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NUCLEAR DIVISION
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CEND CRITICAL FACILITIES

SAFEGUARDS REPORT

BONUS CRITICAL EXPERIMENT

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October 29, 1960

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U. S. ATOMIC ENERGY COMMISSION

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COMBUSTION ENGINEERING, INC.

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CEND CRITICAL FACILITIES

SAFEGUARDS REPORT

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BONUS CRITICAL EXPERIMENT

Prepared by

R. W. KNAPP

October 29, 1960

M Slater

M. Slater
Critical Facility Supervisor

S. Visner

S. Visner, Manager
Physics Design & Experimentation

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I. INTRODUCTION

In August 1960, Combustion Engineering, Inc., Nuclear Division (CEND) was given approval by the AEC to activate CELL #1 of the CEND Critical Facilities at Windsor, Connecticut for performance of critical experiments on a mock-up of the BONUS reactor. The BONUS reactor, a 17 eMw boiling water unit with integral nuclear superheat, is being designed by General Nuclear Engineering Corporation (GNEC), under a prime contract to the AEC. The reactor is now under construction at Punta Higuera, Puerto Rico, for the Puerto Rico Water Resources Authority.

The main objectives of the critical experiment program are to obtain data such as criticality, flooding coefficients, power distribution, and lattice parameters to support the design of the BONUS reactor. The data will be either extrapolated for use in the BONUS design or used to verify the calculational methods used in predicting the nuclear behavior of BONUS under conditions of temperature and power not achievable in the critical assembly.

Critical experiments have been in progress in Cell #2 in connection with the Nuclear Superheat Development Program (NUSU), for which Combustion Engineering, Inc. is prime contractor to the AEC. A variety of configurations of superheater and boiler fuel assemblies are being tested using several types of superheater fuel elements. In particular, the Intermediate Experiments on the NUSU program utilized boiler and superheater fuel rods very similar to the BONUS design. This fuel, in fact, will be transferred to the BONUS Criticals while the NUSU Criticals will continue with another type of fuel element.

The new BONUS facility will be very similar to the existing NUSU facility presently being operated in Cell #2. The layout of the cell, interlocks, instrumentation, and Operating Manual will be practically identical. The fuel used, as just mentioned, will be the same as in the NUSU Phase B and C Experiments. The only major changes will be in the use of coated carbon steel for the tanks, etc.; rather than stainless steel employed in Cell #2, in the use of a smaller dump valve, and in the introduction of non-scrappable shim rods in both cells.

The personnel assigned to the BONUS facility have received thorough training through participation in the NUSU Phase B and C experiments conducted in Cell #2.

The nuclear statics and kinetics of the assemblies to be investigated under the present program are similar to Intermediate Experiment assemblies tested in Phase B of the NUSU program and are in the same general category of low enrichment UO_2 , light-water moderated cores for which there has been extensive experience to date. The usual nuclear hazards based on reactivity additions are controllable with conventional restrictions in operating procedure, automatic interlocks, scram circuits, and the imposition of a maximum reactivity addition rate, viz., $0.04\% \delta k/sec$.

It is shown that the cores to be studied are undamaged if a transient induced by an uncontrolled reactivity insertion at this rate should somehow be produced by some sequence of improbable human and component failures. It is further shown that for the Maximum Credible Accident, which would essentially require sabotage or deliberate suicide, the hazard to the surrounding area would not be serious. It is, therefore, concluded that the BONUS mock-up experiments described herein can be safely carried out without any significant increase in the hazards at the site or to the surrounding area over those approved for similar areas or those approved for this site for previous or present operation of the CEND Critical Facility.

II. SITE DESCRIPTION

A. General Description of Surrounding Area

1. Population

The area surrounding Combustion Engineering's 532-acre site is sparsely populated. Windsor, Connecticut is the nearest town, approximately five miles away, with a population of 19,346 and a population density of 654 persons per square mile. Table 1 shows the distribution of population in the area. Figure 1 is a map of the general area showing the location of the towns listed in Table 1.

2. Industry

The chief occupation of the populace in the immediate vicinity is tobacco farming. Farming is prevalent not only in this vicinity, but throughout the Connecticut Valley. Large industrial plants are widely dispersed in the area. Table 2 shows the distribution of the major defense plants in the vicinity of the Combustion Engineering site. Large business and industrial concerns not engaged in defense work are in general located in Hartford and its immediate vicinity.

3. Climatology and Meteorology

Climatological data for the area is based mainly on measurements made at the U. S. Weather Bureau Station located at Bradley Air Field, about five miles from the site. The following data are taken from the 1959 U. S. Weather Bureau Local Climatological Data for the Hartford, Connecticut area.

The most significant feature of the climate in the area is its rapid changeability. Weather is seldom average or normal for any appreciable length of time.

Mean temperature for 1959 recorded at Bradley Air Field was 49.8°F. Maximum and minimum monthly means for 1959 were 73.6 and 24.0°F, respectively, while the maximum and minimum monthly mean from 1905 to 1959 were 83.0° and 19.5°F, respectively.

Total precipitation for 1959 was 45.70 inches with a maximum of 8.01 inches falling in October. Mean annual precipitation for 1905 to 1959 was 42.70 inches. Maximum monthly mean was 4.02 inches in the period 1905 to 1959.

Average hourly wind speed for 1959 was 8.0 miles per hour. Highest recorded velocity was 70 miles per hour, in November 1950. The prevailing wind direction for six months, May to October, is south, and from November to April, northwest. The average wind velocity at the critical facilities site is 11.2 miles per hour.

The flood level for the area's worst flood, in August 1955, was about 110 feet above mean sea level. Since the critical facilities are located approximately 180 feet above mean sea level, the probability of direct damage resulting from a local flood is very low.

With low-to-moderate wind speeds, inversion conditions exist from sunset to sunrise. A strong lapse rate exists around noon; the temperature difference is a maximum with air flow upward at a maximum rate. As the night

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approaches, weak lapse conditions occur with low air flow.

Hartford's location relative to continent and ocean has a significant influence on the area's meteorological and climatological conditions. With the prevailing west-to-east air flow, continental modifications of the air are important.

However, sudden and oftentimes serious upsets result when storms move north or when other pressure developments produce the strong and persistent northeast winds associated with storms known locally as coastals or northeasters. Seasonally, air-mass characteristics vary from the extremely cold and dry continental-polar quality of winter, to warm, humid maritime tropical character of summer - the one type from Canada, the other from the Gulf of Mexico, Caribbean Sea, or Atlantic Ocean.

Local topography also influences the climate. The Berkshire Mountains to the west and northwest are a source of summer thunderstorms which, when accompanied by wind and hail, sometimes do considerable damage to crops in the Connecticut Valley. Frequently during the winter, when rain falls through cold air trapped in the valley, the resultant icing creates hazardous conditions for transportation and utility installations. On clear nights in late summer or early autumn, cool air drainage into the Valley plus Connecticut River moisture produces ground fog which sometimes becomes quite dense throughout the Valley and hampers ground and air transportation.

4. Geology

The surrounding area has been subjected to the actions of the glacial ice. All dominant geologic features are a result of erosion and depositions so caused during the Pleistocene era.

This state has a favorable earthquake history. Ten earthquakes are listed, the first recorded in 1791 and the latest in 1925. All of these, with the exception of the first, were local in nature and of moderate intensity.

5. Hydrology

The surface drainage in the surrounding area is excellent. The predominantly sandy nature of the soil and heavy forest cover results in very moderate run-off even after heavy prolonged precipitation.

The Farmington River, into which the effluent from the CE industrial waste and sewage treatment plant is discharged, flows along the northwest corner of this property as shown in Figure 2. Two and one-half miles below the property the river flows over the dam of the Farmington River Power Company and approximately six miles below that into the Connecticut River. Minimum recorded flow in the river is 5.1 cubic feet per second.

Public use of the Farmington River is limited to fishing and boating. Analysis of water samples taken at the Windsor Bridge in 1955 by the State Department of Health indicates a mean bacteria count of 11,000 per 100 milliliters which renders it unsuitable for potable water or bathing.

6. Topography

Elevations in this region vary from about 100 feet above sea level at the Farmington River to hilltop heights of from 200 to 280 feet with the level areas being at an elevation averaging approximately 120 feet above sea level. The land rises to the west of the property to a north-south ridge approximately 2 miles distant and averaging 450 feet in elevation. The land slopes gently away from the property on all sides as evidenced by the direction of the flow of small streams which drain the region.

The Farmington River flows along the northerly boundary and the land on all sides is mostly open, level fields tilled for the production of tobacco, except for a small extension of wooded area on the northeast.

TABLE 1

POPULATION AND POPULATION DENSITY OF TOWNS WITHIN 10
MILE RADIUS OF COMBUSTION ENGINEERING, INC.

<u>Town</u>	<u>Approximate Distance of Town Center to Site</u>	<u>General Direction from Site</u>	<u>Square Miles</u>	<u>Estimated Population (1960)</u>	<u>Population Density Persons/Square Mile</u>
Hartford	9	S	18.6	161,077	8,660
Windsor	5	SE	29.6	19,346	654
Bloomfield	5	S	26.9	13,510	502
West Hartford	8	SW	21.6	62,210	2,880
East Hartford	9	SE	18.2	43,845	2,409
Manchester	13	S	27.6	41,907	1,518
South Windsor	7	SE	29.2	9,334	320
East Windsor	6	SE	26.6	7,480	281
Windsor Locks	4	E	9.6	11,386	1,186
East Granby	3	N	17.8	2,430	137
Simsbury	6	SW	24.5	10,094	412
1/3 Avon Incl. Avon Center	8	SW	7.4	1,749	236
Granby Center	6	N	10.0	4,961	496
1/2 Suffield Incl. Suffield Center & West Suffield	6	N	22.0	3,376	153
1/3 Enfield Incl. Thompsonville	8	NE	11.0	10,476	952
1/2 Ellington Excl. Center	12	E	17.0	2,799	165
1/3 Vernon Excl. Center	13	E	6.0	5,661	944
			323.6	411,641	

TABLE 2

DISTRIBUTION OF MAJOR DEFENSE INDUSTRIES
IN THE WINDSOR AREA

<u>Plant</u>	<u>Distance from Site</u>	<u>Direction</u>	<u>Type of Industry</u>
Kaman Aircraft	4 miles	S	Aircraft
Hamilton Standard	5 miles	N	Aircraft Components
Hartford Machine and Screw Co.	7 miles	SW	Machinery & Tools
Allen Manufacturing Co.	5 miles	SW	Machinery & Tools
Flight Enterprises, Inc.	6 miles	N	Aircraft
Ensign-Bickford Co.	6 miles	SW	Explosives
Bradley Air Field	5 miles	N	Air Field

B: General Description of Combustion Engineering Site

1. Geology and Hydrology

The geological and hydrological conditions at the site are essentially the same as discussed above for the surrounding area. Examinations of borings taken in the Combustion Engineering site indicate firm strata of sands and gravel overlying irregularly stratified glacial deposits. Geologically, the conformation suggests the erosional remnant of a glacial kame with the typical sands and gravels of such kames overlying a geologically earlier ground moraine.

The yield of ground water to wells in the vicinity of the Combustion Engineering site, based upon the mean annual rainfall of 42.7 inches, is sufficient to supply continuously all existing and known planned wells. This estimate assumes all wells are pumped 12 to 16 hours per working day for a capacity of 3000 gallons/min. The reliability of ground water supply is thus high barring overdevelopment.

Chemical and radiological analyses of both raw and treated well water and of possibly contaminated waste effluents are made on a routine basis; changes in composition or activity from any causes would be discovered rapidly.

2. Arrangement of Buildings and Concentration of Personnel

The arrangement of the Combustion Engineering site buildings with respect to the critical facilities is shown in Figure 2. Table 3 shows the personnel concentration at the various locations on the site.

C. Critical Facilities Buildings

1. Structure (See Figures 3 and 4)

The critical facilities are housed in a 98 x 98 x 30 foot high steel frame, masonry, and concrete structure. The building has two 35 x 30 x 55 foot high reactor cells, each with a control room. The cell walls are of two and one-half foot thick concrete, with the exception of the walls between the cells and control rooms, which are of four foot thick concrete. The roof over the cells is a four inch thick concrete slab. Sliding concrete access doors are provided at the rear of the cells for the installation of heavy equipment. Serving both cells are one fuel vault, a pre-assembly room, a counting room, a health physics laboratory, an instrument shop, a machine shop, and eight offices.

2. Air Conditioning

Each cell has a separate air conditioning system independent of the building system. Any radioactive dust particles resulting from normal operation are thus confined to the cells and their respective air conditioning ducts and filters. Each cell is also equipped with an emergency exhaust system which exhausts to the outside of the building via C.W.S. absolute filters. These systems can be turned on when needed to clear the cell of any airborne contamination.

TABLE 3

PERSONNEL CONCENTRATION AT COMBUSTION ENGINEERING SITE FACILITIES

<u>Buildings</u>	<u>Approx. Distance in Feet from Critical Facilities</u>	<u>0800-1630</u>	<u>1630-0000</u>	<u>0000-0800</u>
Corporate Administration	3000	800	Normally Secured	Normally Secured
Engineering & Physics	2000	134	Normally Secured	Normally Secured
Fuel Fabrication	1700	170	Normally Secured	Normally Secured
Development Laboratory	1800	120	Normally Secured	Normally Secured
Prototype Facilities	1000	400	100	100
Critical Facilities		20	Normally Secured	Normally Secured
Boiler House	2500	5	2	2
Sewage Treatment Plant	2000	2	2	2
Reproduction	100	<u>25</u>	Normally Secured	Normally Secured
Totals		1676	104	104

The air ducts to and from the reactor cells and other portions of the building are sampled and monitored for contamination on a routine basis by trained health physicists.

Except for a small number of experimental fuel rods, all of the fuel which will presently be used in both facilities is completely clad and no significant contamination problem is expected.

3. Entrances

There are six entrances to the building. Two of these, the massive concrete doors in the rear walls of the cells, are normally secured with heavy steel bars and bolted gaskets. The other four entrances are locked and wired for alarm indication, both at the guard station at the main entrance to the building, and after hours at the main guard house in which personnel are on duty twenty-four hours a day. All personnel enter the building via the main entrance past the armed guard who is on duty during working hours. The security guards and firemen attached to the site have been given training in reactor security, safety and fire hazards.

4. Alarms

a. Security

In addition to the door alarms discussed above, the building is further protected after working hours by an ultrasonic alarm system. The system consists of a group of ultrasonic transmitters and receivers located in both cells and in the fuel storage vault. When the receivers sense a change in frequency (i.e., a moving object), alarm indication is given at the main guard house.

b. Radiation Monitors

Self contained radiation monitors are installed in both cells, the vault, and the pre-assembly room. These units give local alarm whenever excessive radiation levels exist. The units in the cell are turned off automatically during reactor operation, preventing false alarm indication. In addition to local indication, alarms are automatically sounded at other strategic locations in the building to warn personnel of excessive radiation levels.

5. Security Fences

The building is surrounded with a security fence which provides direct access only to the main entrance. Two large, normally locked gates allow vehicle access to the machine shop door and to the massive sliding concrete cell doors. The fence also provides an exclusion area behind the cells; a large hill at the rear of the exclusion area provides additional radiation shielding to the area in the event of a nuclear accident.

III. REACTOR FACILITY

This section describes the basic mechanical, hydraulic, and instrument systems of the CEND critical facility to be used for the BONUS critical experiments. Operational parameters, e.g., rod speeds, flow rates, etc., are discussed in Section VII C.

A. Hydraulic and Mechanical Components

1. Tanks, Pumps and Valves

a. Reactor Vessel

The over-all view of the reactor structure is shown in Figure 5. A coated carbon steel tank eight feet in diameter and seven feet high and of 3200 gallon capacity is located in an opening in the first floor decking. Platforms above and below the main floor facilitate handling and maintenance of rod drives, nuclear instrumentation, etc.

b. Storage Tank

Mounted on the lowest level, twenty-five feet below the first floor level, is a coated carbon steel storage tank of 4000 gallon capacity and associated pumps and demineralizing unit.

c. Dump Valve

A six-inch ball valve located in the six-inch fill/drain line (Figures 5 and 6) serves for moderator dump. The valve is closed by a compressed air operated cylinder acting against a spring so connected that the valve opens on loss of air pressure and/or electrical power due to a scram or component failure. This valve will permit the full reactor to be emptied in 3 minutes at an initial rate of fall of about 0.68 inches per second. It should be emphasized that this system is used only as a "back-up" safety device; the primary (fast) safety shutdown mechanism consists of the rapid insertion of the safety rods.

The reduction in size over the 18" valve used on the NUSU Critical Facility in Cell #2 is not considered a significant reduction in safety because even that large a valve is not sufficiently fast acting to provide more than back-up to the faster acting safety rods. The basic function of a dump valve in the type of control system employed here is to provide complete shutdown of the reactor when personnel are working near it.

No credit is taken for the opening of the dump valve in the hazards analysis presented in Section VIII since the rate and time at which the reactivity subtraction due to dumping the water becomes significant is a complex function of the initial water height, the control rod positions and the core configuration. Furthermore, a detailed analysis of the interlock and safety circuitry has shown that, despite any credible sequence of circuitry or personnel failures, a scram signal which has caused the dump valve to open will always have already caused the safety rods to scram. Thus, rapid shutdown by at least 2% δk is provided by the safety rods falling into the core, further shutdown by another 2% δk is provided at a slower rate by driving the shim rods down, and the water dump provides complete shutdown at a rate inter-

mediate between these two methods.

d. Main Pump

The reactor vessel is filled from the storage tank by a $7\frac{1}{2}$ hp 100 gpm centrifugal pump. This pump is also used to circulate water, via the overflow line, when desired. The flow rate is limited by a locked valve acting as an adjustable orifice which can be set to meet the maximum water addition rates for various cores. The flow rate can be further reduced by a motor operated throttle valve operated from the control room.

The calibration of the locked valve is checked by observing the rate of rise of water in the reactor vessel every time the setting is changed (one inch equals 31.3 gallons in the empty reactor vessel and 25.6 gallons when the full BONUS mockup is installed). The calibration is also checked at regular intervals by the use of an electrical probe for determining water heights. The key for this and all other locked valves is controlled by the Crew Chief, as discussed in Section IV A.

e. Overflow Lines

A permanent 3" overflow line runs from near the top of the reactor vessel to the storage tank. Provision is made for addition of a 3" overflow line inside the tank whose height would be adjustable. At present this line is blanked off.

These lines do not contain any valves or sharp bends and either one of them is adequate to take the full discharge flow from the pump without a serious rise in water level in the vessel. The superheater pressure tubes extend over the top of the vessel so that they can not be accidentally flooded even if the overflow line became obstructed.

f. Microfill/Drain and Millidrain System (Figure 6)

Secondary hydraulic systems, the "microfill/drain" system and the "millidrain" system are employed to produce small changes in water height. The use of these systems for partial water height experiments is discussed in Section IV D.

The microfill system consists of a 20 gpm pump (supplied from the storage tank) whose outlet is connected, via a locked valve, to a line running over the top of the reactor vessel; the line runs into a tube which extends to the bottom of the vessel to prevent wave formation while the pump is being operated. Power for the microfill pump is interlocked with the control rod drives so that power is available and can be utilized only under conditions similar to those applying to control rod withdrawal after the water has been brought above the top of the core. Furthermore, the pump cannot be energized unless the "Partial Water Height" key is turned on. Lower values of flow rate will be maintained, if required, by the setting of the locked valve on the pump outlet.

The microdrain system consists of a normally closed solenoid valve in a one inch diameter drain line in parallel with the dump valve. The millidrain system, which permits more rapid draining than the microdrain sys-

tem, consists of a normally closed solenoid valve in a two inch diameter line, also in parallel with the dump valve. Power for these systems is interlocked so that it is available and can be applied whenever the partial water height key is turned on. Estimated flow rates are 60 gpm for the microdrain line and 250 gpm for the millidrain line.

g. Demineralizer Loop (See Figure 6)

A demineralizer system is provided to maintain the water at an adequate degree of purity. The system consists of a loop which contains a small pump, appropriate valving, and a resin-bed ion exchange demineralizer of 4 gpm capacity. This system is supplied from the building water supply via a free-fall tank or from the storage tank and discharges into the storage tank. It may thus be operated independently of the reactor safety interlocks whenever required to maintain adequate water purity.

2. Neutron Source

The reactor startup source is a doubly encapsulated Po-Be source 2.5 inches long x 0.30 inches diameter and is stored in a 24" diameter x 24" long paraffin and borax filled steel container located on the upper platform among the rod packages. Source strength generally ranges from 20 curies initially to about 1 curie at replacement time. The source capsule is placed in a secondary sealed brass container 4.25 inches long by 0.375 inches diameter attached to a cable, and is guided into its core location by 5/8" diameter stainless steel tubing. The cable is gear driven, and, to give jam protection, a friction clutch is located between the drive unit and the motor-gear reducer unit. Up and down limit control is provided by two microswitches.

A careful check is made when locating the down limit switch to insure that the source will not be lowered below the center of reactivity in the core. By the use of suitable tubing, the source may be guided into any desired location within limitations imposed by core dimensions.

Source position is sensed by a helipot on the drive unit and indicated to ± 1 inch by a voltmeter on the console.

3. Lower Support Structure

Four support columns are used to suspend the lower support structure from the top flange of the reactor tank (see Figure 9). The lower support is approximately six feet square, and is fabricated from coated carbon steel. The bottom grid plates, which support the entire weight of the fuel and other core components, are attached to the support structure with clamps and bolts, permitting flexibility in location. Spacers and jigs are used for final alignment prior to tightening the hold-down bolts.

B. Control Rod Drive System

1. General

The overhead control rod drive packages are supported on a grating and may be positioned at any point over the tank. Connections to the control rods are via turnbuckles and quick disconnects.

The use of shim rods which are non-scrammable represents a major difference between the BONUS facility and previous practice in the NUSU facility where all control rods were scrammable.

2. Safety Rods

The safety rods are moved by scrammable drive units shown in Figure 7. Structurally, the drive consists of an A.C. motor, gear box, magnetic clutch, drive shaft, pinion gear, and rod rack. The upper end of the rack actuates position indication and limit switches. Fine rod position is sensed by a precision helipot geared to the drive shaft after the clutch and is indicated by a sensitive digital ratiometer on the control console. An oil-filled dashpot assembly is incorporated into the rack vertical guide and is suitable for rods from 60 to 100 pounds in weight. Provision is made for the operation of up to eight safety rods. In general, 6 to 8 safety rods will be used; an interlock requires that an absolute minimum of two be employed.

These rod drive packages have proven to be extremely reliable during almost four years of operation in what is now the NUSU facility; there has never been a failure to scram in over 1200 reactor runs and innumerable scram tests.

3. Shim Rods

The shim rods are moved by non-scrammable cable drive units shown in Figure 8. Each unit consists of an A.C. motor, gear box, and drum on which a cable is wound. Limit switches indicate position and prevent overtravel in either direction. Fine position is sensed by a precision helipot. As discussed below, the interlock circuitry permits only one shim rod at a time to be withdrawn and only if the safety rods have first been withdrawn. Furthermore, the shim rods are automatically driven into the core when a scram occurs. Provision is made for the operation of up to twelve shim rods.

4. Safety Considerations

CEND Critical Facility Operating Rules require that an absolute minimum of 2% δk be available as fast safety in the form of two or more raised scrammable control rods whenever there is water in the core. The purpose of this is to provide rapid shutdown if the reactor, by some improbable sequence of events, were to become supercritical even during the initial fill.

Depending on the worth of the rods, the required fast safety can be obtained in two ways: (1) two or more scrammable rods fully withdrawn ("cocked") before water can be added to the core, or (2) all the scrammable rods withdrawn to some intermediate position in the core (safety point) prior

to water addition. Method (1) has the following disadvantages:

- a. There is a period of time before the rods reach the water.
- b. The incremental rod worth as the rods enter the water (the top of the supercritical core) is relatively low.
- c. The initial rate and shutdown margin would be halved if, for example, one of two safety rods were to become stuck.

Method (2), used at the CEND Critical Facilities, has the advantage that it will produce a larger rate of decrease of reactivity in the period immediately after receipt of a scram signal (the time when it is needed most) since more rods are starting at a point where the differential rod worth is much higher than at the top of the core.

In addition to the 2% δk fast safety requirements outlined above, it is also necessary to have a minimum of 2% δk control margin between the safety point and criticality; this is required to ensure that criticality will not be attained during the initial fill. Thus, with all control elements fully inserted, k_{eff} must be $\leq .96$, and with the safety rods at the safety point, k_{eff} must be $\leq .98$. The 2% control margin consists of the worth of the portions of the partially withdrawn safety rods still in the core and/or the worth of additional fully inserted non-scrammable shim rods. The condition in which the safety rods are at the safety point and the shim rods are fully inserted is termed "rods in safety pattern," and is a prerequisite for initial fill. After the water has risen above the core, criticality may be attained by withdrawal of the shim rods and/or further withdrawal of the safety rods from the safety point.

Except for reactor shutdown, none of the safety rods may be driven below the safety point unless the shim rods are down. Thus, an absolute minimum of 2% δk is always available as fast safety. If any safety rod is inadvertently driven below the safety point, pump power and withdrawal power to all rods ("critical power") is automatically removed. To recover it is necessary to take the reactor subcritical by driving the shim rods to the down limit and inserting the startup source.

Because of the absolute minimum of 2% δk fast safety available, it is felt that the use of these shim rods will not cause a significant increase in the hazards associated with the operation of the reactor over those existing for previous operation at the facility.

5. Summary of Restrictions

The rod control circuits provide the following restrictions on the operation of the control rods:

- a. Safety or shim rods may be withdrawn only one at a time by spring return switches.
- b. Any number of rods may be driven down at any time, either individually or as a group. In addition, they may be driven down simultaneously

with the withdrawal of one additional rod. It should be noted, however, that "critical power" is removed if any of the safety rods are driven below the safety point unless all installed shim rods are down. As a consequence of this restriction, when shim rods are present, they are the only rods that may be calibrated over their entire length.

c. An absolute minimum of two safety rods must be installed.

6. Setting of Safety Point

The number of safety versus shim rods, their location in the core, and the location of the safety point limit switches on the safety rods are determined for each new core loading by calculation and/or by previous measurements of rod worth in similar cores. This decision is constantly evaluated during the initial loading of each core.

7. Rod Speed Control

The maximum rod speed, which is controlled by the gear box ratio, is determined by the maximum incremental worth of the control rods so that the reactivity rate of 0.04% δk /sec is not exceeded. Speeds of approximately 5, 10, 15, and 20 inches per minute are available for all rods. The voltage to the drive motors may be reduced by a variac on the control console, thus further reducing the rod drive speed for fine control.

8. Magnetic Clutch Power

Power for the magnetic clutches on the safety rods is supplied from composite safety amplifiers; the current is adjusted to the lowest value for which there is no slippage of a suspended rod. In the event of a scram signal, this power is automatically removed causing the magnetic clutches to disengage, releasing the rods. At the same time, all of the shim rods are automatically driven into the core at their normal rate.

9. Drop Characteristics of the Safety Rods

Drop times have been measured for a fifty pound rod falling twelve inches from rest. The average of measurements made on several drives is 390 milliseconds from release of power to the magnetic clutch. These measurements are made on a periodic basis to ensure that the units are functioning properly.

10. Rod Drop Switches

Switches located on the control console permit manual interruption of current to the electromagnetic clutches on the individual safety rods. These switches may thus be used for dropping any selected safety rod at any time without scrambling the reactor. Note, however, that rod withdrawal power will be lost if all shim rods are not down when this is done. Of course, clutch power to all safety rods is independently turned off in the event of a scram signal.

11. Single Rod Recovery System

As discussed above, "Critical Power" (for withdrawing rods, micro-filling, etc.) is normally lost if any safety rod is dropped or driven below its safety point unless all installed shim rods are down. This is done to insure that the required minimum of 2% δk fast safety is maintained - i.e., to prevent the reactor from being kept critical by safety rod insertion compensated by shim rod withdrawal.

On occasion, however, it is necessary and safe to operate with one of the safety rods below the safety point. This is required for:

- a. Recovery from a rod drop experiment, in which a measurement of the rate of decay of neutron flux can be related to the worth of the rod which has been dropped,
- b. Recovery from inadvertently driving one of the safety rods below the safety point,
- c. Calibration, by period techniques, of a safety rod below its safety point when the insertion of this rod is compensated by withdrawal of another rod.

The Single Rod Recovery System will only be used if the remaining safety rods, at or above their safety point, provide the required 2% δk fast safety. The system will not permit recovery from a condition in which more than one safety rod is below its safety point or in which less than two safety rods remain above their safety points.

The Single Rod Recovery System consists of a key-switch and a rotary selector switch. These switches may only be operated by the Crew Chief. This system changes the "safety rods at or above safety point" requirement for maintenance of "critical power" to, essentially, "all but the selected safety rod at or above safety point."

In summary, this system may only be used if the remaining safety rods maintain the required 2% δk fast safety; does not alter any other required interlocks; and permits recovery only when at least two rods remain at or above their safety points and only one rod is below its safety point.

C. Nuclear Instrumentation and Scram Circuitry

1. Nuclear Instrumentation

The nuclear instrumentation is shown by the block diagram of Figure 10. Seven detectors - four BF₃ counters and three uncompensated BF₃ ion chambers - are used to cover ranges from startup to full power. Of these seven, five are capable of scrambling the reactor, three by excessively low period and two by excessively high power level.

These detectors, contained in 5" diameter aluminum tubes introduced horizontally through the reactor vessel wall, are located in the vessel at various distances from the core. The detectors are surrounded in the tube by polyethylene in order to provide some moderation for increase in sensitivity during initial fill.

The three scram channels outlined below must be operable at all times. Sufficient spare equipment is maintained to permit this.

a. Channel E - linear level scram adjustable range

- (1) Uncompensated BF₃ ion chamber (Reuter-Stokes Type RSN-5)
- (2) Beckman linear ~~ammeter~~ amplifier
- (3) Matching amplifier
- (4) Composite safety amplifier (Honeywell 4639-A)
- (5) 1500 volt chamber supply (Atomic 312)

b. Channel F - Period scram

- (1) Uncompensated BF₃ ion chamber
- (2) Log N and period amplifier (ORNL)
- (3) 1500 volt chamber supply

c. Channel G - linear level scram fixed range

- (1) Uncompensated BF₃ ion chamber
- (2) D.C. Preamplifier
- (3) Composite safety amplifier
- (4) 1500 volt chamber supply

The four information channels, of which at least one log and one linear channel are required to be operable, are constituted as follows:

d. Channels A and C - log level, period indication

- (1) BF₃ pulse counter (RCL 10502)
- (2) Preamplifier
- (3) Linear amplifier (ORNL-A1D)
- (4) Log count rate meter (ORNL)
- (5) Period amplifier (ORNL)
- (6) Simplytrol meter-relay
- (7) 2500 volt chamber supply

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e. Channels B and D - linear level indication, adjustable range instruments

- (1) BF₃ pulse counter
- (2) Preamplifier
- (3) Linear amplifier
- (4) Linear count rate meter (ORNL)
- (5) 2500 volt counter supply

Of these instruments, Channels A, B, C, D, and E are normally on scale during startup; at least Channel A or C, Channel B or D, and Channel E must be on scale at this time. Intermediate power is normally monitored by Channels C, D, E, and F. High power operation is monitored by Channels E, F, and G, none of which may ever be above full scale.

The response times of identical equipment on the NUSU critical assembly have been measured, as follows. Data are given below for the case of a step increase in level starting from zero input signal, the time being from application of the step to removal of clutch magnet voltage.

<u>Channel</u>	<u>Detector</u>	<u>Measuring Instrument</u>	<u>Scram</u>	<u>Time</u>
A	BF ₃ Counter	Log CRM - Period Amp.	Period	2000 msec
C	BF ₃ Counter	Log CRM - Period Amp.	Period	2000 msec
E	UIC	Beckman <i>mm</i> ammeter	Level	275-1370 msec depending on range
F	UIC	Log N - Period Amp.	Period	95 msec
G	UIC	Composite Amp.	Level	20 msec

It can be seen that while the low level startup channels (A and C) are relatively slow (i.e., generally about 2 sec delay), a period scram with 0.1 sec delay covers the medium and high level regions and a level scram with only 0.02 sec delay will initiate a scram if the level should reach the 1000 watt (or lower) point before other instruments can respond.

These measurements will be repeated on the BONUS critical before any fuel is loaded into the core.

2. Rod Drop Instrument Scrams

A scram by rod drop is effected by any one of the following conditions (see Figure 11):

- a. High level on Channel E
- b. Low period on Channel F.
- c. High level on Channel G

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d. Low period on both Channels A and C. This is not an officially required scram circuit and may be bypassed, if either channel is not functioning properly, at the discretion of the Crew Chief.

e. Manual scram (also dump moderator)

Coincident with a rod scram, the following actions are performed automatically:

- a. Shim rods are driven in to the down limit.
- b. The circulating pump is turned off, preventing addition of moderator to the reactor vessel.
- c. Withdrawal power to all rods is cut off, preventing rod withdrawal until the scram is reset.
- d. The microfill system is de-activated, preventing the addition of moderator to reactor vessel.

3. Moderator Dump Instrument Scrams

Fast safety is provided at all times by the rod drop scrams outlined above. Supplementary reactivity removal is provided by dumping the moderator on verification of the single scram signal (which has already dropped the safety rods and caused the shim rods to be driven in) by an independent signal from a separate channel. This prevents loss of moderator on a spurious scram signal from any single channel.

Moderator dump is effected under each of the following conditions (see Figure 11).

- a. High level on E and low period on F
- b. High level on E and high level on G
- c. Low period on F and high level on G
- d. Low period on A and C, and high level on E
- e. Manual scram

Each channel on sensing the existence of a scram condition causes a scram by opening a relay contact. This contact remains open until the scram is reset. Since this is the case, the two and three channel requirements for water dump constitute a verification of scram rather than a coincidence. In addition, it should be noted that water dump is always preceded, if only by a few milliseconds, by a rod scram. It should also be noted that a real transient will always initiate a moderator dump even if one of the officially required channels (E, F, or G) fails to respond at this crucial moment, i.e., any two of the three channels can produce a water dump.

Coincident with a moderator dump scram, power is also removed from any component which was receiving it via a bypass.

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C. Non-Nuclear Instrumentation

1. Rod Position Instrumentation

Rod position is sensed by Helipot in the rod drive packages and is indicated for any one rod at a time either by a sensitive digital ratiometer (to $\pm .01$ inch) or by a coarse position voltmeter (to ± 0.5 inch).

2. Water Flow Instrumentation

The water flow rate in the main fill line is sensed by a pneumatic pressure differential transmitter across an orifice in a bypass line, with remote indication on the control console.

3. Water Level Instrumentation

Liquid level in the storage tank is indicated by a sight glass on the tank. The liquid level in the reactor vessel is sensed by a pressure sensitive pneumatic transmitter located on the bottom of the vessel. Remote indication is given by a gauge on the control console.

Water height in the reactor tank can also be more accurately determined (± 0.01 in.) by an electrical probe attached to a spare non-scrammable control rod drive package. The probe is insulated from the rod drive cable and operates by completing an electrical circuit through the water in the reactor tank. Continuity indication is registered by a microammeter in the reactor console and position is indicated on the digital ratiometer. Variability in water height measurements from surface tension effects is reduced by always approaching the surface from the same direction, preferably from above.

The "water above core" relay in the interlock circuitry is closed by a normally open pressure operated switch which trips when the level in the reactor vessel exceeds a preset height.

4. Annunciators

The annunciator system, which consists of a horn and a panel of lights, announces any abnormal conditions. The following conditions are annunciated:

- a. Scrams (type and instruments responsible)
- b. Bypasses in use
- c. Water flow or pressure higher than normal
- d. Partial water height key on
- e. Pump hold key on
- f. Any circuit breaker tripped

IV. OPERATING PROCEDURES

This chapter summarizes the operating procedures employed at the CEND Critical Facilities. In particular, procedures adopted for partial water height experiments and the approach to criticality following core modifications are described in detail.

A. Administrative Control

1. Experimental Programs and Methods

a. All experimental programs and techniques must have the prior approval of the Physics Department Manager.

b. An Experimental Safety Review Committee, consisting of senior members of the Physics Department, serves as an advisory staff to the Physics Department Manager in evaluating experimental programs and techniques. The Committee includes but is not limited to the following items in its review of each experiment proposed:

- (1) the proposed core loading, number and type of control elements, location of safety point, etc.,
- (2) the method to be employed in the initial approach to criticality,
- (3) maximum rod withdrawal speeds and water addition rates,
- (4) any special experimental arrangement, e.g., void simulation, flooding and voiding of superheater sections, partial water height experiments, etc.,
- (5) the void coefficient of the proposed core,
- (6) the temperature coefficient of the proposed core.

2. Revisions and Deviations

a. Control, Interlock and Scram Circuitry

(1) Major changes in the facility, overall program, operating philosophy or control, will first be reviewed with the CEND Nuclear Safety Committee, composed of senior members of the Nuclear Division and then submitted for approval to the U. S. Atomic Energy Commission.

(2) Minor changes which do not affect the over-all operation or safety of the reactor (e.g., the addition of extra signal lights, replacement of switches or relays) require the approval of the Critical Facility Supervisor.

b. Operating Procedures and Ground Rules

Deviations from the established operating procedures and ground rules for a single experiment or series of experiments must be approved as follows:

(1) Approval must be obtained from the Physics Department Manager, who is advised by the Experimental Safety Review Committee.

(2) At the discretion of the Physics Department Manager, the proposed deviation may be referred to the CEND Nuclear Safety Committee for approval.

(3) If the proposed change is sufficiently basic in nature to affect the over-all hazards evaluation (e.g., maximum reactivity addition rate, new methods of adding reactivity), the matter will be submitted to the AEC for approval.

3. Cell Access

a. The reactor cell is secured with a combination padlock during the evening and weekend hours and in the absence of designated members of the operating crew.

b. The combination of the cell access padlock is known only to persons qualified to act as Crew Chiefs.

c. During approaches to criticality and any subsequent critical operation of the reactor, the cell access door is secured by a doorknob lock. The key is stored in the repository described below and is controlled by the Crew Chief.

4. Reactor Keys

a. All keys are kept in a locked repository in the control room.

b. The key to this repository is kept in the Crew Chief's safe when not in use.

c. The Control Power key will be released to the operator for checkout by the Crew Chief when he removes the cell door security combination padlock and opens the doorknob lock. The Operate key will be released only after the checkout is satisfactorily completed.

d. Interlock Bypass keys will be released, as required, for testing components, by the Crew Chief.

e. The Access Door Interlock Bypass key will be released by the Crew Chief only on prior approval by the Critical Facility Supervisor for each particular use.

f. The partial water height and circulating pump hold keys are to be operated only by the Crew Chief.

g. The doors to the control console are normally kept locked; the

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key is controlled by the Crew Chief and is released only for making authorized changes or repairs in the instrumentation and control circuitry.

h. The keys for the locked valves in the hydraulic system are to be used by the Crew Chief only on prior approval of the Facility Supervisor.

5. Personnel

a. The qualification of personnel as crew chief, reactor operator, etc., is determined by written or oral examination by the Critical Facility Supervisor, with approval of the Physics Department Manager.

b. The designation of personnel to act as crew chief, physicist, operator, etc., for a particular run or series of runs is performed by the Critical Facility Supervisor.

B. Interlock System

A block diagram of the interlock control system is shown in Figure 12. This circuitry acts to allow application of power to various components of the reactor only when certain conditions are satisfied. It has been designed as far as possible to be fail safe; to impose a rigid, safe start-up sequence; and to make certain that unsafe post-startup operations are impossible. A detailed up-to-date wiring diagram is maintained on the wall in the Control Room.

A description of the startup procedure is given below to illustrate the logical sequences enforced by the control system. Deviations from this sequence are prohibited by both the interlock circuitry and the operating rules. The use of jumpers to short out an interlock is absolutely prohibited; however, a keyswitch operated interlock bypass system is used to provide power for testing individual components in a safe manner.

1. Start up Sequence (See Figure 12)

a. Power to drive the source and rods down is available whenever the control power is turned on.

b. Rod clutch power may be obtained when:

- (1) all scram signals have been cleared,
- (2) the cell access door is closed or the dump valve is open,
- (3) and the source is in place.

Note that clutch power is maintained after the source is removed, in order to permit post critical operation.

c. Withdrawal power to the safety rods in the Safety Zone may now be obtained if:

- (1) a neutron count rate meter is on scale,
- (2) no interlocks have been bypassed,
- (3) the source is still in place,
- (4) and all rods have been fully inserted.

d. Dump valve closure power is available:

- (1) after the safety rods have been withdrawn to the safety point,
- (2) if the source is still in place,
- (3) the shim rods are still at the down limit,
- (4) and the cell access door has been closed.

e. The "Reactor On" light and Running Time meter are energized after:

- (1) the dump valve is closed,
- (2) and the Operate key-switch has been turned on.

f. The main flow control valve is normally opened at this time.

g. The main pump may now be turned on to fill the reactor vessel:

- (1) if the rods are still in the safety pattern.

The pump start switch normally acts as a spring return push-button which must be held down for as long as it is desired to run the pump. For some types of Approaches to Criticality (see Section IV C) this switch can be converted to an On/Off two-position system by use of a keyswitch operated by the Crew Chief.

h. The main pump is run until:

- (1) the water is brought above the top of the core.

Note that the setting of this interlock is actually at 6 inches above the top of the active fuel region, insuring that any further addition of water will have a negligible reactivity effect. For some experiments the water is raised an additional 6 inches to provide extra shielding during high level foil irradiations.

i. After the water rises above the top of the core:

- (1) the rods are withdrawn from the Safety Pattern until the reactor is brought critical.

2. Post Critical Operation

a. After the water rises above the top of the core

- (1) the main pump, in conjunction with the reactor vessel overflow, may be run as a circulating system so long as the water remains above the core.

(2) Critical Power is now available for:

- (a) withdrawal of safety rods above the safety point,
 - (b) or withdrawal of shim rods from the down limit,
 - (c) or microfilling (see Section IV D)
- as long as

- (a) the water remains above the core,
- (b) or the circulating pump is turned off and the flow control valve is closed,
- (c) and the safety rods remain above the safety point unless the shim rods are down or no shim rods are installed.

b. Note that only one means of adding reactivity (control rod withdrawal or microfilling or operation of the circulating pump) is available at any one time.

C. Initial Approach to Criticality

For a given core, one of the three following methods will be specified for the initial approach to criticality.

1. Type I - 1/M by Fuel Loading Steps

This method is used for new cores when major changes in instrumentation, interlocks or source-detector-core geometry have been effected. The probability of going critical on any step during water addition when control rods are in the safety pattern is considered remote. The procedure to be followed is outlined below:

a. Use of the "dead man" switch on the water circulating pump is not mandatory; the "hold" key may be used.

b. Beginning with no fuel, source in, water at maximum height, take counts on all channels of adequate sensitivity with rods in the safety pattern and then with all rods out; take water up, source out reading and subtract as background for all subsequent readings.

c. Insert control rods, dump water, and add fuel in initial increments of approximately 20% of the expected core loading; fill vessel with water and take counts as described above.

d. Plot 1/M vs fuel loading at each step, extrapolating to obtain critical loading for each case.

e. Reduce subsequent increments in order to approach criticality in a conservative manner; this will require judgment on the part of the crew, including proper interpretation of the instruments. The last fuel addition shall be limited to a reactivity change no greater than one half of the control margin provided by the rod pattern (i.e., no greater than one half the reactivity difference between the safety pattern with water up and critical).

2. Type II - 1/M on Water Addition and Rod Withdrawal

This method will be applied if the core characteristics are known from previous experiments and it is being re-installed or for new cores or for modifications in existing cores when the change in reactivity associated with the modifications is greater than one half of the control margin of the existing core. A stringent requirement is that no major changes in instrumentation, interlocks or source/detector/core geometry will have occurred prior to or during the core loading. Although the probability of reaching criticality on water, when all control elements are in the safety pattern, is small, the uncertainty inherent in predicting criticality or in loading requires caution in all steps leading to reactivity additions.

Under normal conditions the startup instruments are in the horizontal ports and the source is at the center of the core (16 and 24 inches, respectively from the bottom of the fuel). This configuration ensures that the addition of water in the upper half of the core is seen by the instruments as multiplication and not as attenuation, geometrical or reflector effects. Below this height

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the response is complex and must be known for a given source/detector/core geometry before this method may be used. The presence of the Critical Facility Supervisor or his designee in the control room is required during all approaches to criticality using this method. The procedure to be followed is outlined below:

- a. Load the full core.
- b. With the source in and the control rods in the Safety Pattern, take a zero reading on all channels of adequate sensitivity.
- c. Using the "dead man" switch on the circulating pump, raise water level to the bottom of the fuel, stop filling and take counts on all channels. Note - the use of the "hold" key on the water circulating pump is forbidden during this entire portion of the startup procedure.
- d. Continue filling in increments of about 10% of the core height, pausing long enough between water additions to observe stable neutron levels on the recorders. Measurements of $1/M$ or withdrawal of source at each level will be ordered by the Facility Supervisor on a basis of a comparison of the behavior of level recorders and period meters with the known behavior in previous similar cores.
- e. If criticality with the rods in the Safety Pattern appears imminent while the water is being raised, the reactor will be shut down. After an investigation of both the core condition and of the calculations supporting the loading which was chosen, the loading will be reduced and the procedure will be repeated.
- f. When the tank is full, withdraw control rods individually to the first of a set of incremental positions appropriate to expected core characteristics. The frequency of $1/M$ measurements will again be determined by the Facility Supervisor. The main pump "Hold" key may be used at this time (since the water is above the core) to maintain circulation, if desired.

3. Type III - Quasi-Continuous Approach to Criticality

When criticality has been obtained by one of the preceding methods, and provided the core modifications have been simple and well within the control range of the rods, approaches will be made in a quasicontinuous rather than stepwise manner. All such changes are limited to less than one half of the control margin (i.e., reactivity difference between the Safety Pattern with water up and critical) whether or not compensated.

Progress in this approach will be founded on the level recorders and period meters. The probability of going critical on water addition, with control elements in the Safety Pattern, is again remote; therefore the "hold" key may be used on the water circulating pump.

The main difference between the approach to criticality described in this section and those approaches described in the previous two sections is that the approach is now quasicontinuous rather than stepwise; $1/M$ techniques are not used. The reactor operator will, however, pause and/or attempt to withdraw the source whenever necessary; the frequency of his pauses is now based on previous experience with the particular core.

D. Partial Water Height Experiments

1. Method (See Figure 6)

It will be necessary during each approach to critical to raise the water to the top of the core before control rods are moved from the Safety Pattern. This is done in order to demonstrate that each configuration studied is subcritical (by at least 2% $\beta k/k$) with rods at Safety Pattern and water fully up. Criticality will be obtained with the water below the top of the core only after the reactor has been brought critical by normal means - i.e., by raising the water above the top of the core with the main pump and then pulling control rods. After this has been accomplished, the main pump is turned off and the main flow control valve is closed. The Partial Water Height key-switch is then operated by the Crew Chief and the water is lowered by means of the microdrain and/or millidrain lines. The decrease in reactivity is compensated by withdrawing control rods until the reactor is critical with all rods out of the water. The "micro-fill" pump is used for final adjustment of level and to measure the incremental water worth under these conditions.

2. Interlocks (See Figure 12)

a. After "critical" power is obtained as above, the microfill/drain and millidrain system may be operated if, in addition, the Partial Water Height key-switch has been activated by the Crew Chief. The microfill/drain switch is a three-position switch with spring return from the microfill position to the center position. The microfill position is wired in series with the control rod withdrawal switches, thus ensuring that only one means of adding reactivity is possible at a given time. The microdrain or millidrain operation, as selected by a separate toggle switch, can be carried out simultaneously with control rod withdrawal; the fill/drain switch is not spring returned from the drain position since draining will never add reactivity.

b. Once the water is lowered below the top of the core:

(1) the main pump is automatically turned off if it had not been done so manually;

(2) "critical" power for control rod withdrawal and the microfill/drain system would be lost immediately if the main flow control valve were opened or if the main pump were somehow energized.

(3) the main pump cannot be turned back on until the water is again brought above the core using the microfill pump or unless the reactor is brought subcritical by driving the rods back down to the Safety Pattern.

E. Bypasses (See also Figure 12)

1. Philosophy

The use of jumpers for bypassing any interlock, whether for testing a component or for operating the reactor, is absolutely prohibited. Under certain circumstances, power for testing purposes may be made available to the various reactor components by the use of a key operated interlock bypass system, even though the normal requirements for performing these operations are not satisfied. However, when any interlock is bypassed, additional circuitry makes the remaining interlocks even more restrictive. THE REACTOR CANNOT BE TAKEN CRITICAL IF ANY INTERLOCK IS BYPASSED. This is accomplished by two means:

a. Reactivity cannot be increased since the usual sources of power to the control rods and both pump systems are removed while any bypass is in effect. In addition, power can be removed from the component being tested by the manual scram buttons or by a coincident instrument scram.

b. For most of the bypasses, an additional condition (e.g., dump valve open) must be satisfied before use of the key will yield bypass power to the desired component.

2. Auxiliary Conditions

The auxiliary conditions on the use of each bypass are listed below:

a. The clutch power may be energized by bypass, provided that all scrams are cleared.

b. Dump valve closure power may be obtained by bypass if the circulating pump (which pumps water into the reactor vessel) is off and all scram signals have been cleared.

c. Circulating pump power may be obtained by bypass provided that all scrams have been reset and that either water is above the core or the dump valve is open and the main flow valve is closed.

d. Power to the control rod, microfill/drain and millidrain systems may be obtained if all scrams are cleared, the dump valve is open, the flow control valve is closed, and the circulating pump is off. Note that the "only one at a time" condition still applies in this case and that there is an additional key-switch required to energize each of the systems - the Clutch Power bypass key is required to actually drive the rods up or down, and the Partial Water Height key is required to actually operate the microfill/drain and millidrain solenoids and pump.

3. Access Door Bypass

a. Under certain circumstances it becomes essential to permit cell entry with the water up; typical reasons include, but are not limited to, the following:

(1) Foil removal following irradiation - the water shields personnel from radiation.

(2) Observe existence of bubbles in moderator.

(3) Approved valving changes required during a temperature co-efficient experiment.

b. A door interlock bypass system has been designed which will not allow any unsafe acts to be performed under any foreseeable circumstances. This bypass will be used only at the direction of the Crew Chief and prior permission must be given by the Facility Supervisor for each particular use.

c. The protection afforded by this bypass system is as follows:

(1) Reactivity cannot be increased from the console because the operation of the bypass key removes drive power from the control rods, the microfill/drain system and from the main pump.

(2) The control rods must be in the Safety Pattern, ensuring that the reactor is shut down by at least 2% $\delta k/k$.

(3) The source must be in, insuring that the nuclear instrument system and the audible monitor are on scale.

(4) The conditions above also insure that the automatic scram system is operating and that at least 2% $\delta k/k$ is available for fast shutdown by an instrument or by the manual scram button.

(5) When the bypass key is operated, the access door bypass relay will be energized and the door may be opened without causing a scram. If the door is open and any single one of the required conditions is subsequently violated, both rods and water will be dropped automatically.

It should be noted that a potential safety hazard exists to personnel working near a cocked rod under these circumstances. In the event of a scram signal from, e.g., noise generated by jarring a chamber, the falling rod could inflict serious injury.

The presence of personnel in the cell with the access door closed is absolutely prohibited at this or any other time that there is any fuel in the reactor vessel.

V. OPERATING PHILOSOPHY AND RULES

A. Methods of Insuring Safe Operation

1. The first and foremost prerequisite for safe operation of a critical facility lies in the attitude and level of competence of the personnel who are responsible for the administration and operation of the facility.
2. A second rule to be followed in conducting all operations is that no work will be done under extreme duress of time.
3. The rules and procedures for the facility must not be unnecessarily restrictive but at the same time should provide for safe operation. Non-realistic rules will encourage frequent violation of the letter of the law and if this practice is condoned, the spirit of the law eventually may be disregarded also.
4. All equipment used in connection with critical experimentation must have a high degree of reliability.
5. It will also be required that experimentation be performed only with cores of a type approved in the Safeguards evaluation.
6. Automatic safety devices with a sufficient speed of response and safety margin to accommodate any reactivity changes which may conceivably be encountered will at all times be provided. In addition, the control circuitry of the reactor will include automatic interlocks which will enforce an approved safe sequential startup and operational procedures.
7. It will also be absolutely necessary to ensure that all cores are properly assembled and loaded to criticality.
8. All equipment shall be designed, insofar as possible, to be "fail safe." Thus, failure of electrical power, air pressure, electronic components, etc., should automatically result in shutting down the reactor in a safe manner. Also, the failure of a mechanical component in the reactor should, at worst, not increase reactivity; the failure of an instrument should be immediately obvious or give a pessimistic reading.
9. The design of all circuits, procedures, etc., shall be such that the reactor will be safe after a sequence of three improbable events, with the following definitions:
 - a. "Safe" - protected from going onto a fast transient even though that transient would be terminated by temperature coefficient or water expulsion without any radiation exposure or mechanical damage.
 - b. "Improbable" - unlikely under realistic assumptions.
 - c. The primary event in the sequence, an excessive addition of reactivity, must in itself be improbable.

d. Operator error or inattention may be a primary event.

e. Inattention by the entire operating crew or their failure to institute remedial action may be considered a single improbable event in those sequences where their action could have terminated the sequence. Thus, the usual line of defense is:

(1) Improbability of cause.

(2) Attention by well-trained crew.

(3) Operation of electromechanical shutdown system despite any single failure.

10. It is considered basically impossible to protect the reactor by electromechanical means against sabotage or deliberate suicide by an individual who is reasonably familiar with its operation. The basic safeguard here lies in the selection of personnel and in the physical security measures imposed by fences, armed guards, etc.

11. During approaches to criticality and subsequent operations, the Crew Chief will have absolute authority over the operation of the reactor. However, if any person present, regardless of rank, does not consider an act about to be taken by or under the direction of the Crew Chief to be safe, then all operations will be halted until the disagreement is resolved.

12. There will be a minimum of three qualified persons present at all times during operation of the reactor.

13. It should be understood that the rules listed in this section are necessary but not sufficient as the criteria for insuring safe operation. It should be re-emphasized that the primary safeguard in any critical experimentation lies with the operating personnel and their degree of competence.

B. Summary of CEND Operating Rules

1. The following criteria, predicted by means of calculations or previous measurements, shall apply for all cores:

a. $k_{eff} \leq 0.96$ with water above the core and all control elements fully inserted.

b. $k_{eff} \leq 0.98$ with water above the core and control elements in the Safety Pattern, i.e., safety rods at the Safety Point and shim rods fully inserted.

c. $\delta k/sec \leq 0.04\%$ $\delta k/sec$ for all reactivity addition rates when $k_{eff} > 0.96$.

2. Criticality will normally be first obtained by withdrawing rods and not by raising water.

3. Periods between 30 and 100 seconds or more will be considered normal; a period of less than 10 seconds, if obtained, will require that the operator begin inserting rods; period scrams will be set at no less than 5 seconds.

4. The normal high power level will be approximately 50 watts; the maximum operating level permitted will be 500 watts; the level scrams will be set to scram at no more than 1,000 watts on fixed scale instruments, and no more than a factor of ten above the actual power on variable scale instruments.

5. An absolute minimum of two safety rods will be employed at all times.

6. The sequence of operations will be chosen so that any positive reactivity rate can be overcome by a negative rate of equal or greater magnitude.

7. The interlock circuit will allow the addition of reactivity by only one method at a time, i.e.:

- a. Addition of water by the main pump, or
- b. withdrawal of a single control rod, or
- c. addition of water by the microfill pump.

8. The interlock circuit will allow shim rod withdrawal only if the safety rods are at or above the Safety Point.

9. Cover plates will be installed around all cores to prevent objects from falling in and increasing the water level during partial water height experiments.

VI. REACTOR CORE

A. Core Structure

1. General Arrangement (See Figures 9 and 13)

The core consists of a central 8 x 8 array of ~4 inch square assemblies (boiler) surrounded by four 1 x 8 arrays of ~4 inch x 9 inch assemblies (superheater). The assemblies are supported on grid plates which serve to position them relative to each other and to transmit the weight of the fuel to the core support structure.

The boiler grid plate is ~34.5 inches square, and contains 64 holes, one for each assembly. It is bolted to the support structure at the approximate center of the vessel. The four superheater grid plates each contain tapped holes for bolting down the superheater assemblies. The plates are accurately positioned and bolted to the lower support structure. The boiler/superheater water channel may be varied by relocating the superheater grid plates on the lower structure. Internal boiler or superheater water channels may be changed by dismantling the core and replacing the appropriate grid plate.

The array of assemblies is stabilized and positioned at the top by means of an angle iron frame which is bolted to the rim of the reactor vessel. The assemblies are connected to the frame and to each other using screws, spacer bars and shims to produce a rigid structure with accurate spacing between components.

An additional angle iron frame is provided to support cover plates which are always installed over the top of the vessel to provide a working platform and to prevent objects from falling into the vessel and raising the water level during a partial water height experiment.

2. Boiler Assemblies (See Figures 9 and 13)

Each boiler assembly consists of a four-inch square aluminum box having 0.090 wall thickness and containing 32 fuel rods. The rods are arranged in a 6 x 6 array on a 5/8 inch square pitch with the four central fuel rods missing. The rods are positioned within the boxes by aluminum and nylon "grid tiles" located at the top, bottom, and center of each assembly. The grid tiles also contain four central 1/2 inch holes and 5 rows of 5 each 1/4 inch holes for the insertion of aluminum, stainless steel, or boron rods for water displacement and/or introduction of fixed poison elements. The box walls are formed from two pieces of aluminum angle, tack welded at their intersection. The tiles are each held in the box by 12 screws. The box walls are pierced near the lower ends to allow for water drainage.

These assemblies do not have sufficient mechanical strength or rigidity to permit them to be inserted or removed from the core while loaded with fuel rods; they will be accurately positioned and rigidly fastened before any fuel rods are loaded into the core.

3. Superheater Assemblies (See Figures 9 and 13)

Each superheater assembly consists of a $\sim 4\frac{1}{2}$ inch x $9\frac{1}{4}$ inch "box" containing 32 pressure tubes in a 4 x 8 array. Each pressure tube contains a fuel rod; the ~ 0.116 inch annular gap between the fuel rod and pressure tube mocks up the steam space in BONUS. Each pressure tube is individually sealed by a plug welded to the bottom of the tube; thus, if a plug weld failed, no more than that one element would be flooded. The pressure tubes extend over the top rim of the reactor vessel making it impossible to accidentally flood these tubes by raising the water level in the tank. $1/8$ inch aluminum channels at each end provide a support for the grid tiles which position the pressure tubes.

The general construction is similar to that of the boiler assemblies except that there are no sides to the "box." Aluminum grid tiles are used near the top and bottom of the fuel to accurately position the pressure tubes and an additional tile is used at the top of the assembly to maintain adequate alignment and rigidity.

4. Superheater Flooding and Voiding

Flooding and voiding of the steam spaces in each superheater fuel element will be done manually with the reactor shut down. As discussed in Section VII C this is completely safe so long as there is no water in the core proper - i.e., if the dump valve is open, since the reactor is then subcritical by at least 20% $\delta k/k$.

Flooding is accomplished by manually pouring water into a steam space. The water will be obtained from the storage tank via the microfill pump and line. This operation requires the presence of a reactor operator at the console, use of the "Critical Power Bypass" and "Partial Water Height" keys and is further interlocked to require that the dump valve is open, the main pump is off and the main flow control valve is closed.

Voiding will be accomplished by applying air pressure, via a suitable adapter, to the steam spaces in up to 32 superheater tubes at once, and expelling the water through individual small aluminum tubes permanently installed in the steam space region in each pressure tube.

B. Fuel Elements (Figure 14)

Natural and 1.85% enriched UO_2 fuel clad in 25 mil thick aluminum will be used in the boiler region and 3.41% enriched UO_2 fuel clad in 18 mil stainless steel in the superheater region of the BONUS mockup. This fuel was formerly used in the NUSU Phase B and C experiments. Table 4 lists pertinent data on the three type of pellets and their respective cladding.

The boiler fuel rods are accurately positioned in the grid tiles by the bayonet at the lower end of the rod and the 0.005 inch radial ridge near the top of the rod. The superheater fuel rods are accurately positioned in the pressure tubes by the bayonet in the lower end of the rod and by a plastic spacer held on the top end by a threaded lifting rod.

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TABLE IV

FUEL ELEMENT CHARACTERISTICS*

PELLETS (10.35 gm/cm³)

<u>Type</u>	<u>Used in</u>	<u>Enrichment</u> (w/o)	<u>Length</u> (inches)	<u>Diameter</u> (inches)
I	Boiler	0.714	0.550	0.445
II	Boiler	1.85	0.650	0.445
III	Superheater	3.41	0.625	0.500
IV	Superheater	3.41	0.750	0.500

CLADDING

<u>Type</u>	<u>Material</u>	<u>O.D.</u> (inches)	<u>I.D.</u> (inches)
I & II	6061 Aluminum	0.500	0.450
III	316 Stainless Steel	0.542	0.506
IV	304 Stainless Steel	0.542	0.506

FUEL RODS

<u>Type</u>	<u>Active Length</u> (inches)	<u>UO₂/Rod</u> (kg)	<u>U²³⁵/Rod</u> (gm)	<u>No. of Rods</u>	<u>Total UO₂</u> (Tonnes)	<u>Total U²³⁵</u> (kg)
I	48	1.266	7.96	148	0.19	1.18
II	48	1.266	20.63	2100	2.66	43.32
III	48	1.600	48.07	300	0.48	14.42
IV	48	1.600	48.07	900	1.44	43.26

*All dimensions and quantities are nominal

C. Control Rods

The control rod arrangement is shown in Figure 13. Two types of control rods will be required for the BONUS critical experiments, cruciform and slab rods.

1. Cruciform Rods in Boiler Region

The cruciform rods to be used in the boiler region are 65 inches long and 7 inches wide. They are fabricated from one-eighth inch thick stainless steel containing 2% natural boron. They are held together by 1 x 1 x 1/16 inch aluminum angles riveted in the angle formed by the intersection of the flats.

The rods are connected to the drive packages by hangers and turnbuckles so that they may be positioned vertically as required for proper alignment in the core.

2. Slab Rods between Boiler and Superheaters

The slab rods to be installed between the boiler and superheaters are 65 inches long, and are fabricated from 12 x 1/4 inch stainless steel containing 2% natural boron. The rods are connected to the rod drive packages by a hanger and turnbuckle arrangement, as described above.

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VII. EXPERIMENTAL PROGRAM

A. General Objectives

Measurements will be performed on a nuclear mockup of the BONUS reactor, a boiling water system with integral nuclear superheat and fueled with cylindrical fuel rods containing low enrichment UO_2 pellets. Analytical methods developed with the aid of the BONUS and NUSU experiments will permit extrapolation of the mockup results to the precise reactor system, permitting an accurate determination of the physics characteristics of the final design.

The intention of the BONUS mockup experiments is to derive, within the limitation of cold critical experiments, information that will increase the confidence in analytical methods at all temperatures. These experiments will be concerned with criticality, power distribution, and reactivity coefficients. Criticality will be obtained by control rod adjustment and by critical water height. Power distributions will be measured by foil, wire, and fuel element activation. Reactivity coefficient measurements are to be made by the methods of critical water height, control rod movement, and by the addition or removal of poison, fuel, etc. The experimentation will also yield the cold shutdown margin, initial startup information, and provide training for BONUS operating personnel.

B. Measurements Planned

1. Initial Criticality

a. Before loading fuel into the core, the nuclear instrumentation will be thoroughly checked out with hand sources, pulsers, etc. Scram times will be measured for each scram channel, and compared to those for identical equipment on the NUSU critical. All other circuitry and components in the facility will also be checked thoroughly. A series of dry runs with no fuel will then be made both as an operational check on the system and to familiarize personnel completely with the facility.

b. The boiler will then be loaded by inverse multiplication (Type I Approach) to a 4 x 5 array of boiler assemblies as in NUSU Experiment B-1. The only differences from NUSU Experiment B-1, which used the same fuel, will be in the presence of aluminum boxes around the assemblies, a slightly changed water channel thickness, and in the use of the boron steel cruciform safety rods rather than the cadmium slab safety rods used in NUSU.

With 0.1% to 0.2% $\$k$ excess, another series of operational tests will again be performed at this time, including the observation of instrument behavior at various power levels.

c. The full 8 x 8 boiler will then be constructed and an external superheater added on one side, as in NUSU Experiment B-5. The loading will be attained in several steps, each employing a Type II approach. In accordance with the policy of the Nuclear Division, the detailed plans for this and all experiments will be reviewed by the Experimental Safety Review Committee.

d. The full core, including 64 boiler assemblies, 32 superheater

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assemblies, and 17 control rods will then be loaded and taken critical by a Type II approach.

2. Phase 1 Experiments - Reactor Safety

Measurements will be made of the shutdown margin, the total rod worth of a bank of rods, and the reactivity changes associated with voiding and flooding of the superheater.

a. Measure "cold" shutdown margin by withdrawing rods to make the reactor critical and evaluate the worth of various combinations of withdrawn rods for both superheater voided and flooded.

b. Measure "cold" control rod worth of central boiler rod, eight superheater rods, each group of four boiler rods with superheater voided and flooded.

c. Same as Item b above, but for "hot" condition with aluminum water displacers in the boiler and superheater and with and without stainless steel shims.

d. Measure reactivity change associated with voiding and flooding of the superheater in the "hot" condition for various control rod patterns with and without aluminum water displacers and with and without steel shims.

3. Phase 2 Experiments - Power Distributions and Power Regulation

Measurements will be made of the detailed radial and axial power distributions in the core. The shift in the ratio of the power output of the boiler with respect to the superheater as a function of rod position is to be determined. The local peaking between boiler and superheater is to be measured and a means of reducing it is to be determined experimentally.

a. Measure ratio of the power in the boiler to that in the superheater for the "cold" condition.

b. Measure ratio of power in the boiler to that in the superheater for the "hot" condition for various control rod patterns with and without aluminum water displacers. Very detailed measurements are to be made of local peaking occurring between boiler and superheater fuel rods.

c. Insert combinations of shims between boiler and superheater and between first and second pass of superheater, and repeat Items a and b.

4. Phase 3 Experiments - Burnup Characteristics

These measurements are concerned with reactivity and power distributions, power regulation and power peaking associated with the core as a function of fuel burnup

a. Measure voiding and flooding reactivity changes for "hot" core with boiler simulating reactivity at 10,000 Mwd/ton and superheater at 5000

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Mwd/ton. Also measure power distributions and power regulation for this case.

b. Measure voiding and flooding reactivity changes for "hot" case with fresh boiler fuel and superheater at 5000 Mwd/ton burnup. Also measure power distributions and power regulation.

c. Determine rod programming schedule with burnup that maintains proper power regulation as well as a satisfactory power distribution.

d. Determine schedule for the withdrawal of superheater shims that maintains proper power regulation as well as a satisfactory power distribution.

C. Reactivity Coefficients

1. Physics Analysis

Physics analysis has been performed on the BONUS Mockup Core. The analysis utilizes the three-group, two-dimensional IBM-704 PDQ Program. The method used to determine the three-group constants, reactivity, reactivity coefficients, and control rod worth (as well as comparison with cold critical experiments) is found in CEND-105, Supplement 2.⁽¹⁾ A more detailed description of the analytical methods and the comparison with experiment is found in GNEC-133.⁽²⁾ Furthermore, the preliminary results of the NUSU Phase B Experiments are encouraging by indicating fairly good agreements with predictions using these methods.

The nuclear characteristics of the reference design core at 68°F are the following:

a. Reactivity (k_{eff})

All rods withdrawn, superheater flooded 1.157

All rods withdrawn, superheater voided 1.166

b. Control Rod Worth (Rods Withdrawn from Rod-Filled Core)

Central rod 1.8% δk

Four cruciform rods near core center 4.8% δk (1.2% δk /rod)

Four cruciform rods away from core center 4.4% δk (1.1% δk /rod)

Eight slab rods 8.0% δk (1.0% δk /rod)

c. Total reactor Reactivity Coefficients (Calculated for No Control Rods in Core)

Void coefficient ($\delta k/\%$ void)	-1.40×10^{-4}
Moderator temperature coefficient ($\delta k/^{\circ}\text{F}$)	-0.25×10^{-4}
Doppler coefficient ($\delta k/^{\circ}\text{F}$)	-0.10×10^{-4}

d. Manual Superheater Flooding and Voiding in a Dry Core

The full BONUS mockup core is calculated to have a k_{eff} of ≤ 0.80 when there is no water in the reactor vessel and no control rods are present. If the superheater steam spaces are flooded under these conditions, the maximum reactivity thereby introduced is calculated to be not more than $2\% \delta k$. Conversely, manual voiding of the flooded superheater in a dry core will result in a decrease in reactivity of $2\% \delta k$ in an assembly already subcritical by about $18\% \delta k$.

2. Reactivity Addition Rates

a. Control Rod Withdrawal

The maximum worth of any rod for the configurations considered is estimated to be no more than $3.5\% \delta k$ for small cores of this type. Allowing a factor of two in peak-to-average differential worth, the withdrawal speed for a 48 inch rod would be set at 16 inches per minute to limit the reactivity addition rate to $0.04\% \delta k/\text{sec}$. Rods of lower worth could, of course be safely withdrawn at a correspondingly higher rate of speed - i.e., a rod worth 2.9% or less could be moved at the maximum available rod speed of 20 inches/minute.

b. Water Level Control

For all experiments, it is estimated that criticality cannot occur at water levels below 18 inches with all movable rods withdrawn. At 18 inches, the differential worth of water is conservatively calculated to be $1.0\% \delta k/\text{in}$. The only pumping system normally operable under reduced water height conditions is the microfill system, whose 20 gpm output would result in a reactivity addition rate of only $0.013\% \delta k/\text{sec}$ at a water height of 18 inches. A flow rate corresponding to this low a reactivity addition rate was selected to facilitate fine reactivity control during partial water height experiments.

The flow rate in the main fill line is limited to 90 gpm by means of the locked valve. If criticality were to be attained during initial fill, the reactivity addition rate would be $\leq 0.04\% \delta k/\text{sec}$ at any water height above 21 inches. At any level below this, the maximum reactivity addition would be $\leq 0.06\% \delta k/\text{sec}$, i.e., corresponding to the maximum $1\% \delta k/\text{inch}$ at the minimum possible critical water height of 18".

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The following table gives values of $\partial\rho/\partial t$ for the various hydraulic loops at $\partial\rho/\partial h$ values of 1.0, 0.5 and 0.25% δk /inch.

TABLE 5
Values of $\partial\rho/\partial t$ for Hydraulic Systems

System	Flow Rate(GMP)	Rate of Rise in Vessel (in./Min)*	% δk /sec for $\partial\rho/\partial h$ of		
			1% δk /in.	0.5% δk /in.	0.25% δk /in.
Main Fill	90	+ 3.5	+0.060	+0.030	+ 0.015
Microfill	20	+ 0.78	+ .013	+ .006	+ .003
Microdrain	60	- 2.3	- .038	- .019	- .009
Millidrain	250	- 9.8	- .163	- .082	- .041
Dump	1050	-40.8	- .680	- .340	- .170

*The calculation assumes that the core structure and control rods are installed, and that all displacer, fuel rod, and superheater tube positions are occupied (25.6 gal/inch).

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VIII. HAZARDS ANALYSIS

A. Analysis of Relatively Mild Reactivity Excursions

Light water critical assemblies fueled with UO_2 generally exhibit a high degree of inherent stability and can absorb modest amounts of reactivity insertion without any damage. The Doppler effect, which is associated with the change in fuel temperature, results in a prompt negative temperature coefficient that limits the magnitude of the transient associated with a fast excursion. This prompt limiting of power allows sufficient time for the slower moderator density coefficients to increase in magnitude and further decrease the reactivity. The moderator density coefficients include both thermal expansion and formation of voids within the moderator.

The startup and reactivity addition procedures in the CEND Critical Facilities are effectively controlled by interlocks which allow only one method of reactivity addition at a given time. The rate of reactivity addition by each of these controlled methods does not exceed $0.04\% \delta k/\text{sec}$. It would take more than 15 seconds of continuous reactivity addition at this rate to attain prompt criticality. This rate should therefore allow sufficient time for an operator to scram the reactor manually. Furthermore, the automatic safety circuits should scram the control rods and initiate the moderator dump before a power level of 10 kw* is reached. Finally, if a scram does not occur, the reactor will undergo a mild excursion without any release of radioactivity or significant damage to the fuel. The chain of events leading to the excursion is highly improbable due to the succession of improbable events and circuit failures required to initiate and maintain it, to inactivate the scram circuitry, and to prevent a manual reactor scram.

1. Causes

The three possible causes of an excursion of this type are cited: (a) uncontrolled rod withdrawal, (b) uncontrolled water addition to the reactor vessel and (c) uncontrolled water level change in the superheater.

a. Uncontrolled Rod Withdrawal

The failure of a rod drive control switch or relay could result in the uncontrolled withdrawal of a single control rod. The uncontrolled withdrawal of two or more rods simultaneously is felt to be so highly improbable that it can be neglected here.

*This figure is the calculated maximum power reached after an automatic power level or period scram. The following conditions were assumed: power level scram at 1 Kw, period scram at 5 sec, measured response time as given in Section III C and a ramp reactivity input of $0.04\% \delta k/\text{sec}$. No matter what the initial power level, each of the scram circuits will trip when or well before the reactor reaches 10 kw.

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The maximum rod speed for each core configuration is preset by a suitable choice of gears on the basis of calculated or measured rod worths so as not to exceed a maximum rate of reactivity addition of 0.04% δk /sec. Thus a rod worth 2.9% δk or less can be moved at the maximum available rod speed of 20"/min over its span of 48", allowing a factor of two for the ratio of the peak incremental worth to the average.

b. Uncontrolled Water Addition in the Reactor Vessel

The accidents considered here are caused by (1) continuous operation of the microfill pump during a partial water height experiment, (2) operation of the main fill pump during a partial water height experiment, and (3) a loading error which caused criticality to be attained during initial fill with the rods at the safety pattern.

The minimum critical water height for the proposed core configurations with all movable rods withdrawn is calculated to be 18 inches. The differential worth of water at this height is 1.0% δk /in. Personnel or single component failure could result in the continuous addition of 20 gpm of water to the reactor tank by the microfill pump under these conditions, corresponding to a continuous reactivity addition of 0.013% δk /sec.

If in addition it were assumed that the main pump were inadvertently activated, adding water to the vessel at 90 gpm at this time, the corresponding reactivity addition rate would still be $\leq 0.06\%$ δk /sec if the water height were between 18 and 21 inches, and $\leq 0.04\%$ δk /sec at any height above 21 inches. This would have required a multiple component failure - i.e., relay or switch failure and opening of the main flow control valve, which is required by the interlock circuitry to be closed during partial water height experiment.

Finally, an accident could be caused if a large error were made in loading, and the reactor became critical on initial fill during the approach to criticality. Here, again, reactivity addition would be $\leq 0.04\%$ δk /sec if criticality were attained at any height above 21 inches, and $\leq 0.06\%$ δk /sec at any height between 18 and 21 inches.

It should be noted that the effect of the faster rate of rise of water in the tank due to volume displacement by the core has been taken into account in all of these calculations by assuming that the core structure and control rods are installed and that all displacer, fuel rod, and superheater tube positions are occupied.

c. Uncontrolled Superheater Flooding and Voiding

Accidental flooding of the superheater steam spaces cannot occur even if the reactor vessel overflow were to become plugged while the main pump was in operation because the steam tubes extend above the top of the reactor vessel.

Leaking of the pressure tubes could never cause voiding of a flooded element while there is water in the core. It is predicted that flooding all superheater elements will cause a reactivity decrease of 1%. Even if flooding the superheater led arbitrarily to as much as a 5% increase in k_{eff} , it would require 4 elements to fail simultaneously and flood within one second in

order to introduce reactivity at the rate of $0.04\% \delta k/\text{sec}$. To ensure the accuracy of physics data, the superheater tubes will be allowed to remain overnight in a flooded condition; any leakage will be indicated by a decrease in level in the affected tubes.

As described in Section VI A, the superheater tubes will be manually flooded and voided only when there is no water in the reactor vessel. As shown in Section VII C, the flooded superheater in a dry core is subcritical, and thus no accidents can be postulated while manual flooding or voiding of any or all of the superheater steam spaces.

It should be emphasized that the present design is safe independently of whether flooding or voiding lead to positive reactivity additions.

2. - Results of a Relatively Mild Reactivity Excursion

The effects of a power excursion initiated by an uncontrolled reactivity addition due to any one of the three causes discussed above can be conservatively described by a ramp reactivity input function of $6 \times 10^{-4}\% \delta k/\text{sec}$. An outline of the method employed in the transient analysis is given in Ref. 1. The analysis assumes that the scram system is inoperative, thus the probability that the power excursion would actually exceed $\sim 10 \text{ kw}$ is relatively low due to the succession of improbable events and personnel and circuit failures required to deactivate the automatic scram circuitry and prevent a manual scram.

The results of this excursion are summarized in the table below and illustrated graphically in Figure 15. No melting of fuel or clad is indicated by the maximum temperatures, thus one can conclude that the core is surely undamaged by this very conservative estimate of the consequences resulting from a worst plausible accident of an introduction of only $4 \times 10^{-4} \delta k/\text{sec}$.

TABLE 6

Results of a Mild Reactivity Excursion (See Figure 15)

Input Reactivity ($\delta k/k\text{-sec}$)	Duration (sec)	Integrated Power (mw-sec)	TEMPERATURES ($^{\circ}\text{F}$)			
			Clad (1)		Fuel (2)	
			Max.	Ave.	Max.	Ave.
6×10^{-4}	16	560	890	390	1630	650

(1) Aluminum melting point 1220°F

(2) UO_2 melting point 5000°F

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B. Maximum Credible Accident

1. Causes

The most serious accident conceivable is the rapid insertion of 1.5% δk excess reactivity in a full BONUS mockup. This amount of excess reactivity could be inserted in one of two ways: a manual withdrawal of a central control rod or by an object falling into the reactor vessel. The succession of personnel errors and of failures in control circuitry required are so unlikely that one must almost postulate sabotage to produce this accident.

a. Manual Withdrawal of a Central Control Rod

The insertion of 1.5% excess reactivity can be postulated by the withdrawal of a control rod worth 3.5% $\delta k/k$ from a core initially shut down by 2% $\delta k/k$. Access to the cell when the moderator is above the core and control rods are in Safety Pattern ($k \leq 0.98$ with water up) can be achieved only by use of the access door bypass key. The natural physical hindrances to be deliberately surmounted by a single individual in gaining access to a 5 foot long 50 pound object hanging in the center of the 8 foot diameter reactor tank and effecting its full withdrawal in ~ 0.6 seconds are additional deterrents to the accident.

The act of pulling a control rod from a flooded core is an extreme violation of common sense and all basic safety rules and would require inattention on the part of the remainder of the operating crew. The ensuing accident and its aftermath are considered to have been extremely conservatively calculated for the following reasons:

(1) The quoted rod worth of 3.5% $\delta k/k$ is the estimated maximum worth for a control rod in a small core configuration, whereas the results of the accident have been calculated for the full BONUS mockup core consisting of 64 boiler assemblies and 32 loaded superheater assemblies.

(2) The temperature coefficients for the large core are of a lesser magnitude than those of a small core; consequently the calculated power excursion for a given excess reactivity will be greater than the actual one in a smaller core.

(3) The automatic and manual scram circuitry are assumed to be inoperative. Although the scram system is not capable of nullifying a reactivity ramp of this magnitude, it can be shown, by the following discussion, that it is capable of reducing the magnitude of the initial power transient and preventing a recurrence of the transient. The fast scram circuitry will initiate a period and/or power scram signal within 30-100 milliseconds after the scram conditions exist at the detector. This signal will then trip the rods cocked in the Safety Pattern; the height of these rods is set at a level near their maximum differential worth. Thus in less than 0.1 seconds after receiving a rod scram signal the rods begin their accelerated motion into the core. The resulting negative reactivity insertion begins before the peak occurs in the uncontrolled transient which has been calculated.

(4) The rapid increase in intensity of the audible monitor should indicate, to the person involved, the increased likelihood of a nuclear incident. The reaction time of the individual may not be sufficiently fast to prevent his own exposure to some radiation but would probably reduce the net effect of the excursion.

The moderator dump would also be initiated if the scram circuitry responded; however, due to its relatively slow response time, the moderator dump does not contribute to the suppression of this power transient.

A less severe accident would be caused by the simultaneous withdrawal of all 17 control rods due to a failure of the switches and relays in the rod drive interlock circuitry. This would result in the introduction of 0.68% $\Delta k/\text{sec}$, since each individual rod can contribute at most 0.04% $\Delta k/\text{sec}$. This accident is less severe than the Maximum Credible Accident described above by a factor of ~ 9 .

b. Increase of Water Height

The activation of the main pump through a series of interlock and personnel failures during a partial water height experiment (Section VIII A1) could not add water at a rate sufficient to produce a severe accident. It is necessary to postulate that the water level is raised by an object falling into the reactor vessel. Using the value of 1% $\Delta k/\text{in.}$ for the incremental worth of water at the minimum critical water height, the worth of an object (density $> 1 \text{ gm/cm}^3$) is 0.29% $\Delta k/\text{ft}^3$. Thus, an object with a volume of 5.2 ft^3 falling into the water in one second could produce an accident as severe as the Maximum Credible Accident described above. The probability of this occurring as a single event is negligible, since a full complement of cover plates is always installed over the gap between the top of the core and the vessel walls. Unauthorized attempts to enter the cell during any experiments will automatically result in rod and moderator dump scrams. Furthermore, daily inspections prior to operation insure that all cover plates are in place and no unusual objects are present in the vicinity of the core. Finally, it is unlikely that the reactor would ever be operated at so low a water level because the high axial leakage would make the interpretation of the critical data extremely difficult.

c. Uncontrolled Superheater Flooding or Voiding

As was pointed out in Section A, above, leaks can never cause voiding of flooded superheater elements while there is water in the core. If the same pessimistic assumption is made here that flooding the entire superheater is worth +5% Δk , it is seen that 600 tubes must flood by leakage in one second in order to produce an accident as severe as that described for the Maximum Credible Accident due to manual withdrawal of a control rod.

2. Results of the Maximum Credible Accident

a. Nuclear Reaction

The transient analysis employed in describing the power excursion is described in Ref. 1. The results of this analysis are illustrated in Figure 16 and tabulated below.

TABLE 7

Maximum Credible Accident

Input Reactivity ($\delta k/k$ -sec)	Time Duration (sec)	Power Release (Mw-sec)	TEMPERATURE ($^{\circ}$ F)			
			Clad		Fuel	
			Max.	Ave.	Max.	Ave.
6×10^{-2}	0.25	1200	2200	900	4250	1700

The dose received by the personnel in the control room, through the 4 foot concrete shield wall, is under 0.8 rad.

The maximum fuel temperature is below the presently accepted melting point of 5000° whereas the peak clad temperatures are well above the melting point for aluminum (1200° F). Therefore, no fuel would melt but the loss of cladding would cause a physical deterioration of the fuel rods in the regions of power peaking. The amount of fission products released from the fuel rods depends on the possibility of an aluminum-water chemical reaction and on the integrity of the exposed fuel pellets.

b. Chemical Reaction⁽⁴⁾

The aluminum-water reaction is exothermic (3550 cal/gr-Al) and leads to the formation of Al_2O_3 with the evolution of hydrogen gas (~ 1240 cc/gr-Al). This reaction is capable of more violence than those occurring between water and zirconium, stainless steel, magnesium or uranium. Evidence to date indicates a small degree of reaction between molten aluminum and water at temperatures below $\sim 2200^{\circ}$ F when the aluminum is poured or sprayed into the water. However, above 2200° F the rate of reaction is observed to increase; in particular, for the case of molten metal ($\sim 2800^{\circ}$ F) dispersed in water by an explosive charge the reaction in itself is explosive.

The range of clad temperatures encountered during the Maximum Credible Accident is consistent with an interpretation that neglects the contribution of the heat of reaction or evolution of gas to the power excursion. Any heat generated in this case would most likely be transmitted to the water and would not serve to perpetuate the reaction. Thus, the presence of a moderate heat sink in the form of relatively thick aluminum clad and water moderator further reduces the probability of a violent self-perpetuating reaction so that it can be ignored in evaluating the hazards produced by the maximum credible accident to the surrounding area.

c. Formation of Fission Product Cloud

The fraction of fission products formed in the 1200 Mw-sec incident which is released to the surrounding area is thus determined only by:

- (1) Fraction of cladding which has melted;
- (2) Fraction of fuel pellets exposed by clad failure which disintegrate and release fission products; (3)
- (3) Fraction of fission products in water-steam mixture which are expelled from the core into the air;
- (4) Fraction of fission products in the air which escape from the cell due to leakage around doors, through filters, etc., or due to partial or total rupture of the roof.

Due to the extremely low fission product inventory caused by prior operation in this critical facility, one can afford to be extremely conservative in evaluating the product of these fractions. It has therefore been assumed here that 5% of the fission products generated by the 1200 Mw-sec burst form as a cloud above the cell, i.e., an amount equal to 60 Mw-sec. It is estimated that this is conservative by a factor of 100 to 1000 if, indeed, a fission product cloud would even form.

d. Hazards to Surrounding Area

The methods of calculating the dose rates from this 60 Mw-sec fission product cloud are described in numerous reports*.

YAE-31 contains a set of curves giving the results of conservative calculations of the radiation dosage versus distance from the burst under various meteorological conditions and normalized to an initial fission product release equivalent to a 100 Mw-sec burst. This data was used to provide a conservative estimate of the hazards associated with the fission product release during the Maximum Credible Accident for the BONUS Critical Experiments since the basic assumptions inherent to the analysis are also applicable in this case. The basic assumptions are as follows:

- (1) All fission products released in the accident are air borne in a cloud of 20 foot radius above the cell.
- (2) The center of the cloud is assumed to move along the surface of the ground rather than at some elevation as a hot cloud normally would.
- (3) The cloud is assumed to be reflected giving a concentra-

*See for example: AECU-3066 - "Meteorology and Atomic Energy" and also YAE-31, "Yankee Critical Experiments Hazards Summary Report."

tion double that of a normal Gaussian cloud.

- (4) The center of the cloud moves in a straight line from the source through the point where the observer is standing.
- (5) The velocity of the wind is assumed to be high (30 mph) in the external gamma radiation dose calculations and in the calculations of the radiation dose from deposition on the ground. The wind velocity is assumed low (5 mph) in the inhalation radiation dose calculations. In each of the cases the velocity is chosen to give higher radiation doses.

The radiation hazards fall into three separate categories:

- (1) External gamma radiation due to immersion in a fission products cloud.
- (2) External gamma radiation due to deposition from fission products cloud.
- (3) Internal radiation due to inhalation of radioactive isotopes from fission products cloud.

The immersion beta dose was mentioned in YAEC-31 as being an order of magnitude larger than the gamma dose at 2000 feet. However, it was not included in the tabulation of the external radiation hazards due to the reduced range and lower lethal effectiveness associated with beta rays of this energy. Under certain conditions, e.g., fully exposed, serious burns may be caused by the immersion of fallout beta particles; but, in general, the effects would be less dangerous than those resulting from the gamma dosage.

The nearest site boundary is 1200 feet in the westerly direction and 1900 feet in the easterly direction. The nearest habitable off-site dwelling is ~2000 feet from the Critical Facilities. Table 8 summarizes the maximum integrated radiation dose under adverse meteorological conditions (strong inversion) for 1200, 1900, and 5000 feet from the Critical Facilities.

TABLE 8

Summary of Radiation Hazard

Distance from Burst (feet)	Condition	Immersion	Fallout	Inhalation*, Total	
		Total Gamma Dose rad	3 Hr. Gamma Dose rad	Dose to Lungs, rad. (Short-lived Fission Prod.)	
				Beta	Gamma
1200	Inversion	9	9.4	22.5	-20.4
	Mild Lapse	1.7	1.0	9.6	1.1
1900	Inversion	4.5	4.5	15.4	13.9
	Mild Lapse	0.7	0.4	0.36	0.4
5000	Inversion	1.4	1.2	8.9	7.8
	Mild Lapse	0.15	0.082	0.05	0.06

Therefore, a conservative estimate of the maximum body gamma dose that an offsite person could receive is approximately 9 rad. The contribution of the fallout dose is very conservatively estimated in the table. Evacuation procedures would further reduce the actual dose received. In any event the maximum body gamma dose would be less than the commonly used once in a lifetime emergency dose of 25 rad. The higher doses to the lungs are localized and hence higher doses are allowable under these emergency conditions.

Due to the very conservative nature of the calculation of both the accident and its consequences, the results tabulated in Table 8 indicate doses which are much higher than would be encountered under more realistic conditions. In summary, then, one may state that the hazards to the area as a result of the Maximum Credible Accident are:

- (1) not serious even under the most improbable sequence of events which can plausibly be conceived;
- (2) no worse than those approved for other similar or even more densely populated areas;
- (3) no worse than those for which approval has been granted for previous operation of the CEND Critical Facilities.

*The total integrated doses due to inhalation of short lived fission products, under a strong inversion at the nearest site boundary, are Thyroid: (I-131) ~1.7 rad; Bone: (SR-89) ~ 2.8×10^{-2} rad; Bone: (Sr-90) ~ 5.4×10^{-2} rad.

054

IX. FUEL STORAGE AND HANDLING

The fuel rods are stored in canisters fabricated from heavy walled aluminum pipe, 1/4" thick and 4 inches in inside diameter. The floor of the canister is formed by a welded plug; a "drain hole" is drilled through the wall at floor level. The upper end of the canister has two holes to accommodate a lifting tool and two holes for affixing a cover plate with wire seals. Each canister has a maximum capacity of either 40 fuel rods of 3.5 w/o enrichment or 50 fuel rods of 1.85 w/o enrichment.

The fuel element canisters will be stored vertically in a two-slab array on the balcony of the reactor cell (see Figure 5). The south canister storage racks are approximately 19 feet long and provide room for 84 canisters (42 per row) spaced 5 inches on center within each row and 16 inches on center between each row. The north canister storage racks are of similar twin row construction, and are located in line with the south storage racks, but separated from them by ~3 feet. The north racks are approximately 6 feet long and contain 14 canisters per row for a total capacity of 28 canisters.

Transportation of the canisters between the storage racks and the reactor cell floor is provided by an overhead crane. The primary reason for the choice of a two-slab array near the edge of the balcony was the limited crane travel in this region. Fuel security is insured by a padlocked door in a fence isolating the fuel storage region of the balcony, the ultrasonic alarm in the cell, and by the limited access door leading into the reactor cell (see Section IV A3 for access to cell).

This array is essentially identical to that used to store this same fuel in Cell #2 for the NUSU phase B and C experiments. The nuclear safety of the array, both wet and dry, was demonstrated using the calculational methods outlined in Section VII A of Ref. 1. Details are shown in Sections VII B and C of Ref. 1.

REFERENCES

- (1) Harding, R. S., and Slater, M., Advanced Critical Experiment Safeguards Report, CEND 105, Supplement 2, (1960).
- (2) Deutsch, R. W., Method for Analyzing Low-Enrichment, Light-Water Cores, GNEC-133 (1960)
- (3) Private Communication from W. P. Chernock, Metallurgy Section, Materials Development Lab, CEND.
- (4) Brittan, R. O., Reactor Containment, ANL-5948

FIG. 1
MAP OF WINDSOR AREA

057

COMBUSTION ENGINEERING, INC.
SITE PLAN

058

FIG. 2

ISOMETRIC OF C.E.N.D. CRITICAL FACILITIES

FIG. 3

059

FIRST FLOOR PLAN OF C.E.N.D.
CRITICAL FACILITIES

FIG.4

FIG. 5
061

SYMBOL LIST

- GLOBE VALVE
- GATE VALVE
- GLOBE VALVE WITH LOCK SHIELD
- GLOBE VALVE REMOTE OPERATED
- BALL VALVE REMOTE OPERATED
- FLOAT OPERATED VALVE
- Y STRAINER
- PUSH BUTTON SWITCH @ CONSOLE
- PUSH BUTTON SWITCH LOCAL
- FAUCET
- CONDUCTIVITY METER
- PRESSURE GAUGE
- FLOW MEASURING DEVICE
- WATER TRAP
- SOLENOID VALVE
- PUMP

FIG. 7
SAFETY ROD DRIVE PACKAGE
063

ITEM	QTY	DESCRIPTION	MATERIAL	REMARKS
1	1	CONTROL ROD DRIVE MECHANISM ASSEMBLY		
2	1	POSITION INDICATOR MOUNTING BRACKET ASSEMBLY		
3	1	MOTOR MOUNT BRACKET ASSEMBLY		
4	1	ROD PULLEY AND DOWN LIMIT BRACKET ASSEMBLY		
5	1	UP LIMIT BRACKET ASSEMBLY		
6	1	CABLE AND FITTING ASSEMBLY		
7	1	6.0 LOW INERTIA SERVO MOTOR		DEAL WITH SERVO OR EQUAL
8	1	CABLE SPOOL ASSEMBLY		
9	1	BOLT REGULAR HEXAGON	CARBON STEEL	1/2" DIA. X 1 1/2" LONG
10	1	SPRING LOCK WASHER (LIGHT)	CARBON STEEL	1/8" DIA. X .030" THICK
11	1	SPEED REDUCER, 400 TO 1 RATIO		WITH GEAR AT THE INPUT
12	1	V GROOVE PULLEY	Z017 T4 ALUM	4 DIA. X 1 1/2" TH
13	1	HEX SOCKET SETSCREW - FLAT POINT	CARBON STEEL	#4-32 X 1/4" LONG
14	1	V GROOVE PULLEY	BRASS	1/2" DIA. X 1 1/2" LONG
15	1	ADAPTEE	CARBON STEEL	1/2" DIA. X 1 1/2" LONG
16	1	SPRING PIN	CARBON STEEL	1/8" DIA. X 1 1/2" LONG
17	1	ADAPTEE	CARBON STEEL	1/2" DIA. X 1 1/2" LONG
18	1	ROUND BELT ENDLESS	NEOPRENE	1/2" DIA. X 1 1/2" LONG
19	1	SOCKET HEAD CAP SCREW	CARBON STEEL	1/4" DIA. X 1 1/2" LONG
20	1	CHANNEL	CARBON STEEL	1/2" X 1 1/2" X 1/4"
21	1	BOLT REGULAR HEXAGON	CARBON STEEL	1/2" DIA. X 1 1/2" LONG
22	1	NUT REGULAR HEXAGON	CARBON STEEL	1/2" DIA. X 1 1/2" LONG
23	1	ANGLE BAR	CARBON STEEL	1/2" X 1 1/2" X 1/4"
24	1	STOP BOLT	BRASS	1/2" DIA. X 1 1/2" LONG
25	1	BEAM	CARBON STEEL	1/2" X 1 1/2" X 1/4"
26	1	PHOTO METER WITH THERMOCOUPLE		TYPE 100-1000
27	1	FLUORESCENT LIGHT MACHINE SCREW	CARBON STEEL	1/4" DIA. X 1 1/2" LONG
28	1	SPRING LOCK WASHER (LIGHT)	CARBON STEEL	1/8" DIA. X .030" THICK
29	1	CLIP	CARBON STEEL	1/2" DIA. X 1 1/2" LONG
30	1	BOLT REGULAR HEXAGON	CARBON STEEL	1/2" DIA. X 1 1/2" LONG
31	1	SPRING LOCK WASHER (LIGHT)	CARBON STEEL	1/8" DIA. X .030" THICK
32	1	NUT REGULAR HEXAGON	CARBON STEEL	1/2" DIA. X 1 1/2" LONG
33	1	ANGLE BAR	CARBON STEEL	1/2" X 1 1/2" X 1/4"
34	1	SHOULDER SCREW	CARBON STEEL	1/4" DIA. X 1 1/2" LONG
35	1	SPRING LOCK WASHER (LIGHT)	CARBON STEEL	1/8" DIA. X .030" THICK
36	1	HEXAGON MACHINE SCREW NUT	CARBON STEEL	1/4" DIA. X 1 1/2" LONG
37	1	EYE BOLT	CARBON STEEL	1/2" DIA. X 1 1/2" LONG
38	1	MUSIC WIRE	MUSIC WIRE	1/2" DIA. X 1 1/2" LONG
39	1	WASHER MEDIUM	CARBON STEEL	1/2" DIA. X 1 1/2" LONG
40	1	WIRE NUT	WIRE NUT	1/2" DIA. X 1 1/2" LONG
41	1	V GROOVE PULLEY	Z017 T4 ALUM	1 1/2" DIA. X 1 1/2" TH
42	1	V GROOVE PULLEY	Z017 T4 ALUM	1 1/2" DIA. X 1 1/2" TH
43	1	SOCKET HEAD CAP SCREW	CARBON STEEL	1/4" DIA. X 1 1/2" LONG
44	1	SPRING LOCK WASHER (LIGHT)	CARBON STEEL	1/8" DIA. X .030" THICK
45	1	TAPER PIN	CARBON STEEL	1/2" DIA. X 1 1/2" LONG

5-ALL TAPER PINS TO BE LOCATED AFTER ALIGNMENT OF COMPONENTS
 4-USE ITEM NO. 39 FOR ROD SPEED OF 20 INCHES PER MINUTE
 3-USE ITEM NO. 40 FOR ROD SPEED OF 15 INCHES PER MINUTE
 2-USE ITEM NO. 41 FOR ROD SPEED OF 10 INCHES PER MINUTE
 1-USE ITEM NO. 11 FOR ROD SPEED OF 5 INCHES PER MINUTE

FIG. 8
 SHIM ROD DRIVE PACKAGE
 064

ITEM NO.	DESCRIPTION	QUANTITY	MATERIAL
1	WELDED ASSEMBLY, INCL. COIL	1	
2	GRID PLATE	1	
3	GRID PLATE	1	
4	GRID PLATE	1	
5	GRID PLATE	1	
6	GRID PLATE	1	
7	GRID PLATE	1	
8	GRID PLATE	1	
9	GRID PLATE	1	
10	GRID PLATE	1	

FIG. 9-A
VERTICAL SECTION OF BONUS MOCK-UP
065

066

FIG. 9-B
MANUAL VOID SYSTEM

CHANNEL

NOTE:

CHANNEL 'A'-D MAY BE CONNECTED VIA PATCH CORDS
TO SCALERS ON THE CONSOLE AND IN THE COUNTING ROOM.

NUCLEAR INSTRUMENTATION BLOCK DIAGRAM

FIG. 10

NOTE

1. ALL CONTACTS NORMALLY OPEN - I.E. CLOSED FOR NO-SCRAM CONDITION, OPEN ON SCRAM OR POWER FAILURE.
- 2 SLAVE RELAYS DROP RODS, DUMP WATER, ETC, IF DE-ENERGIZED.

ROD DROP AND WATER DUMP SCRAMS

FIG. II

068

BLOCK DIAGRAM OF CONTROL SYSTEM

NOTE: THE HOLD SYMBOL INDICATES THAT ONCE THE PREREQUISITE CONDITIONS FOR PERFORMING A CERTAIN OPERATION HAVE BEEN SATISFIED, THE OPERATION MAY BE PERFORMED SUBSEQUENTLY (EVEN THOUGH SOME OF THE INITIAL CONDITIONS MAY BE VIOLATED) UNTIL POWER IS REMOVED FROM THE LINE.

CROSS SECTION OF BONUS MOCK-UP

FIG. 13

070

STANDARD FUEL ROD ASSEMBLY

FIG. 14

071

REACTOR RESPONSE TO A RAMP REACTIVITY
INPUT OF $6 \times 10^{-4} \delta k/k$ -SEC FOR 16 SEC.

072

REACTOR RESPONSE TO A RAMP REACTIVITY
INPUT OF $6 \times 10^{-2} \delta k/k$ -SEC FOR 0.25 SEC.

073