy X sheet (Heat X-4762 containing 2.07% cobalt) and tubing (Heat E-9517 containing 0.07% cobalt) were exposed to several different environments in the laboratory to act as controls for the metallographic evaluation of the IB-8T test element. The specimens of interest are discussed below:

- 1) Samples of Hastelloy X sheet were exposed for 5000 and 10,000 hr to reference gas at 1750°F. Figure 13-40 shows the microstructure of the 5000-hr material and Figure 13-41 shows the structure of the material exposed 10,000 hr. In both cases the principal precipitate present is a non-continuous M₆C-type carbide which agglomerates with increasing time at temperature.
- 2) Samples of Hastelloy X tubing material in the form of 0.07% cobalt sheet were exposed for 5000 hr to reference gas at 1750°F. Figure 13-42 shows the microstructure of this material; the appearance is similar to that of sheet exposed under identical conditions.
- 3) Samples of Hastelloy X sheet were exposed to high purity nitrogen for 250 and 2500 hr. Figure 13-43 shows the structure of the 250-hr material; the second phase structure is apparent although not as completely developed as in the 2500-hr material (Fig. 13-44). In this latter case, the second phase has formed into a continuous massive precipitate along the grain boundaries of the matrix.
- 4) Samples of Hastelloy X tubing were exposed for 250 hr to high purity nitrogen at 1750°F. The formation of the second phase is comparable with that observed in the 250-hr sheet specimens (Fig. 13-45). From these observations it was concluded that the cobalt content of the Hastelloy X does not alter the susceptibility of the material to the formation of the second phase during exposure to high purity nitrogen.
- 5) Hastelloy X sheet was exposed to high purity nitrogen containing 0.1 wt% carbon monoxide for 2500 hr at 1750°F. The microstructure of this sample (Fig. 13-46) does not reveal any evidence of second phase formation but greater quantities of carbide have formed than was the case with exposures to the reference gas.

FIGURE 13-40. HASTELLOY X SHEET AFTER EXPOSURE TO REFERENCE GAS FOR 5000 hr AT 1750°F

FIGURE 13-41. HASTELLOY X SHEET AFTER EXPOSURE TO REFERENCE GAS FOR 10,000 hr AT 1750°F

FIGURE 13-42. HASTELLOY X TUBING MATERIAL AFTER EXPOSURE TO REFERENCE GAS FOR 5000 hr AT 1750°F

FIGURE 13-43. HASTELLOY X SHEET AFTER EXPOSURE TO HIGH PURITY NITROGEN FOR 250 hr AT 1750°F

FIGURE 13-44. HASTELLOY X SHEET AFTER EXPOSURE TO HIGH PURITY NITROGEN FOR 2500 hr

FIGURE 13-45. HASTELLOY X TUBING AFTER EXPOSURE TO HIGH PURITY NITROGEN FOR 250 hr

FIGURE 13-46. HASTELLOY X SHEET AFTER EXPOSURE TO 99.9 N₂ + 0.1 CO FOR 2500 hr AT 1750°F

FIGURE 13-47. HASTELLOY X SHEET AFTER EXPOSURE TO ARGON CONTAMINATED WITH OIL VAPOR FOR 100 hr AT 1750°F

6) Specimens of Hastelloy X sheet were exposed to argon contaminated with oil vapor for 100 hr at 1750°F. The internal microstructure of the material is typical of Hastelloy X aged under these conditions (Fig. 13-47). However, a thin layer of chromium carbide (as subsequently identified by X-ray diffraction analysis) appears on the surface.

7) Samples of Hastelloy X sheet were exposed to methane for about 200 hr at 1500°F. The microstructure of the material is typical of Hastelloy X aged under these conditions (Fig. 13-48). However, a carbide surface layer has formed and the carbide phase can be observed extending intergranularly into the matrix.

8) Samples of Hastelloy X sheet were exposed to ammonia for about 200 hr at 1500°F. A heavy surface zone has formed beneath which a zone of second phase, light colored particles can be observed (Fig. 13-49). This structure resembles that observed on the internal surface of some of the IB-8T fuel pins.

2) IB-8T Fuel Cladding Metallography: An extensive metallographic evaluation of samples of the IB-8T-1 fuel pin cladding was conducted. The main objectives of this program were to establish the cause and mechanism of the severe embrittlement detected in the cladding of some of the fuel pins and to observe the changes in the cladding material (other than the severe embrittlement) associated with the exposure.

The fractures in the cladding of fuel pins 1 and 12 were of obvious interest. The fracture in pin 12 (Fig. 13-50) was examined more thoroughly than that in pin 1. The photomicrograph reveals a large quantity of the second phase and the fracture surface appears to follow the grain boundaries where this phase has formed. The sample from which this photograph was made was not polished to the center line and, as a consequence, portions of the structure are distorted, particularly near the inside surface. Figure 13-51, a transverse section taken near the fracture in pin 12, better illustrates the pattern of second phase formation. It can be observed that the second phase is virtually continuous near the tubing inside surface but is somewhat less predominant near the outside surface. This structure more closely resembles the structure found in samples exposed to high purity nitrogen in the laboratory than that of laboratory samples exposed to other atmospheres. Figure 13-52 shows the appearance of the structure of a transverse section taken near the fracture in pin 12. The continuous second phase as well as spherical precipitates located within the grains can be seen in considerable detail. The oxide scale on the outside surface (formed by reaction with the coolant gas) is shown on the left; the inside surface which was in contact with the fuel is shown on the right. The cladding temperature at the fracture point averaged about 1490°F.

A longitudinal section of pin 1 near the crack in the cladding is shown in Figure 13-53. A massive second phase formation can be observed in this photomicrograph although it is less continuous than that illustrated in Figure 13-51. A gray material was observed on the inside surface of the tubing, extending into the cladding matrix (this formation is discussed latter in this section of the report). The cladding temperature averaged 1400°F at the point from which this sample was taken.

500X
FIGURE 13-48. HASTELLOY X SHEET AFTER EXPOSURE TO CH_4 FOR 211 hr
AT 1500°F

500X
FIGURE 13-49. HASTELLOY X SHEET AFTER EXPOSURE TO NH_4 FOR 204 hr
AT 1500°F

FIGURE 13-50. FRACTURE SURFACE OF IB-8T-1, PIN 12, 3.8 in. FROM BOTTOM

FIGURE 13-51. IB-8T-1, PIN 12, 5.3 in. FROM BOTTOM

11.2-64-1415

OUTER SURFACE

500X

INNER SURFACE

FIGURE 13-52. IB-8T-1, PIN 12, 5.3 in. FROM BOTTOM (ETCHED)

OUTER SURFACE

500X

INNER SURFACE

FIGURE 13-53. IB-8T-1, PIN 1, 11 in. FROM BOTTOM (AS POLISHED)

Figures 13-54 through 13-57 show the microstructure of sections taken through the cladding of pin 12 at increasing distances from the bottom of the pin. As the distance from the pin bottom increases, the temperature at which the cladding operated decreases. The estimated average cladding operating temperatures for the samples shown in these figures are: 1440°F at 7.7 in. from the bottom; 1350°F at 13.2 in.; 1240°F at 17.6 in.; and 1040°F at 24.2 in. It must be noted, however, that local variations in temperature exist around the circumference of the tubing depending on the orientation of the test element with respect to the GETR core. Figure 13-54 shows the presence of large quantities of the second phase at this, the hottest portion, of the fuel pin. In Figure 13-55, only a small amount of the second phase can be observed near the inner surface of the cladding. In Figure 13-56, a precipitate of fine M_6C and $M_{23}C_6$ particles can be seen throughout the cladding; the curious banded structure at the inner surface in this figure, from which a slight amount of second phase precipitate appears to be radiating, is typical of the structure observed in many of the fuel pins at points which operated at a temperature of approximately 1250°F. The structure resembles that seen in the out-of-pile control samples exposed to ammonia. Figure 13-57, which was taken at the coldest portion of the pin, reveals no indication of second phase or a gray ring formation.

IB-8T-1 pin 1 was the "twin" of pin 12 (located symmetrically in the IB-8T-1 test element with respect to the GETR core). As a result these two pins experienced identical temperature and radiation exposures. As shown in Figures 13-58 through 13-61, the structures were also nearly identical with those observed in pin 12. Figure 13-58 shows the microstructure of the cladding at the point of maximum average temperature (1530°F). At this point the second phase formed continuously through most of the thickness of the cladding. A minor amount of gray ring can be observed near the inside diameter. In Figure 13-59 the second phase occurs primarily near the inside diameter; the finer dispersed phase noted in Figure 13-55 can also be observed. The gray band with radiating second phase precipitate can be observed in Figure 13-60 and a much thinner gray band is apparent in Figure 13-61.

Figures 13-62 through 13-65 show the microstructure of sections of pin 13 which, together with pin 18, was closest to the GETR in the inner ring of pins. The average temperatures of the cladding at the points from which sections were taken are: 1530°F at 4.4 in. from the bottom; 1430°F at 7.7 in.; 1335°F at 13.2 in.; and 1035°F at 24.2 in. In both Figures 13-62 and 13-63, large quantities of the second phase formation are apparent. The growth of the second phase precipitate has proceeded less extensively in the area shown in Figure 13-64. Figure 13-65, representing the coldest section of the pin, does not indicate the presence of any second phase; only a very fine precipitate of carbides can be observed.

Figures 13-66 through 13-69 show the microstructure of sections of pin 18, the "twin" of pin 13. Figure 13-66 shows the microstructure of the cladding 7.7 in. from the bottom of the pin. The second phase is present in large quantities entirely across the cladding thickness. A gray material can be observed on the inside surface between the cladding matrix and the fuel. Figures 13-67, 13-68 and 13-69 were photographed to show the variation in the appearance of the gray band on the inside surface of pin 18 cladding at three points relatively close together. The temperatures corresponding to the sections

FIGURE 13-54. IB-8T-1, PIN 12, 7.7 in. FROM BOTTOM (AS POLISHED)

FIGURE 13-55. IB-8T-1, PIN 12, 13.2 in. FROM BOTTOM (AS POLISHED)

FIGURE 13-56. IB-8T-1, PIN 12, 17.6 in. FROM BOTTOM (ETCHED)

FIGURE 13-57. IB-8T-1, PIN 12, 24.2 in. FROM BOTTOM (AS POLISHED)

FIGURE 13-58. IB-8T-1, PIN 1, 4.4 in. FROM BOTTOM (ETCHED)

FIGURE 13-59. IB-8T-1, PIN 1, 13.2 in. FROM BOTTOM (ETCHED)

FIGURE 13-60. IB-8T-1, PIN 1, 17.6 in. FROM BOTTOM (ETCHED)

FIGURE 13-61. IB-8T-1, PIN 1, 24.2 in. FROM BOTTOM (ETCHED)

OUTER SURFACE

500X

INNER SURFACE

FIGURE 13-62. IB-8T-1, PIN 13, 4.4 in. FROM BOTTOM (AS POLISHED)

OUTER SURFACE

500X

INNER SURFACE

FIGURE 13-63. IB-8T-1, PIN 13, 7.7 in. FROM BOTTOM (ETCHED)

FIGURE 13-64. IB-8T-1, PIN 13, 13.2 in. FROM BOTTOM (ETCHED)

FIGURE 13-65. IB-8T-1, PIN 13, 24.2 in. FROM BOTTOM (ETCHED)

FIGURE 13-66. IB-8T-1, PIN 18, 7.7 in. FROM BOTTOM (AS POLISHED)

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FIGURE 13-67. IB-8T-1, PIN 18, 19.8 in. FROM BOTTOM (ETCHED)

FIGURE 13-68. IB-8T-1, PIN 18, 22 in. FROM BOTTOM (ETCHED)

FIGURE 13-69. IB-8T-1, PIN 18, 24.2 in. FROM BOTTOM (ETCHED)

from which these photographs were made are: 1150°F at 19.8 in. from the bottom; 1075°F at 22 in.; and 1040°F at 24.2 in. At 18.9 in. from the bottom, the ring varied from 0.002 to 0.003 in. in thickness. At 22 in., the thickness varied from 0.001 to 0.002 in. At 22.4 in. the ring had a maximum thickness of 0.001 in. and vanished completely in several places around the inside surface. The figures do not completely reflect this situation because of the circumferential location from which the samples were taken. Both the thickness of the ring and the amount of second phase material irradiating from the ring appear to be a function of the operating temperatures. This conclusion indicates that the formation of the second phase appears to be diffusion controlled.

Figures 13-70 through 13-72 show the microstructure of the sections of the cladding from the IB-8T-1 pin 6. This pin was located in the outer ring at the greatest distance from the GETR core and, as a result, was subjected to the lowest operating temperatures and neutron flux of any of the IB-8T fuel pins. Figure 13-70 shows the microstructure of the cladding 4.4 in. from the bottom of the pin. The average temperature at this point was 1250°F. The structure closely resembles that observed in sections of pins 1, 12, 13 and 18, which operated at about this temperature. The gray ring is apparent on the inside surface in this figure. Figures 13-71 and 13-72 show structures typical of Hastelloy X annealed at these temperatures; no evidence of second phase formation is apparent.

Figures 13-73 and 13-74 show the appearance of an unetched section of the cladding of pin 10. Although this pin was instrumented, very little nitrogen was detected in the extracted gases after irradiation. As these figures show, there is no evidence of second phase formation or of the gray band detected in the specimens discussed above. This pin operated at temperatures midway between the temperature extremes of the other pins.

Figure 13-75 shows the structure of the cladding of IB-8T pin 17 at 17.6 in. from the bottom. The average temperature at this point was 1180°F. Compare this structure with that of the cladding of pin 18 at 19.8 in. from the bottom which operated at an average temperature of 1150°F (Figure 13-6/). Pin 18 contained nitrogen after irradiation but pin 17 did not. As shown in Figure 13-75, no evidence of second phase formation or gray ring at the inside diameter was observed in the cladding of pin 17, thus further substantiating the hypothesis that the second phase formation is dependent on the presence of high purity nitrogen at the metal surface.

13.2 Fuel Metallography

The nuclear fuels used in the IB-8T-1 fuel element were high density UO_2 pellets and high density 40 wt% BeO-60 wt% UO_2 pellets. The pellets were loaded into Hastelloy X tubing as described earlier in this report. The fuel pellets in certain of the fuel pins were cored to permit the installation of thermocouples.

The UO_2 pellets were procured from the Nuclear Materials and Equipment Company of Apollo, Pennsylvania. The ceramic grade UO_2 powder utilized in the fabrication of these pellets was produced by ammonium diuranate precipitation and ranged in size from 1 to 5 microns. The vendor used conventional cold pressing techniques followed by hydrogen sintering at 1700°C to produce pellets

FIGURE 13-70. IB-8T-1, PIN 6, 4.4 in. FROM BOTTOM (ETCHED)

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FIGURE 13-71. IB-8T-1, PIN 6, 7.7 in. FROM BOTTOM (ETCHED)

FIGURE 13-72. IB-8T-1, PIN 6, 24.2 in. FROM BOTTOM (ETCHED)

FIGURE 13-73. IB-8T-1, PIN 10, 7.7 in. FROM BOTTOM (ETCHED)

FIGURE 13-74. IB-8T-1, PIN 10, 17.6 in. FROM BOTTOM (ETCHED)

FIGURE 13-75. IB-8T-1, PIN 17, 17.6 in. FROM BOTTOM (AS POLISHED)

with densities from 94 to 97% of theoretical. The solid BeO-UO₂ pellets were fabricated in the ceramics laboratory of AGN at San Ramon. The starting materials were Spencer Ceramic Company's ceramic grade UO₂ and Brush Beryllium Company's LOH ceramic grade beryllia powder. The pre-dried homogenized powders were blended for 2 hr in Vee blenders and then wet-mixed with a solution of water, castor oil and CD108 (a commercially available binder). The wet-mixed material was partially dried in a vacuum oven and granulated by passing through a double screen. Subsequent drying and removal of the fines yielded 20 mesh BeO-UO₂ granules. The pellets were cold pressed from the granules in a Stokes Model F-4 automatic press and sintered in hydrogen at 3000°F. Pellets with densities from 95 to 97% of theoretical were produced routinely by this process. Fabrication techniques similar to those described above, with minor modifications to binder and blender techniques, were used by BMI to fabricate cored BeO-UO₂ pellets.

Numerous specimens of the fuel from the IB-8T-1 test element were prepared and evaluated by metallographic techniques to determine the behavior of the fuel in this test element. The samples were ground in an Automat unit adjusted to apply a 30 lb load. The samples were ground on 240 grit paper for 10 minutes; a paraffin-kerosene lubricant was added during the last 5 minutes of this grinding. Specimens were next ground on 400 grit papers for 5 minutes with the lubricant. The next step consisted of grinding on 600 grit paper for 20 minutes with the lubricant. This step was repeated three times with fresh papers. The samples were then polished on a Syntron polishing unit for 2-1/2 hr using 0.5 micron diamond dust in a 2% chromic acid solution. Etching was accomplished by swab application of a sulfuric acid-hydrogen peroxide solution to define the UO₂ structure and an ammonium bifluoride solution to define the BeO structure where applicable.

Figure 13-76 shows unirradiated UO₂ fuel in the as-etched condition. The structure shown is considered to be characteristic of sintered UO₂ based on previous examinations at AGN and structures reported in the literature (Ref. 13-1). Figure 13-77 shows the change in structure with irradiation as revealed at various points in fuel pin 13 (a pin in which maximum fuel temperature and burnup were experienced). The cracking evident in the 50X photographs is characteristic of the appearance of UO₂ after significant irradiation. The gradual change in structure from the cooler (24.2 in. from bottom) portion of the fuel pin to the hotter (4.4 in. from bottom) areas is evident. The nature of the porosity apparent in the as-fabricated body has changed; minor porosity appears in the grain boundaries, particularly in the higher temperature, higher burnup sections. Very little if any change in grain size is evident; this is attributed to the relatively low (2000°F) central fuel temperatures.

Figure 13-78 shows a section through pin 17 which operated at somewhat lower temperatures than pin 13 but experienced similar burnup. The structures shown in this picture are quite similar to those shown in Figure 13-77 and are representative of the appearance of UO₂ fuel.

Figure 13-79 shows an unusual structure observed at the inside diameter of the cored UO₂ pellets in pins 13 and 18. The layers shown are believed to be oxides of the fuel which formed when the coolant gas leaked into the fuel pin. Identification of this material was attempted but the location, high

FIGURE 13-76. TYPICAL STRUCTURE OF UNIRRADIATED SINTERED UO_2 FUEL (ETCHED)

radiation level and thinness of the deposit made conventional identification techniques (X-ray diffraction or microprobe analysis) unsuitable.

Another unusual formation, which appears as a metallic fiber or splinter thrusting into the fuel, is illustrated in Figure 13-80. These formations appear to originate from the second phase located at the fuel cladding interface but they have not been positively identified.

Figure 13-81 shows the structure of unirradiated BeO-UO_2 fuel of the composition used in the IB-8T test elements. The relatively uniform dispersion and distinct delineation of the two constituents is quite apparent. Figure 13-82 shows the appearance of the BeO-UO_2 fuel at various levels in pin 1 (which accumulated the maximum burnup and was exposed to the highest temperatures of all the BeO-UO_2 fuel bearing pins). Some cracking has occurred in the fuel although significantly less than that observed in the UO_2 . Part of the damage apparent in the 50X photograph at the 13.2 in. level was the result of preparation of the metallographic sample. As evidenced in this figure, the UO_2 particles have agglomerated into more discrete and larger particles. This change was also reflected in increasing difficulty in polishing and etching the more highly irradiated samples which accounts for the difference in appearance of the photographs at the 13.2 and 4.4 in. levels. Figure 13-83 shows the microstructure of BeO-UO_2 fuel from pin 6 which operated at lower temperatures and accumulated a lower burnup than pin 1.

24.2 in. from Bottom
Operating Temperature, 1370°F

1.90×10^{20} fissions/cc

13.2 in. from Bottom
Operating Temperature, 1890°F

3.16×10^{20} fissions/cc

4.4 in. from Bottom
Operating Temperature, 2100°F

3.24×10^{20} fissions/cc

ETCHED

50X

ETCHED

500X

FIGURE 13-77. UO₂ STRUCTURE AS A FUNCTION OF TEMPERATURE AND BURNUP (IB-8T-1 PIN 13)

FIGURE 13-78. SECTION THROUGH PIN 17, AT 17.6 in. FROM BOTTOM.
BURNUP 2.88×10^{20} fissions/cc; SURFACE TEMPERATURE, 1720°F

11.2-64-1430

Section from Pin 13, 4.4 in. from Bottom

Section from Pin 18, 7.7 in. from Bottom

FIGURE 13-79. FORMATIONS OBSERVED ON INSIDE DIAMETER OF CORED FUEL PELLETS FROM IB-8T-1 FUEL PINS

FIGURE 13-80. SECTIONS THROUGH IB-8T-1, PIN 13, SHOWING FORMATIONS OBSERVED OF FUEL CLADDING INTERFACE

11.2-64-1432

FIGURE 13-81. TYPICAL STRUCTURE OF UNIRRADIATED
SINTERED BeO-UO_2 FUEL (ETCHED)

11-2-64-1433

24.2 in. from Bottom
Operating Temperature, 1180°F

2.50×10^{20} fissions/cc

13.2 in. from Bottom
Operating Temperature, 1540°F

3.34×10^{20} fissions/cc

4.4 in. from Bottom
Operating Temperature, 1705°F

3.24×10^{20} fissions/cc

FIGURE 13-82. BeO-UO₂ FUEL STRUCTURE AS A FUNCTION OF TEMPERATURE AND BURNUP (IB-8T-1 PIN 1)

Pin 6, 24.2 in. from Bottom: Etched;
Burnup 1.67×10^{20} fissions/cc; 1060°F
Operating Temperature

Pin 6, 4.4 in. from Bottom; Etched;
Burnup 1.62×10^{20} fissions/cc; 1350°F
Operating Temperature

FIGURE 13-83. IRRADIATED BeO-UO_2 FUEL STRUCTURE, IB-8T-1, PIN 6

13.3 Thermocouple Sheath Metallography

During the metallurgical evaluation of the IB-8T-1 test element, a second phase formation was observed in the Inconel 600 thermocouple sheathing. Figure 13-84 shows the structure of the thermocouple sheath at the top of pin 12. The average temperature of the sheath at this point during operation was 1210°F. It can be seen that a reaction has occurred on the outside surface (fuel side) of the sheath although the balance of the material exhibits a structure typical of Inconel 600. Figure 13-85 shows the structure of a section of sheath which operated at an average temperature of 1565°F. In this photograph the second phase has agglomerated throughout the sheath. The ragged appearance of the outside surface may indicate that the material has separated from the Inconel 600 matrix during fuel pin cooling. This mechanism is substantiated by the appearance of the sample shown in Figure 13-86 where obvious discontinuities appear on the outside surface. Figure 13-87 shows another section of thermocouple sheath where the second phase has apparently agglomerated but appears to be located predominantly on the inner portion of the sheath.

No attempt was made to determine the nature or cause of the apparent second phase since such activity was not required for the successful prosecution of AGCRSP work.

13.4 Summary of IB-8T-1 Metallurgical Findings

a. Cladding: A review of the metallurgical evaluation of the IB-8T fuel pin cladding reveals that an unusual second phase has formed in some pin cladding but other pins exhibit a structure which closely resembles that of aged Hastelloy X exposed in high temperature environments. The second phase appears to originate from the inner surface of the fuel pins and is characterized by the formation of a ring in cooler parts of the pin which gradually broadens and eventually becomes a continuous intergranular structure in the high temperature regions. Since phases of this sort are generally hard and brittle (confirmed, in this case, by the microhardness measurements) the continuous intergranular structure of this phase is considered to be the primary cause for the extreme embrittlement observed in some of the pins and the reason for the cracking of pins 1 and 12 during hot cell handling.

The reduced ductility of pins in which the second phase was not present is attributed to use of incompletely annealed material, resulting in unusual carbide precipitation, and to the difficulty associated with testing a limited number of samples of highly irradiated material.

Table 13-9 presents data on the nitrogen content of the gas extracted from the IB-8T-1 fuel pins, the presence of a second phase in the fuel pin cladding, and the location of thermocouples in the fuel pins. The correlation shows that massive amounts of the second phase occurred only in the cladding of fuel pins which contained nitrogen after operation (pins 1 and 12 could not be analyzed since fracture occurred before the gas samples could be taken; based on all other evidence, it seems highly probable that these pins contained nitrogen). It appears, therefore, that the presence of nitrogen within the fuel pins promotes the formation of the second phase; the cladding of the three

FIGURE 13-84. INCONEL 600 THERMOCOUPLE SHEATH
IB-8T-1, PIN 12, 24.2 in. FROM BOTTOM

FIGURE 13-85. INCONEL 600 THERMOCOUPLE SHEATH
IB-8T-1, PIN 12, 13.2 in. FROM BOTTOM

INSIDE SURFACE

500X

FIGURE 13-86. INCONEL 600 THERMOCOUPLE SHEATH IB-8T-1, PIN 13, 17.6 in. FROM BOTTOM

OUTSIDE SURFACE

500X

FIGURE 13-87. INCONEL 600 THERMOCOUPLE SHEATH IB-8T-1, PIN 13, 17.6 in. FROM BOTTOM

TABLE 13-9 - CORRELATION BETWEEN NITROGEN CONTENT, SECOND PHASE
FORMATION AND INSTRUMENTATION IN IB-8T-1 FUEL PINS

<u>Pin No.</u>	<u>Nitrogen in Extracted Gas, %</u>	<u>Second Phase Present</u>	<u>Thermocouple Installed</u>
1	(1)	Yes	Yes
2	0.05	No	No
3	0.02	(2)	No
4	n.d. ⁽³⁾	(2)	No
5	n.d.	(2)	No
6	95.6	Yes	Yes
7	n.d.	No	No
8	0.20	(2)	No
9	0.03	(2)	No
10	0.07	No	Yes
11	n.d.	No	No
12	(1)	Yes	Yes
13	90.0	Yes	Yes
14	n.d.	(2)	No
15	(4)	(2)	No
16	0.11	(2)	No
17	0.09	No	Yes
18	82.0	Yes	Yes

(1) Pin broken during hot cell handling.

(2) Not examined metallographically.

(3) n.d. = not detectable.

(4) Extraction equipment leaked.

pins known to contain nitrogen all exhibited massive formations of the second phase while the cladding of the five pins definitely known to contain principally helium did not show any evidence of second phase formation.

The role of nitrogen in promoting second phase formation can be inferred from laboratory control tests. Samples exposed to high purity nitrogen (or nitrogen with oil vapor) at 1750°F for as little as 6 hr exhibited the initiation of second phase formation along the surface. On the other hand, this phase did not form in Hastelloy X exposed to argon or to oxidizing atmospheres (99.5 wt% nitrogen plus 0.5 wt% oxygen, or air) or to carburizing atmospheres (nitrogen with CO or CH₄).

The origin of the nitrogen in some of the IB-8T-1 fuel pins can be inferred from Table 13-9. As indicated in this table, every pin which contained nitrogen was instrumented. The installation of a thermocouple in the fuel pin involves a brazed joint at the point where the thermocouple enters the fuel pin sheath. It seems highly probable that a leak could occur in the sheath as a result of heat sensitization of the sheath material in the vicinity of the braze. However, two such leaks would be required to permit the coolant gas to enter the fuel pin. It seems more probable that porosity in the brazed joint was responsible for the introduction of nitrogen into the pin. Highly porous brazed joints were observed metallographically during the development of the thermocouple installation procedures. Once coolant entered the fuel pin, the small amount of oxygen would be rapidly "gettered" by the UO_2 or by the metal components and a pure nitrogen atmosphere would remain. This mechanism appears to reasonably describe the method by which nitrogen was introduced into the fuel pins and is substantiated by the fact that significant amounts of nitrogen were not detected in any uninstrumented pin.

The identification of the second phase was exceptionally difficult. The Ni-Cr-Mo-Fe system is an extremely complex quaternary system on which new information is constantly appearing in the metallurgical literature. The complexity of the system increases significantly when reactions of the alloying constituents with nitrogen and carbon are considered. It was not surprising, therefore, that a search of the literature did not provide an exact identification of the second phase observed in the IB-8T-1 cladding sections and in Hastelloy X sheet specimens exposed to high purity nitrogen.

The X-ray diffraction and electron microprobe data indicated that the second phase was a Ni-Cr-Mo-Fe compound with a lattice parameter which varied from 6.35 to 6.40 Å. Apparently the variation in lattice parameter is associated with variations in the chromium and molybdenum content which do not alter the structure. Based on the information available in the literature, the second phase was tentatively identified as a beta-manganese structure in the Ni-Cr-Mo-Fe system analogous to the pi-phase reported in the Fe-Cr-W system by Goldschmidt (Ref. 13-2). Goldschmidt reported that the pi-phase in the Fe-Cr-W system had a beta-manganese structure with a lattice parameter which varied between 6.275 and 6.39 Å which is in good agreement with the lattice parameters observed for the second phase in the Hastelloy X samples. As a consequence, the second phase observed in the IB-8T-1 has been categorized as a variant of the pi-phase and is referred to as such in the balance of this report. Goldschmidt also reports that the pi-phase in the Fe-Cr-W system is stable only at very high temperatures and that carbon played an important part in stabilizing the phase. This observation suggests the possibility that nitrogen could act in a manner similar to carbon to stabilize the pi-phase in the Ni-Cr-Mo-Fe system.

It is not clear whether the nitrogen actually forms a part of the pi-phase. It is possible that nitrogen could promote the formation of the phase without entering into the structure by changing the solubility of the various components so that the pi-phase would precipitate out of the austenite solid solution. A survey of literature on binary, ternary and quaternary systems

containing nickel, chromium, iron and molybdenum* did not disclose any further data which was helpful in the identification of the second phase. Further studies in this area were not considered justified under the AGCRSP inasmuch as the fuel elements intended for extended exposure in the ML-1 reactor will not contain thermocouples and consequently embrittlement is not anticipated.

b. Fuel: A review of the metallurgical findings of the IB-8T fuel evaluation indicates that this material will perform satisfactorily for the 10,000-hr lifetime specified for the ML-1 fuel elements. Some cracking and structural changes were observed and some minor evidence of fuel-cladding interactions; however, none of these processes appear to be of sufficient significance to affect the performance of the material.

c. Thermocouple Sheathing: The appearance of a second phase in the normally single-phase Inconel 600 was observed but the subject was not studied in detail. It appears likely that this phase formation is associated with the same circumstances which caused the formation of the pi-phase in the Hastelloy X fuel pin cladding.

*A selected bibliography on this subject is included below.

Ref. 13-1: J. Belle, "Uranium Dioxide, Properties and Nuclear Applications," USAEC, 1961

Ref. 13-2: H. J. Goldschmidt, "Occurrence of the Beta-Manganese Structure in Transition Metal Alloys and Some Observations on Chi-Phase Equilibria", Metallurgia, July 1957, p. 17-26

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C. C. Clark and J. S. Iwanski, "Phase Changes in Precipitation Hardening Nickel-Chromium-Iron Alloys During Prolonged Heating," Transactions of the Metallurgical Society of AIME, Vol. 215, pp. 648-651 (August 1959)

P. Duwez and S. R. Baen, "X-Ray Study of the Sigma Phase in Various Alloys Systems," Symposium on the Nature, Occurrence, and Effects of Sigma Phase, ASTM-STP-110 (1959)

P. V. Guthrie and E. E. Stansbury, "X-Ray and Metallographic Study of the Nickel-Rich Alloys of the Nickel-Molybdenum System II," ORNL-3078 (July 17, 1961)

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M. Kaufman, "Control of Phases and Mechanical Properties in Nickel-Base Alloys of the Rene 41 Type," Transactions of the Metallurgical Society of AIME, Vol. 227, pp. 405-422 (April 1963)

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R. J. Radebaugh and E. P. Sadowski, "Microstructural Changes in a 42 Pct Ni, 30 Pct Cr, 26 Pct Fe Alloy During Creep Rupture Testing," Transactions of the Metallurgical Society of AIME, Vol. 215, pp. 23-29 (February 1959)

D. P. Shoemaker, "Study of Crystal Structures of Complex Metallic Phases," ASTIA Report No. AD209926 (November 1958)

J. H. Westbrook, "Temperature Dependence of the Hardness of Secondary Phases Common in Turbine Bucket Alloys," Journal of Metals, pp. 898-904, July 1957

14.0 IB-8T-2

The IB-8T-2 test element was evaluated metallurgically to detect any changes induced by the 10,000-hr irradiation which would affect performance and lifetime of the ML-1 fuel elements. Considerable emphasis was placed on the evaluation of structural changes in the fuel pin cladding in view of the embrittlement found in the cladding of some IB-8T-1 fuel pins. In addition, the oxidation penetration of the fuel pin cladding after 10,000 hr was evaluated as was the extent of the collapse of the cladding on the fuel stack and the appearance of the fuel.

14.1 Cladding

The location from which samples of the fuel pins were removed for metallurgical evaluation of the IB-8T-2 cladding is shown in Figure 14-1. The specimens were prepared for metallographic evaluation using the techniques described earlier for the IB-8T-1 test element specimens.

a. Metallography:

1) Control Sample Metallography: Numerous Hastelloy X sheet specimens were exposed in the laboratory to conditions which closely approached those experienced by the IB-8T-2 test element to provide control data for metallurgical evaluation. The control specimens were exposed for 7500 hr to reference gas at 1750°F followed by an exposure for 3500 hr to air at 1600°F (the IB-8T-2 was exposed for 6400 hr in reference gas and 3700 hr in air). The change in the temperature for the air exposure was made to reflect refined calculations of the IB-8T-2 average maximum cladding temperature. This work indicated that the cladding was operated at approximately 1600°F.

The metallographic appearance of the control samples shows that the oxidation attack has reached a maximum penetration of approximately 0.002 in. (Figures 14-2 and 14-3). The penetration of the precipitates along the grain boundary appears more extensive than that observed in the IB-8T-1 control samples.

2) IB-8T-2 Cladding Metallography: The extent to which oxidation penetrates the cladding of ML-1-type fuel elements is of considerable importance to the performance of the reactor inasmuch as a single penetration of the cladding

KEY - NUMBERS DENOTE DISTANCE FROM BOTTOM OF PIN IN INCHES

▨ TENSILE TEST SAMPLES

--- DENOTES CUTS

✓ DENOTES FACE EXAMINED METALLOGRAPHICALLY

ALSO TENSILE SAMPLES:

T- 5	PIN 7	2.2 TO 8.2 in.
T- 6	PIN 7	8.2 TO 14.2 in.
T-10	PIN 16	2.2 TO 8.2 in.
T-11	PIN 16	8.2 TO 14.2 in.
T- 9	PIN 15	2.2 TO 8.2 in.

FIGURE 14-1. SCHEMATIC DIAGRAM OF LOCATION OF METALLOGRAPHY AND TENSILE SAMPLES FROM IB-8T-2 CLADDING

FIGURE 14-2. SECTION THROUGH THE OXIDIZED SURFACE OF A HASTELLOY X SHEET SPECIMEN

(Exposed for 7500 hr at 1750°F to 99.5% N with 0.5% O₂ followed by exposure for 3500 hr at 1600°F to air)

FIGURE 14-3. INTERNAL STRUCTURE OF SPECIMEN SHOWN IN FIGURE 14-2

could permit release of fission products into the coolant gas, oxidation of the UO_2 fuel with associated swelling of the pin, or introduction of nitrogen into the interior of the pin which could result in cladding embrittlement. The results of the measured maximum oxide penetrations and surface oxide thicknesses observed in the IB-8T-2 fuel pin cladding are presented in Table 14-1.

TABLE 14-1 - IB-8T-2 FUEL PIN CLADDING OXIDATION

<u>Pin No.</u>	<u>Distance From Bottom, in.</u>	<u>Calculated Average Temperature, °F</u>	<u>Maximum Oxide Penetration, in.</u>	<u>Maximum Surface Oxide Thickness, in.</u>
1	4.4	1530	0.0022	0.0010
1	13.2	1350	0.0020	0.0006
1	17.6	1240	0.0016	0.0003
1	24.2	1035	0.0	0.0
12	7.7	1425	0.0024	0.0004
12	24.2	1035	0.0	0.0
10	7.7	1350	0.0020	----
10	17.6	1180	0.0006	0.0001
6	4.4	1250	0.0008	----
6	7.7	1240	0.0010	0.0005
6	24.2	960	0.0	0.0
13	4.4	1530	0.0040	0.0012
13	7.7	1430	0.0044	0.0018
13	13.2	1335	0.0030	0.0012
13	24.2	1035	0.0	0.0
18	7.7	1430	0.0030	0.0008
17	17.6	1180	0.0003	0.0002

In general, it can be observed from the data in the table that the oxide penetration and thickness of the oxide surface varies directly with the temperature at which the cladding operated. The maximum penetration observed was 0.0044 in.; this value is approximately twice that which would have been predicted from out-of-pile experiments but is sufficiently small to eliminate concern about the ability of the element to operate satisfactorily for 10,000 hr.

Photomicrographs were prepared at the outer surface of the cladding samples to illustrate the extent and mode of oxidation. Figure 14-4 shows the appearance of the oxidation attack of the cladding of IB-8T-2 pin 13; this pin was the most severely attacked of all the pins in the test element. The maximum penetration observed occurred at 7.5 in. from the bottom where the deepest point of attack was measured at 0.0044 in. However, at 4.4 in. from the bottom, where the calculated average temperature was highest, the penetration was slightly less than 0.004 in. In all cases the penetration took the form

CLADDING

24.9 in. from Bottom

1035°F

No Detectable Oxidation

13.7 in. from Bottom

1335°F

Maximum Oxide Penetration = 0.003 in.

CLADDING

7.5 in. from Bottom

1430°F

Maximum Oxide Penetration = 0.0044 in.

4.4 in. from Bottom

1530°F

Maximum Oxide Penetration = 0.004 in.

FIGURE 14-4. OUTER SURFACE OF IB-8T-2, PIN 13 AT VARIOUS LOCATIONS

of intergranular oxidation typical of that observed in laboratory control specimens.

Figure 14-5 shows the oxidation of the outer surface of the cladding of IB-8T-2 pin 1. The cladding of this pin operated at the highest temperatures (1530°F average) in the outer ring of fuel pins. The oxide penetration increased with increasing temperature and reached a maximum of 0.0022 in. The attack in this pin was somewhat less than that observed in pin 13 although the calculated operating temperatures for the two pins were approximately equal.

Figure 14-6 shows the oxide attack on the outer surface of the IB-8T-2 fuel pin 10 and Figure 14-7 shows the appearance of the outer surface of pin 12. The degree of attack in these cases agreed closely with that predicted from the appearance of fuel pin 1 at locations which operated at equivalent temperatures.

Figure 14-8 shows the appearance of the surface of pin 6. This pin operated at the lowest temperature in the fuel element and, as indicated in the photographs, experienced only a very minor attack. Figure 14-9 shows the appearance of the surface of pin 18; this pin was the twin to pin 13. As revealed in the photomicrographs, oxidation penetration was approximately 0.003 in.

b. Microstructure: A comprehensive metallographic evaluation of the microstructure of the IB-8T-2 fuel pin cladding was performed to determine if structural rearrangements had occurred which could result in embrittlement of the cladding, to compare the structures with those observed in out-of-pile exposures of similar materials, and to provide information to assist in the evaluation of tensile tests and microhardness measurements.

All of the IB-8T-2 cladding specimens examined revealed structures which were essentially identical with those of Hastelloy X aged out-of-pile under similar temperature conditions. Figures 14-10 through 14-13 show the microstructure of four sections taken through the cladding of IB-8T-2 pin 1 at different vertical locations. Pin 1 operated at the highest cladding temperature in the outer row of pins. It can be seen from these photomicrographs that the intergranular precipitate agglomerates with higher temperature and tends to collect along the grain boundaries of the metal.

Figures 14-14 through 14-17 show the structures at various sections through the IB-8T-2 pin 13. This pin operated with the highest cladding temperature of any of the inner ring of fuel pins. The structures in pin 13 are quite similar to those of pin 1; no evidence of second phase formation or any other unusual condition can be observed.

The structures of sections of pins 1 and 13 were analyzed with the techniques developed by Fullman (Ref. 14-1) to provide a quantitative evaluation of the precipitation phenomena. The analysis developed data on the amount of precipitate, the mean free path between the precipitates, and the average precipitate size in each specimen. This technique involves the use of a random line drawn across the photomicrographs of the section. The volume percentage of a precipitate was determined by measuring the total length of the random

24.2 in. from Bottom

1035°F

No Detectable Oxidation

17.6 in. from Bottom

1240°F

Maximum Oxide Penetration = 0.0016 in.

13.2 in. from Bottom

1350°F

Maximum Oxide Penetration = 0.002 in.

4.4 in. from Bottom

1530°F

Maximum Oxide Penetration = 0.0022 in.

FIGURE 14-5. OUTER SURFACE OF IB-8T-2, PIN 1 AT VARIOUS LOCATIONS

FIGURE 14-6. OUTER SURFACE OF IB-8T-2, PIN 10

FIGURE 14-7. OUTER SURFACE OF IB-8T-2, PIN 12

11.2-64-1449

FIGURE 14-8. OUTER SURFACE OF IB-8T-2, PIN 6

FIGURE 14-9. OUTER SURFACE OF IB-8T-2, PIN 18

INNER SURFACE

OUTER SURFACE

FIGURE 14-10. IB-8T-2, PIN 1, CLADDING AT 24.2 in. FROM BOTTOM (Average Temperature = 1035°F)

INNER SURFACE

OUTER SURFACE

FIGURE 14-11. IB-8T-2, PIN 1, CLADDING AT 17.6 in. FROM BOTTOM (Average Temperature = 1240°F)

INNER SURFACE

OUTER SURFACE

FIGURE 14-12. IB-8T-2, PIN 1, CLADDING AT 13.2 in. FROM BOTTOM (Average Temperature = 1350°F)

INNER SURFACE

OUTER SURFACE

FIGURE 14-13. IB-8T-2, PIN 1, CLADDING AT 4.4 in. FROM BOTTOM (Average Temperature = 1530°F)

INNER SURFACE

OUTER SURFACE

FIGURE 14-14. IB-8T-2, PIN 13, AT 24.2 in. FROM BOTTOM (Average Temperature = 1035°F)

160

INNER SURFACE

OUTER SURFACE

FIGURE 14-15. IB-8T-2, PIN 13, CLADDING AT 13.2 in. FROM BOTTOM (Average Temperature = 1335°F)

INNER SURFACE

OUTER SURFACE

FIGURE 14-16. IB-8T-2, PIN 13,CLADDING AT 7.7 in. FROM BOTTOM (Average Temperature = 1430°F)

161

INNER SURFACE

OUTER SURFACE

FIGURE 14-17. IB-8T-2, PIN 13,CLADDING AT 4.4 in. FROM BOTTOM (Average Temperature = 1530°F)

line intercepted by particles, A, and the total length of the random line, B. The volume percentage was then calculated by dividing A by B. The mean free path between precipitates was determined by counting the number of particles per unit length intersected by the random line, N_L ; the mean free path is then equal to $(1-f)/N_L$, where f is the volume fraction of the precipitate. The average precipitate particle size was determined by measuring the Martin's diameter of individual particles and multiplying the arithmetic mean of the measured diameters by 4π . Martin's diameter is the distance across particles measured on a line bisecting the particles where all bisecting lines are parallel. The data developed by this analysis is presented in Table 14-2.

TABLE 14-2 - AMOUNT, SPACING, AND SIZE OF PRECIPITATES
LOCATED IN THE CENTER OF SECTION FROM IB-8T-2 PINS 1 AND 13

<u>Pin No.</u>	<u>Distance From Bottom, in.</u>	<u>Temperature °F</u>	<u>Amount of Precipitate, vol%</u>	<u>Mean Free Path, in.</u>	<u>Average Precipitate Size, in.</u>
13	4.4	1530	13.2	0.00112	0.00026
13	7.7	1430	6.8	0.00144	0.00014
13	13.2	1335	9.6	0.00074	0.00009
13	24.2	1035	2.7	0.00072	0.00004
1	4.4	1530	12.4	0.00066	0.00013
1	7.7	1350	6.9	0.00088	0.00007
1	17.6	1240	4.5	0.00084	0.00005
1	24.2	1035	4.2	0.00046	0.00003

Examination of the data in Table 14-2 reveals that the amount of precipitate increases with increasing temperature (except for the value derived for 7.7 in. from the bottom of the pin for pin 13), that the mean free path increases to a maximum and subsequently decreases with increasing temperature, and that the average particle size increases with increasing temperature.

c. Tensile Testing: Tensile tests of specimens of the 10,000-hr IB-8T-2 fuel pin cladding were conducted to determine the changes in mechanical properties as a result of the extended thermal and irradiation exposures. In addition, the data generated by these tests were used to verify the assumptions incorporated in the design of the fuel element and to expand the data available on the effect of extended thermal exposure on the mechanical properties of Hastelloy X. The physical properties explored in these tests were the ductility and strength of the irradiated material. Inasmuch as the major physical shocks imposed on the ML-1 fuel elements would occur at ambient conditions (as a result of relocation and transportation), most of the tensile tests were performed at room temperature. However, it was desirable to verify the properties of the material at higher temperatures to ensure that no unanticipated reduction in strength or ductility had occurred. Consequently, a significant number of tensile tests were performed at 1200° and 1400°F, temperatures representative of in-reactor operating conditions.

Test specimens were prepared by the same procedures used for preparation of the IB-8T-1 tensile specimens and the tests were performed at BMI in the same machine under the same conditions used for the IB-8T-1 tests. The specimens were taken from the locations shown in Figure 14-1; the condition of the specimens is shown in Table 14-3. The problem of dissolving the fuel (discussed in the IB-8T-1 section of this report) was also encountered in the processing of the IB-8T-2 samples. The dissolution time for the UO_2 fuel varied between 7 and 18 hr but in some cases nearly 100 hr of immersion in the reagent was required to dissolve the BeO-UO_2 .

TABLE 14-3 - CONDITION AND PRE-TEST LENGTH
OF IB-8T-2 TENSILE TEST SPECIMEN

<u>Specimen No.</u>	<u>Fuel Pin No.</u>	<u>Fuel Type</u>	<u>Pre-Test Specimen Length, in.</u>	<u>Visual Condition</u>
T-1	1	BeO-UO_2	7.2	Bent at one end
T-2	6	"	4.8	Straight
T-3	12	"	5.0	Severely S-shaped
T-4	18	UO_2	6.0	Straight
T-5	7	BeO-UO_2	6.0	Straight
T-6	7	"	4.8	Straight
T-7	12	"	4.9	Slightly S-shaped
T-8	13	UO_2	6.0	Slightly bent
T-9	15	"	6.0	Straight
T-10	16	"	6.0	Straight
T-11	16	"	6.0	Straight
T-12	18	"	6.0	Straight

The test was complicated and, in all probability, the accuracy of the results was affected somewhat by the condition of the samples and the need to perform the work remotely. The extreme hardness of the samples caused the grips to slip in some cases, necessitating a repetition of the test. On some occasions the relatively rough oxide film interfered with the proper operation of the extensometer. In spite of these problems, the data is considered to be consistent and sufficiently accurate to describe the characteristics of the material (Table 14-4). In general, the data of Table 14-4 show that the Hastelloy X material retained a significant amount of the strength and ductility of the unirradiated material. The reduction in yield strength is attributed largely to the extended thermal exposure; the ductility changes were probably caused by a combination of thermal and irradiation effects although the selection of the predominant effect is not possible from the available data.

Figure 14-18 shows photomicrographs of the fracture of typical specimens under room temperature tensile testing. Figure 14-19 shows similar photographs of specimens tested at elevated temperatures. General yielding is not obvious in these photographs and, from the elongation data, would not be expected. However, some local yielding is apparent, particularly in the photographs of specimens tested at elevated temperatures.

TABLE 14-4 - MECHANICAL PROPERTY DATA - IB-8T-2 ELEMENT

Specimen No.	Pin No.	Operating Temp, °F	Testing Temp, °F	Neutron Dose x 10 ²⁰ , nvt	Ultimate Tensile Strength, psi	0.2% Yield Strength, psi	Elongation in 3-in., %	Remarks
B-102	---	Unexposed	Room	---	114,800	67,000	37.5	Failed in middle of gage section
B-102	---	Unexposed	1200	---	67,000	36,800	40.7	
T-1	1	1455	Room	10.0	67,000	41,900	8.3	
T-3	12	1375	Room	9.8	86,300	48,800	*	Broke in gage section
T-6	7	1195	Room	6.5	116,300	59,400	15.4	Broke at end of center pin
T-7	12	1480	Room	10.1	98,400	53,600	9.1	
T-9	15	1300	Room	6.3	95,070	46,700	17.8	Broke at center of gage section
T-10	16	1300	Room	6.3	101,000	46,100	28.8	Broke in gage section, 1 in. from center pin
T-11	16	1200	Room	7.3	107,700	49,700	17.4	Broke in gage section
T-12	18	1480	Room	10.6	94,490	44,400	15.5	Broke in gage section
T-2	6	1195	1200	6.0	39,000	29,600	6.1**	Broke in grips
T-5	7	1240	1200	5.5	54,100	31,900	9.2**	Broke in grips
T-4	18	1385	1400	10.2	34,900	27,600	3.1**	Broke in grips
T-8	13	1385	1400	11.0	35,500	27,800	11.0**	Broke in grips

*Specimen slipped in grips, elongation data suspect.

**Broke in grips, elongation data invalid.

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Specimen T-12, Pin 18
Operating Temperature, 1480°F;
Irradiation Dose 10.6×10^{20} nvt_f

6X

Specimen T-11, Pin 16
Operating Temperature, 1200°F;
Irradiation Dose, 7.3×10^{20} nvt_f

6X

FIGURE 14-18. FRACTURE OF IB-8T-2 SPECIMENS TESTED AT ROOM AND ELEVATED TEMPERATURE

Specimen T-2, Pin 6
Operating Temperature, 1195°F;
Irradiation Dose 6.0×10^{20} nvtf
Testing Temperature, 1200°F

6X

Specimen T-8, Pin 13
Operating Temperature, 1385°F
Irradiation Dose 11.0×10^{20} nvtf
Testing Temperature, 1400°F

6X

FIGURE 14-19. FRACTURE OF IB-8T-2 SPECIMENS TESTED AT ELEVATED TEMPERATURE

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Figures 14-20 and 14-21 show microstructures of longitudinal samples of specimens tested at room temperature and at elevated temperature respectively. These microstructures do not differ greatly from those of irradiated IB-8T-2 tubing or from those of tensile tested sheet specimens exposed 10,000 hr out-of-pile. The predominant differences are the twinning and cracking of M_6C particles and the incipient grain boundary cracking near the specimen surface in Figure 14-21.

d. Microhardness Measurements: Microhardness measurements were made on the tensile-tested specimens in accordance with procedures described for the IB-8T-1 test element cladding. The results of this testing are shown in Table 14-5.

Although the data in Table 14-5 are badly scattered, it can be concluded that the hardness of the specimens from the lowest temperature regions (near the top of the pin) is very close to the as-received hardness (180 to 190 DPH) and that the hardness generally decreases with increasing exposure temperature. This latter conclusion is supported reasonably well by the plots shown in Figure 14-22.

TABLE 14-5 - MICROHARDNESS OF IB-8T-2 CLADDING AFTER IRRADIATION
(Diamond Pyramid Hardness)

IB-8T-2 Pin No.	Distance From Pin Bottom, in.	Temp, °F	(O.D.) Sampling Location (I.D.)					Avg
			1	2	3	4	5	
1	24.2	1035	166	166	198	197	197	191
1	17.6	1240	208	209	209	206	216	210
1	13.2	1350	228	226	212	229	221	223
1	4.4	1530	172	197	205	201	206	196
6	24.2	960	177	183	180	190	189	184
6	7.7	1240	239	226	208	194	208	215
6	4.4	1250	265	234	231	231	217	236
10	17.6	1180	259	255	228	239	235	243
10	7.7	1350	231	200	200	189	198	204
17	17.6	1180	229	253	247	253	237	244
18	7.7	1430	158	186	189	195	204	186
12	24.2	1035	308	274	241	247	241	262
12	7.7	1425	194	206	198	194	200	198
13	24.2	1035	163	197	214	200	204	216
13	13.2	1335	206	198	189	200	190	197
13	4.4	1430	216	208	206	198	194	204
13	7.7	1530	192	179	195	183	168	183

FIGURE 14-20. IB-8T-2, SPECIMEN T-12 (PIN 18) TESTED AT ROOM TEMPERATURE

FIGURE 14-21. IB-8T-2, SPECIMEN T-5 (PIN 7) TESTED AT 1200°F

FIGURE 14-22. MICROHARDNESS OF IB-8T-2 CLADDING SPECIMENS VERSUS CALCULATED OPERATING TEMPERATURES

e. Fuel-Cladding Compatibility: The interface between the fuel and the cladding of the IB-8T-2 test element was examined metallographically to determine if any reaction detrimental to the performance of the fuel pin had occurred. The appearance of typical fuel cladding interfaces is shown in Figures 14-23 through 14-29. As shown in these photographs, minor reactions have occurred but these have not progressed sufficiently during the 10,000-hr exposure to cause concern about the integrity of the fuel element.

The reaction was characterized by two distinct metallographic structures. Figure 14-23, illustrating the fuel cladding interface of pin 13 at several locations, is typical of one type of reaction, wherein small, randomly-oriented voids appear in the cladding at the interface. Figure 14-24 shows the second type of metallographic structure at 17.6 in. from the bottom of pin 1. Here the reaction area seems to consist of a dark gray material (not the fuel) alternating with a light, metallic-appearing material. Figures 14-25 and 14-26 both show layers of the dark material with some evidence of the initiation of the void structure. Figures 14-27 through 14-29 all show variations of the non-continuous gray material and metallic structure mentioned above.

Neither of the observed structures resemble the gray deposit which occurred on the inner surfaces of the IB-8T cladding after embrittlement in the nitrogen atmosphere. The identity of the material was not established since this activity was not considered within the scope of the AGCRSP fuel development activity. The nature or extent of reaction product formation does not appear to be a function of the operating temperature although some tenuous correlation with fuel burnup could be established. The reaction was apparently not dependent on the type of fuel involved; both types of structures occurred with both BeO-UO_2 and UO_2 fuels.

f. Cladding Collapse: The possibility of the fuel pin cladding collapsing on the stack of fuel pellets in the ML-1 fuel elements was recognized during the developmental phase. This collapse would occur under the pressure differential which exists across the cladding if the strength of the cladding were insufficient to support the pressure loading. Such a collapse would cause problems only if the fuel stack contained large enough voids to permit bending of the cladding with resultant stress concentrations.

Cladding collapse has been studied by several workers (Ref. 14-2 and 14-3) in reactor programs which employed free-standing tubular fuel cladding. In particular, extensive experimental and theoretical work has been performed at the Oak Ridge National Laboratory with Inconel 600 and stainless steel cladding. These studies indicate that the cladding will collapse by a creep mechanism when subjected to external pressure at elevated temperatures.

Two series of tests were performed with Hastelloy X tubing at San Ramon. In the first series, ML-1 Hastelloy X fuel pin cladding was collapsed on widely spaced ceramic (Al_2O_3) pellets. In the second series, the tubing was collapsed on a fuel stack which contained ML-1 fuel pellets with size tolerances typical of those used in the assembly of the fuel elements. These tests revealed that non-uniform collapse caused high tensile strains in the cladding which resulted in accelerated oxidation.

24.9 in. from Bottom

1035°F

No Reaction

13.2 in. from Bottom

1335°F

Voids Extending 0.0008 in.

7.5 in. from Bottom

1430°F

Voids Extending 0.0012 in.

4.4 in. from Bottom

1530°F

Voids Extending 0.0014 in.

FIGURE 14-23. FUEL (UO_2)/CLADDING INTERFACES OF IB-8T-2, PIN 13 AT VARIOUS LOCATIONS

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FIGURE 14-24. FUEL (BeO-UO_2)/CLADDING INTERFACES OF IB-8T-2, PIN 1, AT VARIOUS LOCATIONS

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FIGURE 14-25. FUEL (BeO-UO₂)/CLADDING INTERFACES OF IB-8T-2,PIN,10 AT VARIOUS LOCATIONS

FIGURE 14-26. FUEL (BeO-UO₂)/CLADDING INTERFACES OF IB-8T-2,PIN 12, AT VARIOUS LOCATIONS

FIGURE 14-27. FUEL (BeO-UO_2)/CLADDING INTERFACES ON IB-8T-2, PIN 6, AT VARIOUS LOCATIONS

FIGURE 14-28. (UO_2) /CLADDING INTERFACE OF IB-8T-2, PIN 17

FIGURE 14-29. FUEL (UO_2)/CLADDING INTERFACE OF IB-8T-2, PIN 18

11.2-6A-1462

Two techniques were employed to evaluate the extent of cladding collapse in the IB-8T-2 test element. Specimens of the fuel cladding interface were examined metallographically and measurements of the outside diameter of the cladding were made for comparison with the original diameters.

The results of the metallographic examination are presented in Table 14-6.

TABLE 14-6 - RESULTS OF METALLOGRAPHIC EXAMINATION OF
FUEL-CLADDING INTERFACE, IB-8T-2

<u>Pin No.</u>	<u>Distance From Bottom, in.</u>	<u>Fuel Type</u>	<u>Average Operating Temperature, °F</u>	<u>Average Fuel/Cladding Gap, in.</u>
1	4.4	BeO-UO ₂	1530	0.0004
1	13.2	"	1350	0.0004
1	17.6	"	1240	0.0002
12	7.7	"	1430	0.0006
12	24.2	"	1135	0.0022
10	7.7	"	1350	0.0010
10	17.6	"	1180	0.0020
6	4.4	"	1250	0.0035
6	7.7	"	1240	0.0009
6	24.2	"	960	0.0020
13	4.4	UO ₂	1530	nil
13	7.7	"	1430	nil
13	13.2	"	1335	nil
13	24.2	"	1135	---
18	7.7	"	1430	nil
17	16.6	"	1180	0.0015*

*Extensive fuel drop out.

Typical photomicrographs of the sections examined are shown in Figures 14-30 through 14-33. In interpreting these results, the difference in the coefficient of linear expansion of the material involved must be considered, inasmuch as the collapse occurred at high temperature while the metallographic evaluation was performed at room temperature. The total linear expansion of the materials involved is shown in Figure 14-34. If it is assumed that the normal average temperature of the Hastelloy X cladding was 1600°F and that of the UO₂ pellets was 2400°F, cladding which was fully collapsed on the fuel at operating temperatures would be separated from the fuel by a diametral gap of 0.0003 in. after cooling to room temperature. However, no gap should be apparent between the BeO-UO₂ and the cladding upon cooling because the cladding

FIGURE 14-30. LONGITUDINAL SECTION THROUGH IB-8T-2, PIN 12, (BeO-UO₂ FUEL, 14 in. FROM BOTTOM) SHOWING FUEL-CLADDING INTERFACE

FIGURE 14-31. LONGITUDINAL SECTION THROUGH IB-8T-2, PIN 11, (BeO-UO₂ FUEL, 6.0 in. FROM BOTTOM) SHOWING FUEL CLADDING INTERFACE

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FIGURE 14-32. TRANSVERSE VIEW OF FUEL-CLADDING INTERFACE OF IB-8T-2, PIN 13, AT 4.4 - in. FROM BOTTOM

FIGURE 14-33. TRANSVERSE VIEW OF FUEL-CLADDING INTERFACE OF IB-8T-2 PIN 11 (BeO-UO₂ FUEL) AT 5.5 - in. FROM BOTTOM

11.2-64-1464

FIGURE 14-34. TOTAL LINEAR EXPANSION VERSUS TEMPERATURE FOR CORE MATERIALS

diameter would decrease 0.0007 in. more than the pellet diameter. The metallographic examination indicated that the cladding in the IB-8T-2 element had collapsed uniformly over the fuel and that non-uniform collapse was not likely even though significant cracking of the fuel had occurred. No evidence of accelerated oxidation was observed at any point in the fuel/cladding interface.

Measurements of the outside diameter of the fuel pin cladding of the IB-8T-2 test element is presented in Table 14-7. The diameter of the tubing at the time of fabrication was 0.241 ± 0.0015 in. It is apparent from an inspection of the data of Table 14-7 that the scatter is too large to permit any conclusion concerning the collapse of the fuel pin cladding.

14.3 Inner Liner Evaluation

Metallographic evaluation of the IB-8T-2 Hastelloy X inner liner disclosed that a structure had formed on both the inner and outer surfaces which closely resembled that noted on the inside surface of the cladding of certain IB-8T-1 fuel pins. A surface oxide film also formed on both the inside and outside surfaces.

The appearance of the inner liner at a point 2 in. from the bottom of the element is shown in Figure 14-35. The oxide scale on the inside surface is 0.0003-in. thick and that on the outside surface is 0.0004-in. thick. The gray structure resembling that seen in certain IB-8T-1 fuel pins was 0.0008-in. thick on the outside surface and varied from 0- to 0.0006-in. thick on the

TABLE 14-7 - OUTSIDE DIAMETERS OF IB-8T-2 FUEL PIN CLADDING

Measurement Location(1) In. From Top	IB-8T-2 PIN NUMBER - DIAMETER IN INCHES																		
	1	2	3	4	5	6	7	8	9	10	11(2)	12	13	14	15	16	17	18	19
4	2408	2405	2406	2391	2399	2400	2404	2410	2420	2399	2395	2403	2401	2399	2400	2398	2402	2412	2396
	2405	2409	2408	2393	2401	2403	2415	2405	2418	2395	2398	2402	2404	2395	2410	2400	2402	2408	2398
	2405	2407	2406	2385	2395	2400	2418	2402	2420	2395	2383	2404	2408	2392	2409	2405	2400	2410	2400
8	2406	2410	2410	2393	2400	2406	2412	2404	2419	2395	2397	2404	2402	2402	2428	2396	2402	2406	2399
	2404	2405	2410	2395	2400	2401	2411	2408	2420	2400	2400	2402	2402	2406	2422	2399	2405	2405	2395
	2404	2408	2409	2398	2402	2402	2413	2410	2420	2400	2402	2400	2409	2405	2419	2400	2405	2405	2393
12	2408	2406	2410	2403	2400	2400	2409	2405	2412	2396	2385	2402	2409	2400	2403	2400	2406	2402	2400
	2405	2405	2412	2391	2400	2400	2409	2410	2414	2398	2385	2400	2400	2403	2398	2400	2403	2404	2396
	2404	2408	2410	2405	2402	2401	2412	2408	2415	2398	2397	2403	2402	2405	2400	2402	2405	2405	2400
16	2401	2404	2402	2385	2400	2401	2412	2410	2410	2400	2395	2402	2398	2395	2412	2402	2402	2400	2404
	2400	2406	2408	2405	2398	2403	2410	2408	2414	2400	2398	2402	2400	2399	2406	2404	2402	2400	2400
	2420	2409	2405	2392	2395	2403	2414	2409	2413	2400	2397	2403	2400	2401	2410	2402	2402	2400	2403
20	2402	2406	2400	2398	2400	2403	2411	2408	2410	2396	2402	2400	2395	2400	2405	2400	2395	2400	2402
	2404	2409	2400	2385	2402	2405	2412	2410	2410	2400	2395	2399	2395	2394	2400	2400	2392	2396	2399
	2401	2408	2402	2398	2400	2402	2411	2410	2410	2404	2397	2402	2398	2398	2398	2400	2395	2395	2400
24	2404	2406	2400	2385	2400	2405	2410	2408	2412	2416	2390	2420	2392	2395	2400	2396	2392	2392	2390
	2405	2405	2399	2389	2399	2408	2411	2409	2412	2414	2385	2418	2395	2398	2403	2398	2395	2410	2395
	2401	2405	2402	2388	2400	2405	2412	2410	2414	2413	2398	2416	2398	2395	2398	2400	2395	2418	2398
28	2399	2398	2410	2397	2400	2405	2410	2406	2409	2400	2400	2410	2412	2406	2406	2399	2390	2403	2405
	2400	2392	2408	2399	2400	2408	2412	2410	2410	2399	2380	2410	2415	2405	2400	2400	2389	2405	2400
	2400	2395	2410	2403	2400	2404	2413	2406	2410	2396	2375	2408	2412	2405	2402	2400	2390	2408	2399
				2391							2383								
				2387							2385								
				2377							2380								

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(1) Three measurements taken 120° apart, accuracy of measurements: ± 0.0015 in.; to read dimensions multiply by 10^{-4}

(2) Eight measurements taken at equidistant intervals

FIGURE 14-35. IB-8T-2 HASTELLOY X INNER LINER, 2 in. FROM BOTTOM OF ELEMENT

FIGURE 14-36. IB-8T-2 HASTELLOY X INNER LINER, 8 in. FROM BOTTOM OF ELEMENT

inner surface. As shown in Figure 14-36, the extent of oxidation was reduced and no evidence of the gray structure is apparent in a cooler section of the liner.

These findings are typical of the results of evaluation of several liner sections taken in different locations. It was apparent that a reaction had occurred between the metal and the atmosphere in the lower, hotter portions of the inner liner. The extent to which the reaction proceeded is not considered to affect the integrity of the liner. The cause of the reaction is considered to be the same as that which resulted in embrittlement of some IB-8T fuel pin cladding. Reference gas entered the region between the liners and, after the oxygen was depleted by reaction with the metal surfaces, the initiation of the pi-phase occurred in the relatively pure nitrogen atmosphere.

14.4 Burnable Poison Evaluation

The ML-1-I burnable poison is a solid solution alloy of 1% cadmium and 99% copper clad with Kanigen nickel plating to eliminate oxidation and galvanic corrosion effects. Laboratory tests had demonstrated that the material was satisfactory for use in the fuel element when placed on the inside surface of the outer liner and centered at the mid-point of the fuel element. To facilitate fabrication of the IB-8T test elements, however, the burnable poison was placed directly against the outside surface of the inner liner at the bottom of the element. As a consequence, the poison foil experienced temperatures significantly above those predicted for the poison in the ML-1-I fuel elements.

A photograph of the metallographic section of the burnable poison foil taken longitudinally 13 in. from the bottom of the element is reproduced in Figure 14-37. Although the foil operated at temperatures exceeding those predicted for the ML-1-I burnable poison, the cadmium-copper matrix was not severely attacked during the 10,000-hr exposure. The nickel plating was still intact although several cracks were observed and the plating appeared to have separated from the matrix in one location. It appears that any corrosion arising from cracks in the plating proceeds along the cladding-matrix interface rather than into the matrix, thus leaving the burnable poison in good condition.

14.5 Fuel Evaluation

The nuclear fuels in the IB-8T-2 test element were identical with those used in the IB-8T-1. However, since internal thermocouples were not installed in this element, cored pellets were not used.

Samples of the fuel were prepared for metallographic evaluation from the specimens removed at the locations shown in Figure 14-1. Observed significant variations in the fuel structure were correlated with the burnup experienced at the location from which the specimen was taken. Figure 14-38 shows typical structures in the UO_2 fuel. Considerable cracking is evident in these photographs; these specimens were more severely cracked than those observed in the IB-8T-1 evaluation. A major point of interest in these photographs is the heavily "pulled-out" area on the left side. This side of the pin was closest to the GETR reactor and, because of the power skewing, experienced a higher neutron flux and consequently higher radiation damage than the opposite side of the fuel pin. The radiation damage is manifested as embrittlement which results in pull-out during polishing operations. Figure 14-39 illustrates

FIGURE 14-37. IB-8T-2 BURNABLE POISON 13 in. FROM BOTTOM OF ELEMENT

Approximate Fuel Temperature, 2100°F - 14.4 in. from Bottom

100X

Approximate Fuel Temperature, 1890°F - 13.2 in. from Bottom

100X

FIGURE 14-38. REPRESENTATIVE STRUCTURES OBSERVED IN THE UO_2 FUEL IN IB-8T-2 AT THE LEVELS INDICATED

24.2 in. from Bottom - Approximate
Burnup, 3.9×10^{20} fiss/cc

7.7 in. from Bottom - Approximate
Burnup 4.1×10^{20} fiss/cc

Views of Pin 13 showing variation in damaged area with distance along pin axis

Pin 17, 17.6 in. from Bottom - Approximate
Burnup, 4.5×10^{20} fiss/cc

Pin 14, 5 in. from Bottom - Approximate
Burnup, 3.6×10^{20} fiss/cc

FIGURE 14-39. LOW MAGNIFICATION VIEWS OF UO_2 FUEL FROM THREE IB-8T-2 PINS SHOWING VARIATION IN DAMAGE WITH BURNUP

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variations of damage, as evidenced by the degree of pull-out, along the axis of pin 13. By way of comparison, the photographs of sections of pin 14 (a low burnup pin) and a low burnup section of pin 17 are also presented in this figure. The high magnification views presented in Figure 14-40 show that essentially no variation in grain size exists between the edge and central portions of the fuel, indicating that the central temperature was not great enough to cause grain growth and that the temperature gradient from the center to the edge was small. This figure also shows that the size of the voids at the grain boundary has increased in the lower temperature sections of the fuel.

Representative microstructures of the BeO-UO₂ fuel are shown in Figure 14-41. It is evident from these photographs that the BeO-UO₂ fuel bodies cracked much less severely than was the case with the UO₂ fuel. It is also evident that, with increasing temperature and burnup, increasing difficulty was experienced in developing definite microstructures. In the highest temperature regions, it was exceedingly difficult to develop the structure of the UO₂ particles by etching; it was virtually impossible to develop the BeO structure except in extremely low temperature, low burnup regions. This problem is further illustrated in Figure 14-42 which shows the structure at the edge and center of the fuel at three locations in IB-8T-2 pin 1. Closer to the bottom of the pin, the detail is greatly reduced and, specifically, the definition of fine UO₂ particles is virtually absent. The larger UO₂ particles in these photographs appear as bubble-like formations. This appearance was also observed in samples prepared from high-burnup static capsule tests. The majority of the voids present in the photomicrographs of the central fuel were caused by specimen preparation.

Extensive evaluation of ceramic fuels has been performed by Westinghouse-Bettis in support of PWR technology (Ref. 14-4). In this work, great emphasis was placed on the behavior of UO₂ after extensive irradiation. In consideration of the differences which existed in operating temperatures and burnup rates, the IB-8T-2 data is substantially in agreement with the information published by Bettis.

14.6 Conclusions of Metallurgical Evaluation

The metallurgical evaluation of the IB-8T-2 fuel pin cladding revealed that oxidation penetration of the material was slightly greater than that which would have been predicted from the laboratory experiments but small enough (less than 0.005 in.) not to affect the performance of the cladding. The examination of the microstructure of the cladding alloy revealed no evidence of the π -phase observed from the IB-8T-1 cladding; the structures were essentially as predicted by the laboratory experiments. The tensile test data revealed that the cladding retained significant strength and ductility after irradiation. The microhardness data indicated that the material was somewhat harder than would be expected from the data developed in the laboratory but retained sufficient ductility to perform satisfactorily for 10,000 hr. An evaluation of the compatibility of the fuel and the cladding revealed that a slight reaction had occurred in this interface in some locations. In no case did the reaction proceed far enough to affect the performance of the materials. It was observed that the cladding collapsed uniformly about the fuel pellet, bridging gaps and cracks in the fuel without introducing strains in the cladding.

Edge

Nominal Temperature, 1560°F

Center

Nominal Temperature, 2100°F

Pin 13, 4.4 in. from the bottom 500X

Edge

Pin 14, 5 in. from the bottom

Center

FIGURE 14-40. EDGE AND CENTER VIEWS OF UO₂ FUEL FROM IB-8T-2

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Nominal Fuel Temperature, 1705°F

100X

Pin 1, 4.4 in. from Bottom

Nominal Fuel Temperature, 1540°F

100X

Pin 1, 13.2 in. from Bottom

FIGURE 14-41. REPRESENTATIVE STRUCTURES OBSERVED IN THE BeO-UO₂ FUEL FROM IB-8T-2 AT THE LEVELS INDICATED

4.4 in. from Bottom

Nominal Operating
Temperature, 1560°F

13.2 in. from Bottom

Nominal Operating
Temperature, 1365°F

24.2 in. from Bottom

Nominal Operating
Temperature, 1065°F Nominal Central
Temperature, 2100°F Nominal Central Fuel
Temperature, 1920°F Nominal Fuel Central
Temperature, 1370°F

FIGURE 14-42. STRUCTURAL CHANGES OBSERVED IN BeO-UO_2 FUEL (IB-8T-2 PIN 1) AT OPERATING TEMPERATURES INDICATED

The evaluation of the inner liner indicated that a reaction had occurred at the surface in some locations. It appears that the observed reaction is typical of the initiation of the formation of the pi-phase in Hastelloy X. The extent of the reaction was not sufficient to affect the performance of the liner. The evaluation of the burnable poison indicated that this material had performed very satisfactorily despite exposure to conditions significantly more severe than anticipated in the reactor.

The evaluation of the fuel revealed that, while substantial cracking had occurred in the UO_2 bodies, the fuel had performed satisfactorily during the 10,000-hr test. The appearance of the material correlated closely with that observed in specimens from static in-pile capsules and reasonable agreement with the Westinghouse-Bettis data was observed.

Ref. 14-1: R.L. Fullman, "Measurement of Particle Size in Opaque Bodies," Transactions of the AIME, Vol. 197, pp. 447-452, 1953

Ref. 14-2: "An Investigation of the Instantaneous and Creep Buckling of Initially Out-of-Round Tubes Subjected to External Pressure," ORNL-3299, January 30, 1963

Ref. 14-3: Maritime Gas-Cooled Reactor Quarterly Progress Report, GA-1259, March 31, 1960

Ref. 14-4: M. Bleiberg, "Effects of High Burnup on Oxide Ceramic Fuels," WAPD-T-1455, March 1962

APPENDIX A

AGCRSP BACKGROUND INFORMATION

This background of the Army Gas-Cooled Reactor Systems Program includes a short history of the Program, a description of the ML-1 power plant, and a selected bibliography.

A. HISTORY

The purpose of the Army Gas-Cooled Reactor Systems Program (AGCRSP) is to develop a mobile nuclear power plant for military field use. The current primary goal of the Program is the fabrication and test operation of a demonstration model of such a plant.

In 1955, at the request of the USAEC Division of Reactor Development, the Oak Ridge School of Reactor Technology performed a study which established the feasibility of the concept of a mobile, closed-cycle, gas-cooled nuclear power plant. Following this work, the Corps of Engineers Nuclear Power Field Office authorized the Sanderson and Porter Company to evaluate power conversion equipment and to prepare a conceptual design for the projected plant. At the conclusion of the Sanderson and Porter work, responsibility for development of the power conversion equipment for the plant was assigned to the Corps of Engineers and the development of the reactor was assigned to the USAEC.

As a result of the above arrangement, parallel programs were undertaken as follows:

- 1) The Corps of Engineers directed the Stratos Division of Fairchild Engine and Aircraft Corporation to develop a turbine-compressor set suitable for use in the projected plant. The construction of a test facility (GTTF) to evaluate the power conversion equipment was assigned to Aerojet-General Corporation. (The design of the facility was completed by Sanderson-Porter.)
- 2) Under the direction of the USAEC, Aerojet studied the feasibility of several concepts for the reactor to be incorporated in the power plant. The water-moderated concept was selected as the basis for development because of the modest extrapolation of technology required. Aerojet was

awarded the contract for the design and testing of a reactor based on this concept. At the same time, Aerojet was assigned responsibility for the design and construction of a test facility (GCRE) at NRTS.

Aerojet was designated as systems contractor and performance specifications for the demonstration power plant were evolved in June 1959. Fabrication of the reactor skid was completed in April of 1961, fabrication of the power conversion skid was completed in June 1962, and the power plant first operated as a unit (ML-1) in September 1962. Following a modification and checkout, the ML-1 power plant operated successfully for 101 hours in February and March 1963.

B. THE ML-1

The ML-1 is a closed cycle, gas-cooled nuclear power plant developed under the AGCRSP to demonstrate the feasibility of such a plant for military field use. During the design and construction, every reasonable effort was made to incorporate features into the plant which would be directly usable in the design of a field unit. However, since evaluation of the performance of the plant was a major requirement for the ML-1, a significant amount of additional instrumentation was provided. The physical arrangement of the equipment is such that the "prototype" components are easily identified as the:

- 1) Reactor Skid - a 15 ton unit containing the nuclear reactor and associated shielding and controls;
- 2) Power Conversion Skid - a 15 ton unit containing the power conversion equipment; and,
- 3) Control Cab - a 2-1/2 ton unit containing all the instruments and controls for operation of the field plant.

The ML-1 reactor consists of a calandria-type pressure vessel with appropriate inlet and exit gas ducts and plenums. Sixty-one pin-type BeO-UO₂ fuel elements are located in the tubes of the calandria. The demineralized water moderator surrounds the calandria and the six semaphore-type control rods operate in the moderator spaces between the calandria tubes. The entire reactor structure is supported inside a nine-foot diameter tank which contains heavy metal shields to permit relocation of the reactor within 24 hours after shutdown from extended operation, and a drainable (borated water) shield to attenuate radiation during reactor operation.

The plant working fluid (99.5 vol% nitrogen, 0.5 vol% oxygen) enters the reactor at 800°F and approximately 300 psia. The gas is heated to 1200°F in a single pass over the hot surfaces of the fuel elements. The moderator water is maintained at a temperature of 180°F; energy deposited in the moderator is removed in an water-to-air heat exchanger mounted on top of the power conversion skid. Provision is made for circulation, filtration and demineralization of the moderator water and for circulation and cooling (by exchange with the moderator water) of the shield water.

The hot gas leaving the reactor is expanded in a gas turbine which drives the compressor and alternator. The gas leaving the turbine passes through a regenerative heat exchanger (recuperator), through the system heat sink (air-to-air precooler) and to the compressor suction. The compressor discharges through the recuperator to the reactor inlet, thus completing the closed (modified Brayton) cycle. A lubrication system (including provision for recovery and removal of lubricating oil from the working fluid which leaks past the turbine compressor seals), the electrical switch gear, and miscellaneous power conversion hardware are mounted on the power conversion skid. An a-c, two-speed motor is coupled to the turbine compressor shaft to provide starting power to the set.

The following auxiliary systems are provided for the ML-1:

- 1) A deoxygenation system which removes dissolved oxygen from a bypass stream to maintain the moderator system oxygen content below 0.7 ppm.
- 2) An emergency cooling system which automatically injects a supply of coolant gas into the reactor in the event of a complete stoppage of working fluid flow.
- 3) A working fluid makeup system to compensate for normal leakage and to provide for initial charging of the system.
- 4) Waste gas storage facilities to accommodate the charge of radioactive gas in the loop in the event of an emergency.

C. BACKGROUND BIBLIOGRAPHY

The following bibliography provides information on sources of additional background to the material presented in this report. These documents trace the technical evolution of the AGCRSP from inception but do not, in general, document programmatic decisions. Such activity may be inferred from the technical approaches pursued and from the general background information presented in the reports.

The following reports were published by Aerojet-General Nucleonics, San Ramon, California under the Army Gas-Cooled Reactor Systems Program.

DOCUMENT NO.	TITLE	CLASSIFICATION
IDO-28505	<u>GCRE Semiannual Report, 1 November 1956 Through 30 June 1957, 20 February 1958</u>	CRD
IDO-28506	<u>GCRE-I Hazards Summary Report, December 1958, with three addenda, March 1959, February 1960, May 1960</u>	U
IDO-28519	<u>GCRE Semiannual Progress Report, 1 July Through 31 December 1957, 26 June 1958</u>	CRD
IDO-28526	<u>GCRE Semiannual Progress Report, 1 January Through 30 June 1958, 17 October 1958</u>	CRD

Report No. IDO-28616

DOCUMENT NO.	TITLE	CLASSIFICATION
IDO-28533	<u>AGCRSP Semiannual Progress Report, 1 July Through 31 December 1958, 28 February 1959</u>	CRD
IDO-28537	<u>Preliminary Hazards Summary Report for the ML-1 Nuclear Power Plant, 22 April 1959</u>	U
IDO-28542	<u>AGCRSP Semiannual Progress Report, 1 January Through 30 June 1958, July 1958</u>	U
IDO-28549	<u>AGCRSP Semiannual Progress Report, 1 July Through 31 December 1959, 22 December 1959</u>	U
IDO-28550	<u>The ML-1 Design Report, 16 May 1960</u>	U
IDO-28555	<u>ML-1 Transportability Studies, 23 March 1960</u>	U
IDO-28558	<u>AGCRSP Semiannual Progress Report, 1 January Through 30 June 1960, 11 July 1960</u>	U
IDO-28560	<u>Final Hazards Summary Report for the ML-1 Nuclear Power Plant, with four supplements, 5 August 1960</u>	U
IDO-28567	<u>AGCRSP Semiannual Progress Report, 1 July Through 31 December 1960, 17 December 1960</u>	U
IDO-28573	<u>AGCRSP Semiannual Progress Report, 1 January Through 30 June 1961, 10 August 1961</u>	U
IDO-28581	<u>AGCRSP Semiannual Progress Report, 1 July Through 31 December 1961, 31 January 1962</u>	U
IDO-28590	<u>AGCRSP Semiannual Progress Report, 1 January Through 30 June 1962, 24 August 1962</u>	U
IDO-28597	<u>AGCRSP, Study of the GCRE Tube Bundle Failure, 14 December 1962</u>	U
IDO-28602	<u>AGCRSP Semiannual Progress Report, 1 July Through 31 December 1962, 22 February 1963</u>	U
IDO-28607	<u>AGCRSP Quarterly Progress Report, 1 January Through 31 March 1963, 15 May 1963</u>	U
IDO-28612	<u>AGCRSP Quarterly Progress Report, 1 April Through 30 June 1963, 15 August 1963</u>	U
IDO-28617	<u>AGCRSP Quarterly Progress Report, 1 July Through 30 September 1963, 15 November 1963</u>	U

Report No. IDO-28616

<u>DOCUMENT NO.</u>	<u>TITLE</u>	<u>CLASSIFICATION</u>
IDO-28621	<u>AGCRSP Quarterly Progress Report, 1 October</u> <u>Through 31 December 1963, 27 January 1964</u>	U
IDO-28626	<u>AGCRSP Quarterly Progress Report, 1 January</u> <u>Through 31 March 1964, 15 May 1964</u>	U
IDO-28629	<u>AGCRSP Quarterly Progress Report, 1 April</u> <u>Through 30 June 1964, 27 May 1964</u>	U

APPENDIX B

ML-1 MILITARY CHARACTERISTICS

This appendix presents the subject characteristics as of 30 November 1960. No revisions have been made since that date.

- I. Title: Mobile Nuclear Power Plant, 300-500 kw, ML-1
- II. Security Classification:
 - A. Existence of Project: Unclassified
 - B. Military Characteristics: Unclassified
 - C. Plant Design and Performance: Unclassified
- III. Proposed Use: There are requirements within the military services for general purpose portable nuclear power plants with electric power outputs in the range from 300 to 500 kw. The ML-1 is the prototype of field units which will be procured to meet these requirements. Examples of military operations requiring a power source without dependence on continuous fuel or cooling water supply are: support of tactical missile systems, air head operations including field hospitals, remote locations, etc.
- IV. Environmental Characteristics: The plant shall have the inherent capability for acceptable performance under the Basic Operating Conditions, Extreme Cold Weather Operating Conditions, and Extreme Hot Weather Operating Conditions as established in Paragraphs 7a, 7b, and 7c of AR 705-15; and shall be capable of safe storage and transportation under conditions as established in Paragraph 7d of AR 705-15. A cold weather kit may be utilized for operation under extreme cold weather conditions.

V. Power Characteristics:

A. Nominal Output at "Design Conditions" (Ambient temperature -65°F to 100°F)

- (1) Electrical power output in the range of 300-500 kw, at 0.8 Power Factor
- (2) 2400/4160 volts, 3 phase, 60 cycles per second

B. Output at "Off Design Conditions"

- (1) Net output of the system in the event of hot weather (>100°F) or high altitude operation shall be limited only by the plant heat rejection capacity.
- (2) Operation shall be possible up to 500 kw net output except as limited by plant heat rejection capacity.

C. Power Quality

- (1) Objective: Steady state and transient frequency and voltage control characteristics as specified by MIL-G-14609 (CE).
- (2) Minimum requirement: Steady state and transient frequency and voltage control characteristics as specified by MIL-G-10328A (CE).

D. Other Requirements:

- (1) Automatic power level control is required.
- (2) The power plant shall operate satisfactorily in parallel with other units of similar rating.
- (3) The power plant shall be capable of 50 cycle operation with correspondingly reduced power output.
- (4) The power plant shall be treated for the elimination of interference with radio communication in accordance with applicable Signal Corps specifications.

VI. Field Operating Characteristics:

- A. The plant unit (connected reactor and power conversion packages) shall be assembled on a standard military semi-trailer (M-172 or M-172A1) and all preparations completed for relocation to an operating site within six (6) hours after unloading from any aircraft.
- B. The plant shall be capable of being installed and delivering rated power within twelve (12) hours after arrival at an operating site.

- C. The plant shall be capable of relocation to a new site beginning twenty-four (24) hours after reactor shutdown following operation for extended periods at full power. Completion of disassembly and loading shall be within thirty-six (36) hours after reactor shutdown.
- D. Nuclear Radiation allowed:
 - (1) Allowable radiation, twenty-four (24) hours after shutdown at twenty-five (25) ft from the reactor in the direction of the cab (without water shield), is fifteen (15) mr/hr with the power conversion package in place.
 - (2) No personnel shall receive greater than 3 rem/quarter.
- E. Equipment may be operated in any plane within five degrees (5°) from level.
- F. Control of the power plant will be from a separate shelter which shall be transportable as a unit on a standard $2\frac{1}{2}$ ton truck (M-35) which, during plant operation, shall be connected to the power plant by quick disconnect electrical cables of sufficient length (up to 500 ft) to permit flexibility in the relative locations of the power plant and control shelter.
- G. Plant design shall include necessary safety features for protection of personnel in the immediate area from results of reasonably conceivable mechanical, electrical, or nuclear malfunctions.
- H. During plant operation, radiation shielding integral to the plant may be supplemented by field expedient materials (earth, gravel, wood, water) sufficient to reduce radiation levels outside the combined shield to within dosage levels prescribed by the Surgeon General D/A. The integral plant shield (excluding supplemental expedient shielding) shall be adequate to reduce the residual radiation level following full power operation for extended periods of safe levels in time to permit relocation twenty-four (24) hours after reactor shutdown.
- I. The operating crew for three-shift operation of the field unit shall consist of no more than seven (7) men (one operator per shift plus mechanic-electrician, radiological safety technician and officer or NCO in charge). The crew will be specially trained for operation, maintenance, and plant installation at the site.

VII. Transportability Characteristics:

- A. Objective: Capability of being transported overland (highway and cross-country), by rail, by water, and by aircraft in accordance with AR 705-8, Transportability.
- B. Requirements: The ML-1 plant shall be consistent with the following transportation requirements:

- (1) General: The "power plant unit" is defined for transportability as the complete integral power plant including reactor, shutdown shielding, power conversion equipment and skid mounting. Control unit, auxiliary power unit, auxiliary equipment packages and trailer are not included in this definition. Required modes of transport are indicated below.
- (2) Primary mode of transportation:
 - (a) The primary mode of transport of the power plant unit shall be overland on standard military semitrailer. Power plant unit integrity is required.
 - (b) The control unit shall be one package which can be transported intact and which is suitable for both operation and transport on a standard military 2½ ton truck.
- (3) Secondary modes of transportation: (power plant unit integrity required).
 - (a) Secondary modes of transport for the power plant unit where unit integrity is required include:
 1. Normal railway freight service U.S. and foreign.
 2. Water freight by ship or barge.
 - (b) In addition, when necessary and where satisfactory clearances can be arranged, the power plant unit shall be transportable trailer-mounted on the above transit types.
 - (c) Similarly, when truck-mounted, the truck and control unit shall be transportable as one package when necessary and where satisfactory clearances can be arranged for the above transit types.
- (4) Secondary modes of transportation: (power plant unit integrity not required).
 - (a) The power plant unit may be separated into two packages for transport by:
 1. USAF C-130.
 2. USAF C-124C.
 3. Sled.
 - (b) Trailer mounting is not required for (a) above.

- (c) The control unit shall be transportable intact. Truck mounting is not required for (a) above.
 - (d) The complete power plant including the control unit and auxiliary equipment shall be transportable in one USAF C-133.
- (5) Other Requirements:
- (a) The power plant shall be transportable in Phase III of airborne operations.
 - (b) Design for air transport shall conform to requirements of USAF Specification MIL-A-8421A for transportation of material in military cargo aircraft.
 - (c) Provisions shall be made to permit rapid loading and unloading of each unit by standard weight handling techniques.
 - (d) Shock and vibration protection shall be provided integral with the plant consistent with each transport type as indicated below:
 - 1. Maximum shock transmitted to cargo bed of transport vehicle without loss of serviceability, utilizing a shock isolation tiedown system:

<u>Direction</u>	<u>Acceleration (g)</u>	<u>Duration of versed sine pulse, (sec)</u>
Fore & Aft	15	0.030
Forward Only	8	0.1
Fore & Aft	3	Static
Vertical Up	4	0.030
Vertical Down	4.5	0.1
Lateral	2	0.030

- 2. Maximum shock transmitted to cargo bed of transport vehicle without loss of plant serviceability, utilizing rigid tiedown:

<u>Direction</u>	<u>Acceleration (g)</u>	<u>Duration of versed sine pulse, (sec)</u>
Fore & Aft	5	0.1
Fore & Aft	3	Static
Vertical	4.5	0.1
Lateral	1.5	Static

3. Steady vibrations in transit without loss of serviceability.

<u>Source</u>	<u>Peak Amplitude (Inch)</u>	<u>Frequency (cps)</u>
Railroad Flatcar	0.12	20 - 30
Semi-trailer	0.05	>20
Ship	0.05	15 - 20
Aircraft	0.05	>20

(e) Weight:

1. Heaviest package for air transport is 15T.
2. Trailer load (M-172 or M-172A1) is 30T.
3. Control shelter is 2.5T.

(f) Size:

1. Largest package shall be transportable on C-124, C-130, and C-133 aircraft; and on railroad with limits as prescribed by the Berne International Clearance Diagrams.
2. Reactor and power conversion skids shall be transportable as a unit on the M-172 or M-172A1 semi-trailer.
3. Maximum height when trailer-mounted is 150 inches. Power plant height when trailer-mounted may be reduced to 132 inches for the purpose of passing low clearance obstacles outside CONUS.
4. Length and width shall be compatible with a standard military semi-trailer (M-172 or M-172A1).

- (g) When trailer-mounted and in transit, the plant shall have the capability of shallow fording in fresh or salt water, as defined in AR 705-2300-8. Extreme caution must be taken to avoid a nuclear accident due to core flooding.

VIII. Logistical Characteristics:

- A. The plant shall operate with no more than periodic field maintenance service for 10,000 hr between major overhaul.
- B. Routine plant maintenance shall be performed every 30 days or at greater intervals, as determined by operational testing.

- C. The plant shall have 50,000 hr total life.
- D. The plant shall be capable of full power operation for 10,000 hours between refueling operations. Refueling may be accomplished in the field.
- E. The plant shall be capable of operation without a continuous water supply. Any initial quantity of water (for shielding or reactor moderator) will be kept to a minimum and of a quality which can be supplied by standard military purification equipment capable of delivering 600 gph.
- F. Radioactive waste from the plant will be kept to a minimum. Disposal of radioactive waste will be in accordance with field procedures approved by the Chief Chemical Officer. (NOTE: Disposal will be subject to AEC approved procedures at NRTS.)
- G. The plant will be completely self-sufficient to permit steady state operation without the need for auxiliary equipment, except for process fluid storage and makeup. For startup and shutdown operations, the plant should require no more than a 45 kw diesel generator and necessary construction equipment for erection of the supplemental expedient shield. Equipment for expedient shield construction shall not be part of the overall plant equipment.
- H. The power plant shall be equipped with all special purpose tools required for normal maintenance and startup, but not for refueling. Special equipment and procedures required for field refueling shall be specified. The prototype plant shall be equipped with a Maintenance Package as referenced in AR 705-5 and AR 750-6. The field plant shall be equipped with complete operating manuals defining detailed operating, maintenance, installation and relocation procedures. The manuals shall conform to the Publication Standards and Style Guide for Corps of Engineers Equipment Publications.

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The information herein is regarded as preliminary and subject to further checking, verification and analysis.

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