Argonne National Laboratory

EBR-II DRY CRITICAL EXPERIMENTS

Experimental Program,
Experimental Procedures,
and Safety Considerations

by

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Contributions from the EBR-II Project Staff
Reactor Engineering Division

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PREFACE

This Report describes a proposed Dry Critical Experimental Program to be conducted in the EBR-II. It describes the revisions in the reactor system and operating procedures which will be made and considers the safety aspect of the program.

This Report is to be used in conjunction with the "Hazards Summary Report - Experimental Breeder Reactor-II (EBR-II)," which was published as ANL-5719 in May, 1957. Specific references to supplementary information in ANL-5719 are denoted in brackets; e.g. [Section V], [pages 65-69], [Fig. 23].

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EBR-II DRY CRITICAL EXPERIMENTS

I. INTRODUCTION

The Dry Critical Experiments are zero power experiments (<1 kw) to be conducted in the EBR-II Reactor prior to filling the Primary System with sodium coolant. These experiments have two general objectives:

(1) To determine information relating to the performance of the system without sodium coolant, for future comparison with information derived with sodium coolant. By comparison, it will then be possible to ascertain various sodium effects on the neutronics of the system.

(2) To determine and/or verify certain operational data to permit modification or improvement of the system (neutron shield, instrumentation, etc.) prior to the introduction of sodium into the system. After sodium filling it will be difficult to alter, modify, or correct many of the system components.

Installation of reactor system equipment is in progress with the expectation that all equipment associated with the conduct of these experiments will be installed by about the end of April, 1961. It is proposed to then begin the final check-out operations and to prepare the reactor system for the dry critical experimental program.

Construction work will continue in other areas of the plant associated with the secondary sodium system, steam system, and the fuel cycle facility. No construction work will be permitted within the areas involved in the dry critical experiments (Reactor Plant and Control Room).

Some of the operational and safety considerations relating to these experiments are different from those relating to power operation of the reactor. The experimental operations will be conducted on a one-shift basis and will not require the transfer of information from shift to shift. Because of the nature of the experiment, the operations will be performed by Laboratory Staff (Physicists and Engineers) intimately familiar with the system. Other significant considerations are:

(1) The maximum reactor power level will not exceed 1 kw.

(2) There will be no initial contamination of the fissionable and fertile materials with fission products or plutonium, nor will a significant quantity of either be generated during these experiments.

(3) The reactor critical mass is larger, and the core will be at least 20% larger in diameter than contemplated for the reactor with sodium

present. The reactivity worth of individual fuel subassemblies is consequently smaller particularly the worth of each subassembly at the periphery of the core as the critical configuration is approached. The reactivity worth of each control rod and safety rod is also smaller.

The reduction in safety rod worth is approximately proportional to the reduction in subassembly worth; however, the worth of the control rods is not reduced proportionally because in a larger core they are disposed inside the core periphery.

(4) It will be possible to install viewing apparatus to permit observation of operations in and around the reactor.

(5) All experiments will be conducted within a primary structure and a building containment system designed for a power reactor.

Similar experiments have been conducted in ZPR-III and the significant parameters have been measured. These include the EBR-II Mock-up Experiments described in Appendix A. Although these experiments were not an exact mock-up of the EBR-II Dry Critical configuration, they provided a sound basis for extrapolation. These results are particularly significant with respect to control, neutron source, and instrument response at low power level.

II. SUMMARY

The EBR-II Dry Critical Experiments will be conducted in the EBR-II Reactor prior to filling the primary system with sodium. The experiment will employ the same components and fuel to be used subsequently for normal power operation. Many operations will be simplified, however, because of the absence of sodium and the low power level of operation.

Special in-core instrumentation will be added to obtain additional information and to provide additional operational reliability (including period trip for automatic shutdown) at low power level; particularly at source level and at low neutron multiplication. The reactor shutdown system has been revised to incorporate manual scram of the safety rods during reactor operation as well as manual and automatic scram during fuel-handling operation.

Because of the larger critical mass of the "dry" reactor configuration, a larger number of enriched subassemblies are required. A significant number of these (about 25) will be loaded in the inner blanket region of the reactor. To enhance administrative control during loading, the two types of inner blanket subassemblies will be color coded (enriched - red; depleted - blue), in addition to the series identification of their serial numbers. Also, the loading operations will be revised to load the reactor directly from the operating floor to the reactor (the storage rack will not be employed during the loading operations). In addition, during these experiments, the transfer and loading operations within the primary tank and the reactor will be visible. All fuel-handling operations will be observed visually through periscopes and viewing ports.

The reactor containment system will be operative and, in fact, will be in a somewhat more secure status than will obtain during normal power operation. Many of the building services will not be required during these experiments, and the quick-closing valves (which normally must be closed to effect containment) will be locked closed. The air locks will be operative in the normal manner, with interlocking of the doors.

Similar experiments have been conducted on ZPR-III including mock-ups of the EBR-II reactor configuration. These have provided a thorough understanding of the system and have established the more significant parameters relating to control, neutron source, instrument response, etc. As a result, the characteristics of the Dry Critical Experiments have been predicted in considerable detail. These predictions provide a basis for the experimental program and a valuable check on experimental procedures.

These experiments do not present a hazard to the general population because of the containment system, the absence of long-lived fission products and/or plutonium, and the plant location. These same factors (except plant location) apply to the evaluation of hazards to the immediate surroundings and

to the on-site non-operating personnel; employees of the Construction Contractors, the Commission, and the Laboratory. The Commission's emergency evacuation procedures apply to all of these personnel as well as the Laboratory Operating Staff. The assessment of plant safety indicates that the design as developed for power operation provides safety features and imposes operational restrictions which enhance plant safety.

III. DESCRIPTION OF FACILITY

The EBR-II Facility is described in detail in ANL-5719 [Section III].* These descriptions are augmented in this report with particular emphasis on the systems and components employed in the conduct of the dry critical experiments. The EBR-II is an experimental nuclear power plant and certain facilities and operations will be altered for performing the dry critical experiments. Many features of the plant result from criteria established for power operation. Some of these are also assets to the dry critical experiments. No water or steam is brought into the building as a precaution against initiating a sodium-water reaction. This, of course, also precludes the possibility of water entering the reactor during the critical experiments and the attendant hazards associated with moderating the reactor. Similar precautions have been taken with respect to the inadvertent entrance of lubricants into the Primary Tank because of their deleterious reactions with sodium. These same precautions preclude the possibility of inadvertent introduction of moderating lubricants into the reactor. The design criteria for the reactor building containment system were based on a hypothetical accident involving a large energy release resulting from a sodium-air reaction. The system is very much over-designed for these experiments in which no sodium is present.

A very large part of the plant will be inoperative during these experiments since their functions are not essential to the conduct of the experiments; e.g., the sodium systems, steam system, power generation equipment, etc.

The experiments will be conducted in the reactor vessel which is located at the bottom center of the primary tank [Fig. 7 and 9]. The major components and equipment will have been installed in the primary tank, with the exception of the sodium to sodium intermediate heat exchanger. The heat exchanger shell and all primary sodium piping will be installed, but the tube bundle and plug will be omitted to permit access into the tank during and subsequent to the dry critical experiments. A removable shield plug will be provided for this opening.

The reactor vessel has been modified to eliminate discontinuities in the vessel wall, and the thermal baffling arrangement has been revised to minimize thermal stresses in the vessel. These revisions do not affect the reactor configuration or arrangement. The design as it relates to these experiments is unchanged.

The reactor will be loaded with fuel and blanket subassemblies produced for normal (power) operation of the reactor. The subassemblies and the corresponding fuel and blanket elements are as shown [Figs. 10, 11, and 12].

*See PREFACE for explanation of use of [Brackets].

Only minor revisions have been made in a few of the "hardware" parts; none of which affect operation. The fuel pin, fuel pin tubing spacing etc. are unchanged. The coolant flow orificing arrangement in the subassembly lower adapters has been revised, but obviously this does not affect these experiments.

The control and safety subassemblies [Figs. 13 and 14] have undergone minor modification relating to coolant orificing and coolant flow through the subassemblies. These changes are not relevant to the dry critical experiments.

All of the necessary components in the primary system associated with reactor control and fuel handling will be installed and made operative for these experiments. The reactor will be operated from the control room (in the Power Plant), while fuel loading will be effected in the Reactor Plant. Additional instrumentation provided for these experiments (in-core fission counters) will be displayed in the control room and on the reactor building operating floor.

Centralized administrative control over reactor operation and fuel-handling operation is provided by a key switch for "control power." This switch is located on the console and controls power to the control rod drives, safety rod drives, and fuel-handling components. The key can be removed only in the "off" position. When the control power system is de-energized, the control and safety rods scram, and the drives and fuel-loading equipment become inoperative. This switch does not affect power for instrumentation, indicating lights and annunciators; therefore, sensing, information gathering, and annunciation will continue even though operation is prevented.

A. Instrumentation and Control

The eleven channels of neutron flux-monitoring instruments for normal reactor operation will be used (as applicable) for these experiments. The general purpose of these instruments remains unchanged:

(1) Monitor neutron flux level and reactor period from source level through the power range.

(2) Initiate automatic scram of control rods (during reactor operation), or automatic scram of safety rods (during fuel-handling operations), whenever the neutron flux level exceeds a preset value, or the reactor period becomes shorter than a preset value.

Two additional channels of in-core instruments are provided for the Dry Critical Experiments. To increase the assurance of measuring neutron flux level and reactor period at low power levels, a startup flux interlock withholds power to raise the control rods and safety rods until the in-core

counting channels yield a neutron count rate in excess of a minimum preset count rate (about 10 count/sec). In addition, the period trips and low power level trips normally used for actuation of the safety rods during fuel handling will provide actuation of the control rods during reactor operation. This is done to take maximum advantage of the low-power level instrumentation available. It is convenient to do so because the operating limitations are essentially the same for "reactor operation" and "fuel-handling operation."

The safety rods are tripped automatically only during fuel-handling operation, but can also be scrammed manually during reactor operation. (Manual scram of the safety rods during reactor operation will also cause the control rods to scram.)

Audio units are provided for the in-core channel scalers in the reactor plant and for the log count rate channels in the control room and the reactor plant.

1. Instrument Channels

The reactor power range over which the neutron flux is measured by each of the eleven channels of normal instrumentation and the two channels of in-core instrumentation is shown in Fig. 1. The Block Diagrams of the normal nuclear instrumentation (Fig. 2), and in-core instrumentation (Fig. 3), show the associated equipment and indicate the physical location of the neutron flux level and reactor period indicators and recorders.

The general arrangement of the normal nuclear instrument thimbles and their associated fission counters and compensated ion chambers is shown in Fig. 4. Eight thimbles are provided (the three thimbles originally planned external to the primary tank have been eliminated). The four "J" thimbles are imbedded in the graphite neutron shield outside the reactor vessel, while the four "O" thimbles are located immediately adjacent to the neutron shield.

The in-core instrument thimbles and their associated fission counters are located as shown in Fig. 5. The in-core instrumentation will provide additional sensitivity at low neutron flux (subcritical conditions), and a measure of the sensitivity of the normal instrumentation channels in the "J" thimbles.

Further description of the normal and in-core channels of instrumentation follows (summarized in Table I):

a. Channels A and B (fission counters) are in-core log count rate channels which measure period and log count rate from source level to about 100 watt. They indicate reactor period in the reactor plant and in the control room. Log count rate (flux level) is recorded in the reactor plant and indicated in the control room. These channels provide period trips (trip setting: 15 sec), to automatically scram the control rods during reactor

operation, or the safety rods during fuel-handling operation. Any one channel period trip will scram the control rods or safety rods. A startup flux level interlock withholds control power to raise the control rods or safety rods until these channels see a neutron count rate (flux level) in excess of a preset level (about 10 count/sec). The low count rate interlock (startup flux level interlock, Fig. 6), can be bypassed by operation of a key switch on the operating console. This bypass is required to permit removal of the neutron source. A warning light on the console will continue to flash and alarm as long as the bypass key switch is "ON."

b. Channels 1, 2, and 3 (fission counters) are log count rate channels which measure reactor period and log count rate (flux level) from source level to about 2×10^3 watt. They indicate reactor period and record log count rate (flux level) in the control room and the reactor plant. They also provide period trip (trip setting: 15 sec), and power level trip (trip setting: 1 kw). Any one channel period trip or level trip will automatically scram the control rods during reactor operation, or will automatically scram the safety rods during fuel-handling operation. Channels 1, 2, and 3 provide additional period trips which are connected in coincidence, requiring any two of the three channel period trips to scram the control rods during reactor operation. (The two sets of period trips - one coincidence and one a single trip - result from the incorporation of the "fuel-handling" trips in the "reactor operating" circuit. For normal power operation, two out of three coincidence trips will scram the control rods during reactor operation, while one out of three will scram the safety rods during fuel-handling operation. Combining these two functions in the reactor operation "shutdown circuit" results in two trips, actuated by independent trip contacts, initiating scram of the control rods.)

c. Channels 4, 5, and 6 (compensated ion chambers) are log N power channels which measure reactor period and log N (flux level) from about 100 watt to full power. They indicate reactor period and record log N in the control room. During reactor operation, any two out of three coincidence period trips (trip setting: 15 sec in each channel) will automatically scram the control rods. These channels also provide additional period trips (trip setting: 15 sec) for fuel-handling operation, which automatically scram the safety rods. No low level bypass of the period trips is provided in these channels.

d. Channel 7 (compensated ion chamber) is a linear power channel which measures flux level from about 100 watt to full power. Indication and recording in the control room of reactor power level is given on an expanded scale obtained by manual range switching. Linear level trip is provided with automatic adjustment of trip level integral with range switching. This level trip on the most sensitive range (trip setting: 1 kw) will automatically scram the control rods during reactor operation.

e. Channel 8 (compensated ion chamber) is used for automatic rod control from 10% to full power. This channel will not be used during critical experiments.

f. Channels 9, 10, and 11 (compensated ion chamber) are high-flux safety channels which measure flux level from about 60 kw to reactor full power. They indicate and record reactor flux level in the control room. During reactor operation a two out of three coincidence flux level trip (trip setting: 60 kw) will automatically scram the control rods. Although these channels are designed for use at higher reactor power, they will serve as a back-up for the low power level trips in Channels 1, 2, and 3. The power level trips in Channels 9, 10, and 11 will be set at the lowest power level consistent with a reliable neutron flux level signal (approximately 60 kw).

Table I

SUMMARY OF PERIOD AND LEVEL TRIPS

Channel No. and Range	Reactor Operation ¹				Fuel Handling ²			
	Period Trip	Setting (sec)	Level Trip	Setting (watt)	Period Trip	Setting (sec)	Level Trip	Setting (watt)
In-Core Channels A & B Source to 100 watt	Any 1 of 2	15	-	-	Any 1 of 2	15	-	-
Log Count Rate Channels 1, 2, & 3 Source to 2×10^3 watts	2 out of 3 or Any 1 of 3	15	Any 1 of 3	10^3	Any 1 of 3	15	Any 1 of 3	10^3
Log N Channels 4, 5, & 6 100 watt to full power	2 out of 3	15	-	-	Any 1 of 3	15	-	-
Linear Channel 7 100 watt to full power	-	-	1 of 1	10^3	-	-	-	-
High-Flux Safety Channels 9, 10 & 11 6×10^4 watt to full power	-	-	2 out of 3	6×10^4	-	-	-	-

¹ During Reactor Operation, either or both the period trips or level trips shown will automatically scram the control rods.

² During Fuel Handling, either or both the period trips or level trips shown will automatically scram the safety rods.

Loss of counter high voltage in any one of Channels 1, 2, or 3, or loss of the compensated ion chamber high voltage in any two of Channels 4, 5, or 6, will cause the control rods to scram during reactor operation, or the safety rods to scram during fuel-handling operation. Loss of the compensated ion chamber high voltage in any two of Channels 9, 10, and 11, or in Channel 7, will cause the control rods to scram.

2. Instrument Response and Trip Calibration Check

Although all the normal instrumentation channels have been calibrated using electrical signal sources and low-level neutron sources before installation, an additional check on the calibration of power level and reactor period response will be made on the instrument channels with all equipment installed ready for use in the critical experiments. These post-installation calibrations will be made with a pulse signal generator and an exponential pulse rate generator (in lieu of the fission counters) to check the log count rate (1 to 10^5 count/sec) and the reactor period (at 3, 5, 10, 15, and 20 sec) in the log count rate Channels A and B (in-core), and Channels 1, 2, and 3. Also, a current test source and an exponential current generator will replace the compensated ion chambers to check calibration of power level (100 watt to full power) and reactor period (at 3, 5, 10, 15, and 20 sec) in the log N Channels 4, 5, 6, linear level Channel 7 (no period), and high-flux safety Channels 9, 10, and 11 (no period).

In addition to these checks of the instrument response, the power level trip and reactor period trip setting calibrations will be checked.

A 10-millicurie polonium-beryllium neutron source will be used after check-out with the aforementioned electrical signal generators.

Additional calibration checks of the instrument channels against reactor flux are described in Section V-D.

3. Reactor Shutdown Circuit

The reactor period and flux level trips which are provided to automatically scram the twelve control rods during reactor operation, or the two safety rods during fuel handling, are shown on the block diagram Fig. 6. The interlocks which permit transfer of control power between fuel-handling operation and reactor operation are also shown. The prerequisite conditions to be met prior to reactor startup, as well as conditions other than nuclear trips which can cause scram, are also shown on Fig. 6. All nuclear trips causing reactor scram are annunciated in the control room.

All nuclear instrument channel period and level trips are summarized in Table I.

4. System Abnormalities Causing Reactor Scram or Alarm

A complete list of abnormalities causing reactor scram and/or alarm (which identify the system abnormality) which will be employed in the Dry Critical Experiments is given in Table II.

Table II

SYSTEM ABNORMALITIES CAUSING REACTOR SCRAM AND/OR ALARM

Abnormality	Alarm	Normal Operating Scram & Alarm ¹	Fuel Handling Scram & Alarm ²
<u>Nuclear System</u>			
1. Log Count Rate Period, Channel A		X	X
2. Log Count Rate Period, Channel B		X	X
3. Log Count Rate Level, Channel 1	X	X	X
4. Log Count Rate Level, Channel 2	X	X	X
5. Log Count Rate Level, Channel 3	X	X	X
6. Log Count Rate Period, Channel 1	X	X	X
7. Log Count Rate Period, Channel 2	X	X	X
8. Log Count Rate Period, Channel 3	X	X	X
9. Log Count Rate Counter High Voltage, Channel 1	X	X	X
10. Log Count Rate Counter High Voltage, Channel 2	X	X	X
11. Log Count Rate Counter High Voltage, Channel 3	X	X	X
12. Log N Period, Channel 4	X	X ³	X
13. Log N Period, Channel 5	X		X
14. Log N Period, Channel 6	X		X
15. Chamber High Voltage, Channel 4	X	X ³	X ³
16. Chamber High Voltage, Channel 5	X		
17. Chamber High Voltage, Channel 6	X		
18. Linear Flux Level, Channel 7	X	X	
19. Chamber High Voltage, Channel 7	X	X	
20. High Flux Level, Channel 9	X	X ³	
21. High Flux Level, Channel 10	X		
22. High Flux Level, Channel 11	X		

¹ Control Rods only automatically scram.

² Safety Rods only automatically scram.

³ Any two out of three trips will cause Scram and Alarm.

Table II (Cont'd)

Abnormality	Alarm	Normal Operating Scram & Alarm ¹	Fuel Handling Scram & Alarm ²	
<u>Nuclear System (Cont'd.)</u>				
23. Chamber High Voltage, Channel 9	X	X ³	X	
24. Chamber High Voltage, Channel 10	X			
25. Chamber High Voltage, Channel 11	X			
26. Control Rods Shock Absorber-Low Oil Level	X			
27. Control Rods Unlatched	X	X		
28. Control Rods Compressed Air Tank - Low Pressure	X	X		
29. Safety Rods Unlatched	X	X		
30. Cable Routing Room Nuclear Inst. Racks - High Temperature	X			
31. Instrument Center No. 2 Nuclear Inst. Racks - High Temperature	X			
<u>Auxiliary System</u>				
1. Constant Power (Process) - Abnormal	X			
2. Constant Power (Nuclear) - Abnormal	X			
3. Instrument Air - Low Pressure	X			
4. Fuel Element Clad - High Temperature	X			
5. Containment Vessel Internal Pressure - High	X			
6. Containment Vessel Internal Pressure - Low	X			
7. Operating Floor - High Temperature	X			
8. Personnel Air Lock Emergency Signal	X			
9. Personnel Air Lock Outside - Open	X			
10. Personnel Air Lock Inside - Open	X			
11. Emergency Air Lock - Open	X			
12. Equipment Air Lock - Open	X			
13. Operating Floor - Radiation High	X			
14. Reactor Building Isolation Valve Trip	X			

Table II (Cont'd.)

Abnormality	Alarm	Normal Operating Scram & Alarm ¹	Fuel Handling Scram & Alarm ²
<u>Radiation Monitoring</u>			
1. Reactor Plant - Personnel Air Lock	X		
2. Reactor Plant - Primary Tank Top	X		
3. Reactor Plant - Operating Floor	X		
4. Reactor Plant - Air Exhaust	X		
5. Reactor Plant - Operating Floor (Air)	X		
6. Reactor Plant - Sub-Basement (Air)	X		
7. Liquid Waste Monitor (EBR-II Site)	X		
<u>Electrical System</u>			
1. 138-kv Tie Circuit Breaker	X		
2. Transformer 1 - 13.8-kv Secondary Trip	X		
3. Transformer 2 - 13.8-kv Secondary Trip	X		
4. Transformer 5 - 13.8-kv Primary Trip	X		
5. Transformer 5 - 480-volt Secondary Trip	X		
6. Transformer 5 - Temperature High	X		
7. Transformer 6 - 13.8 kv Primary Trip	X		
8. Transformer 6 - 480-volt Secondary Trip	X		
9. Transformer 6 - Temperature High	X		
10. 13.8-kv Bus Tie Trip	X		
11. 13.8-kv Bus Section 1 - Lockout Trip	X		
12. 13.8-kv Bus Section 2 - Lockout Trip	X		
13. 13.8-kv Bus Section 1 - Undervoltage	X		
14. 13.8-kv Bus Section 2 - Undervoltage	X		

Table II (Cont'd.)

Abnormality	Alarm	Normal Operating Scram & Alarm ¹	Fuel Handling Scram & Alarm ²
<u>Electrical System (Cont'd.)</u>			
15. 480-volt Emergency Switchgear E1 Feeder Trip	X		
16. 480-volt Power Plant MCC-P2A Feeder Trip	X		
17. 480-volt Reactor Plant MCC-R1A Feeder Trip	X		
18. 480-volt Lighting Distribution Center E2 Feeder Trip	X		
19. 480-volt Power Plant MCC-P2B Feeder Trip	X		
20. 480-volt Fuel Handling and Sodium Purification MCC R3 Feeder Trip	X		
21. 480-volt Power Plant MCC-P1 Feeder Trip	X		
22. 125-volt d-c Battery Bus Ground	X		
23. 400-kw Diesel Generator - Fail to Start	X		
24. 400-kw Diesel Generator - 480-volt Circuit Breaker Closed	X		
25. 400-kw Diesel Generator - Water Temperature High	X		
26. 480-volt Emergency Bus E-1 - Undervoltage	X		
27. 400-kw Diesel Generator - Oil Pressure Low	X		
28. 400-kw Diesel Generator - Overspeed	X		
29. 480-volt Emergency Bus MCC R2A - Undervoltage	X		

5. Neutron Source

The antimony-beryllium neutron source is contained in a specially designed subassembly shown in Fig. 7. Two 200-curie sources and one 20-curie source will be provided. The 20-curie source will be used for reactivity measurements. The location of the neutron source

for the Dry Critical Experiments is shown in Fig. 5 (the inner blanket, Sector E, Row 7, Position 3). The source material is located at the same elevation, and is equal in length to the core.

The source subassembly is loaded into the reactor in two parts: the source thimble (beryllium cylinder), and the source rod (antimony rod), to facilitate handling. The activated antimony rod may be separated from the beryllium cylinder, thus eliminating the need for a neutron-shielded container. The location, handling, and storing of the neutron sources is described in Section V.

6. Temperature of Fuel Element Clad

Fourteen thermocouples will be used to measure the temperature of the fuel element clad on a number of elements in the central core subassembly (Row 1, Position 1, Fig. 5). The fourteen thermocouples will be attached to the elements at various locations within this subassembly. A maximum of two thermocouples will be used on any one element: one thermocouple will be on the centerline of the element, and the other thermocouple either 3 in. above, or 3 in. below the centerline. The output from each thermocouple will be recorded in the control room. Any thermocouple measuring a temperature greater than 180°F will trip an alarm in the control room.

B. Fuel Handling and Reactor Loading

1. Operational Considerations

The fuel-handling system as described [pages 34 to 37] is designed for power operation of the system, and for operations conducted under sodium. Also, provisions are made for cooling subassemblies after irradiation to remove fission product decay heat. The requirements for the Dry Critical Experiments are quite different and are reflected in the operational procedures to be used. The absence of sodium and the absence of cooling requirements simplify the procedures, while the larger number of subassemblies required for the larger core introduce additional procedural requirements.

The reactor grid is designed to accommodate the three basic types of subassemblies comprising the reactor loading: core, inner blanket, and outer blanket subassemblies. As described [page 21], each type of subassembly must be loaded into the appropriate reactor zone. Improper loading is prevented physically by the different hole sizes in the reactor grid.

The Dry Critical loading requires a larger enriched core volume with the result that approximately 25 enriched subassemblies must be loaded into the inner blanket region of the reactor. To permit this, both enriched and depleted subassemblies (which will fit in the inner blanket region) are

required. Although they are clearly marked, the physical prevention of error does not obtain during this experiment; loading errors in the inner blanket region must be prevented by administrative control.

To enhance administrative control, the loading procedure will be altered (and simplified). Normal procedures (for power operation) consist of inserting subassemblies into the primary tank and, via the transfer arm, depositing them into the storage rack. This is a relatively slow procedure involving a shielded transfer machine, gas purging to avoid contamination of the argon cover gas (sodium blanket), and provisions for subassembly cooling during all operations. These operations are completely independent of reactor operation and can be performed with the reactor in operation. Reactor-loading operations are then performed between the reactor and storage rack with the reactor shut down.

This two-step loading procedure will not be employed during the Dry Critical Experiments. All subassemblies for the core and inner blanket region of the reactor will be loaded directly from the operating floor (where they are properly segregated and identified - see Section V), to the transfer arm, and then directly to the reactor. The subassemblies will be lowered manually (with mechanical assist to support the weight), through the refueling port to the transfer arm in the primary tank. (The subassembly identification number will be read by the "loader" at that time.) The transfer arm will deliver the subassembly directly to the gripper mechanism which inserts it in the reactor. The components involved are shown in Fig. 8. Also shown are typical temporary viewing devices which will be employed during the Dry Critical Experiments. The features provided in normal system components are unchanged; the gripper and hold-down functions [Fig. 30], and the transfer arm functions [Fig. 31], are retained.

Some of the outer blanket subassemblies will be loaded by this procedure to verify the operation of the loading equipment; however, to conserve time (510 outer blanket subassemblies are to be loaded), the majority will be loaded by hand directly into the reactor through a spare port in the small rotating plug and reactor cover (the opening employed subsequently for the periscope shown in Fig. 8). It should be noted that only one type of outer blanket subassembly exists (Table III - Section V) and the physical loading control described above applies (no other type of subassembly can be loaded in the outer blanket region).

The philosophy of subassembly "replacement" will apply to the Dry Critical Experiments. Before any enriched subassemblies are loaded, all subassembly positions in the reactor grid will be filled with depleted subassemblies, dummy steel subassemblies, rod thimbles, instrument thimbles, etc. Loading will consist of removing a subassembly and replacing it; only one position will be empty at any time. The detailed loading procedure is described in Section V.

The control and safety rod thimbles are an exception to the replacement loading procedure. These thimbles are locked in place by rotation of the thimbles, which requires that the six surrounding subassemblies be removed for installation or removal (rotation is prevented by the surrounding subassemblies). These thimbles will be installed initially prior to the introduction of enriched assemblies and will not be altered except in the event of malfunction. In the event a thimble is to be removed, the six surrounding subassemblies will be removed in clockwise sequence, separately stored, and returned in the same sequence after the thimble has been replaced. (It should be noted that in the Dry Critical Experiments all of these operations are visible and can be observed.)

Although the storage rack will not be employed during the reactor-loading operations, it will be available for storage of subassemblies removed from the reactor in the event the radiation level of a subassembly is too high to permit manual handling. This is not anticipated in the normal conduct of the experiments (loading changes will be very infrequent); however, it may be desirable to remove a subassembly or a control rod without requiring several days of cooling to permit manual handling. If a subassembly (or control rod) is stored in this manner it will be subsequently removed from the primary tank; it will not be returned directly to the reactor.

Other operations associated with fuel handling will be simplified. For example, the "freeze seals" on the rotating plugs will not be filled with seal metal alloy, thereby removing the need for melting the seals prior to plug rotation. (The seals are required to contain the argon cover gas which blankets the sodium. Without sodium present there is no need for the seals.)

2. System Design Description

Subassemblies are inserted into, and removed from, the reactor by a gripper mechanism which is mounted in the small rotating plug. The plug, in turn, is eccentrically located in the large rotating plug. A hold-down mechanism operates in conjunction with the gripper mechanism, and these units can be positioned above any subassembly location by rotation of the plugs. A subassembly is transferred from the gripper mechanism to the transfer arm. The transfer arm, in turn, rotates in a horizontal arc to move the subassembly to the periphery of the primary tank at the "transfer position" of the rotating plugs. The control rod drives also are mounted in the small rotating plug and therefore an "operate position" exists for the rotating plugs.

The fuel-handling and reactor-loading operations involve the use of several independent components and require the proper setting of associated equipment. Because of the critical interrelationship of these devices, they are extensively interlocked to prevent maloperation.

The design features incorporated in the various components are described in conjunction with the functions to be performed. Only the functions relating to the Dry Critical Experiments are included: those relating to the sodium system, temperatures, etc., are omitted.

The fuel-handling operations are controlled from the fuel-handling console in the reactor plant. This system is energized from the main control room by placing the control switch on the operating console to the "fuel loading" position. (This is a three position key switch: "reactor operate" - "off" - "fuel loading"; the key can only be removed in the off position - neither system operative.) The control key switch on the fuel-handling console must also be actuated to energize the fuel-handling system. The fuel-handling procedures consist of two categories of functions (1) preparation of the reactor system prior to fuel handling (and the reverse procedure following fuel handling); and (2) fuel-handling operations.

The fuel-handling operations are not automatic; they are manually initiated with "permissive" control provided by the system. This is accomplished by push buttons, indicating lights, position indicators, and circuit interlocks. The operator must take the initiative after each step is performed (the completion of an operation is indicated on the panel). If an incorrect action is attempted, it is prevented by the interlock circuits.

a. Preparations Prior to Fuel Handling

(1) Disconnect control rods from drives: This is a multi-step procedure. The control rods are lowered to shut down the reactor. When all twelve drives are in their "down" position, the control selector switch is turned to "fuel handling" (if selector switch is turned prior to shutdown, the control rods scram). The disconnecting operations are effected from the fuel-handling console, and the safety rods must be in their "up" position (rods in). The control drive gripper jaws are opened and the control rods drop ($\frac{3}{4}$ in.) in their thimbles. The rod-sensing tips drop to indicate that the rods have been released from the gripper (Fig. 9 - Position No. 2). The control drive lifting platform is then raised 3 in. to the "up" position (Fig. 9 - Position No. 3). In this position, the ends of the control rod drives clear the tops of all subassemblies in the reactor grid. All indicating lights (jaws open, rod-sensing tips down), are checked visually by the operator (he confirms this check of the indicating lights by actuating a push button which constitutes one of the interlocks that permits subsequent operations).

The grippers are then closed and opened to guarantee that the rods are not stuck to the rod-sensing tips. The relationship of the end of the control rod, the rod-sensing tip, and the gripper jaws is such that the jaws will not close if the rod has not fallen clear [Fig 23]. This check procedure is accomplished by the operator through the actuation of push buttons (a latching relay in the interlock sequence prevents circumventing this procedure).

(2) Raise Reactor Cover: The three cover-locking mechanisms are released and the reactor cover is raised (interlocks actuated by position switches on the three cover-locking mechanisms permit the cover-raising operation). A large number of interlocks must be satisfied prior to, and during, the cover-lifting operation. For example:

- (a) Transfer arm at "pre-transfer" position (clear).
- (b) Control drive lifting platform "up."
- (c) Reactor cover unlocked.
- (d) Two reactor cover-lifting screws "in phase."
- (e) Fuel gripper and hold-down shafts are free in their guides in the cover (the units do not move while cover is raised). (Note: The same protection is provided for each of the control drives by means of the "down limit" switches in the Fuel-Handling Shutdown Circuit - Fig. 6).
- (f) The control rod grippers have been checked and the control rods have been dropped.

The reactor cover is raised at a speed of $18\frac{1}{4}$ in./min.

(3) Disconnect Operating Cables: The cable connections to each of the twelve control rod drives are disconnected (these are multiconductor, plug-in type receptacles). This disconnect deactivates the control rod drives completely and they remain in their prescribed positions. The cables for the cover-lifting mechanism are similarly disconnected (the "cover - up" position switch signal is not disconnected). (For power operation, the molten seals on the rotating plugs require heating and thermocouples must indicate that the seals are molten at this point in the sequence. This requirement is omitted in the Dry Critical Experiments.)

b. Fuel-Handling Operations

The rotating plugs and the subassembly gripper mechanism, are rotated to the proper index point for the removal of the selected subassembly. Many interlocks must be satisfied prior to, and during, plug and gripper unit rotation, including:

- (1) Reactor Cover "up."
- (2) Transfer arm at "pre-transfer" position.

If gripper is empty (not transporting a subassembly):

- (3) Gripper at "operate" level.
- (4) Gripper jaws "open."
- (5) Gripper jaws "empty" (rod-sensing tips down).
- (6) Hold-down at "operate" level.

If gripper is loaded (transporting a subassembly):

- (3a) Gripper at "transfer" elevation.
- (4a) Gripper jaws "closed."
- (5a) Gripper jaws "loaded" (rod-sensing tips up).
- (6a) Hold-down at "up" elevation.

The hold-down tube is lowered over the selected subassembly and seated on the adjacent subassemblies. The gripper unit is lowered through the hold-down tube and affixed to the subassembly. (The gripper unit is similar to the units on the control rod drives, employing a two-jaw gripper and a sensing rod to indicate that a subassembly has been grasped.) The downward travel of the gripper unit drive is stopped at the core by a force-sensing limit switch. The switch serves to prevent damage by overdriving, and to assure firm contact with the subassembly. The "down" position interlock signal is derived from coincident actuation of the force-sensing limit switch and an "approximately down" switch.

The gripper unit is raised, to remove the subassembly from the reactor (at a speed of 6 in./min for 21 in., and then 72 in./min for the remainder of the travel). During the removal of the subassembly from the reactor, the hold-down tube remains firmly on the adjacent subassemblies preventing inadvertent raising of these units. After the subassembly has been removed, the hold-down tube is raised to the "up" position approximately $4\frac{1}{2}$ ft below the top of the subassembly), to provide support for the subassembly during lateral movement to the transfer position.

The small and the large rotating plugs, and the gripper unit are each rotated to the proper angular orientation to effect transfer to the transfer arm - the "transfer" position. The transfer arm is rotated through an arc to engage the intermediate collar on the subassembly adapter. (Note: The transfer arm is rotated manually and engagement with the subassembly can be felt by the operator.) With the transfer arm in "transfer" position, the gripper unit is lowered, permitting the subassembly collar to rest in the recessed socket. A second force-sensing switch is employed here to stop the downward travel of the gripper drive, and to detect the joint engagement of the gripper and transfer arm with the subassembly. The locking bar is closed to prevent disengagement of the subassembly from the transfer arm. Interrelationship of these mechanisms at the transfer point is shown [Fig. 31]. The gripper jaws are opened and the gripper unit is raised to release the subassembly. (The sensing rod drops to indicate that disengagement has been effected.) The hold-down tube is lowered to the "operate" position to permit the subassembly to clear as the transfer arm is rotated.

The various operations of the gripper mechanism and hold-down mechanism are closely interrelated with respect to relative motion and relative position. The above description is a simplification of these operations to clarify the basic functions of these devices. Figure 10 describes the motions and positions in greater detail.

The transfer arm is rotated and positioned beneath the refueling port in the primary tank (Fig. 8). The manual grappling device is lowered through the refueling port to engage the subassembly. The manual grappler also employs a gripper head and gripper jaws to engage the subassembly. When engagement is effected, the locking bar in the transfer arm is retracted. The manual grappler (and the subassembly) is lifted slightly to permit the transfer arm to be rotated clear of the subassembly. The subassembly is withdrawn from the primary tank.

Consistent with the "replacement" loading philosophy, a new subassembly is loaded in the vacated core position by a series of operations in which the order of succession is reversed, and in which each step is approximately the converse of the corresponding step in the unloading sequence. Again, interlocking permits the execution of each step only under safe conditions and in the proper sequence.

Upon completion of loading operations, the reactor is returned to the "reactor operating" condition: the rotating plugs are indexed to the "operate" position; the reactor cover is lowered and locked; the control rod drive units are plugged in; and the control drive-lifting platform is lowered to the "down" position (Fig. 9 - Position No. 4). (The sensing rods raise as the drives engage the control rods.) The gripper jaws on the twelve control rod drives are closed, and the control rod drive-lifting platform is raised ($\frac{3}{4}$ in.) to the operating position. The reactor is now prepared for operation and the selector switch on the control console is set on the "reactor operate" position. (The fuel-handling control power key switch on the fuel-handling console must be turned to "off" to permit energizing the reactor control power. The key is removable only in the "off" position.)

C. Reactor Control

1. Operational Considerations

Reactor operation is controlled by the twelve control rods as described [pages 30 to 34, 57, and 61]. (Automatic shutdown of the reactor is effected as described in Section III-A).

Certain inherent features in the control rod drives provide added reliability in the conduct of reactor experiments. Although many of these features are incorporated to meet the requirements of power operation of the reactor system, they are available for the Dry Critical Experiments.

Noteworthy among these is the requirement for disconnecting the drives from the control rods as a routine operation (enforced by appropriate interlocks), to effect fuel-loading operations. This provision results in the completely independent installation, removal, etc., of the drives and the rods. In no case is it necessary to remove, move, alter, perform maintenance operations, etc., on the drive while it is engaged to the control rod. Secondly, since the engagement and disengagement operations are routine, very reliable sensing and interlocking of these operations are provided [Fig. 23].

Refueling operations require that the control rod drives be raised 3 in. to provide clearance between the bottom ends of the disengaged drive shafts and the top of the subassemblies. The raising operation is accomplished by the control rod drive-lifting platform completely independent of normal control rod motion. Motion of the platform is absolutely restricted to the 3 in. of travel. It is impossible to inadvertently overtravel during the lifting operation, either as a result of operator error or equipment malfunction. In addition, as described [page 33], the lifting platform is interlocked electrically and mechanically, and cannot be raised until the control rods are released from the drives, and the sensing devices indicate that release has been effected. The various control drive functions are shown schematically in Fig. 9.

2. Description of System Design

Although the design of the control rod drives has been described [pages 31 to 33], the salient features are repeated here.

Reactor control is effected manually from the operating console in the control room. Only one of the twelve control rods can be moved at a time, as controlled by a 12-position selector switch. Rod selection can be changed at any time and the rod movement pattern can involve raising each rod through its full stroke, or raising each rod in steps; but in no case can more than one rod be moved at a time except during scram when all twelve rods drop.

Figure 11 is a composite photograph and schematic of the control rod drive which describes the mechanisms involved and supplements the following description.

Rod motion is effected by a constant speed (5 in./min), instantly reversible, polyphase drive motor. The drive tube (rack tube) is engaged to the main drive shaft through a fast-acting magnetic latch. The main drive shaft is attached to a piston in a pneumatic cylinder. The upper end of the cylinder contains compressed air at a pressure of 35 psi acting against the piston and always available to drive (scram) the rod downward when the magnetic latch is disengaged. Latch disengagement is effected by interruption of power to the magnet, whether in response to an actuating signal, or

due to power failure (fail safe). Also, scram is effected completely independent of normal drive motion: the drive shaft moves through the rack tube irrespective of the direction of the rack tube travel.

The compressed air for the pneumatic cylinders is supplied from an accumulator tank which is isolated from the compressed air supply by check valves. The accumulator tank is equipped with a pressure switch which causes a scram if the pressure drops to 25 psi. In the event that the compressed air supply fails, and the associated low pressure-initiated scram also fails, the rods will drop under the force of gravity. Each control rod drive has been tested at various compressed air pressures, including 0 psig. These tests and the time-displacement curves are reported in Appendix A.

Position indication for each rod is given in the control room. The position indicator consists of a moving tape which provides graphical position indication, and a digital indicator which provides accurate position indication to 0.01 in. Limit switches indicate the extremities of the travel (up and down), and automatically interrupt the circuit to the drive motor. Added reliability in preventing overtravel is provided by the magnetic latch. In the event that the upper limit switches fail (there are two in series), and the drive motor continues to raise the drive tube, the piston engages the end of the air cylinder, restricting further movement of the main drive shaft (and therefore of the control rod). The continued upward motion of the drive tube automatically disengages the latch (mechanically) and effects a scram. The rod cannot be raised inadvertently out of the core, and if the normal preventive devices fail, an automatic mechanical scram results.

The twelve control rod drives are identical and the power and control circuits to each are carried through a cable to a multiconductor plug. This plug-in arrangement is provided to permit electrical disconnection of the control drives for fuel-handling operations that require rotation of the plugs on which the control rod drives are mounted. Again, a feature incorporated for routine operation can be used to provide reliability for experimental operations. Some of the experiments to be performed require the removal of a control rod drive (oscillator rod calibration, wet critical in-core instrumentation calibration, etc). Physical removal is facilitated by pulling a plug to effect all electrical disconnections (no manual terminal disconnections are required). More important, this arrangement affords a very simple method of "simulating" control rod drive positions for the removed drive. The various position indicators on the control drive must be properly actuated to permit appropriate actions in the control drive- and fuel-handling operations. Extensive interlocking and sequence controls are involved. When a drive is removed this will be accomplished by plugging in a temporary switch box which can simulate with appropriate switches, each position indicator and thereby permit the appropriate subsequent operations (for example: gripper open, sensing rod down, etc., for fuel handling). Two such units will be available for the Dry Critical Experiments.

Minor modifications have been made in the basic design which alters slightly the performance contemplated. The air pressure in the scram cylinders has been reduced from 50 psi to 35 psi resulting in an acceleration of 1.5 g instead of 2.0 g. The accelerating force was lowered to reduce the acceleration and deceleration loads on the rods. Also the magnetic force on the magnetic latch was increased to improve reliability. This increased the latch release time from 0.003 - 0.007 sec, to approximately 0.014 sec. Although scram is somewhat slower than anticipated, a higher degree of reliability has been achieved.

The safety rod system has been revised to permit the manual scram of the two safety rods during reactor operation by depressing a button on the operating console. The primary function of the safety rods is to provide available shutdown reactivity during fuel-handling operations [page 31]. The safety rods are scrammed automatically during these operations. These functions and system interlocks are unchanged. The rods have now also been made manually operative during reactor operation which makes an additional manual scram operation available to the operator. This is not a "normal" operation but has been added to provide additional reliability. Reactor scram, therefore, may be effected by automatic actuation of the control rods, manual actuation of the control rods (normal manual scram), and manual actuation of the safety rods.

D. Reactor Building Containment

The proposed dry critical experiments will be conducted within a "building containment system" designed, constructed, and tested to criteria based on full-power (62.5 Mw) operation of the EBR-II reactor, and where the major energy release is due to sodium-air reaction. This system consists basically of a steel shell which encloses the complete Reactor Plant, and which is designed to be leak tight at an internal pressure of 24 psig. A relatively detailed description of the system has been presented [Appendix E]. This description remains valid except for minor revisions relating principally to the number and function of the "penetrations" discussed below.

The building shell is provided with numerous penetrations for personnel, equipment, freight, electrical and instrument conductors, and service fluid. Three air locks are available for personnel and equipment access. Each air lock features two doors with latching mechanisms interlocked to ensure that at least one door is closed at all times. Double seals - one compression, and one inflatable - are employed on each door.

A large opening is provided for freight access. The opening is fitted with double doors that are not interlocked but are required (administratively) to be bolted closed during reactor operation or during fuel-handling operation. The inner door employs double seals, both of the compression type. The outer door utilizes a single, compression-type seal.

Electrical and instrument conductors are brought through leak tight pressure connectors (pressure rating: 30 psi). Twelve conductors carry large current; each consists of a short length of MI cable brazed to a nozzle plate. The plate is fitted with terminals at each end of the center conductor and the insulation at each end is sealed with epoxy resin. The remainder are A-N-type multiconductor pressure connectors which provide solid, leak tight, through-conductors with terminals at each end. Each electrical cable begins or ends at these terminals; no cable (other than the 12 MI cables) is required to penetrate the containment shell.

There are 25 service fluid pipes of various sizes that penetrate the shell. They are seal welded either directly to the shell, or to a blind flange bolted to a thermal sleeve which is welded to the shell. Eighteen of these pipes are equipped with an isolation valve which is normally open but which closes automatically in event of a major nuclear incident. The other seven pipes require no isolation valve because the exterior portions of the piping systems involved are capable of withstanding conditions more severe than those anticipated subsequent to a nuclear incident. These include: four NaK pipes for the two shutdown coolers; two secondary sodium pipes; and a sodium fill line for the primary sodium system (which is subsequently capped after filling). One pipe, for the air bleed-in system, is equipped with a valve which is normally closed; it opens automatically only to provide bleed-in of air to prevent excessive reduction in building pressure due to consumption of oxygen in the event of a major sodium-air reaction.

Two series of pressure tests and leakage tests have been conducted on the EBR-II building containment system. The first series was run upon completion of construction of the shell proper, with the airlocks and freight door installed, but with no other penetrations. At that time, the shell was bare - without paint, bitumastic, insulation, etc., - and was supported by temporary steel columns, so that the entire shell surface was accessible for inspection. A preliminary leak test first was effected with 10 psig internal air pressure by soap-suds testing the full length of every weld on the shell, both airlocks, and the freight door. A pressure (strength) test then was conducted at 30 psig internal pressure. Finally, a leak rate test was performed at 24 psig over a period of more than 72 hr. The results showed a total leak rate from the system of less than 560 ft³/day (32°F, 36.2 psia).

The second series of tests was run upon completion of construction of the Reactor Plant except for installation of certain mechanisms and equipment inside the building. All penetrations had been incorporated, the shell had been set on its permanent foundation, and all painting, insulating, backfilling, etc. had been completed. These tests were conducted, in essence, on the "integral" containment system. Three of the total number of penetrations were temporarily sealed off: the two penetrations for the secondary system pipes, because their seal bellows were not yet available; and one electrical penetration, because a connector was known to leak and

required replacement. (The two secondary piping penetrations will remain sealed off, with blank flanges, until the Dry Critical Experiments are completed. The faulty electrical connector has been replaced and the penetration successfully tested.) A pressure test was first conducted at 30 psig. Then a final leak rate test was performed at 24 psig over a period of more than 72 hr. The results of this test, more accurate than those of the earlier test, showed a total leak rate from the system of less than 100 ft³/day (32°F, 36.2 psia). This is equivalent to less than 0.03% of the building air volume, at the same conditions, per day.

During the proposed Dry Critical Experiments, most of the service pipes discussed earlier need not be operative. Accordingly, all isolation valves except two (one each on the "instrument air" and "plant service air" lines) will be closed as an added safety precaution. The two valves left open will be closed automatically by the building isolation system in the remote event of a significant nuclear incident. The trip signal will be initiated in the normal manner by high radioactivity level, high temperature, or high pressure within the building. Between experiments, the ventilating air pipe isolation valves will be opened occasionally to freshen up the building air for personnel comfort.

The proposed utilization of the containment system as described above should provide a degree of containment for these experiments substantially higher than that normally provided for a critical facility. This results primarily from the fact that the containment system design is based on criteria more severe than those represented in the proposed experiments.

IV. EXPERIMENTAL PROGRAM

The dry critical program has two basic objectives:

- (1) to obtain fundamental information relating to the neutronics of the reactor system; and
- (2) to verify certain design information.

The program will include the following experiments which will be conducted approximately in the order listed:

- (1) Neutron source and instrument response.
- (2) The approach to criticality.
- (3) Neutron flux and power calibration.
- (4) Reactivity measurements.

A. Neutron Source and Instrument Response

The relationship between neutron source strength and neutron detector response is important to the conduct of the Dry Critical Experiments, and to the basic design of the reactor system as it relates to normal operation. The Dry Critical Experiment will permit establishing this information reliably and safely because of the allowable use of special "in-core" instrumentation to augment the normal instrumentation which is located external to the reactor. The additional instrument sensitivity available will permit flux measurement at source levels and very low multiplications, and simultaneous determination of the response of the normal instrumentation. This, in turn, will provide an assessment of the normal instrumentation at a time when modifications can be made, if necessary. Similar measurements were made in the EBR-II Mock-up critical experiment conducted in ZPR-III. These results are described in Appendix A.

B. The Approach to Criticality

This phase of the program is an essential sequence of operations to permit a safe approach to the critical reactor configuration. The procedure will provide useful information pertaining to the critical approach when sodium is present. The extra in-core instrumentation for the Dry Critical Experiments will provide an extra margin of safety during the conduct of these experiments, in addition to more detailed information relating to the characteristics of the system.

Viewing devices will permit observation of the various mechanisms associated with fuel handling and reactor operations. Observation of these mechanical operations will provide assurance that these operations can be performed reliably later under sodium.

The determination of the dry critical mass when compared later with the wet critical mass, will provide a measure of the total reactivity worth of the sodium coolant in EBR-II. Attempts were made to measure sodium worth in the EBR-II Mock-up critical experiment in ZPR-III. Results of sufficient accuracy and reliability were not obtained. This uncertainty should be resolved with the performance of these experiments in EBR-II.

C. Neutron Flux and Power Calibration

These experiments have four primary objectives:

- (1) To afford a precise calibration relating instrument response to dry reactor power level.
- (2) To calibrate high-temperature fission counters relative to an absolute fission counter which will provide a calibrated high-temperature in-core detector for the wet critical experiments.
- (3) To determine leakage neutron flux from reactor blanket and neutron shield.
- (4) To determine fission distributions; principally in the reactor blanket.

A precise power calibration is necessary to provide knowledge of, and to control, power level of the dry reactor system. This information is required to assess the response of the normal reactor instrumentation and to prevent overheating of the reactor during the experiments.

It is necessary to calibrate the high-temperature fission counter relative to an absolute fission counter to establish an absolute power calibration during the wet critical experiments. The absolute fission counter used in the Dry Critical Experiments cannot be used during the wet critical experiments because molten sodium temperatures exceed the temperature limits at which the counter can operate.

Leakage neutron flux from the reactor blanket and neutron shield is measured principally to determine the effectiveness of the neutron shield. The neutron shield effectiveness will determine the level of activation of removable components in the primary tank, and activation of secondary sodium (as it passes through the heat exchanger). These measurements will be related to reactor power to permit extrapolation to power operation.

Fission rates are required in radial blanket subassemblies relative to total reactor power. Power distributions in the radial blanket are uncertain because of the geometric asymmetries inherent in the hexagonal reactor structure.

D. Reactivity Measurements

Several types of reactivity effects will be measured to afford a means of evaluating design parameters as well as to achieve understanding of the neutronics of the reactor. These measurements will provide:

- (1) Total reactivity worth of all twelve control rods.
- (2) Total reactivity worth of the two safety rods.
- (3) Total reactivity worth of a single control rod.
- (4) Incremental calibration of a single control rod.
- (5) Total reactivity worth of a single "special" control rod.
- (6) The dry isothermal temperature coefficient of reactivity.
- (7) A static calibration of the oscillator rod.

The various rod calibrations are determined to establish the reactivity worth for reactor operation and shutdown. They are also required to provide a method of measuring reactivity changes for experimental purposes; i.e., to determine the isothermal temperature coefficient, to calibrate the oscillator rod, etc.

A "special" control subassembly containing "poison" as well as "fuel" has been designed to provide operational flexibility for the experimental program; it is not intended for normal plant operation. Certain experimental procedures require the removal of one, but not more than two, control rods. For instance, the oscillator rod is to be calibrated in a control rod location, and there will be only eleven control rods available for shutdown. By replacing one (or two) of these eleven control subassemblies with "special" control subassemblies, the amount of shutdown reactivity can be maintained relatively constant for all operations.

The calibration of the "special" control subassembly will provide data applicable to the design of "stronger" control rods for future operation of the plant should this become necessary. This will be particularly relevant if the measured reactivity worth of the normal control rods is significantly less than indicated by the ZPR-III experiments (Table IV and Appendix A).

The additional reactivity held by the "special" control subassemblies is obtained by placing a $B_4^{10}C$ follower in the coolant void section above the fuel elements. The follower is comprised of stainless steel-clad $B_4^{10}C$ rods (pellets), 7 in. long. The bottom of the $B_4^{10}C$ is 7 in. above the upper extremity of the fuel alloy.

The total reactivity worth of the "special" control subassemblies will not exceed $0.6\% \Delta k/k$ (ZPR-III results indicate approximately $0.55\% \Delta k/k$). Therefore, the rate of reactivity insertion does not exceed the rate used to specify initially the reactor control rod drive speed [Section IV - page 57].

The dry isothermal temperature coefficient is a parameter of basic interest, and, when compared to the wet isothermal temperature coefficient, provides a very interesting measurement of the sodium coefficient under uniform heating.

The static calibration of the oscillator rods will provide a verification of the design several months prior to operation. This will permit modifications to be made, if necessary.

V. OPERATIONAL AND EXPERIMENTAL PROCEDURES

A. General Considerations

The experiments will be performed roughly in the order listed in Section IV. However, it will be both convenient and necessary to perform certain experiments before the sequence of Section IV might indicate. For example, in order to establish that 1 kw reactor power is not exceeded, an absolute power calibration will be performed at high multiplication (~ 30) prior to the completion of the initial critical approach.

The following sections describe the detailed sequence of these experimental procedures with reference to Fig. 5. (Figure 4 shows the relative location of the normal and in-core nuclear instrument thimbles.) The J1 and J2 thimbles contain the low-power fission counters (Channels 1, 2, 3). Subassembly location is identified by row, sector, and position in the sector. The row designation defines the hexagonal row in which the subassembly is located. Thus the central core subassembly is in (and is) row 1. The safety and control subassemblies are in rows 3 and 5, respectively. The sector designation (A, B, C, D, E, F) in conjunction with position in the sector (counting clockwise), locates the exact position of a subassembly.

Initially, the reactor will be filled with subassemblies and thimbles containing only steel, depleted uranium, or natural uranium. Subsequently, the enriched uranium-bearing control and safety subassemblies will be loaded into the reactor. After these initial operations have taken place, all subassembly changes in the reactor will be effected by replacement. A subassembly must be removed before a different subassembly can be added. As described in Section III-B, fuel-handling operations will be conducted so that only one reactor grid plate position will be empty at any one time (except for the special case of safety or control rod thimble removal). The reactor grid plate is completely filled during reactor operation.

All measurements are restricted in that:

- (1) Reactor power level does not exceed 1 kw.
- (2) An Sb-Be neutron source is in the reactor during all fuel-handling operations and reactor operating procedures.
- (3) Reactor period will be longer than 15 sec.

To assure adequate instrument response and to operate the reactor at low power level, three neutron sources are provided for the experimental program:

(1) Two strong neutron sources each having 100-200 curie of Sb activity.

(2) A weak neutron source having 10-20 curie of Sb activity. A strong neutron source will be used during the critical approach. The other strong neutron source is a standby. Subsequent to the critical approach, the strong neutron source will be replaced by the weak neutron source with the reactor subcritical. (The antimony rod is moved by the fuel-handling system which is only operative with the reactor shut down; also, with the low count rate interlock bypassed with key switch.) The use of a strong neutron source during the critical approach assures high instrument response at low reactor multiplication. The use of a weak neutron source subsequent to the critical approach permits reactor operation at low (<10 watt) powers. Low-power, steady-state operation is required to prevent excessive activation or heating of the fuel.

A further restriction requires that the fuel element sodium bond temperature not exceed 180°F at any time. This requirement does not encumber the program since the reactor can be operated at 1 kw for 2 hr (without heat dissipation) before this temperature is reached. Contemplated steady-state operation will not exceed 100 watt. If no heat is lost from the fuel elements, bond temperatures approach 200°F after 17 hr of steady-state operation at 100 watt.

The central subassembly (No. 1 - Fig. 5) will contain seven fuel elements each with two thermocouples attached to the outside surface of the fuel tube. Temperatures will be recorded on a multipoint recorder with an alarm set point of 180°F .

The relationship of reactor grid hole size and subassembly adapter size precludes most loading errors. In addition, each subassembly is identified by letter and number stamped on the top end fixture. The letter refers to the type of subassembly. As each subassembly is inserted into the reactor, a record is made of the particular subassembly and the grid plate location. When a subassembly is removed from the reactor it will be recorded by type, number and the reactor location it occupied. The amount and disposition of enriched uranium is available at all times during the experimental program. The serial numbers used to identify the various subassemblies are listed in Table III.

As a further precaution a special color code will distinguish between enriched and depleted inner blanket subassemblies. Both types of subassemblies are accepted by the inner blanket grid plate locations. A large number of enriched inner blanket subassemblies (about 25) are required because the dry critical mass is considerably larger than the wet critical mass which determined the reactor grid configuration. To reduce the possibility of inadvertent interchange, the top end fixtures of all inner blanket subassemblies

are colored red for enriched uranium and blue for depleted uranium. This will provide an easily visible identification during loading and will permit visual review of the subassembly distribution in the inner blanket at all times.

Table III
SUBASSEMBLY IDENTIFICATION SCHEME
(First EBR-II Core Loading)

Type of Subassembly	Type Identification	Number Sequence	Color of Top End Fixture
Natural Uranium Core	N	6 - 26 (Omit 21)	Blue
Enriched Uranium Core	C	100 - 154	-
Enriched Uranium Inner Blanket*	B	300 - 317	Red
Enriched Uranium Inner Blanket*	D	350 - 373	Red
Control Rod	L	400 - 413	-
Special ($B_4^{10}C$) Control Rod	T	500 - 502	-
Safety Rod	S	600 - 602	-
Depleted Inner Blanket	A	700 - 771	Blue
Depleted Outer Blanket	U	1000 - 1547	-
Dummy Core (No Uranium)	E	50 - 81	Blue

*The B and D series are identical. However, the total number of enriched inner blanket subassemblies (42) is far in excess of requirements for power operation. The "D" series will be refabricated after the Dry Critical Experiments are completed. They are identified by a different letter series to permit easy separation for salvage; special fabrication procedures were followed to simplify the salvage operation.

The natural uranium and the stainless steel dummy core subassemblies are also colored blue as indicated in Table III. After the initial loading of subassemblies (prior to the introduction of enriched subassemblies) the entire core and inner blanket region (rows 1 through 7) will be filled with "blue" subassemblies. After the 61-subassembly core has been filled with enriched subassemblies (not colored) the inner blanket region will stand out as a two row band of blue subassemblies. The subassembly pattern in this region will then be very easily visible and identifiable.

Another significant operational consideration for the Dry Critical Experiments is that subsequent to the initial critical approach there are no operations that require handling of fuel subassemblies within the nominal 61-subassembly core. Therefore, fuel subassembly handling is restricted

to subassemblies holding less than $0.5\% \Delta k/k$. The only exception would arise if it should become necessary to remove a control or safety rod thimble (as described in Section III-B). The latter operations are not contemplated.

Subsequent to the initial critical approach, the loading of the reactor is reviewed and altered, if necessary, to provide less than one dollar of excess reactivity by control rod motion. Furthermore, a single control rod holds less than a dollar of reactivity. (Normal control rods each hold about one-half dollar of reactivity).

All plant procedures are controlled and initiated in the control room. Fuel handling operations are carried out in the reactor plant but only after control power is made available by a control room supervisory key switch.

B. Procedures Prior to the Critical Approach

An extensive series of preoperational tests will precede preparations for the critical approach. All mechanical components pertinent to the Dry Critical Experiment will be tested in place. Control and safety rod drives will be subjected to normal operation and scram procedures. Fuel-handling equipment including the rotating plugs, will be operated through each step involved in the loading and unloading of subassemblies into the reactor.

Fission counters, ion chambers and associated preamplifiers, amplifiers, scalars and recorders, etc., will be placed into operation and their performance evaluated. Period and power level trips will be set and tested for response (see Section III-A).

Each component of the reactor shutdown circuit will be tested for proper function upon scram signal. These tests apply to both reactor operation and fuel handling circuits.

Subsequent to the preoperational testing of the reactor components a sequence of operations is required to prepare the reactor system for the critical approach. These consist of loading the reactor with depleted and natural uranium-bearing subassemblies, a number of dummy steel subassemblies, and the control and safety rod thimbles. Subsequent to these operations, the in-core instrumentation is installed. The strong-neutron source is assembled and enriched uranium bearing control and safety subassemblies are mechanically tested.

Detailed procedures are as follows (identification of subassembly locations are shown in Fig. 5).

- (1) Special thermocouple core subassembly is loaded into central position (row 1) and thermocouple leads are brought out through the bottom of the reactor, under the neutron shield and to the operating floor of the reactor building.

- (2) Control and safety rod thimbles are installed.
- (3) Dummy steel¹ core subassemblies are loaded into rows 2, 3, and part of row 4 (26 subassemblies). Natural uranium² core subassemblies are loaded to fill rows 4 and 5 (20 subassemblies).
- (4) Normal inner blanket subassemblies (depleted uranium) are loaded in rows 6 and 7 (66 subassemblies).
- (5) Outer blanket subassemblies (depleted uranium) are loaded (510 subassemblies).
- (6) Reactor grid plate is filled at this point.
- (7) Inner blanket subassembly 6-B-1 is removed and replaced by in-core instrument thimble.
- (8) Outer blanket subassembly 8-B-6 is removed and replaced by in-core instrument thimble.
- (9) Inner blanket subassembly 7-E-3 is removed and replaced by a source thimble.³
- (10) Two outer blanket subassemblies (11-F-5, 6) are removed and replaced by lead-lined source shield thimbles for future storage of the antimony rod.
- (11) In-core instrumentation is inserted into the in-core thimbles (6-B-1, 8-B-6). Each thimble will contain one high-temperature fission counter (Section III-A), and one absolute fission counter. The preamplifiers are located in the entrance to the coolant exit pipe leaving the coolant outlet plenum. The instrumentation cables are routed between the thimbles and preamplifiers over the outer blanket region. Cable installation is arranged to prevent interference with subsequent fuel-handling and reactor-operating procedures.
- (12) The active (>100 curie) antimony rod is located and fixed in the source subassembly thimble in position 7-E-3. Source power is not expected to exceed 0.01 watt.

¹These are subassemblies filled with stainless steel. Their function is simply to fill the grid plate and provide for a solid high density system prior to loading elements containing enriched uranium.

²These are core subassemblies containing natural instead of enriched uranium.

³This thimble contains the Be cylinder of the Sb-Be neutron source.

(13) Count rates are determined in the "J" thimble fission counters and the in-core instruments. If at least 3 count/sec are observed in J1 and J2 thimble counters, loading of enriched elements can proceed. However, if less than 3 count/sec are noted by more than two in-core counters, the second strong neutron source will be assembled in position 7-E-5. (Counting rates approximately 10 times as high are anticipated).

(14) The two safety rods are loaded into the safety rod thimbles and attached to the safety rod drive mechanism. The safety rod drive mechanism is tested for operation and scram. The safety rods are raised to permit loading to proceed.

(15) The twelve control rods are loaded into the control rod thimbles. (In their loaded position, the active sections of the control rods are below the core - their least reactive position - and disengaged from their drives.)

Note: A total of about 29 kg U^{235} is contained by control and safety rods, plus the central thermocouple subassembly. The fuel alloy contained in these subassemblies is less than 50% enriched in U^{235} . In optimum spherical geometry with a high-density (~ 18 gm/cc) uranium core and reflector, about 25 kg U^{235} are required to achieve a critical configuration. Unreflected, the critical configuration would require about 75 kg U^{235} at high density. Neither the geometry nor the disposition of the fuel in the reactor is spherical. The reflector permeating and surrounding these subassemblies is effectively low density. If it is assumed that these fifteen subassemblies were to interact with each other, such interaction is most pronounced when tightly packed against each other. This is not physically possible within the reactor configuration. However, such a tightly packed configuration is calculated to be less than 20% of a critical mass. Therefore, there are no criticality problems associated with the even less reactive disposition of the fuel alloy as contained in the control, safety, and thermocouple subassemblies.

(16) The Sb-Be neutron source is disassembled by removing the antimony rod from the source thimble. (The low count rate start-up interlock is bypassed with the key switch described in Section III-A.) The antimony-bearing rod is then lowered into the source shield thimble (11-F-5). This procedure permits personnel access to the primary system to observe control rod operation. If two neutron source subassemblies are required, both sources are disassembled. The second source shield subassembly is located in position 11-F-6.

(17) The reactor cover is lowered and the control rods are engaged.

(18) Each control rod is checked out individually. The other eleven are in the "down" position while individual checkout proceeds. Checkout consists of:

- (a) Rod is raised to maximum "up" position.
- (b) Rod is operated up and down as though regulating power level.
- (c) Rod is raised to maximum "up" position.
- (d) Rod is scrammed.
- (e) Rod is raised to maximum "up" position.
- (f) Rod is lowered to the "down" position.

(19) With the twelve control rods in the "down" position the safety rods are manually scrammed. The safety rods are raised to the "up" position.

(20) The control rods are disengaged and the reactor cover is raised.

(21) The active antimony rod is removed from position 11-F-5 and returned to and fixed in the source thimble located at 7-E-3. The low count rate interlock bypass is removed by turning the key switch. The key is removed.

(22) At this point the preparations prior to the critical approach are complete.

C. The Approach to Criticality

The approach to criticality will be conducted in a series of steps. Figure 12 shows the expected variation of inverse count rate as a function of the number of enriched uranium-bearing subassemblies loaded into the reactor core. It is to be noted that all fuel subassemblies are loaded into the reactor while the system is in its least reactive configuration (the control rods disengaged and in their lowest - least reactive - position).

The upper limit on the reactivity worth of fuel, control, and safety subassemblies has been established on the basis of measurements at ZPR-III. Table IV compares the expected reactivity worths, and reactivity insertion rates associated with control rod, safety rod, and fuel handling mechanisms in the Dry Critical Experiment with the "predicted" and "measured" values. The "predicted" values are given in ANL-5719 [Appendix A]. The "measured" values are as obtained in ZPR-III and described in Appendix A. It can be seen that the total worth of a control rod will be about one-half dollar of reactivity.

The highest anticipated multiplication during fuel-loading procedures is about 30. This applies during the last stage of adding fuel subassemblies prior to raising the control rods.

Table IV

NORMAL RATES OF REACTIVITY INSERTION BY VARIOUS DRIVE MECHANISMS

	Control Subassembly			Two Safety Subassemblies			Core Subassembly Loading Mechanism (Central Core Subassembly)		
	Predicted ¹	Measured ²	Estimated ³ for Dry Critical	Predicted ¹	Measured ²	Estimated ³ for Dry Critical	Predicted ¹	Measured ²	Estimated ³ for Dry Critical
Total Reactivity Worth, %Δk/k	0.5	0.37	0.32	2.0	1.36	1.1	~2.0	1.5	1.2
Drive Speed, in./min	5.0	-	5.0	2.0	-	2.0	6.0	-	6.0
Effective Stroke, in.	14.0	14.0	14.0	14.0	14.0	14.0	-	-	-
Rate of Reactivity Addition, %Δk/k/sec									
Average	0.003	0.0022	0.0020	0.0050	0.0032	0.0026	0.015	0.011	0.0088
Maximum	0.005	0.0038	0.0034	0.0086	0.0055	0.0045	0.025	0.018	0.014

¹Predicted for wet critical reactor [Appendix A]. These reactivity insertion rates were used to specify control rod, safety rod, and fuel-handling mechanism drive speeds.

²Measured on EBR-II Mock-up on ZPR-III. The mock-up has the configuration of the wet critical reactor except that low-density aluminum represents the sodium coolant.

³These values are inferred from the "Measured" values. Reduced reactivity insertion rates and total reactivity worths prevail because the Dry Critical core is at least 20% larger than the "Measured" core.

The smallest estimated critical configuration consists of an 80-subassembly core, including control and safety rods. The more probable critical configuration is between 85 and 92 subassemblies. There are two basic uncertainties in the critical determination:

(1) Variation of reactivity worth of core subassemblies in row 6 as a function of position in the row which is a hexagonal configuration.

(2) Lack of precision in predicting the reactivity associated with the large (~50%) void in the reactor core, and in the structural axial gap regions between core and axial blanket.

The first increment of loading will not include more subassemblies than would yield a critical system if all the enriched uranium in the core subassemblies were highly enriched. A further margin of safety is that this prediction is based on a sodium-filled system. The two following loading increments consist of 18 and 12 core subassemblies. At this point the nominal 61-subassembly core including control and safety subassemblies contains about 160 kg U²³⁵. This is about the same amount of U²³⁵ contained in the EBR-II mock-up of ZPR-III (Appendix A). The dry reactor is known to be substantially subcritical at this point.

Subsequent loading increments shall consist either of six fuel subassemblies, or the indicated number of fuel subassemblies required for a critical configuration (with the control rods in). The smaller number of subassemblies of the two alternatives shall constitute the loading increment. The limitation is to insure that the reactivity held by the twelve control rods is at least twice that held by the loading increment.

The detailed procedures of loading fuel subassemblies follow. All nuclear instrumentation (Section III-A) is operative. Trip settings are:

Reactor Period - 15 sec

Reactor Power Level - 1000 watt

Reactor neutron flux is monitored at all times. Count rates to establish the experimental equivalent of Fig. 12 will be taken during, and subsequent to, each step in the loading sequence. In particular, if there is an appreciable time lag (on the order of 4 hr or more) between the completion of one step and the initiation of the next, initial condition count rates are taken and compared with those taken subsequent to the completion of the previous step.

All loading of fuel subassemblies will be symmetrical about the core periphery; subsequent to the addition of any fuel subassembly, the fuel will be in the most symmetric and compact configuration, with respect to the hexagonal array.

The "critical approach" will begin with the reactor system in the condition described in Section V-B. It will include the following steps:

(1) Replace dummy steel assemblies in rows 2 and 3 with enriched core subassemblies. Number of subassemblies: 16. Obtain count rate.

(2) Lower reactor cover; engage and raise control rods. Obtain count rate.

(3) Lower control rods; disengage and raise reactor cover.

(4) Replace natural uranium and dummy core subassemblies in row 4 with enriched uranium core subassemblies. Number of subassemblies: 18. Obtain count rate.

(5) Lower reactor cover; engage and raise control rods. Obtain count rate.

(6) Lower control rods; disengage and raise reactor cover.

(7) Replace natural uranium core subassemblies in row 5 with enriched uranium core assemblies. Number of subassemblies: 12. Obtain count rate.

(8) Lower reactor cover; engage and raise control rods. Obtain count rate.

(9) Lower control rods; disengage and raise reactor cover. At this point the nominal 61-subassembly reactor core is filled.

Subsequent loading increments consist of six subassemblies until the critical approach is apparent on the basis of the experimental equivalent of Fig. 12. Then the loading increment shall consist of the number apparently required to achieve criticality; however, the number of subassemblies shall not exceed six at any time.

(10) Replace depleted inner blanket subassemblies in row 6 with six enriched uranium inner blanket subassemblies. These are loaded symmetrically, one on each hexagonal sector (position 3 - row 6 - each sector). Obtain count rate.

(11) Lower reactor cover; engage and raise control rods. Obtain count rate.

(12) Lower control rods; disengage and raise reactor cover.

(13) The sequence of steps (10), (11), and (12) is repeated (position 4 - row 6 - each sector, then position 2, then position 5, etc.), until the critical approach is apparent from the experimental multiplication curve. The loading increment is then reduced as suggested by the experimental inverse count rate determinations.

(14) When 73 elements are loaded into the reactor core, the absolute fission counters are used to calibrate reactor source power. The absolute counters will register about 25 count/sec for a source power of 0.2 watt at a multiplication of about 20. If the power calibration suggests that these estimates are appreciably exceeded (and the count rate is significantly higher), the neutron source will be relocated at a greater distance from the center of the reactor core. (The availability of two strong sources and one weak source makes it possible to effect this relocation with a neutron source in the reactor at all times.)

(15) The final subassembly increment necessary to achieve criticality with control rods inserted is loaded as follows:

- (a) Drop safety rods.
- (b) Raise safety rods.
- (c) Insert the final loading increment (less than six subassemblies).
- (d) Lower reactor cover; engage and prepare to raise control rods.

(16) Successive count rate determinations during the rod raise sequence will establish whether criticality can be attained without loading more fuel subassemblies. If criticality is not realized with all control rods raised, step (15) is repeated.

(17) As the approach to critical becomes apparent, the control rod is moved in increments of not more than 3 in. This will assure a reactor period of not less than 50 sec. The exact period will depend on the location of the rod as well as the degree of insertion prior to divergence.

(18) After the reactor has demonstrated divergence as exhibited by a period of about 50 sec or longer, the control rod is moved to give a stable power level at less than 50 watt. It is to be noted that this will not be the true critical configuration, nor will the observed period necessarily be the true asymptotic period. This results from the rather low power level of operation at which the Sb-Be source supplies an appreciable number of neutrons. For example, with a 150 to 200-curie Sb neutron source, the reactor will be subcritical by about $\Delta k/k \approx 0.1\%$ (~ 40 inhours) at 10 watt of stable power.

(a) At this point the reactor loading is reviewed to provide less than one dollar of excess reactivity. If necessary, enriched inner blanket subassemblies are unloaded and replaced by depleted inner blanket subassemblies. If this procedure proves necessary, steps (17) and (18) are repeated. The reactivity worth of the last fuel subassemblies added in step (15) is estimated to be between 0.2 and 0.3% $\Delta k/k$.

(b) The reactor power is again calibrated as described in step (14).

(19) Lower and disengage control rods; raise reactor cover.

(20) Locate neutron source thimble in position 7-E-5 by replacing an inner blanket subassembly.

(21) Lower weak (10-20 curie) antimony source assembly into and fix in source thimble at position 7-E-5.

(22) Remove strong (100-200 curie) antimony source assembly from source thimble at position 7-E-3 and transfer to source shield thimble at position 11-F-5.

(a) Determine instrument response of normal and extra in-core instrumentation. Re-normalize critical approach curves on the basis of this determination.

(23) Lower reactor cover; engage and raise control rods. The rod-raising sequence will be essentially the same as that described in step (17).

(24) The point at which the reactor diverges will not be the same as in step (18). With the weak source in 7-E-5 and no other arrangement of fuel subassemblies, the reactor will require between four and five more inches of control rod insertion than in step (18).

(25) After divergence on a period longer than 50 sec, the reactor is brought to a stable power level at less than 10 watt. With the weak neutron source, the reactor is $\Delta k/k \approx 0.01\%$ (~ 4 in hours) subcritical.

(26) The reactor will be brought subcritical, the reactor loading will be reviewed and altered, if necessary, to provide less than one dollar of excess reactivity from control rod motion. It is not intended to operate the reactor during the Dry Critical Experiments with more than one dollar of excess reactivity available from control rod motion at any time.

D. Neutron Flux and Power Calibration

These experiments pertain to the spatial distribution of the neutron flux and the absolute power level of the dry reactor. Some of these experiments will be carried out simultaneously with those of V-B, V-C, and V-E. For instance, the first experimental power calibration will be performed during the critical approach (Section V-C), while the first instrument response measurements will be performed during the procedures prior to the critical approach (Section V-B). Such experiments will be described here in some detail.

1. Neutron Source and Instrument Response

The determination of the instrument response in the "J" thimbles is of paramount importance in establishing reliable neutron flux monitoring during fuel-handling and reactor-startup operations, and for subsequent power operation of the reactor system. Such experiments were performed on the EBR-II critical experiments on ZPR-III (Appendix A). Similar experiments on the Dry Critical will verify the final design of the J1 and J2 thimbles and associated instrument channels. The ZPR-III experiments have shown that fission counter response in these thimbles can be increased by a factor of about three by surrounding the fission counters with moderating material. Special aluminum cans filled with ZrH powder have been provided for the Dry Critical Experiments. These fit into the "J" thimbles and accommodate the fission counters. The cans are longer than the sensitive length of the counter, and each can has a moderator cross-sectional area of about 8 in.². Count rates obtained during the operations described in Section V-B will determine whether moderating material will be required during these experiments and/or subsequent plant operation.

2. Calibration of High-Temperature Fission Counters

Each of the in-core thimbles contains one of the high-temperature fission counters intended for use during the wet critical experiments. An absolute fission counter will also be located in each in-core thimble. Thus, the calibration of the high-temperature counters consists of comparing count rates in the four instruments. This is then refined on the basis of spatial variation of neutron flux relative to the actual location of the counters. The absolute fission counter in the 6-B-1 thimble is expected to yield 60 cps/(watt of reactor power) (~ 1 mg of U^{235} in counter). Similarly, the fission counters in the "J" thimble are expected to yield approximately 600 cps/(watt of reactor power).

3. Power Calibration

The reactor power will be continuously monitored and periodically calibrated relative to predicted and measured power distributions. The purpose of the periodic calibrations is to assure that reactor power does not exceed expectations. Furthermore, they will enable a realistic 1 kw reactor power level trip setting.

Predicted power level is based on a multigroup two-dimensional calculation correlated with measurements obtained on ZPR-III. The absolute fission counters which were calibrated in ZPR-III and AFSR, are the basic correlation in these determinations.

Initially, power level trip settings are based on ZPR-III measurements. These suggest that an absolute fission counter in the in-core thimble will record about 60 cps/(watt of reactor power)/(milligram of highly enriched uranium in the absolute counter). Under the same conditions the "J" thimble fission counters are expected to record about 600 cps.

When the reactor multiplication is about 20, the response of the absolute in-core counters is used to establish the power level on the basis of 60 cps/(watt)(milligram). Power level trip settings of "J" thimble and in-core high-temperature counters may have to be altered on the basis of this determination. Subsequent calibrations will be performed in the same manner.

Reactor power level trip settings will be altered only while the reactor is shut down. Under these conditions the reactor is at least 3% $\Delta k/k$ subcritical.

The power level will be limited throughout the Dry Critical Experiments. The power calibration affords a correlation between desired power of reactor operation and actual reactor operation. The ZPR-III measurements provide the initial basis for determining operating power levels and trip settings.

4. Leakage Neutron Flux

The purpose of this measurement is to determine the effectiveness of the neutron shield. This will be accomplished by measuring U^{238} fission flux and total neutron flux on both the inside (in the reactor vessel), and the outside (in the primary tank) of the radial neutron shield.

These measurements will be obtained by fission counters and foil activation.

5. Fission Distributions in Inner Blanket

Power distribution in the inner blanket adjacent to the core will be determined by activating foils distributed throughout a dummy inner blanket subassembly. This subassembly is an empty hexagonal can with inner blanket adapter and normal top end fixture. The objective in this investigation is twofold:

(a) To determine the variation in power distribution as a function of position within a hexagonal row of the inner blanket.

(b) To determine the variation of the neutron flux within a single inner blanket subassembly when located adjacent to a fuel subassembly in the inner blanket of the reactor.

E. Reactivity Effects

All measurements described in this section assume initial conditions as those at the conclusion of Section V-C unless specifically described otherwise. The determination of reactivity associated with control rods and temperature are the primary objectives of these measurements. The following experimental procedures will be employed:

(1) With the reactor critical between 5 and 10 watt, the control rods are manually scrammed. (It is assumed that the control rods were inserted more than 80% prior to scram in order to achieve a critical configuration.) The change in instrument response will determine the reactivity worth of the control rods. This experiment will be repeated with a different critical reactor control rod configuration (relative position of the 12 control rods prior to scram).

(2) With the reactor critical between 5 and 10 watt, the safety rods are manually scrammed. (Control rod scram will automatically accompany this scram.) The reactivity worth of the safety rods is established in conjunction with the results from step (1).

(3) With the reactor critical between 5 and 10 watt, a single control rod (5-C-3) is moved in 2-in. increments from fully inserted (up) to fully removed (down). Count rates are taken after each increment of movement.

(4) The reactor is brought to criticality between 5 and 10 watt. The same control rod (5-C-3) is calibrated by period measurements. It is intended that these measurements be made with periods longer than 35 sec. Two inches of rod insertion will cause the reactor to diverge on about a 100-second period. Three inches of rod insertion are expected to give about a 50-second period.

(5) Step (4) is repeated on the control rod at 5-D-1.

(6) The dry isothermal temperature coefficient is determined by noting the change in a calibrated control rod (5-C-3) setting when changing the isothermal temperature of the reactor by non-nuclear heating. Two criticality determinations between 5 and 10 watt will be carried out at two different reactor system temperatures. The highest temperature will not exceed 120°F. This limit is imposed to avoid overheating the reactor instruments. The minimum contemplated reactor system temperature is about 50°F (the minimum which can probably be achieved). A maximum 70°F temperature change is expected to yield between 2.5 in. and 4.0 in. of shift in control rod setting, depending upon the degree of control rod insertion during the experiment.

(7) The normal control rod at 5-E-1 is replaced by a special control rod (described in Section IV). Criticality is attained between 5 and 10 watt. This will be compared with the initial configuration (step 4). Step (4) is repeated with the special control rod. This rod holds more reactivity than the normal control rod (approximately 0.55% $\Delta k/k$). Therefore, increments of rod insertion will be only about one-half to two-thirds of those in step (4) to yield similar periods.

(8) The normal control rod in 5-A-1 is removed. The oscillator rod is installed for calibration. The oscillator rod will be installed in the initial angular position of maximum reactivity worth (boron cylinder at maximum radial displacement from core). Rotational motion of the rod will reduce reactivity. The control rods are engaged and raised with the special control rod (5-E-1) fully inserted. (It is possible that the reactor cannot be brought critical with ten rods fully inserted and the calibrating rod at 5-C-3, about half inserted. If this is the case, the reactor is shut down and one - or possibly two - depleted inner blanket subassemblies in row 6 or 7 are replaced with enriched inner blanket subassemblies.)

The calibrated control rod at 5-C-3 is used to determine the worth of the oscillator rod. This is accomplished by bringing the reactor subcritical (multiplication ~ 100) and slowly rotating the oscillator rod in

angular increments of about 30° . The peak-to-peak reactivity will not exceed $0.10\% \Delta k/k$ (approximately 40 inhour). The reactor is then brought to the initial critical condition as evidenced by power level and by noting the shift in the position of the calibrated control rod at 5-C-3. This completes the calibration for the fully inserted position of the oscillator rod (maximum reactivity effect).

The reactor is brought subcritical by lowering all control rods. The oscillator rod is moved upward so that only 10 inches of the 14-inch length of $B_4^{10}C$ is adjacent to the core. The previous measurements are repeated. The same determination is carried out at 6-inch insertion and perhaps at 2-inch insertion (where the peak-to-peak reactivity will not exceed 10 inhour).

(9) The reactor is shut down by lowering the control rods. The oscillator rod is removed from 5-A-1. The reactor control rods are disengaged and the reactor cover is raised. (If one or more fuel subassemblies were added in row 6 or 7 of the inner blanket for step (8), they are removed and replaced by inner blanket subassemblies.)

A normal control rod is placed in 5-A-1. The special control rod is removed from 5-E-1 and is replaced by a normal control rod.

(10) The critical configuration is investigated as a function of power level. This is to determine the non-negligible effect of the neutron source at these low powers. Stable power between 50 and 100 watt is expected to require about an inch more control rod insertion than stable power between 5 and 10 watt.

Neutron flux will be continuously monitored throughout the experimental program which extends over an appreciable length of time with respect to the half-life (~ 60 day) of the neutron source. A systematic record of critical reactor configuration as a function of time, power level, and reactor system temperature will be maintained.

F. Procedures after Completion of Experimental Program

Subsequent to the completion of experiments described, the reactor will be disassembled. The sequence of unloading is the reverse of that described in Sections V-B and V-C. First the enriched inner blanket subassemblies are removed and replaced by depleted inner blanket subassemblies. Then the enriched core subassemblies are removed, proceeding inward from the radial core boundary. The active antimony rods will not be removed from the source thimble before all enriched subassemblies have been removed from the reactor (except for the central thermocouple subassembly which must be removed by hand with the thermocouple leads).

VI. HAZARDS EVALUATION AND PLANT SAFETY CONSIDERATIONS

In evaluating the possible hazards associated with this experiment it is concluded that no hazard exists with respect to either the surrounding area or the general population. This conclusion is warranted because:

- (1) No long-lived fission products (or plutonium) will be present in the reactor.
- (2) Comprehensive "containment" is provided.
- (3) The reactor is located at the NRTS (approximately 35 miles from the nearest population center).

A. Plant Safety

1. General Considerations.

Because of the large capital investment in the facility and its primary function as an experimental nuclear power facility, extraordinary precautions have been taken to insure plant safety during the conduct of the Dry Critical Experiments. These are:

- (a) Loading flexibility is limited (by reactor configuration and subassembly composition).
- (b) Extensive instrumentation is provided.
- (c) Maximum power level is limited to 1 kw.
- (d) Conservative temperature limitations are imposed on fuel and reactor system.

These and other considerations relating to the reliability and safety of these experiments are discussed in this Section.

The experimental procedures to be followed are also very conservative with respect to loading increments, control of enriched subassemblies, etc. This will reduce the pace at which the experiments can be conducted; however, this is warranted to achieve the maximum safety to the facility.

The dry critical experiments will be conducted at less than 1 kw reactor power. Fuel element clad temperature will not exceed 180°F at any time and the reactor system temperature will not exceed 120°F.

The absence of coolant sodium in the primary system eliminates all hazards associated with the presence of sodium [V-D, V-F]. Furthermore, the absence of sodium will permit observation of all mechanical operations within the reactor primary system.

Perhaps the most significant aspect pertaining to reactor safety is that the neutronics of the system have been investigated extensively at ZPR-III (Appendix A), using a critical configuration of a system similar to that contemplated during the Dry Critical Experiment. The results have provided upper limits on reactivity effects associated with control, safety, and fuel subassemblies. Furthermore, the detailed measurements made on ZPR-III provide a reliable basis for calibrating reactor power in terms of nuclear instrument response prior to the performance of any experiments in EBR-II.

In particular, it has been established experimentally that no single normal reactor control rod will hold more than about one-half dollar of reactivity. A definite lower limit on critical mass has been established on the basis of ZPR-III measurements.

Extra in-core instrumentation is being installed as an additional precautionary measure along with an Sb-Be neutron source of sufficient strength to monitor neutron flux prior to the insertion of any enriched uranium bearing subassemblies into the reactor.

There is no liquid moderator in the reactor building. No water or steam is brought into the building as a precaution against initiating a sodium-water reaction. This precludes the possibility of water entering the reactor during the Dry Critical Experiments and the attendant hazards of moderating the reactor. Similar precautions have been taken with respect to the inadvertent entrance of lubricants into the primary tank because of their deleterious reaction effects with sodium. These same precautions preclude the possibility of inadvertent introduction of moderating lubricants into the reactor.

Administrative control of reactor loading is enhanced by direct loading to the reactor (the storage rack will not be employed during reactor loading), and the positive identification of all subassemblies (Section V). In addition to providing for detailed over-all fuel and blanket subassembly control, it is possible to visually identify the most important subassemblies during the loading operation, and after loading into the reactor.

In summary, the absence of sodium and fission products, the known neutronics of the system, the presence of in-core instrumentation, and the provision for special administrative controls assure a safe and methodical conduct throughout the experimental program. Furthermore, these experiments will be conducted at a power level less than 1 kw, in a containment system designed for a reactor producing 62.5 Mw of heat.

2. Safety Features of Mechanical, Nuclear, and Control Design

The safety features of the reactor design are as previously given [V-A]. The control drive mechanism is designed to prevent a rod from being raised above core level, or from failing to unlatch without knowledge of the operator. The design makes it impossible inadvertently to raise all rods as a unit. The loading mechanism can accommodate only one subassembly. Subassemblies will fit only in preassigned regions (core, inner blanket, or outer blanket).

In addition to the twelve control rods which are fail-safe, two safety rods are provided so that about 1.1% $\Delta k/k$ is available for shutdown during fuel-loading operations, should gross mismanagement and operational errors combine to make the reactor critical at this time. (The procedures to be followed should make this impossible.) The safety rods may be manually scrammed (followed by control rod scram), when the reactor is operating. This provides a total shutdown reactivity of about 4.5% $\Delta k/k$.

The dry critical core will be at least 20% larger than the contemplated wet critical core. Therefore, a large number (42) of enriched inner blanket subassemblies will be available for the experiment. Since these are designed to fit into the inner blanket, the grid plate cannot distinguish between these and depleted inner blanket subassemblies. Because of the possibility of inadvertently interchanging these subassemblies, the loading of the reactor will proceed from the operating floor; the storage basket will not be used. (If a subassembly has been placed into the storage basket, it will be removed to the operating floor prior to being inserted into the reactor.) Administrative identification of these subassemblies is enhanced by color as well as serial number (Section V).

Experimental results from the EBR-II mock-up on ZPR-III suggest that the originally predicted rates of reactivity insertion are generally higher than observed. This applies to control and safety rods as well as fuel-handling equipment.

The rates of reactivity addition based on prediction were used to specify the various drive speeds for inserting control and safety rods as well as subassemblies during fuel-handling operations. These drive speeds are even more conservative on the basis of the experimental measurements. They are further reduced because the dry critical reactor core is larger than the ZPR-III core.

The normal rates of reactivity insertion by various reactor mechanisms are summarized in Table IV. These values will vary depending on the actual core size.

Special control rods containing boron-10 and holding more reactivity (approximately 0.55% $\Delta k/k$) are available to provide operational flexibility without compromising safety. For instance, during the calibration of the oscillator rod at least one control subassembly must be removed from the system to provide a location for the oscillator rod. A relatively fixed amount of shutdown reactivity can be provided by replacing one or two normal control rods by special control rods.

The maximum rate of normal reactivity addition can occur during fuel-handling operations. The system is expected to be more than 3% $\Delta k/k$ subcritical (multiplication <30) under these conditions. However, if through some gross mismanagement the system is at delayed critical and reactivity is added at the maximum rate, it will require at least 50 sec to insert one dollar of reactivity. This example applies to the insertion of the central core subassembly. However, it should be noted that the central core subassembly will be loaded during the approach to criticality (Section V-C). The experimental program (Section V) does not require further handling of fuel subassemblies within the 61-subassembly core. The only subassemblies involved in fuel-handling procedures after the initial critical approach, are control rods and enriched inner blanket subassemblies located in row 6 or 7. The reactivity insertion rates for these subassemblies are calculated to be approximately one-fourth that in the above example (200 sec to insert one dollar of reactivity; however, the total reactivity worth of this subassembly is less than one dollar).

B. Nuclear Accidents

The means of generating a nuclear accident during the Dry Critical Experiments are similar to those described previously - summarized in [V-C], and detailed analysis [Appendix A] - and involve the simultaneous failure of administrative controls and equipment. The postulated accidents require the malfunction of control and safety rod drives, and fuel-handling equipment, concurrent with operational mismanagement. The same types of accidents [V-C], can be considered during the Dry Critical Experiments. The potentially most serious nuclear accidents are those generated by dropping of fuel into a near-critical reactor. However, the consequences will be less severe during the Dry Critical Experiment because:

(1) The prompt shutdown coefficient of $-1.3 \times 10^{-6} (\Delta k/k)/^{\circ}\text{C}$ appropriate during the more serious excursions [Appendix A - p.209] is unchanged for the Dry Critical Experiment, because this coefficient is due only to the axial expansion of the fuel alloy and includes the inertial effects which inhibit expansion of the fuel under extremely rapid rates of reactivity addition. The magnitude of this coefficient is approximately one-third that which obtains during low rates of reactivity addition (no inertial effects).

(2) The reactivity insertion rates are about one-half the rates in the previous analysis [Appendix A]. The reactivity insertion rates are smaller because ZPR-III experiments have established lower total subassembly reactivity worths than had been predicted (Table IV - Section V) and because the Dry Critical mass is at least 20% larger than the wet critical reactor. Therefore, approximately twice as much time is available for safety rod scram to be actuated (Table IV - Section V).

(3) Initial fuel alloy temperature is less than 200°F, while the postulated accidents [Appendix A] are initiated at 600°F.

Because of the larger number of enriched subassemblies available for the Dry Critical Experiment, the loading accident resulting from inadvertent continued loading to criticality has been examined. It is assumed that the loading procedure described in Section V is disregarded and that loading is continued until the reactor is critical in the fuel-loading phase of operations. (It is assumed that the six enriched subassembly limitation per loading sequence is ignored - an addition of more than twelve subassemblies would be required.) It is also assumed that the operating staff disregards the operative nuclear instrumentation. This then assumes a condition where the reactor is at essentially delayed critical and that a depleted inner blanket subassembly is removed and an enriched subassembly is inserted in its place.

The "replacement" reactivity worth of an enriched inner blanket subassembly is about 0.3% $\Delta k/k$. At the normal fuel-loading drive speed of 6 in./min, the reactivity addition rate is about $0.21 \times 10^{-4} (\Delta k/k)/\text{sec}$. About 140 sec are required for effective subassembly insertion.

A power level of 1 kw is reached about 2 min after the excursion is initiated. If the safety rods are operative, a power-level trip will shut down the reactor. The maximum fuel alloy temperature attained is well below 200°F. If the safety rods are not operative, the power level will increase until it reaches an equilibrium due to the negative temperature coefficient of the reactor. The operative negative temperature coefficient with this slow reactivity addition rate is conservatively assumed to be about $0.45 \times 10^{-5} (\Delta k/k)/^\circ\text{C}$, of which 0.39×10^{-5} is due to axial expansion of the fuel, and 0.06×10^{-5} is due to density change in the stainless steel. (Note that the operative temperature coefficient assumed is only one-quarter of the total dry isothermal temperature coefficient which is predicted to be $1.8 \times 10^{-5} (\Delta k/k)/^\circ\text{C}$.) Under these assumed conditions, the equilibrium fuel alloy temperature will be approximately 1180°F.

The foregoing accident was also examined with the assumption that the enriched subassembly being added to the reactor is dropped accidentally and falls freely under the force of gravity (instead of being driven in at the rate of 6 in./min). If it is assumed that the reactivity (0.3% $\Delta k/k$) is added

essentially instantaneously, the reactor power would increase on about a 9-second period. At least 40 sec would elapse before a power level of 1 kw is reached. If the safety rods are operative, the period trip would terminate the excursion before this power level is reached; if not, the fuel would reach about the same equilibrium temperature (1180°F).

Although the maximum fuel temperature attained is much higher than the 200°F limitation established for these experiments, it does not appear that any significant damage would result. The integrity of the fuel element sodium bond might be impaired; however, no fuel melting would occur, nor would any eutectic alloy be formed (stainless steel-uranium $\sim$ 1340°F). Also, the consequences will be tempered by any amount which the reactor is subcritical at the time the loading error is made.

Under normal power operating conditions, it is not believed feasible that a significant quantity of moderating material can be added to the reactor core because the entire reactor is submerged in approximately 80,000 gal of sodium. The sodium pumps and piping are also submerged. It is assumed that it would be necessary to introduce the moderator into the sodium to effect an inadvertent addition of moderator to the reactor. The chemical activity of sodium is believed to preclude the possibility of significant additions of water, hydro-carbons, etc.

During the Dry Critical Experiments, the primary tank will contain air, and the addition of water is not "chemically" excluded. Although no water is available in the reactor building (there are no water or steam lines into the containment vessel), the possibilities of flooding the reactor with water have been examined.

The only time when moderating material may be effectively inserted into the reactor is during fuel-handling operations (the reactor cover is raised and the reactor is accessible). The reactor is at least 3% $\Delta k/k$ subcritical. The reactor can be brought above delayed critical by the optimum distribution (for a finite period of time) within the core of about one liter of H_2O . Because of the large number of openings in the reactor this is an exceedingly difficult condition to realize. The consequences of such an accident depend upon the manner in which water (or other moderator) is inserted; partial flooding of the reactor core (to the equivalent of one liter effectively distributed) would require:

- (1) The introduction of about 1,000 gal of water into the reactor vessel; however, the water will not be retained permanently because of the leakage paths from the reactor vessel to the primary tank.

- (2) The introduction of 30,000 gal of water into the primary tank; the reactor core would be flooded gradually and permanently by seepage through the same leakage paths.

It is inconceivable that such quantities of water could be brought into the reactor building undetected. Since there are no water supplies in the building, it would be necessary to bring pipes (or a hose) from one of the adjacent buildings.

However, it should be emphasized that a much smaller quantity of water can be effective under optimum conditions. If one or two subassemblies at the center of the core are filled with water during fuel-handling operations (the reactor at least 3.0% $\Delta k/k$ subcritical), the reactor would become critical. The entry of additional water into the system (into other subassemblies) could produce a subcritical or super prompt critical burst. The water would, of course, flow through the subassemblies and a flow rate of approximately 25 gpm/subassembly would be required to maintain a water level in the subassemblies. This would require a large water supply (of the order of 100 gpm), or a very effective dispersal of a small quantity of water (a few gallons) in a very short time period. Either of these possibilities would require the transport of water and an optimum use of the water.

C. Maximum Credible Nuclear Accident

The general characteristics of a maximum possible nuclear explosion are as given [Appendix C]. However, more recent calculations which represent more accurately some of the physical reactor conditions suggest higher explosive yields for given reactivity insertion rates.

The maximum possible nuclear explosion is postulated to be the result of a core collapse under gravity [V-E]. The pessimistic hypotheses to describe the course of the accident are:

- (1) The uranium from the center of the core has melted and flowed down into the lower part of the core and is retained there, producing a region abnormally dense in enriched uranium at the bottom of the core, with a large gap at the center of the core.
- (2) At the most effective time, the upper portion of the core falls as a single unit, producing a prompt critical configuration at the highest possible insertion rate.

The major conclusions to be drawn from the re-evaluation of postulated accident are that a given reactivity insertion rate derived from core collapse now suggests a higher nuclear energy yield than had been calculated previously [V-E]. Table V summarizes the calculated results by comparing explosive energy yield with reactivity insertion rate.

Table V

EXPLOSIVE ENERGY YIELD AS A FUNCTION
OF REACTIVITY INSERTION RATE

Reactivity Insertion Rate (\$/sec)	Explosive Energy Yield (lb of TNT)	
	Previous Analysis [V-E]	Present Analysis
200	280	370
600	660	850
1000	1050	1240

A reactivity insertion rate of \$200/sec, with an explosive yield of less than 400 lb of TNT, can now be assumed for this gravity-induced accident [V-E]. Higher reactivity insertion rates may be postulated with correspondingly higher explosive energy yields; however, it is necessary to assume conditions requiring many coincidences in timing and motion of fuel prior to, and during, the excursion. It is believed that these assumptions are unrealistic.

The same accident (which assumes the absence of sodium) can be postulated for the Dry Critical Experiment. In addition, the Dry Critical reactor has a critical mass which is at least 20% larger than the wet (sodium-filled reactor) critical mass. The presence of the larger amount of fissionable material will tend to increase the nuclear energy yield over that calculated for the smaller critical mass of the sodium filled system. Therefore, the nuclear energy yields in Table V must be multiplied by ~ 1.2 to provide realistic yields for the Dry Critical Experiment.

A reasonable upper limit for the maximum credible accident for the Dry Critical Experiments, therefore, would appear to result in an explosive yield equivalent to approximately 450 lb of TNT.

D. Containment of Maximum Credible Accident

The EBR-II reactor system is provided with two distinct containment systems. The first, termed the "primary containment system," surrounds the primary tank. The reactor is located near the bottom of this tank. The second, termed the "building containment system," surrounds the primary containment system as well as the entire reactor plant [Appendix E].

The primary containment system has been analyzed on the basis of a 300-pound TNT equivalent nuclear energy release at the center of the core. The rate of energy release assumed was that which calculations indicate to be the worst possible with respect to containment by this particular system (energy release time of 0.0018 sec). The primary containment system was found to be adequate to contain such an explosion, with the primary tank filled with sodium [Appendix E]. The sodium contributes significantly to the energy absorption of the system and in attenuating the resultant shock wave. The analysis concludes that the primary containment system would not be ruptured (although breaching with some expulsion of sodium was contemplated). The very large safety factors employed in the design suggest that the primary containment system is actually capable of containing a nuclear energy release several times larger than the 300-pound equivalent used as a basis for design [Appendix E - pp. 328, 329]. However, this conclusion cannot be extended to the Dry Critical Experiments because of the absence of sodium. The containment analyses were not repeated for the "dry" primary system; therefore, it is assumed that the primary containment system may be ruptured even though this is not necessarily the case.

The "building containment system" has been designed to withstand a maximum interaction between the primary system sodium and the air in the reactor building resulting from failure of the primary containment system and the expulsion of sodium into the atmosphere [Appendix E - p. 332].

The maximum energy yield believed possible from the sodium-air reaction was estimated to be equivalent to 29,000 lb of TNT [Appendix E - p. 332]. This value represents an extrapolation of experimental results representing idealized reaction conditions [Appendix D] and would require the ejection of 14,000 to 23,000 lb of 750°F-sodium with optimum dispersal characteristics. This energy release would result in an internal pressure of 75 to 80 psig and exceed the design capability of the containment vessel (not its ultimate strength). However, an energy release equivalent to approximately 10,000 lb of TNT would not exceed the 24-psig design rating of the vessel (which has been tested twice at 30 psig). Therefore, it can be assumed that the containment vessel will not be ruptured due to the maximum credible accident (450 lb of TNT equivalent) during the Dry Critical Experiments, even if the primary containment system fails. There is no secondary source of rapid energy release of significant magnitude such as exists with sodium present.

The analysis of possible missiles generated in a nuclear accident [Appendix E - p. 325] revealed a very low probability of the efficient ejection of "effective missiles." More recent studies have indicated that a control rod drive shaft, considered to be the component possessing greatest potential as a missile, could not be ejected to sufficient height

even to make contact with the building containment shell (with 300 lb of TNT energy release and sodium-filled primary tank). Under conditions of the proposed experiments (450 lb of TNT and dry primary tank), potential missile velocities are possibly enhanced. However, substantial missile shielding is provided for the shell in the form of reinforced concrete, which should afford adequate protection. (The concrete false ceiling originally planned has been replaced by a reinforced concrete lining applied by the Guniting Process directly to the inner surface of the shell dome.)

It is concluded that the assumed maximum credible accident will be contained and that the containment vessel will not be breached by pressure or missiles which might arise as a consequence of such an accident.

FIG. 1
NUCLEAR INSTRUMENT RANGE CHART

FIG. 3
EBR II IN-CORE INSTRUMENTATION

FIG. 4
LOCATION OF NUCLEAR INSTRUMENT THIMBLES

THIMBLE NO.	NUCLEAR INSTR. CHANNEL NO.	NEUTRON DETECTOR
01	11	C.I.C.
02	8	C.I.C.
03	10	C.I.C.
04	9	C.I.C.
J1	1 & 2	F.C. & F.C.
J2	3 & 4	F.C. & C.I.C.
J3	7	C.I.C.
J4	5 & 6	C.I.C. & C.I.C.

F.C. = FISSION COUNTER
C.I.C. = COMPENSATED ION CHAMBER

FIG. 5
SUBASSEMBLY LOCATING DIAGRAM

NOTES:

1. "OPENS TO TRIP" OR "OPENS" MEANS THE 12 CONTROL RODS WILL SCRAM WHEN ANY CONDITION SHOWN OPENS THE OPERATING SHUTDOWN CIRCUIT... OR
2. "OPENS TO TRIP" OR "OPENS" MEANS THE 2 SAFETY RODS WILL SCRAM WHEN ANY CONDITION SHOWN OPENS THE FUEL HANDLING SHUTDOWN CIRCUIT.
3. ALL CONDITIONS CAUSING CONTROL ROD SCRAM WILL ALARM AND ANNUNCIATE ON THE OPERATING CONSOLE.

FIG. 6
REACTOR SHUTDOWN CIRCUIT FOR DRY CRITICAL EXPERIMENTS

FIG. 8
FUEL LOADING SYSTEM

FIG. 9
CONTROL DRIVE ACTUATING POSITIONS

NOTE: 1. PAIRS OF ENCIRCLED NUMBERS INDICATE COINCIDENCE IN TIME AT WHICH GRIPPER AND HOLD-DOWN MOTIONS ARE INTERLOCKED.

2. EXCEPT AS NOTED, MECHANISMS STOP ONLY BETWEEN MOTIONS IN OPPOSITE DIRECTIONS.

3. FAST $\cong 72$ in./min; SLOW $\cong 6$ in./min.

FIG. 10
SEQUENCE OF GRIPPER AND HOLD-DOWN MOTIONS DURING REMOVAL AND REPLACEMENT OF SUBASSEMBLY

FIG. 11
CONTROL ROD DRIVE MECHANISM

FIG. 12
DRY CRITICAL APPROACH CURVE

APPENDIX A

SUPPORTING EXPERIMENTAL AND OPERATIONAL DATA

A. Critical Experiments on ZPR-III

Two series of critical experiments were performed on ZPR-III to provide detailed information concerning the neutronics of the EBR-II.¹ These were mock-ups of the EBR-II reactor configuration simulating the normal power loading (with sodium coolant). In addition, a large number of experimental investigations were made on systems similar to the EBR-II.¹ The neutronics of the EBR-II are well understood as a result of both analysis and experiment.

The ZPR-III critical experiments utilized low-density aluminum as a substitute for sodium in the mock-up of the coolant volume of the reactor system. Gross substitution experiments with aluminum of different effective densities confirmed that removal of aluminum from the reactor core and reflecting regions resulted in a loss of reactivity. The critical experiments established a definite lower limit on the critical mass of the Dry Critical reactor configuration of 165 kg U²³⁵.

The EBR-II mock-up was smaller than the predicted Dry Critical configuration. Therefore, the experimental results represent upper limits on the reactivity worth of fuel, control and safety subassemblies. This, in turn, establishes upper limits on the rates at which reactivity can be inserted in the Dry Critical reactor.

1. Critical Mass and Reactivity Effects

The critical mass of the reactor configuration, which most closely resembles the composition and geometry of the EBR-II reactor, was measured to be 165 kg U²³⁵. This critical configuration can be, and was predicted to within a few kg U²³⁵.

The average reactivity worth of a normal reactor control rod was measured to be 0.37% $\Delta k/k$.

The reactivity worth of two safety subassemblies was measured to be 1.36% $\Delta k/k$.

¹ J. K. Long, W. B. Loewenstein, C. E. Branyan, G. S. Brunson, F. S. Kirn, D. Okrent, R. E. Rice, and F. W. Thalgott, "Fast Neutron Power Reactor Studies with ZPR-III," Proceedings of Second International Conference on Peaceful Uses of Atomic Energy, Geneva (1958) Vol. 12, p. 119.

The reactivity worth of a core subassembly, relative to low-density aluminum (simulating the displacement of sodium), is principally a function of radial position in the core. These measurements are reported in Table A-1. The reactivity worth of inner blanket subassemblies relative to low-density aluminum are also given in Table A-1.

Table A-1

REACTIVITY WORTH OF CORE AND BLANKET SUBASSEMBLIES
AS A FUNCTION OF POSITION (ZPR-III MOCK-UP EXPERIMENTS)

Reactor Row Number (Fig. 5)	Type of Uranium	Reactivity Worth (% $\Delta k/k$)
1	Enriched	1.53
2	Enriched	1.48
3	Enriched	1.31
4	Enriched	1.05
5	Enriched	0.77
6	Depleted	0.18
7	Depleted	0.04

A number of detailed experiments were performed to determine the reactivity worth of various materials uniformly distributed throughout the reactor core and reflector regions. The importance of these worths is to effect a transition from the "aluminum-cooled" ZPR-III to the Dry Critical reactor. Results instrumental in effecting this transition are given in Table A-2.

Table A-2

REGIONAL MATERIAL REPLACEMENT EXPERIMENTS

Material	% ($\Delta k/k$)/kg of material ($\times 10^2$)			
	Reactor Region ¹			
	Core	Axial Structural Gaps ²	Reflectors	
			Radial	Axial
Aluminum	7.4	5.8	3.7	1.6
Stainless Steel	2.3	2.7	2.0	0.8

¹ Core contains enriched uranium. Reflector regions contain depleted uranium. Structural gaps contain no uranium.

² The axial structural gaps are located between the core and the axial reflector.

2. Power Distribution

Uncertainties in the predicted power distribution and calibration may be estimated by comparing predicted with measured fission distributions from ZPR-III. Predicted fission distributions

$$F(r,z) \equiv \int_0^{\infty} \sigma_f(E) \phi(E,r,z) dE$$

are obtained from a multigroup two-dimensional (r,z) analysis. Figures A-1 and A-2 compare experimental with theoretical radial and axial fission distributions, respectively. Since these experimental data are not adequate to determine an absolute normalization, the experimental data are normalized to the predicted U^{235} fission density at the center of the core.

The slight dip at $r \approx 11$ cm for the experimental U^{238} fission distribution on Fig. A-1 is attributed to the presence of the safety rods. These were not properly described in the two-dimensional (r,z) analysis. Since mostly U^{238} fissions occur in the portions of the radial blanket shown on Fig. A-1, it can be seen that dominant radial reflector fissions (U^{238}) relative to the dominant core fissions (U^{235}) are relatively well predicted. The data points in the outer portions of the radial reflector exhibit large uncertainties because of the low power and thus the low flux intensity in ZPR-III.

The rather striking deviations between theory and experiment in the gap regions on Fig. A-2 must be attributed to an inadequate understanding of the neutron cross sections of aluminum and stainless steel. However, since very few fissions ($<2\%$) occur in the axial reflectors these deviations between theory and experiment are not significant in determining the reactor power distribution.

3. Neutron Source and Instrument Thimble Experiments

Special experiments dealt with the attenuation and spectral shift of neutron flux from the center of the core to the "J" and "O" thimbles (Fig. 4) where the normal fission counters and ion chambers are located. The "J" and "O" thimbles are located about 58 and 78 in., respectively, from the reactor centerline in the central core plane. The unperturbed neutron flux can be estimated within and outside of the graphite neutron shield under the postulate of no thimble and uniform shield. However, the presence of the thimbles of up to 15 in. O.D. appreciably complicates such an estimate and reduces the accuracy of the predicted result. The other aspect of these investigations dealt with determining the neutron yield of an Sb-Be neutron source of the type to be employed in EBR-II. Estimates of neutron yield are extremely uncertain. Thus the experiments were designed to yield data relevant to:

(1) The required Sb activity in the Sb-Be neutron source.

(2) The geometric and compositional arrangement outside of the radial blanket and around the "J" thimbles to optimize low-power fission counter response.

Two series of experiments were conducted. The first dealt with the early design of the neutron shield and reactor vessel as shown [Figs. 8, 20, 21].

The second series of experiments investigated several geometrical and compositional arrangements adjacent to the "J" thimbles in the final design configuration. Only those results pertaining to the final reactor configuration are cited here.

An absolute fission counter was placed at the core center and another at the radial core boundary during each investigation. Additional counters were placed throughout the radial blanket and neutron shield. However, the information of primary interest is that of J1 and J2 thimble fission counter response relative to the absolute fission counter response, and thus relative to reactor power. One experiment to enhance J1 and J2 thimble fission counter response included moderating material in the interior of these thimbles adjacent to the fission counters. This was accomplished by inserting 4.1 kg ZrH in the form of 10 in. long aluminum cans having a cross-sectional area of 9.8 in.² Table A-3 summarizes experimental counter response data for a critical reactor producing 1 watt of thermal power.

Table A-3

INSTRUMENT THIMBLE FISSION COUNTER RESPONSE
(Critical Reactor; 1 watt)

Location	ZrH	Counter Response (cps)	Cadmium Ratio	Comment
J1, J2	Yes	2020	5.5	A
	No	730	1.9	A
J3, J4	No	395	2.4	A
O1,2,3,4	No	~20	~5	A
Center of Core	No	170	1.0	B
Radial Core Boundary	No	82	1.0	B

^A Fission Counter (Same as those in Channels 1,2,3 described in Section III-A)

^B Absolute Fission Counter (contains ~0.8 mg U²³⁵)

The neutron shield configuration adjacent to the J1 and J2 thimbles was the object of a subcritical investigation. With a 10-curie Sb-Be neutron source located at the reactor core-inner blanket interface, subcritical count rates were obtained in the mock-up J1 thimble, as well as at the core-inner blanket interface. The relative location of source, in-core detector, and J1 thimble are essentially as described in Section V-B. Table A-4 summarizes the experimental results.

Table A-4

SUBCRITICAL COUNT RATES IN J1, J2 AND IN-CORE THIMBLES

Approximate		J1 and J2 Thimbles ¹ (cps)		In-Core Thimbles (cps)
k _{eff}	Multiplication	With ZrH	Without ZrH	Absolute Counter ²
0.97	33	29	11	~3
0.94	16	16	6	~2

¹ Fission Counters (Same as Channels 1, 2, 3 described in Section III-A)

² Absolute Fission Counter (contains ~0.8 mg U²³⁵)

4. Other Experiments

The experiments cited in the previous sections represent a small fraction of the experimental investigations pursued on the EBR-II mock-up on ZPR-III. Other experiments dealt with the effect of inhomogeneity, neutron lifetime, spectral indices and a large number of material replacement experiments. Two experiments pertinent to the Dry Critical Experiments are summarized:

a. A special control rod (Section IV) mock-up was calibrated. This reactivity determination definitely established that the reactivity held by such a rod is less than one dollar of reactivity. The worth of such a rod depends on the amount of B₄¹⁰ C contained in the poison follower section. However, the special control rod to be measured in the Dry Critical Experiment will hold less than 0.60% $\Delta k/k$.

b. An oscillator mock-up rod was calibrated. This measurement limits the reactivity effect, due to 180-degree rotation (peak-to-peak), to considerably less than 0.10% $\Delta k/k$ during the Dry Critical Experiment.

B. Tests on Control and Safety Rod Drive Mechanisms

1. Control Rod Drive Mechanisms

Two control rod drive mechanisms were built for tests under a variety of operating conditions. The first of these drive units was similar to the final design except for a slightly different gripper mechanism, and a somewhat shorter drive shaft. Also, the gripper-sensing device was not included. A control rod and control rod thimble, similar to those of the final design, were included as part of the test equipment. This drive mechanism was cycled approximately 300 times in air. One complete cycle consisted of raising, and then lowering, the control rod the entire 14-inch stroke permitted by the drive unit. The control rod drive mechanism was then installed in a sodium loop (Fig. A-3) and tested for a period of approximately four months. The temperature of the sodium in the loop was varied from 300° to 900°F during these tests. A total of 14,000 cycles were successfully completed during the test period.

The second control rod drive mechanism tested was identical to the final design with the exception of having a shorter drive shaft and other minor modifications to accommodate the experiment in the available laboratory space. The control rod and thimble employed was identical to the final design. This drive unit successfully completed 19,900 cycles in air and 1,700 cycles in sodium. The temperature of the sodium was varied from 300° to 960°F during these tests. In addition to these cycle tests, approximately 35 scram cycles (a portion of these in sodium) and 15 rod gripping operations were accomplished. A scram cycle consisted of raising the rod the full 14 inches inside the thimble and then effecting a scram release which rapidly ejected the rod drive downward by the force of gravity plus the force derived from the scram assist pneumatic cylinder. In this scram cycling, assist pressures up to 50 psig were used. The rod-gripping operations consisted of gripping and releasing the cone-shaped upper end of the control rod adapter by means of the gripper mechanism.

Although only twelve control rod drive mechanisms (Fig. A-4) are required for the EBR-II, fourteen drives have been fabricated and tested briefly, by the vendor, prior to shipment. This testing was accomplished in air by the previously described methods of stroke cycling, scram cycling, and rod-gripping operations.

Certain of the time-displacement data obtained on the control rod scram tests are summarized in Fig. A-5. Time is computed from the instant a scram signal is received at the rod drive. The scram tests were made with air cylinder pressures of 0, 25, and 50 psig. The data from the 0-psig and 50-psig tests, only, are presented in the figure (as a pair of curves at each pressure, the data from all 14 units falling between the two curves). It will be noted that the pair of curves for the 50-psig air pressure case converge at approximately 0.3 sec after receipt of the scram signal and show that all rods had completed their full 14-inch stroke within

this time. The two curves for the 0-psig air pressure case, however, do not converge within the time scale of the graph. This is due to the longer time required in this case to traverse the last 5 in. of the stroke during which the rod is being decelerated by the hydraulic dashpot. The dashpot is designed to decelerate the total rod and drive mass in 5 in. after being accelerated at about 2 g. At the lower acceleration rate experienced with zero pressure, the rod enters the dashpot zone at considerably lower velocity and, because of the fixed characteristic of the dashpot, requires much more time to traverse this portion of the stroke. The result is that more than 0.5 sec are required for the rods to drop the full 14 in.; however, the rod travels the first 10 in. in approximately 0.23 sec (about .07 sec longer than with 50-psig assist). It is intended that the scram-assist air pressure employed in the EBR-II operation provide a rod acceleration of 1.5 g. It may be seen from the displacement-time curves that the required air pressure is approximately 35 psig. (The exact pressure will be obtained from curves based on subsequent tests in sodium.)

Of the fourteen drives fabricated, thirteen have been shipped to the construction site and one has been retained at the Laboratory for further testing. This test drive mechanism is identical in every respect to those to be installed in the reactor. The tests will be accomplished in air (later in sodium) and will consist of the cycling, scrambling, and gripping operations described previously.

After installation of the twelve rod drives in the reactor, each will be tested in place to check the stroke function, scram, gripper operation and correctness of all control wiring. This testing will be conducted immediately prior to approach to criticality, and in the dry condition (without sodium).

2. Safety Rod Drive Mechanism

The complete safety rod drive mechanism was tested prior to shipment to the EBR-II site. These tests were performed in air at room temperature. The mechanism was scrambled approximately 100 times, using simulated safety rods. The safety rod displacement vs time during scram is plotted in Fig. A-6. The total time required for a complete scram is about 0.320 sec with 9 in. of travel in approximately 0.2 sec. The shock absorbers of the drive mechanism were tested separately, in addition to tests of the complete unit. In these tests, the shock absorbers were scram cycled (approximately 50 times) using weights equal to the safety rods.

Installation of the safety rod drive mechanism in the EBR-II has been completed. The mechanism will be retested in the reactor prior to approach to criticality. This will be done to insure proper mechanical operation and correctness of control wiring.

FIG. A-1
RADIAL FISSION DISTRIBUTIONS AT CORE MIDPLANE

FIG. A-2
AXIAL FISSION DISTRIBUTIONS AT THREE RADIAL POSITIONS

FIG. A-3
CONTROL DRIVE TEST FACILITY

FIG. A-4
TEST ASSEMBLY OF TWELVE CONTROL DRIVES

FIG. A-5
DISPLACEMENT VS TIME FOR CONTROL RODS DURING SCRAM

FIG. A-6
DISPLACEMENT VS TIME FOR SAFETY RODS DURING SCRAM