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ADDENDUM TO HAZARDS SUMMARY REPORT

for the

GCRE CRITICAL-ASSEMBLY EXPERIMENTS

by

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TABLE OF CONTENTS

	<u>Page</u>
ABSTRACT	1
INTRODUCTION	1
GENERAL DESCRIPTION OF SITE AND FACILITY	5
SUMMARY OF PREVIOUS EXPERIMENTS	10
Experimental Core	10
Clean Core.	10
Lead-Reflected Core	13
Simulated-Flooded Core	13
Modified Lead-Reflected Core	13
Uniform Radial Power Distribution Core	14
ML-1A Core	14
Results of Previous Studies Pertaining to Operations and Safety	14
ML-1B EXPERIMENTS	14
ML-1B Power Reactor	16
ML-1B Critical Assembly.	16
Control and Operation	17
Safety Characteristics	18
General Limitations on Experiments	18
Control Limitations	18
Power Limitations	20
Limitations on Core Assembly Operations	20
Precritical Operation Checkout	20
EXPERIMENTAL PROGRAM.	21
Experimental Procedures	22
Approach to Criticality	22
Critical Mass	23
Reactivity Effects	23
Temperature Coefficient of Reactivity	24
Flux and Power Distributions	24
Control-Blade Evaluation	24
ANALYSIS OF SAFETY ASPECTS	25
Hazards During Normal Operation	25
The Maximum Credible Accident	26
Other Possible Incidents	27
Conclusions From Safety Analysis	27
APPENDIX	
REACTOR BEHAVIOR UNDER ABNORMAL OPERATION	A-1

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Critical experiments are described for the ML-1B, a gas-cooled, water-moderated reactor. The ML-1B is part of the Army Gas-Cooled Reactor Program and is a continuation of previous GCRE critical-assembly studies.

The experiments performed with other cores in the GCRE program are listed and results which are applicable to safety are presented. The experimental program and procedures for the ML-1B study are outlined.

The safety aspects of the present system have been reviewed and several power transients resulting from the sudden insertion of rather large amounts of reactivity have been carried out. It appears quite evident that this critical-assembly program can be conducted with reasonable assurance of safe operation and that no public persons are jeopardized by its operation.

Consequently, approval from the Commission is requested to proceed with this experiment as described.

INTRODUCTION

This report reviews the safety aspects of critical experiments for the ML-1B. These experiments are an extension of the studies reviewed in BMI-1240 (Rev.), "Hazards Summary Report for the GCRE Critical-Assembly Experiments", July 22, 1958.

Battelle has been conducting critical-assembly experiments since November, 1957, for Aerojet-General Nucleonics as a part of the Army's Gas-Cooled Reactor Program. In this program a test-reactor facility, the Gas-Cooled Reactor Experiment (GCRE), will be constructed at the National Reactor Testing Station in Idaho to study the reactor components at operating conditions.

The core of the GCRE-1 consists of two gas plenums connected together by 61 pressure tubes which contain the fuel elements. The entire assembly is submerged in water which serves as moderator and primary radiation shield. A fast-neutron reflector of lead 4 in. thick will be placed around the core assembly as a result of the initial critical-assembly experiments. The plenums and pressure tubes are insulated on the gas side and cooled by the surrounding, unpressurized water.

In the original design (GCRE-1), the fuel elements consisted of four concentric fueled cylinders varying in diameter from 0.5 to 1.25 in. The cylinders were formed from plates of UO_2 dispersed in stainless steel clad on both sides with stainless steel. The plates were 0.045 in. thick with a 0.033-in. core of 30 w/o UO_2 -stainless steel dispersion. The plates were cut to proper width and rolled to form 120-deg sectors of the cylindrical fuel tubes; each fuel cylinder was made up of three of these sectors held together with a mechanical-clip arrangement.

Fuel elements consisting of four fueled cylinders were contained in the pressure tubes and suitably insulated to reduce heat transfer to the water moderator. The entire assembly was 1.875 in. in OD with an active length of 28 in.

The GCRE-1 was to be controlled by blades of tungsten inserted horizontally between the pressure tubes. A 6-in.-long cylinder of steel containing samarium oxide placed around the insulation close to the reactor midplane was to serve as a burnable poison.

The core assembly, fuel elements, and control systems of the GCRE-1 were mocked-up in a critical assembly. The fuel tubes were simulated by wrapping 1-mil highly enriched uranium foil on stainless steel tubes of four different diameters. A concentric arrangement of these "fuel" tubes and other tubes which mocked-up structural components was placed within a tube simulating the pressure-tube wall. These assembled fuel elements were positioned at top and bottom by aluminum support plates. The gas-plenum assemblies were mocked-up at one end by a stainless steel cylinder 2 ft in diameter and 2 ft high located under the core and at the other end by the air space above the core. Figure 1 shows the core of the critical assembly with several fuel elements in place. This entire assembly was contained in a tank mounted on the assembly stand shown in Figure 2.

Mock-up GCRE-1 control blades were supported rigidly at the periphery of the critical-assembly core and were moved manually. The critical-assembly control rods were vertically acting driven rods; a sandwich construction of stainless steel and cadmium sheets was used. For loading and making other adjustments to the core, shutdown safety was assured by draining water from the core tank.

The critical assembly was normally operated at room temperature but facilities for heating the moderator water were incorporated so that temperature coefficients could be obtained.

Four different core configurations were constructed during the course of the initial program. These cores were the (1) Experimental Core, (2) Clean Core, (3) Lead-Reflected Core, and (4) Simulated-Flooded Core. All these reactors had a 4-1/2-in. lower axial water reflector surrounding the fuel-tube extensions. Except for one reflector-worth experiment, the upper 2-3/4-in.-thick aluminum positioning plate had its lower surface aligned with the top edge of the fuel, which was also the water height in normal operation. The cores differed only in details and in the radial-reflector design. The Simulated-Flooded Core was used to investigate the refueling condition in the GCRE-1 when the gas plenum is removed and all fuel elements are flooded. The shutdown worth of the safety blades must be sufficient to maintain the core in a non-critical condition.

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FIGURE 1. INSTALLATION OF GCRE-1 CRITICAL-ASSEMBLY CORE

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FIGURE 2. CRITICAL-ASSEMBLY STAND SHOWING MAJOR COMPONENTS

In an attempt to improve the performance of the basic GCRE-1 design Aerojet-General calculated a fuel-element spacing arrangement which was intended to flatten the radial power distribution. For these calculations, it was assumed that the fuel-element construction and core-material composition were unchanged, and that the lead reflector was uniformly at a distance of 1/2 in. from the peripheral fuel elements. The fuel-element lattices for the (calculated) Uniform Radial Power Distribution Core and the Lead-Reflected Core are compared in Figure 3. The moderator-to-fuel ratio in the Uniform Radial Power Distribution Core increases with radius. Compared with the constant value of this ratio for the Lead-Reflected Core, it is smaller in the central region and larger in the outer regions.

The ML-1 (Mobile, Low power) reactor plant represents the first merger of the high-temperature, gas-cooled, water-moderated reactor with compact power-conversion equipment in a closed-cycle, gas-turbine power plant. The ML-1A was designed to use the tubular fuel elements previously described which were to be tested in the GCRE-1 facility prior to their use in the ML-1A. The core configuration for the ML-1A is a slight modification of the Uniform Radial Power Distribution Core as can be seen from Figure 4. The critical assembly for the ML-1A core used essentially the same fuel elements and core components. Since the new core modification did not represent a change in the operational or safety characteristics of the GCRE-1 experiment, no formal hazards analysis was made. Of course, to meet laboratory rules each specific experiment was analyzed and written up prior to operation.

A list of experiments on GCRE-1 and ML-1A which have been completed is presented in a later section.

A recent decision to reduce plant weight has necessitated a decrease in core length from 30 in. (28-in. active length) to 24 in. (22-in. active length). The reduced core size requires a larger critical mass and, consequently, a higher fuel loading. Since the fuel loading (30 w/o UO_2) was already quite high for a metal dispersion, it was decided that the 1B fuel (UO_2 pins) presently under development should be substituted for the 1A dispersion element. A sketch of the ML-1B fuel element is shown in Figure 5.

It appears probable that certain operating and safety characteristics of the ML-1B critical assembly may differ from those of the previous assemblies. Consequently, an analysis of the procedures and shutdown mechanisms has been made and the results are reported in this addendum. To place the new experiment in proper perspective, a list of the previously conducted experiments and some results are presented.

GENERAL DESCRIPTION OF SITE AND FACILITY

A complete description of the facility and site along with details on meteorology and hydrology have been presented in several previous reports. The site information is summarized here; the complete description can be found in BMI-1240 (Rev).

Briefly, the Critical-Assembly Laboratory is located on a 500-acre tract at the Battelle Nuclear Research Center about 2 miles northeast of West Jefferson, Ohio. The

FIGURE 3. COMPARISON OF FUEL-ELEMENT LATTICES FOR THE LEAD-REFLECTED AND CALCULATED UNIFORM RADIAL POWER DISTRIBUTION CORES

Shaded portions are those of the Uniform Radial Power Distribution Core.

FIGURE 4. ML-1B CRITICAL-ASSEMBLY

A-32454

FIGURE 5. ML-1B FUEL-ELEMENT ASSEMBLY

Burnable poison is not shown.

surrounding area is farm land with about 60 year-around residents in an area enclosed by a circle of 1-mile radius. The closest dwellings are about 3000 ft away.

The laboratory is a two-story building made of aluminum Q-metal and brick. It contains a large (40 by 40 by 50 ft high) assembly room, vault, counting room, control room, shop, and offices. The vault, assembly room, and control room form a security area which is shut off from the remainder of the building. The entire area is supplied with the usual fire and safety equipment and film and indium-foil detectors.

At the site the prevailing-wind direction is from the southerly quadrant blowing 41 per cent of the time from the SE through SW. Expressed as one of the 16 compass points, the most frequent wind directions are from the S and SE, both occurring 10 per cent of the time. The wind speed is quite variable during a 24-hr period; averaged annually 58 per cent of the winds occur in the wind-speed group between 4 and 12 mph. The annual average is 8.4 mph.

The conclusion of a hydrology and geology report was that "the Battelle site seems to be almost entirely safe for the operation of a nuclear reactor with respect to the effects on ground-water resources".

SUMMARY OF PREVIOUS EXPERIMENTS

Table 1 lists the major experiments performed during the past 15 months. Seven basic core configurations have been investigated. These are described briefly below to indicate the experience gained with this type of assembly. The procedures used in conducting these experiments will be used again in the present program. Results of the experiments which are significant to operation and safety are also presented.

Experimental Core

The Experimental Core consisted of an array of fuel elements without the strip of burnable poison. There were 69 fuel-element positions available. A water reflector surrounded the core. The Experimental Core provided immediate data to aid in "normalizing" physics calculations and evaluating the worth of the simulated burnable-poison sheets.

When the burnable-poison sheets became available, this core configuration was dismantled to set up the Clean Core.

Clean Core

The simulated burnable poison was added to the fuel elements in this core. A cadmium layer approximately 0.00012 in. thick was plated on a steel foil and wrapped in the insulation-liner tube of each fuel element. The midplane of foils was 1 in. above the horizontal midplane of the fuel.

TABLE 1. EXPERIMENTS PERFORMED IN THE CRITICAL-ASSEMBLY PROGRAM

Experimental Core	Clean Core	Lead-Reflected Core	Simulated-Flooded Core	Modified Lead-Reflected Core	Uniform Radial Power Distribution Core	ML-1A Core
Worth of	Worth of	Worth of	Worth of	Worth of	--	Worth of
1. Upper axial annular reflector	1. Peripheral fuel elements	1. Peripheral fuel elements	1. Peripheral fuel elements	1. Peripheral fuel elements		1. GCRE-1A production elements
2. Upper axial reflector		2. Polyethylene, paraffin, and mineral oil as compared with water in an element position		2. Additional fuel (101 g per element)		2. GCRE-1B element (19 UO ₂ pins)
3. Peripheral fuel elements		3. Axial reflector materials		3. Additional stainless steel (385 g per element)		
4. Peripheral empty fuel cans		4. Mock-up blade guides				
M ² in two configurations	--	--	--	Migration area	--	Intracell flux and power distribution in GCRE-1B element
--	Temperature coefficient of reactivity (20 to 75 C)	Temperature coefficient of reactivity (20 to 75 C)	--	--	--	Temperature coefficient of reactivity
Worth of mock-up shim and shutdown blades at two positions in the core	Effects of changes in size and position of a mock-up shutdown blade	Worth and geometric effect of mock-up shim and shutdown blades at one location in the reactor	Critical configuration with a variety of mock-up shutdown-blade configurations	--	--	Worth of scissor-action control blade
2-D flux distribution with bare and cadmium covered wire	2-D flux distribution with bare and cadmium-covered wire	2-D power distribution	Axial flux distribution in the center fuel element	2-D flux measurements with bare and cadmium-covered wire	2-D flux measurements with bare and cadmium-covered wire	2-D flux measurements with bare and cadmium-covered wire
2-D power distribution with catcher foils	Axial, radial, and azimuthal power variation in the vicinity of the cadmium band on two fuel elements	Detailed power distribution within fuel elements at the center of the core and the periphery	Detailed flux distribution within and adjacent to the central fuel element with bare and cadmium-covered wires	--	2-D power distribution with catcher foils	2-D power measurements with catcher foils

TABLE 1. (Continued)

Experimental Core	Clean Core	Lead-Reflected Core	Simulated-Flooded Core	Modified Lead-Reflected Core	Uniform Radial Power Distribution Core	ML-1A Core
Detailed flux distribution within and adjacent to fuel element with bare and cadmium-covered wire		Detailed perturbed power distribution near a mock-up shim blade Power distortion caused by mock-up blade guides		Thermal utilization	Thermal utilization	

This core was a good mock-up of the GCRE design at the time of the experiments, and therefore, a considerable amount of testing was done with the Clean Core. However, the excess reactivity of this core was not sufficient to provide for poison override, burnup, and operating control, so a modified reactor design was suggested to overcome this difficulty. This modification led to the construction of the Lead-Reflected Core.

Lead-Reflected Core

The fuel elements in this core were identical with the Clean Core elements. A 4-in. -thick lead reflector was added at the core perimeter 1/2-in. from the fuel elements. The lead billets making up the reflector were 24 in. long and their midplane coincided with the fuel midplane.

Two lead-reflected configurations were constructed. In the first, the lead surrounded 51 fuel elements. When this did not provide a critical configuration, the reflector was rearranged to roughly surround 61 elements. After preliminary tests with this arrangement the lead was reoriented to surround the 61 elements with a more uniform 1/2-in. gap between the lead and perimetral fuel elements. The predicted reactivity effect of the lead reflector was borne out, so this reactor design became the new reference GCRE-1 concept. Accordingly, a number of experiments, primarily of an engineering nature, were performed with this reactor configuration.

Simulated-Flooded Core

Whenever fuel elements are changed in the GCRE-1 core the upper gas plenum will be removed and all the fuel elements will be flooded. The reactivity worth of the control blades must be sufficient to maintain the core in a noncritical condition. To simulate flooding, each fuel element was filled with 1100 cm³ of mineral oil with a molecular formulation approximating C₂₀H₄₂ and a density of 0.83 g per cm³. A water-simulating material was used because uranium foil corrodes rapidly in water. The volumetric hydrogen density of mineral oil is about 10 per cent greater than that of water. The tests were performed with the lead reflector surrounding 61 element positions.

Modified Lead-Reflected Core

In separate experiments, the Lead-Reflected Core was modified by adding fuel to each fuel element and then, with the initial fuel loading, stainless steel to each element. The fuel was added to each of the four fuel cylinders by wrapping on an additional 1-mil-thick uranium foil, increasing the fuel content per element from 303 g to 404 g. A total of 385 g of stainless steel was added to each fuel element (with 303 g of uranium) by placing 0.010-in. stainless steel sleeves around each fuel tube.

Uniform Radial Power Distribution Core

This core configuration was calculated to give a flat radial power distribution. The separation between fuel elements increases with radius and, consequently, the moderator-to-fuel ratio increases with increasing radius. Compared with the constant value of this ratio in the Lead-Reflected Core, the ratio in the URPD core is smaller near the center and larger in outer regions. A schematic of this core configuration is shown in Figure 3.

ML-1A Core

The ML-1A core is a slight modification in lattice spacing from the URPD core, as can be seen by comparing Figures 3 and 4. The upper and lower tube sheets are stainless steel as are the pressure tubes. Initially, these parts were made of aluminum. A new lead reflector was machined to fit with greater accuracy around the hexagonal core.

Results of Previous Studies Pertaining to Operations and Safety

The experiments listed in Table 1 were conducted using techniques outlined in BMI-1240 (Rev.) and summarized in a later section. The experimental techniques have proven effective and safe and the reactor behaved as predicted.

The results of a number of the experiments in Table 1 provide a comparison of values calculated for the safety analysis prior to operation. Table 2 summarizes these comparisons. Data are included for the GCRE-1 (Lead-Reflected Core) and the ML-1 critical assemblies. It can be seen from this table that reasonable predictions were made for all important parameters. Where differences exist they were generally predicted conservatively. One exception was the slow temperature coefficient which was found to be larger than predicted. This difference does not increase the hazards since the slow coefficient does not contribute to the course of a fast excursion.

The last column of Table 2 gives estimates of the quantities important to the safety of the ML-1B critical assembly. The parameters depend almost exclusively on the fuel-element lattice spacing. Accordingly, it is understandable that the new predictions are similar to the measured values for the ML-1A.

ML-1B EXPERIMENTS

The ML-1B reactor is described below. Following this, the ML-1B critical assembly is described and its characteristics and safety features discussed.

TABLE 2. COMPARISON OF SOME SAFETY CHARACTERISTICS OF THE GCRE-1 AND ML-1 CRITICAL ASSEMBLIES

Quantity	Value Predicted for GCRE-1 and ML-1A	Value Measured		Value Predicted for ML-1B
		GCRE-1	ML-1A	
Void Coefficient at Top of Core	--	--	$-6 \times 10^{-7} \Delta k/k/cm^3$	$-6 \times 10^{-7} \Delta k/k/cm^3$
Moderator Temperature Coefficient	$10^{-5} \pm 1.5 \times 10^{-5} \Delta k/k/C$	$8 \times 10^{-5} \Delta k/k/C$ at 25 C 0 at 67 C	$1 \times 10^{-4} \Delta k/k/C$ at 25 C	$10^{-4} \Delta k/k/C$ at 25 C
Fuel Temperature Coefficient	$-5.9 \times 10^{-6} \Delta k/k/C$	0 at 120 C	--	Less than 10^{-6} $\Delta k/k/C$
Water-Height Reactivity Coefficient (Reactor Tank Filled)	$0.0045 \Delta k/k/in.$	$0.0037 \Delta k/k/in.$	$0.0037 \Delta k/k/in.$	$0.004 \Delta k/k/in.$
Time to Uncover Core by Dumping Water	20 sec	24 sec	--	20 sec
Fuel-Element Worth at Core Edge	$0.003 \Delta k/k$	$0.0039 \Delta k/k$	$0.0045 \Delta k/k$	$0.004 \Delta k/k$
Maximum Worth of Flooding a Fuel Element	$0.003 \Delta k/k$	Less than $10^{-4} \Delta k/k$	--	$10^{-4} \Delta k/k$
Worth of Doubling the Quantity of Fuel in an Element	$0.0045 \Delta k/k$	--	$0.003 \Delta k/k$	$0.003 \Delta k/k$
Air-Borne Fission-Product Release During Normal Operation	Negligible	Negligible	Negligible	Negligible
Operator Gamma Dose Rate for 40 W-Hr Operation	0.047 MPE	0.03 MPE	--	0.03 MPE

ML-1B Power Reactor

The general features of the ML-1B reactor core will be similar to those of the GCRE-1 and ML-1A. The only significant changes from previous designs will be in the fuel element and in the length of the core. The core structure will consist essentially of two gas plenums with 3-in.-thick steel tube sheets separated by stainless steel pressure tubes which contain the fuel elements. The volume around the pressure tubes will be filled with water which will serve as a moderator. The entire core will be surrounded by a thick lead annulus to provide fast-neutron reflection and gamma shielding.

The fuel element of the ML-1B reactor will consist of 19 Inconel X tubes 0.170 in. in ID arranged in a hexagonal array and containing UO_2 as fuel. The uranium in the center 7 pins will be enriched to 93.4 per cent in the isotope uranium-235. The outer 12 pins will contain 50 w/o highly enriched UO_2 mixed with BeO . The oxides will be compacted to about 90 per cent of theoretical density.

The core length between tube sheets will be 24 in. with an active length of 22 in. The top of the fuel will be at the same level as the bottom of the upper tube sheet. The lower end of the fuel will be 2 in. above the tube sheet which will provide a 2-in.-thick axial water reflector around the tube extensions. The operating core will contain about 61 elements and roughly 45 kg of uranium-235.

Control and shutdown will be provided by six pairs of scissor-action blades which will enter the core between the pressure tubes as shown in Figure 4. The blades will be stainless steel 3 in. wide wrapped with 0.020 in. cadmium. A 4-in.-wide band of samarium oxide will be wrapped inside the pressure tube at the reactor midplane as a burnable poison.

ML-1B Critical Assembly

The fuel pins in the critical assembly will consist of 0.025-in.-wall Inconel X tubes 0.250-in. OD by 24 in. long with a 22-in. fueled section (see Figure 5). Nineteen of these tubes will be arranged in a hexagonal array to form a fuel element. The fuel in the center 7 pins will be fully enriched UO_2 compacted to about 75 per cent of theoretical density. (The actual density will depend on the value obtainable without sintering.) The outer 12 pins will contain 50 w/o of enriched UO_2 mixed with alumina (Al_2O_3) with the mixture compacted to about 75 per cent of theoretical density. The combination of inside tube diameter and density will be chosen to duplicate the fuel loading in the ML-1B pin. The density and diameter given here correspond to a fuel loading of 76 g of uranium-235 in each of the inner 7 pins and 19 g of uranium-235 in the outer 12 pins.

The fuel pins will be placed in stainless steel pressure tubes (1.875-in. OD by 0.030-in. wall) used previously in the ML-1A after they have been cut down to a 24-in. length. The tube sheets will be mocked-up by 3-in.-thick steel plates which will support and position the pressure tubes containing the fuel pins. The general arrangement is shown in the elevation in Figure 4. Externally, the core structure will look like Figure 1 and be located on the multilevel stand shown in Figure 2.

Control and Operation

The critical assembly will be controlled by three safety blades 5-3/4 in. wide and one regulating blade 1-1/4 in. wide. The blades have been used in the other experiments and are sandwiches of steel and cadmium 0.2 in. thick. These blades will be driven vertically and can be released by interrupting the current to a magnetic clutch. In previous calibrations, the wide blades have been worth 1.2 to 1.7 per cent and the narrow ones 0.4 to 0.5 per cent $\Delta k/k$.

The motor-clutch-type control-blade drive unit will include two Selsyn position indicators, a rack-and-pinion gear, and a hydraulic shock absorber. Although each blade will have a drive unit, there will be only one power supply; a selector switch will make power available to one rod at a time.

In operation all blades will normally be withdrawn (with the possible exception of a small amount of regulating blade) which will limit the available excess reactivity. The maximum drive speed will be adjusted so that reactivity will be added at less than 0.04 per cent $\Delta k/k/\text{sec}$. As an added safety procedure, operation will be restricted to cores where the core (with water) will be subcritical with safety blades worth 2 per cent or more completely withdrawn.

Water will be pumped into the reactor tank by a 30-gpm (maximum capacity) centrifugal pump. When the water level approaches the top of the core, this rate will be reduced to about 20 gpm because of the water head. Water will be dumped through two quick-acting 4-in. valves (air-to-close, spring-to-open) into the storage tanks. The core can be completely uncovered in 24 sec. Water level will be indicated to $\pm 1/4$ in. by a Tank-o-meter pressure-type instrument. More accurate readings, to 0.01 in., will be available from an electronic liquid-contact device. Both units have been operated previously.

The water level in the critical condition is supposed to be even with the top of the fuel. Consequently, an overflow pipe has been installed at this level to prevent accidentally adding water reflection above the core. However, the 3-in. steel plate at the top will serve as a fast reflector and will also prevent water from rising around the tops of the pressure tubes (see Figure 4).

Care will be taken to insure that a critical condition is not obtained by pumping water but by withdrawing blades. The interlock system will be arranged so that blades of sufficient reactivity worth to maintain the reactor in a subcritical condition must be full-in while water is being added. The interlock system will also demand that rods worth 2 per cent or more be completely withdrawn to provide shutdown in case of error.

Because of this method of operation, accurate data were not obtained on reactivity as a function of large changes in core height in the previous experiments. Estimates based on the effect of small changes in water height on reactivity indicate that the reactivity is increasing at less than 0.02 per cent per sec as the water level approaches the top of the core.

The critical assembly will be operated from the console in the second-floor control room. In this console all meters and controls required to operate the assembly are located on a panel directly in front of the operator. The rod-position indicators are on an inward-sloping panel immediately above the meters. Control instrumentation

consists of three neutron-sensitive ion chambers and a scintillation detector for gamma-ray monitoring. Two of the ion chambers are connected to logarithmic electrometers and a period signal is obtained as well as a $\ln n$ signal. All four detectors are connected into scram circuits and any one can trip all rods. The two linear circuits are connected to meters on the console to give a visual indication of power level.

The laboratory rules, as stated in BMI-1240 (Rev.) require that three level and one period scram will be operative at all times.

Safety Characteristics

The scram and interlock circuits are entirely switch and relay circuits. They are designed to actuate when relays open except for the contact-meter relays mounted on the console. Electrical interlocks prevent:

- (1) Entering the assembly room when the reactor is operating or in an operating condition
- (2) Startup without a neutron source in the reactor
- (3) Pumping water without a safety rod cocked
- (4) Moving rods and pumping water at the same time.

Figure 6 is a schematic of the entire interlock and safety system. The only interlock that can be bypassed is the one on the main door to the assembly room. The door interlock is bypassed mechanically during instrument checkout. The door cannot be closed properly with the mechanical interlock in place and a flashing light on the console indicates this condition; thus, there is adequate visual indication of this bypass.

The prompt-neutron generation time for this core has been estimated to be 45×10^{-6} sec. However, this parameter does not enter into the calculations on shutdown characteristics. Previous experiments indicate that the moderator temperature coefficient probably will be positive ($\sim 10^{-4} \Delta k/k/C$) at room temperature but will be negative above 60 to 70 C. In a transient, this coefficient has little effect because of poor heat transfer from the fuel. The prompt temperature coefficient (heating the fuel elements only) is probably very small; for the other cores it was 0 at 120 C within the limits of experimental error.

The probable shutdown mechanism in this assembly is radiolytic gas formation in the moderator. The higher melting point and heat capacity of the fuel (UO_2 compared to the uranium foil in the other assemblies) saves the fuel from vaporizing before shutdown by gas formation. This subject is discussed in a later section.

General Limitations on Experiments

Control Limitations

A basic limitation on all experiments and operations will be a maximum rate of change of reactivity of 0.04 per cent $\Delta k/k/\text{sec}$. This limitation will apply to all controls

(A) To scram relay causing a scram if door is opened during reactor operation. (B) To warning light on console to indicate door open at any time. (C) To interlock light on console to indicate open interlock circuit. (D) To interlock relays to prevent withdrawal of control rods. (E) To interlock relays to prevent filling the core tank with water. (F) To interlock relays which prevent operation of core-material reactivity changing device.

FIGURE 6. FUNCTIONS OF INTERLOCK AND SAFETY SYSTEM

which may be operated remotely. This includes control blades and the moderating liquid when the reactor k_{eff} is greater than about 0.95.

The reactor will not be taken critical with less than 2 per cent $\Delta k/k$ available for quick shutdown. This limitation can be satisfied by interlocking so that adequate control blades must be withdrawn from the core and poised above the fuel region before the startup procedure can continue.

Power Limitations

The ML-1B critical assembly will normally be operated in the power range from 1 megawatt to 1 w. For activation experiments the core may be operated at 20 to 40 w for short periods of time.

The power-level scrams are normally preset at 0.1 to 1 w depending on the requirements of the particular experiment. In experiments where foils or wires are activated, the power-level scrams will be set at 50 to 100 w. In all cases the period trips will be set to scram the reactor if the period is less than 4 sec. The period trips are set at the same value regardless of the experiment.

Limitations on Core-Assembly Operations

Assembly operations are considered a part of the normal operations since it is anticipated that several different core configurations will be required during the course of the experiment. Procedures and controls will be adopted to guard against criticality hazards during the construction. The procedures to be used during construction are described in Supplement No. 4 to Procedures Manual BMI-PM-607. Only general features of the procedures are outlined here.

- (1) The amount of uranium-235 handled at any one time will be limited to the quantity contained in two fuel elements.
- (2) The location of the uranium and the amount assembled at any one point will be carefully controlled by having a flow and balance record at each point where material is handled.
- (3) A supervisor will be appointed who will have responsibility for each fuel-handling operation.

In addition to these precautions, the multiplication of the reactor will be measured, recorded, and evaluated during assembly of the core. Since assembly will be carried out with all liquids completely drained from the reactor tank the multiplication is expected to be very small.

Precritical Operation Checkout

Prior to each day's reactor operation, and more often if the supervisor deems it necessary, a complete check will be made of the instrumentation and safety mechanisms of the reactor. Each radiation detector will be checked with a neutron source. Each scram circuit will be checked and each scram mechanism will be operated using a source

to initiate the scram. The interlocks will be checked by attempting operation contrary to the design procedure. Following this, a visual check will be made of the facility and the room, the door to the assembly room will be closed, and the checkout form will be signed by the person making the check.

EXPERIMENTAL PROGRAM

In the following sections, the experimental program planned for the ML-1B critical assembly is outlined. A brief description of procedures is also included. In general, the program is the same as the previous program for the GCRE and the techniques have been used numerous times.

Table 3 is a complete listing of the experiments planned for the ML-1B. Most of these are rather routine experiments such as critical fuel loading, two-dimensional flux plots, intracell flux measurements, etc. The void-coefficient studies will be conducted to investigate the effect of voids at the center of the core and at the water-lead reflector interface. Fuel-element-exchange studies will investigate changes in composition of the fuel element, i. e., additional fuel and structural materials.

TABLE 3. EXPERIMENTS WITH THE ML-1B CORE MOCK-UP

"Dry" Core	"Wet" Core ^(a)
Worth of	
1. ML-1B control and safety blades with and without cadmium	Shutdown worth of six scissor-action blades
2. Peripheral fuel elements	
3. Additional fuel and structural material	Reactivity effect of borating water
Moderator temperature coefficient of reactivity	
2-D flux distributions with bare and cadmium-covered wires	
2-D power distribution with catcher foils	
Detailed intracell flux measurements	
Detailed power measurements around control blades	
Void coefficient	

(a) Wet applies to simulated flooded case.

The term "dry core" is used to describe the reactor in a normal operating condition, i. e., air inside the pressure tubes representing the gas coolant. The "wet core" is a simulation of the flooded condition when the gas channels contain water during fuel-element-loading operations.

Flooded-core experiments will check the reactivity of water in the pressure tubes. Pressure tubes will be filled with water individually in the vault area and placed one at a time into the core tank. The experiment will be treated as an initial approach to criticality with a new core as outlined below.

Experimental Procedures

Prior to each day's reactor operation, and more often if the supervisor deems it necessary, a complete check will be made of the instrumentation and safety mechanisms of the reactor. Each radiation detector will be checked for calibration with a neutron source. Each scram circuit will be checked and each scram mechanism will be operated using a source to initiate the scram. The interlocks will be checked by attempting operation contrary to the design purpose. Following this, a visual check will be made of the facility and the room, the door to the assembly room will be closed, and the checkout form will be signed by the person making the check.

Approach to Criticality

Each approach to criticality will follow a set procedure which is described below and which, to a large extent, is enforced by the interlock system. The initial approach, which is described first, will be used the first time the reactor is made critical and on any other occasion when significant changes have been made in the reactor or in its instrumentation (for example, the flooded core). Once the critical state of a given reactor configuration has been determined subsequent approaches to criticality will be made with less data taking and plotting.

Initial Approach to Criticality. Each approach to criticality will be preceded by the operational checkout described above. Following the checkout, everyone will leave the assembly room and the assembly-room door will be closed. The source will be inserted and the control rod which is interlocked into the water-pumping system will be fully withdrawn. At least one of the remaining control rods will be partially withdrawn to a point where it will be immediately effective on scram. A number of control rods, sufficient to make the reactor subcritical with the full water height, will be interlocked so that they must be in their full-in position before the water can be pumped. Normally this rod full-in interlock will be satisfied with one of the wide control rods. This will leave the positioning of several control rods to the option of the operator.

With the control rods set up in the positions demanded by the interlock circuits and with the remaining control rods partially withdrawn, the dump valve will be closed and the water pump and filling valve operated. The counting rates and the meter readings will be taken on all of the nuclear detectors with no water in the tank. The water will be pumped to the level of the bottom of the active core region and data again taken. The water height will then be raised in steps of approximately 5 in. each, with detector readings taken at each height. The plot of the inverse of the readings as a function of water height will be extrapolated after each change to predict when the reactor will go critical with the chosen rod configuration. Normally full water height will be reached without criticality being attained. The remaining control rods will be withdrawn step-wise with the continuation of the plotting of the inverse of the detector readings with each step. The steps used in raising water height or withdrawing rods will be made smaller as criticality is approached.

These steps apply to the approach to criticality for each fuel loading. Normally a number of fuel loadings are required in order to develop a curve of inverse count versus fuel loading. On the basis of this curve the critical condition of the reactor can be predicted with accuracy. This critical condition is apparent when the reactor power rises into the operating range and the neutron source can be withdrawn without changing the power level. Final adjustments on the shim control rod are usually required at this time.

Subsequent Approach to Criticality. Once the critical state of the reactor has been determined by the method described above a simpler means of making the reactor critical will be used. In the subsequent approaches to criticality, the changes in water level and rod position will be continuous rather than stepwise, and counts on the pulse-counting channels will not be plotted. However, the pulse-counting channels will be in operation so that there will be an additional visual and an aural signal of the power level. The progress in this approach to critical will be founded on the power-level recorder and the meters on the console, which will be in operation during all of the reactor experiments.

The reactor will be taken critical first without the burnable-poison bands and axial flux distribution will be measured. Subsequently, criticality will be obtained with the burnable-poison bands placed on the fuel elements. In the event the reactor is subcritical with 61 fuel elements in either case, the following changes may be made.

- (1) Remove the cadmium bands from the elements on the outer perimeter
- (2) Add 5 additional fully enriched uranium-dioxide pins which together with the center 7 pins from each element can be used to make up an outer 12-pin ring of fully enriched pins. Complete the fuel element with 7 center pins of 50 weight per cent uranium-dioxide in alumina.

In general, the experimental procedures used in this study have been used in previous critical-assembly work.

Critical Mass

In a heterogeneous reactor system such as the ML-1B critical assembly, the exact critical mass is not easily determined. Experimentally, the number of fuel elements and the excess reactivity in the critical core can be measured and will be the quantities reported.

Reactivity Effects

Reactivity effects will be determined by measuring the change in the reactor period or the change in the position of calibrated control rods. Since the control rods will be calibrated by period measurements, all reactivity measurements will be based on the usual period-reactivity relationship. Reactor period will be measured by several detectors.

Temperature Coefficient of Reactivity

The temperature coefficient of reactivity will be measured by raising the water temperature in increments of 5 to 10 C and noting the change in position of a calibrated control rod. To degas the water it will be heated and allowed to cool to room temperature prior to the measurements. Each water temperature will be maintained for a time to insure that all reactor components are at the same temperature. However, in previous work the position of the calibrated control rod changed only slightly while awaiting temperature equilibrium; consequently, at the higher temperatures, which are difficult to maintain, this was not always done. Two stirrers will be used to circulate the water through the tank.

The moderator temperature will be measured by two Thermohm resistance thermometers. One will be positioned in or near the fuel; the other will be located in the reflector region.

A third Thermohm will be located in a fuel element, about 8 in. from the top of the core. The temperature at this point generally lags the other readings by 2 to 6 C because of poor heat transfer between the fuel cylinders and the moderator. Previous tests have shown that the fuel-element temperature does not affect the results.

Flux and Power Distributions

The flux will be obtained by measuring the manganese activity in irradiated copper-manganese wires (P-metal). The wires will be activated by operating the reactor at about 10 w for 30 to 60 min. These wires will be 0.010 and 0.033 in. in diameter and will contain approximately 80 w/o manganese and 20 w/o copper. For two-dimensional core plots several long lengths of wire will be inserted in individual fuel elements and in 0.038-in. - ID aluminum tubes located in the water midway between the fuel elements. For some studies the individual wires will be positioned by plastic holders in the water-moderator region.

Several fuel pins will be made in four pieces which can be separated to remove a capsule containing two UO_2 pellets for counting purposes. A central hole in the capsule will accept short pieces of wire for activation. In operation, the fuel-pin pieces will be fitted together and taped before assembly into the fuel cluster. After irradiation the capsules will be removed, the wires counted for flux distribution and the encapsulated uranium-dioxide pellets counted to measure power distribution.

Epicadmium flux will be obtained by irradiating the same type wires in cadmium tubes having a wall thickness of 15 mils.

The power at points within a fuel capsule will be determined by the catcher-foil technique. Aluminum foil placed adjacent to a uranium foil will catch the recoiling fission products, which can then be counted. This procedure will give a direct measure of power since the fission-product activity in the catcher foil is directly proportional to the fissioning in the uranium fuel.

Control-Blade Evaluation

The critical-assembly operational control rods will be calibrated to insure the safety of the operations and to provide means of measuring other changes in reactivity.

For example, temperature coefficient of reactivity can be measured in terms of the equivalent change in a calibrated rod position.

The reactivity worth of each type of prototype control rod will be measured and the worth of various combinations of these rods will be determined. Their worth will be evaluated by intercomparing them with the critical-assembly operational rods, by sub-critical multiplication measurements, and by evaluating the rod in terms of the change in fuel mass required to maintain criticality. In general, these studies will determine the integral worth of the rods. A few determinations of differential rod worth will be made on a selected number of prototype rods.

ANALYSIS OF SAFETY ASPECTS

Hazards During Normal Operation

During normal operations, potential hazards exist from direct irradiation, including sky shine, and from air-borne fission products. Radiation surveys taken during operation of other assemblies indicate that the dose from direct irradiation is not a significant amount. The gamma dose rate in the control room and at other positions in the building is such that for 40 w-hr of operation, personnel receive only 0.03 of maximum permissible exposure. Monitoring of fast neutrons with a Hurst dosimeter gives no reading.

Little information is available on fission-product release from UO_2 which is directly applicable to these conditions. Experiments at higher temperatures do indicate that no diffusion of fission products would be expected at temperatures encountered in the critical assembly (120 C maximum). However, some of the gaseous fission products will recoil out of the UO_2 particles. Because of the low density of the bulk UO_2 (75 per cent of theoretical) gas products recoiling from particles may find it easy to escape through the interconnected voids.

Experiments conducted at Battelle indicate that 4 or 5 per cent gaseous fission products may recoil and escape from 150- μ particles of UO_2 in graphite. This figure has been used to estimate the gaseous-fission-product release from the ML-1B UO_2 pins. Based on 1 w-hr of operation, it has been calculated that the critical-assembly room (volume $2 \times 10^9 \text{ cm}^3$) would contain radioactive gas at approximately 0.25 maximum permissible concentration. This figure is based on the release of krypton isotopes 85, 87, and 88. The radioactivity of xenon isotopes, which may be released, would peak sometime later because of the decay time of their precursors. At the time of peak xenon release, the krypton would have decayed so the biological hazard would be less than at peak krypton. For this reason xenon isotopes have been neglected. Gaseous fission products having extremely short half-life (less than 3 min) have also been neglected since the assembly room is normally not entered for about 30 min after a run.

The above calculations refer to unsealed fuel elements. However, in operation the UO_2 is placed in Inconel X tubes and end plugs brazed in place. The pins are placed in a pressure tube with one end closed by a stainless steel disk welded in place, and the other end closed with a Teflon cap. Consequently, the gas activity is contained by two barriers and the hazard appears negligible.

A small number of UO_2 pellets will be counted for fission-product activity to determine power distribution. However, the amount of unjacketed UO_2 is only several grams and the gas release should be reduced by a factor of at least 1000.

The Maximum Credible Accident

In postulating credible accidents for these experiments both the critical-assembly design and the experimental program were carefully reviewed. It was apparent that the only mechanically available reactivity addition lay in the radially mounted scissor-acting mock-up control blades. The maximum credible accident for this critical assembly is therefore taken as the instantaneous removal of one blade of a scissor control unit.

In the experimental program, one scissor unit will be mocked up with two separate blades and reactivity worth as a function of separation angle measured. The angle of separation is varied manually, i. e., two studs which secure each of the blades individually are removed, the angle changed, and the studs relocated. Only one mock-up blade pair is constructed this way. For separate experiments six scissor-blade mock-up units will be constructed each in a one-piece-type assembly and, hence, the angle cannot be varied. These fixed blades are secured by six studs to a support piece which is fastened to the top and bottom tube sheets as shown in Figure 4. The fixed blades are used in only one experiment; namely a measurement of their shutdown worth in the flooded core. The method of securing these blades, their nonadjustable construction, and their use in only one experiment removes them from the credible-accident category.

In measuring the reactivity as a function of angle (in the adjustable-blade mock-up) it is "conceivable" that the experimenter would fail to replace the studs properly. If this happened the rod might fall to the bottom of the core during operation. Previous experiments indicate that one blade is worth about 0.5 per cent $\Delta k/k$. (Shearing off one of these blades is considered unlikely since they are made of stainless steel 3 in. wide, 5/8 in. thick at the base and 1/8 in. thick at the tip.) It is shown in the Appendix that a step increase of 0.5 per cent $\Delta k/k$ would cause a power excursion with a total energy release of 20 megawatt-sec. Since the heat is not readily transferred to the moderator, the energy goes into heating the fuel. Assuming no heat loss, the final temperature of the fuel is 360 C.

The total time for the transient is about 100 milliseconds, i. e., the time from initiation until the power returns to essentially zero. The water is accelerated so violently that it is expelled forcibly from the tank and cannot return to cause a second pulse. Thus, the reactor is permanently shut down. In addition, since the heat is confined to the fuel pins, there is no buckling or expansion of the core structure and the critical-assembly control rods will drop into the core providing a second positive shutdown mechanism. The rods drop completely into the core in 400 to 500 milliseconds.

The melting points of UO_2 and Al_2O_3 are 2600 C and 2050 C, respectively. Consequently, there will be no melting in the fuel even if a sizable hot-spot factor is applied. Inconel X melts at approximately 1400 C, considerably above the maximum temperature reached during the transient. It is doubtful that much heat would be transferred to the metal sheath during the transient and, hence, the heat capacity of the Inconel X was not included in the calculations. The additional heat capacity would tend to reduce the maximum temperature.

The thermal expansion of the fuel [linear-expansion coefficient 10^{-5} in./(in.)(C)] is not sufficient to close the gap between fuel pellets and cladding so no strain is induced in the Inconel X tube. The gap will be at least 0.002 in. for easy pellet loading. The increase in gas pressure due to fission-gas formation or temperature change (to about 2 atm) is not sufficient to exert any significant pressure on the tube.

Thus, this transient results in only a high direct-irradiation level of short duration in the assembly room. Operating personnel possibly would receive 5 to 10 MPE. The large exclusion area would protect all public persons from direct irradiation. Fission products would be contained in the sealed fuel pins or by the secondary barrier, the corked pressure tubes. It is unlikely that any fission products would escape and, consequently, no public persons would be endangered.

Other Possible Incidents

Incidents other than those described under the "Maximum Credible Accident" have been investigated as possible causes of a power transient in this assembly included:

- (1) Continuous withdrawal of control rods
- (2) Continued addition of water to a reactor critical at less than normal water height
- (3) Flooding a fuel element
- (4) Dropping a fuel element into a critical core.

All of these incidents are improbable but nevertheless procedures have been adopted to prevent their occurrence. Even if they do occur the reactivity addition is either very slow or quite small compared with the amount required for prompt criticality. Incidents giving rise to step increases in reactivity as large as 1 per cent are difficult to postulate. However, the calculated energy release resulting from such increases does not appear to be of sufficient magnitude to melt the fuel. Consequently, persons in areas adjacent to the Battelle Nuclear Research Center are almost certainly free of radioactive hazards from this assembly.

Conclusions From Safety Analysis

The conclusions reached after analyzing the ML-1B critical assembly is that the experiment can be conducted with reasonable assurance of safe operation.

JWC:HME:WSH:DAD/nb

APPENDIX

REACTOR BEHAVIOR UNDER ABNORMAL OPERATION

To investigate the behavior of the ML-1B critical assembly under abnormal conditions, a series of transients was solved on the Battelle Electronic Operational Analog. The calculations assumed the first negative reactivity effect "felt" by the assembly after a positive step insertion was due to radiolytic gas formation. The gas was assumed to be formed predominantly at the center of the core and to expel the cylindrical water volume above the midplane in the manner of a piston. Because of poor thermal contact between water and fuel, the transients occur with little or no heat transfer to the moderator.

The dynamic equations which were solved are

$$P = \frac{Mg}{A} + M \frac{d^2x}{dt^2} + 1.013 \times 10^6 \quad (1)$$

$$PAx = NRT \quad (2)$$

$$k = k_0 - aAx \quad (3)$$

$$\frac{dN}{dt} = bp \quad (4)$$

where

$$\frac{dp}{dt} = \frac{K_{ex} p}{\ell}$$

M = mass of water, 5.9×10^4 g

g = acceleration due to gravity, 980 cm per sec²

A = piston area, 980 cm²

R = gas constant, 8.3×10^7 ergs/(mole)(K)

T = 300 K

b = gas formation rate = 3.1×10^{-3} mole per megawatt-sec

a = void coefficient, 5×10^{-6} per cm³

x = piston travel

N = moles of gas formed

P = gas pressure

p = reactor power.

The mathematical model assumes that

- (1) Each 100 ev of ionizing radiation produces one molecule of hydrogen
- (2) Three per cent of the fission energy is effective as ionizing radiation
- (3) Hydrogen produced during the first 15 megawatt-sec of the excursion is dissolved in the water; after this the hydrogen produces voids.

The value of the void coefficient used in the calculations was an estimate based on an experimentally determined coefficient measured at the upper reflector-core interface in the ML-1A core. The experimental value was 0.6×10^{-6} per cm^3 . The effect of a void would be expected to vary as the product of the flux and the adjoint flux, i.e., approximately as ϕ^2 . An investigation of flux distributions indicates that the void coefficient at the center of the core should be about 20 to 80 times the coefficient at the perimeter. This fact is borne out by temperature coefficient data.

Values of k_{ex} were chosen arbitrarily at 1.0, 1.5, and 2.2 per cent. No credible accidents with this core would be expected to produce an excess of this magnitude. For example, one pair of scissor blades (two blades) shearing off and falling instantaneously from the core would insert only about 1 per cent k_{ex} .

Results of the calculations are shown graphically in Figure A-1.

FIGURE A-1. ENERGY RELEASE IN A POWER TRANSIENT VERSUS k_{ex} ABOVE DELAYED CRITICAL

This curve shows an energy release of 70 megawatt-sec for a step increase of 1 per cent Δk . The energy required to melt the fuel in the core (50 kg UO_2 + 30 kg Al_2O_3) is estimated to be 135 megawatt-sec; to vaporize the fuel would require an additional 200 megawatt-sec. These estimates are based on rather incomplete physical-property data but are believed to be conservative. The energy requirements do not include melting or vaporizing the Inconel tubes; the heat transferred from the loosely fitting pellets to the tubes in a fast transient probably would be very small. If heat were transferred to the metal tubes, the energy required to melt the fuel elements would be still larger.