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SUPPLEMENT NO. 2

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ADVANCED CRITICAL EXPERIMENT
SAFEGUARDS REPORT

SUPPLEMENT NO. 2
(NUSU PHASE B AND C)

MARCH 17, 1960

CONTRACT NO. AT(11-1)-795
U.S. ATOMIC ENERGY COMMISSION

NUCLEAR DIVISION
COMBUSTION ENGINEERING, INC.
WINDSOR, CONNECTICUT

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PENDING PATENT CLEARANCE

CEND-105

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SAFEGUARDS REPORT

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(NUSU PHASE B AND C)

Prepared by:

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March 17, 1960

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NUCLEAR DIVISION
COMBUSTION ENGINEERING, INC.
WINDSOR, CONNECTICUT

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I. INTRODUCTION

On July 2, 1959 Combustion Engineering, Inc. was awarded contract AT(11-1)-795 with the Atomic Energy Commission to conduct a development program on a boiling water reactor incorporating nuclear superheat (NUSU). Approval to operate the Advanced Critical Experiment (ACE) facility and conduct Phase A experiments was obtained from the AEC Reactor Hazards Evaluation Branch in January 1960. These experiments, presently in progress, are primarily for the purpose of calibration and testing of new equipment and for the evaluation of new experimental techniques. The neutron flux requirements for these experiments were met by a core consisting of highly enriched plate type assemblies.

This report is Supplement #2 to CEND-105, "Report on Advanced Critical Experiment Safeguards"; Supplement #1 having been prepared for the Phase A experiments. A description of the site location, meteorological conditions, etc., may be found in the above mentioned original report and also in CEND-SLC-141, "SLC Hazards Summary Report." In the interest of clarity and continuity, a certain amount of descriptive information is repeated in this supplement, especially in the sections on the facility and operating procedures where modifications have been made to carry out effectively and safely the critical experiments in the present program.

The critical experiments for the NUSU program will provide data for a comparison with present day nuclear theory and design techniques as applied to a boiling water reactor with integral nuclear superheat. In order to provide as broad a basis for this comparison as possible, several types of superheater elements will be studied, e.g., steam cooled solid rod, single and double annular rods, etc. Various core geometries will also be investigated; these cores may be characterized by the location of the superheater section, e.g., internal, external, or combined boiling and superheating assemblies. It was felt that the most expeditious means of delineating, as early in the program as possible, those areas in which theory and experiment are in disagreement was to start the Superheat program with fuel elements and geometries of a type for which extensive analytical studies had already been carried out, viz. - the BONUS reactor core. This reactor consists of a central boiling water region fueled with zirconium clad 1.85 w/o enriched UO_2 and surrounded by an external water-moderated superheating region. The superheater elements consist of stainless steel clad 3.5 w/o enriched UO_2 fuel rods; at normal operating temperatures these solid rods would be cooled by steam flowing through a concentric stainless steel tube.

The initial NUSU experiments thus consist of a water moderated core with slightly enriched UO_2 in cylindrical fuel rods, "steam" flow tubes concentric with solid fuel rods, aluminum and stainless steel cladding and dimensions and spacings generally similar to those in BONUS.

The objectives of these Phase B and C experiments of the NUSU program are to study the nuclear properties of cores of this general type, including, however, geometries with internal superheaters. The Phase B experiment has a basic geometry which is "intermediate" between that of an exact mock-up and that of the "clean geometry" Phase C experiment which will follow it.

The basic difference between these two geometries is that the former will include internal water channels, water holes and fixed control elements while the latter will have a uniform pitch slab type geometry. Both phases of the NUSU program will use the same fuel elements with no mock-up of assembly box walls, steam insulating systems, etc. Phase B will concentrate on the over-all characteristics of this type core: loading for criticality, power split between boiler and superheater, flooding and voiding effects in the superheater, etc. Phase C will concentrate on those areas in which discrepancies between theory and experiment become apparent. Detailed parameter type measurements will be made under conditions where theory and experiment can be compared with higher precision and greater economy.

The basic reactor safety problems of the Phase B and C experiments are the same and are discussed below. An additional report will be issued at a later date to describe any differences which may be encountered during later Phases of the NUSU program.

The nuclear statics and kinetics of the assemblies to be investigated under the present program are not dissimilar to the general category of low enrichment UO_2 , light-water moderated cores for which there has been extensive experience to date. Furthermore, the usual nuclear hazards based on uncontrolled reactivity addition are controllable with conventional restrictions in operating procedure, automatic interlocks, scram circuits, and the imposition of a maximum reactivity addition rate, viz. - $0.04\% \Delta k/sec$.

There is however an additional hazard that is peculiar to the present experiment which involves the flooding and voiding of the "steam" annulus around the superheater fuel rods. It is shown in this report that, although flooding or voiding of the complete steam space could be worth $\sim 1\% \Delta k$ in the worst case to be studied, the system which has been designed to control this will limit the reactivity addition rate to $0.04\% \Delta k/sec$ under any set of reasonably plausible conditions.

It is also 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 experiments described herein can be carried out without any 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 operation of the ACE critical facility.

II. REACTOR FACILITY

This section describes the basic mechanical, hydraulic and instrument systems of the ACE critical facility. Specific details on operational parameters, e.g., orifice diameters, rod speeds, etc., for the Phase B and C experiments are discussed in Section VI C.

A. Hydraulic and Mechanical Components

1. Tanks, Pumps and Valves

a. Reactor Vessel

The overall view of the reactor structure is shown in Figure 1. A cylindrical stainless steel tank six feet in diameter and seven feet high and of 1500 gal capacity is located in an opening in the first floor decking. Two platforms eight feet above and below the first floor facilitate handling and maintenance of rod drives, nuclear instrumentation, dump valve, etc.

b. Dump and Storage Tanks

Mounted on the lowest level, twenty-five feet below first floor level, are two 2100 gallon dump and storage tanks with associated circulating pumps and demineralizing and heating units. The water can be demineralized or heated at any time in the operation. The dump tank has a vent sufficiently large to accommodate the overflow should it be inadvertently full at the time of a dump from the reactor.

c. Dump Valve

An 18" diameter pipe, terminated by the dump valve, connects the reactor vessel and dump tank. The dump valve is a circular plate held upward against a gasketed flange on the previously mentioned dump pipe. The necessary force to hold this valve closed is developed by three compressed air actuated cylinder and piston units, so connected that the valve opens on loss of pressure and/or electrical power due to scram or component failure.

The following response times have been measured:

Manual scram to dump valve full open.....	0.25 sec
Manual scram to water level drop 1/16".....	0.45 sec
Manual scram to water level drop 12".....	2.0 sec

d. Pumps and Valves (See Figure 2)

Water is circulated by one of two $7\frac{1}{2}$ hp centrifugal pumps. The maximum flow rate, with all valves wide open and only the flow rate measuring orifice in the line, is approximately 200 gpm. This flow rate is reduced to an absolute maximum of 100 gpm by a padlocked valve in the main flow line; this valve setting is not changed under any circumstances since the upper limit of the flow rate meter is 130 gpm.

The flow rate into the reactor tank is limited, independently of the setting of the padlocked valve, by a calibrated orifice in the fill line. The orifice can be replaced to meet the maximum water addition rates for various cores. A 50 gpm orifice corresponds to a rise of 2.84 inches/minute in an empty tank. The flow rate can be further reduced by a motor-driven throttle valve operated from the control room.

The calibration of the orifice, flow meter, manometer and padlocked valve are checked by observing the rate of rise of water in the empty reactor tank every time the orifice is changed. The calibrations are also checked at regular intervals, correcting for the core volume, by use of an electrical probe for determining the water height.

e. Microfill/drain and Flood/Void Systems

A secondary pumping system, the so-called "microfill/drain" system, and the superheater flood/void system are also illustrated in Figure 2. The use of these systems is described in Sections IV E and F, "Partial Water Height Experiments and Superheater Flooding and Voiding Experiments."

The microfill/drain system consists of a 20 gpm pump, three manually operated and two solenoid operated valves. Power for the pump and solenoid valves 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 is above the core. The 20 gpm pump will permit a maximum increase in water level of 1.15 inches/minute in an empty tank; lower values will be maintained, if required, by the setting of the padlocked valve, A. This pump, by the use of manual valves, B and C, is also used, with a length of plastic tubing, to supply water to test the superheater manifolds for obstructions or leaks. This test requires the dump valve in the main tank to be open and is described in Section IV H.

The superheater flood/void system uses gravity feed to flood each superheater plenum individually from the reactor tank and to drain each superheater plenum to the dump tank. Power for the solenoid valves is interlocked with the microfill/drain and control rod drives as described above. The maximum fill and void rates are determined by orifices in the superheater manifolds, as described in Section VI C.

The solenoid valves in the microfill/drain and superheater flood/void systems are all normally closed valves; the status quo in both systems is thus maintained on loss of power due to scram or component failure.

2. Control Rod Drives

a. Locations and Connections to Rods

The overhead control rod drives used at the facility are shown in Figure 3. The drive packages are supported on two concentric rails and may be positioned at any point over the tank. 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 and coarse rod position is determined by precision helipots geared to the drive shaft after the clutch. Connection of the rack to the control rods is through a turnbuckle. An oil-filled dash pot assembly is incorporated into the rack vertical guide, and is suitable for rods from 50 to 100 pounds in weight.

b. Rod Speed Control

Maximum rod speed is determined by the gear box ratio, which can be changed to meet maximum 5k/sec restrictions. The voltage delivered to the drive motors may be reduced by a variac on the control console, thus further reducing the rod drive speed for fine control.

c. Magnetic Clutch Power

Power for the magnetic clutches is supplied from composite amplifiers (see Section IIB1); the current is adjusted to the lowest value for which slippage of a suspended rod is zero. In the event of a scram signal, this power is removed, thus causing magnetic clutches to disengage and release the rods.

d. Rod Drop Characteristics

Drop times have been measured for a fifty-pound rod falling from an initial withdrawal of twenty-four inches. The results are as follows:

<u>Inches Insertion</u>	<u>Time after Magnet Voltage Cut-Off</u>
1	0.18 sec
6	0.30 sec
12	0.40 sec
18	0.45 sec
22 (contacts dash pot)	0.47 sec

3. Neutron Source

The reactor startup source is a doubly encapsulated Po-Be source and is stored in an 18" diameter x 18" long paraffin and borax filled steel container located on the upper platform among the rod packages. Source strength generally ranges from 20 curies initial to about 1 curie at replacement time. The source capsule is placed in a secondary sealed container attached to a teleflex cable, and is guided into its core location by 5/8" diameter stainless steel tubing. The teleflex 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 trip balls on the cable which actuate limiting microswitches.

A careful check is made when locating the down limit trip ball 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.

An add/subtract revolution counter, actuated from the gear which drives the teleflex cable, gives source position indication to within $\pm 3"$.

B. Nuclear Instrumentation and Scram Circuitry

1. Nuclear Instrumentation

The nuclear instrumentation is shown by the block diagram of Figure 4. 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 excessive power level.

These detectors, contained in sheaths introduced horizontally through the reactor vessel walls or vertically in tubes suspended from above, are located in the vessel at various distances from the core.

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

a. Channel E - linear level scram, adjustable range

- (1) Uncompensated BF₃ ion chamber
- (2) Beckman linear ~~μ~~ammeter
- (3) Matching amplifier
- (4) Composite safety amplifier
- (5) 1500 volt chamber supply

b. Channel F - Period scram

- (1) Uncompensated BF₃ ion chamber
- (2) Log N and period amplifier
- (3) Composite safety amplifier
- (4) 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 any two are required to be operable, are constituted as follows:

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

- (1) BF₃ pulse counter
- (2) Preamplifier
- (3) Linear amplifier
- (4) Log count rate meter
- (5) Period amplifier
- (6) 2500 volt chamber supply

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
- (5) 2500 volt counter supply

The response time of each channel has been measured. 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 loss 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	1100 msec
C	BF ₃ Counter	Log CRM - Period Amp.	Period	1100 msec
E	ULC	Beckman μ ammeter	Level	275-1370 msec depending on range
F	ULC	Log N - Period Amp.	Period	95 msec
G	ULC	Composite Amp.	Level	30 msec

It can be seen that while the low level startup channels (A and C) are relatively slow (i.e., generally about 1 sec delay), a period scram with 0.1 sec delay covers the medium and high level regions and a level scram with only 0.03 sec delay will initiate a scram if the level should reach the 1000 watt (or lower) point before other instruments can respond.

2. Rod Drop Instrument Scrams

A scram by rod drop is effected by any one of the following conditions (see Figure 5):

- a. High level on Channel E.
- b. Low period on Channel F.
- c. High level on Channel G.
- 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. Circulating pumps are turned off, preventing addition of moderator to the reactor vessel.
- b. "Up" power to the rods is cut off, preventing rod withdrawal until the scram is reset.
- c. Microfill/drain system is de-activated preventing the addition of moderator to reactor vessel. Both solenoid valves are normally closed without power.
- d. Non-preselected flood/void solenoid valves are closed maintaining the status quo and preventing further changes in super-heater region.

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 on the single scram signal, which has dropped the rods, 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 5):

- 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, the pre-selected flood/void solenoid valves are closed, maintaining the status quo of all the super-heater manifolds. Power is also removed from any component which was receiving it via a Bypass (see Section IV G).

C. Non-Nuclear Instrumentation

1. Rod Position Instrumentation

Rod position is determined by sensing Helipot in the rod drive package. Coarse position is indicated at the console on individual meters to the nearest 1/2 inch. Fine position is indicated by a sensitive digital voltmeter to an accuracy of ± 0.01 inch for any one rod at a time.

2. Water Flow and Pressure Instrumentation

Gauges in the console give indication of the circulating pump pressure and flow rate and the flow rate through the demineralizer.

3. Water Level Instrumentation

Liquid level in the storage and dump tanks are indicated by means of six-foot sight glasses at the tanks. Coarse indication of the liquid level in the reactor vessel is given by a manometer, graduated in 1/4-inch steps, located in the control room. This instrument requires an air supply for its operation.

The same manometer system is used to determine the water height in each of the superheater manifolds. Solenoid operated valves switch the manometer from one system to another. One of the solenoid valves (the one leading to the reactor tank) is normally open in order to prevent the manometer fluid from backing up in the lines if a power failure occurs.

Water level in the reactor tank is indicated and can be controlled by means of signals originating in a capacitor probe. The instrument controls can be adjusted to regulate at any level from the tank overflow to a level near the bottom of the core. The desired water level, as set by a pointer on the indicating instrument (Fielden) is automatically maintained to within $\pm 1/8$ inch by means of a pneumatically operated valve in a special drain line. The drain line is closed during Partial Water Height experiments. This instrument also operates the "water above core" relay (see Section IV C).

Water height in the reactor tank can also be more accurately determined (± 0.01 in.) by an electrical probe attached to a spare control rod drive package. By operating this package as a "B" rod, no changes or modifications to the control rod interlock circuitry are required. The probe, however, must be fully inserted ("all rods down") before water is admitted to the vessel and can only be moved up when "critical" power is available and only when no reactivity is being added. The probe will also scram on a rod drop signal. These limitations are not serious enough to warrant modifying the rod interlock circuitry since the water level is also indicated on other instruments.

The probe is insulated from the rod drive rack and is operated 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 voltmeter. The uncertainty in water height measurements will be reduced by always approaching the surface from the same direction, preferably from above.

4. Annunciators

The annunciator system, which consists of a horn and a panel of lights, announces any off-normal conditions. A flashing red light plus a green one indicates that the condition has returned to normal. The following conditions are annunciated:

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

III. REACTOR CORE

A. Core Support and Grid Plates

Four jackscrews are used to suspend the lower support structure from the top flange of the ACE tank (see Figure 6). The lower support structure is fabricated from $5/8"$ x $6"$ stainless steel plates and extends across the ACE tank on a $9"$ square mesh. The bottom grid plates rest on this support structure and bear the entire weight of the fuel. An upper support frame is also attached to the jackscrews; this frame supports only the weight of the top plates. The bottom and top grid plates are fabricated from $1"$ thick aluminum. Provisions for adding intermediate frames and lucite grids were made in the design in the event that bowing of the fuel rods was encountered. To attain maximum flexibility for the various two-region cores to be studied, the grid plates, which span the tank, are made in sections. These sections are either $4"$ or $8"$ wide and are 62 inches long for the central region of the tank. In addition to the $1/2"$ diameter fuel rod holes on a $5/8"$ square pitch, the plates have additional holes of $\sim 1/4"$ diameter for water displacers, fixed poison elements, or foil holders (see Figure 7).

Six bottom grid plates will be available; two $4"$ wide plates having six bayonet sockets across the narrow dimension and four $8"$ wide plates having 14 bayonet sockets across the narrow dimension. All plates will have 77 sockets at a constant pitch ($5/8"$) along the other direction ($\sim 48"$). The $8"$ plates will permit the assembly of uniform pitch cores in both directions whereas the $4"$ plates will permit water channels and provide the flexibility required for various core configurations.

Eight $4"$ wide top grid plates having 6 holes in the narrow direction and 77 holes in the long direction will be available. This set of plates is designed to provide water channels for fixed or movable control elements. For the Clean Geometry Experiments the edges of the $4"$ upper grid plates will be machined to allow the assembly of uniform pitch boiler regions.

The grid plates are accurately positioned by means of replaceable dowel bars; the dowel bars are, in turn, positioned on the permanent frames by means of dowel pins (see Figures 8 and 9). Any combination of grid plates of different widths can be accommodated by the expediency of changing the dowel bars.

The present design will take cores up to 4 feet high and provide a $6"$ water reflector above the top plug of the longest fuel rod used. It will be possible to accommodate slabs 42 inches wide and up to 41 inches deep, slabs 47 inches wide and up to 38 inches deep, and slabs 54 inches wide and up to 34 inches deep - all with a minimum water reflector of 6 inches.

In order to considerably reduce the possibility of an object falling into the tank and raising the water level while the core is critical during a Partial Water Height experiment, grid plates will always be installed over the entire area of the upper frame, regardless of the size of the core being studied.

B. Superheater Tubes and Manifolds

The design of a typical single superheater manifold is illustrated in Figure 10. The tube spacing is shown in Figure 7. Four such manifolds will be available for the Intermediate Geometry (Phase B) Experiment. Each manifold consists of 64 stainless steel superheater "steam tubes" soldered into a stainless steel plenum. A soft "silver" solder is used to permit re-use of the "steam" tubes if other superheater geometries are studied. This solder has a melting point of 430°F and a shear strength of ~15,000 psi. The manifold serves two purposes: it permits the positioning of each section via the dowel bars and pins, and it contains a drain or fill plenum for the purpose of remotely controlling the water level in the "steam space." At the bottom of the plenum are bayonet receptacles for positioning the lower end of the superheater fuel elements; the upper end of each fuel element is centered in the tube by means of a plastic spacer element.

The requirement for flooding and voiding in the superheating region necessitated the fabrication of superheater manifolds which span the tank and thus do not require grid plates. The positioning of the superheater manifolds is also accomplished by means of dowel bars and pins.

The superheater plumbing diagram was described in Section II A 1 (see also Figure 2). The fill and void rate is determined by an orifice in the plenum exit. Present design is based on a 30 second time interval for the water level to move past the active four foot length of fuel in the superheater. Before the core is loaded, an experimental determination of this interval will be made with corresponding modifications in the orifice, if warranted. This system will allow the "steam space" in the superheater region to be flooded or voided by remote control from the reactor console, thus ensuring safety, increasing precision and reducing the time required for experiments.

The steam tubes extend above the top edge of the reactor vessel. This will prevent the accidental flooding of the steam space in the event that the vessel overflow drain line is not functioning properly.

C. Fuel Elements

Figure 11 illustrates the type of standard fuel rod construction for Phase B and C experiments. The fuel consists of high density (~ 10.25 gm/cc) sintered uranium dioxide. The bayonet at the lower end of the rod and the 0.005 inch radial ridge near the top of the rod permit the accurate positioning in the fuel grid plates. Table 1 lists the pertinent data on the three types of fuel pellets and their respective cladding.

Figure 12 shows the type of construction employed in the "experimental" fuel rods which will be used in limited numbers for these experiments. The access windows and inter-window slots in the fuel clad permit the replacing of fuel pellets, bare and cadmium covered foils, etc., with minimum contamination hazard and fuel pellet damage. Water tight seals on the access windows and slots are achieved by the use of one mil mylar tape. For the boiler region present plans call for use of the experimental rods along with standard rods in a module; the module will contain a total of approximately 4 - 9 rods. Special top grid plates will be available to accept the module. In the superheater region the experimental rods may be used in any one of the pressure tubes. The rods will be assembled and disassembled in a large vented box. This box has interior lighting and sloping lucite side walls with hand holes mounted in sliding panels. The box is vented through absolute filters to the exterior of the building at a rate of approximately 750 cfm.

TABLE 1

Fuel Element Characteristics*PELLETS (10.25 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	3.50	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	304 Stainless Steel	0.542	0.506

FUEL RODS

<u>Type</u>	<u>Active Length</u> (inches)	<u>Total 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	62 $\frac{1}{4}$	1.254	7.89	175	0.22	1.38
II	48	60 $\frac{1}{4}$	1.254	20.45	2112	2.65	43.18
III	48	61 $\frac{1}{4}$	1.583	48.84	389	0.62	19.00

*All dimensions and quantities are nominal.

**Includes pellets for 11, 33 and 22 experimental rods.

D. Control Rods

1. Intermediate Experiment

Four types of control rods will be available for use in the various core configurations:

a. Movable Blade Type - Nine movable blade type rods will be available. These rods contain a sheet of 21 mil cadmium (8" x 44") sandwiched between two sheets of 1/8" thick stainless steel. The active length of the rod is 4" less than the 48" active fuel length. The lower 4" is used as a follower to guide the rod when it is withdrawn to the uppermost position. This will permit removal of the poison from the upper reflector for certain types of measurements. Provisions are made to allow addition of cadmium to the lower 4" of the rod, if necessary. Two (2 w/o) boron-steel blades, 1/4" thick by 12" wide, will also be available.

b. Fixed Control Elements - Three control slabs having dimensions of 36" x 65" x 1/8" are fabricated from 2 w/o boron steel. Nine cruciform type rods, having a span of 7 inches, will then be fabricated out of the control slab material. The cruciform rods and control slabs will not be connected to the control rod drives during Phase B and C experiments. The control slabs will be rigidly and securely mounted by spacer bars and clamps in the water channels; the cruciform type will be mounted between the upper and lower grid plates and held in position by pins simulating the ends of the fuel rods.

The control slabs will be positioned by the overhead crane since they will weigh approximately 83 pounds, exclusive of spacers and clamps. The cruciforms will be positioned by hand (weight: ~32 lbs) during the assembly of the core. These rods will be immovable unless grid plates and fuel are removed.

2. Clean Experiments

Present design plans call for up to 9 blade type rods consisting of a 10 mil thick layer of cadmium plated on both sides of 1/16" stainless steel. These rods will, it is expected, be operable in the 1/8" space between rows of fuel elements. One rod is being fabricated for proof testing; these rods will be used only on demonstration that they are completely reliable and foolproof in operation. In the event that the thicker cadmium sandwich, blade type rods must be used, it will be necessary to increase somewhat the size of the water channels in the "clean" slab cores.

IV. OPERATING PROCEDURES

This chapter summarizes the operating procedures for the Intermediate and Clean Geometry experiments. Certain modifications in past operating procedures were made to facilitate the use of experimental techniques to be employed in NUSU experiments. In particular, procedures adopted for partial water height experiments, the approach to criticality following core modifications and the flooding/voiding and testing of superheater regions 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 will include, but not be limited to, the following items in its review of each experiment proposed:

- (1) the proposed core loading, number 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 the Reactor Hazards Branch of 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 ACE 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 rate, new methods of adding reactivity), the matter will be submitted to the Reactor Hazards Branch of the U. S. AEC.

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 qualified 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 he is not in the building.

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 (see Section IV G).

e. The Access Door Interlock Bypass key will be released by the Crew Chief only on prior approval by the ACE Facility Supervisor for each particular use (see Section IV G).

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

g. The doors to the control console and cubicle are normally kept locked; the key is controlled by the Crew Chief and is released only for making authorized changes or repairs in the instrumentation and control circuitry.

B. Control Rod Circuits

1. Rod Selection

In some critical facilities safety rods have to be fully withdrawn ("cocked") before water can be added to the core. The purpose of these safety rods is to provide a rapid shutdown mechanism if the reactor, by some improbable sequence of events, became supercritical while the water was rising during the initial fill. This system 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 became stuck.

The method used in the ACE facility, where all rods are dropped on a scram signal, requires that some number of rods (generally more than two) be partially withdrawn to a preset "Safety Point" while the water is being raised. This will produce a larger rate of decrease of reactivity in the period immediately after receipt of a scram signal (the time when it is most needed) since more rods are starting at a point where the differential rod worth is much higher than at the top of the core.

The number of rods and the location of the Safety Point limit switches in the rod drive packages is adjusted so that the total rod worth available while water is being raised is both adequate and equal to or greater than that which would be attained under the system of two fully cocked rods. The number of rods required and the location of the Safety Point is 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. As an example, consider a typical core to be critical when all of its four rods are almost fully withdrawn. The alternate system would require two of these rods to be fully withdrawn as safeties while raising the water level. The ACE system requires all four to be withdrawn approximately one half of their total travel, achieving about the same shutdown margin after the water is up (see also Section VI B). Thus a much greater initial reactivity insertion rate is available if the reactor, by some very improbable sequence of events, became critical while the water was being raised on initial fill.

The rods that are cocked by partial withdrawal to the preset Safety Point are designated as "A" rods. The region below the Safety Point is called the "Safety Zone"; that above, the "Control Zone."

For experimental purposes, certain additional movable rods, designated as "B" rods, may also be used. The condition required for initially raising the water in the tank, namely, "A" rods at Safety Point, "B" rods down, is designated as "Rods in Safety Pattern." This interlock also requires that there be a minimum of two "A" rods. Additional interlocks prevent the withdrawal of the "B" rods until the conditions for withdrawal of the "A" rods beyond the safety point are met (e.g., water above core).

Selector switches inside the console determine whether a rod drive mechanism operates under "A" or "B" control. In any event, both "A" and "B" rods are dropped on receipt of a scram signal.

2. Restrictions

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

- a. Rods may be withdrawn, one at a time, by spring return switches.
- b. Any number of control 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.
- c. Any of the rods can be selected as type "A" or type "B" rods subject to the restriction that at least two must be selected as type "A" rods. It is to be noted that all rods scram identically whether selected to operate under type "A" or type "B" control.

3. Rod Drop Switches

Switches located on the cubicle in the control room permit manual interruption of current to the electromagnetic clutches on the individual rod drive packages. These switches may thus be used for dropping any selected rods at any time without scramming the reactor. An additional circuit prevents use of the access door bypass until all the switches are closed, providing power to all the clutches, inhibiting manual raising of the rods. Of course, clutch power to all rods is independently turned off in the event of a scram signal.

C. Interlock System

A block diagram of the interlock control system is shown in Figure 13. 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 startup 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. A keyswitch operated interlock bypass system is used, as described in Section IV G, to provide power for testing individual components in a safe manner.

1. Startup Sequence

a. Power to drive the source and rods down and energize the preselected flood/void valves 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 "A" 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 "A" rods have been withdrawn to the safety point,
- (2) if the source is still in place,
- (3) 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 ("P-K") valve is normally opened at this time.

g. The circulating pump may now be turned on:

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

The pump start switch normally acts as a spring return pushbutton 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 D) this switch can be converted to an On/Off two-position system by use of keyswitch operated by the Crew Chief.

h. The circulating 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 control rods are withdrawn from the Safety Pattern until the reactor is brought critical (see Section IV D).

2. Post Critical Operation

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

(1) the main pumps, in conjunction with the Fielden regulated drain and/or the tank 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 both "A" and "B" rods, both above and below the Safety point,

(b) or superheater flood/void manual control (see Section IV F),

(c) or microfill/drain control (see Section IV E),

(3) 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.

b. Note that only one means of adding reactivity - control rod withdrawal or superheater flooding or voiding or microfilling or the circulating pump is available at any one time.

D. 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 shall be 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 all control rods in safety pattern and then with all control rods out.
- 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 (14 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 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 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 elements 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 quasi-continuous 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 sequence of rod withdrawal will, in general be such that the rods will be pulled individually to the expected critical point, beginning with the most valuable rod, and thus approaching criticality with the weakest rod.

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 quasi-continuous rather than stepwise; l/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.

E. Partial Water Height Experiments

1. Method (See Figure 2)

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% (see Section VI B) 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 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" line. The decrease in reactivity is compensated by withdrawing control rods until the reactor is critical with all rods out of the water. The "microfill" pump is used for final adjustment of level and to measure the incremental water worth under these conditions.

2. Interlocks (See Figure 13)

- a. After "critical" power is obtained as in Section IV C1, the microfill/drain solenoid valves and pump 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 and superheater flood/void switches, thus ensuring that only one means of adding reactivity is possible at a given time. The microdrain operation can be carried out simultaneously with control rod withdrawal and the switch is not spring returned from this position since draining will never add reactivity.
- b. Once the water is "microdrained" below the top of the core:
 - (1) the circulating pump is automatically turned off (if, indeed, the water level can be lowered with the microdrain-line while the pump is pouring water into the tank);
 - (2) "critical" power for control rod withdrawal and flood/void operation as well as for the microfill/drain system would be lost immediately if the main flow control valve were opened or if the main pump were energized;
 - (3) the circulating 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.

F. Superheater Flood/Void Experiments

1. Method (See Figures 2 and 19)

In general, one of the solenoid valves on each superheater section will be locked open; this is done to reduce the possibility of reactivity changes due to small leaks. The initial locking of the valves in the desired configuration (e.g., plenums 1 and 2 flooded, 3 and 4 voided) is carried out prior to raising water in the reactor vessel. It is controlled by four "Preselector" key switches, one for each section; these key switches are operated only by the Crew Chief. After criticality is reached, the Crew Chief places a single key-switch in its "unlocked" position and the "Selector" is switched to the corresponding plenum; operation of the flood and void solenoids of the selected plenum can now be controlled manually by the operator. The reactor is maintained critical by compensating with rod motion or the microfill/drain system. After the plenum which is being controlled manually has been completely voided or flooded, the Crew Chief operates the appropriate Preselector key-switch to lock this plenum in its new state.

The status of the Preselector switches is indicated by lights above them and by the Annunciator which indicates an "abnormal" condition whenever any of the Preselector switches is in its unlocked position - i.e., when control of any of the sections is in the hands of the operator. Prior to using the Preselector key-switch the Crew Chief will make an announcement of his intentions and receive verification from the cognizant physicist.

After the change of state of each superheater manifold has been completed and its reactivity effect evaluated by both the Crew Chief and the cognizant Physicist, they will verify that the change of state of the superheater has not violated the following operating criteria:

- a. $k_{eff} \leq 0.98$ for rods in Safety Pattern, water above core.
- b. $k_{eff} \leq 0.96$ for rods down, water above core.

2. Interlocks

a. Power for operation of the flood/void solenoids is available after the water has been raised above the core. Initial criticality will always be achieved, however, by control rod withdrawal. Only one superheater section will be unlocked at a time; even if more than one were unlocked, only one may be operated at a time because of the rotary plenum selector switch.

b. The "Manual" flood/void toggle switch used by the operator to control the unlocked plenum is spring returned to its "off" position from both the flood and void positions; no assumption has been made as to the sign of the superheater flooding coefficient of reactivity. This switch is connected in series with the rod withdrawal and microfill switches so that only one method is available at a given time for adding reactivity.

G. Bypasses (See also Figure 13)

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, flood/void 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, flood/void and microfill/drain 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 three systems - the Clutch Power bypass key is required to actually drive the rods up or down, the Partial Water Height key is required to actually operate the microfill/drain solenoids and pump and the Preselector key is required to change the state of the superheater plenums.

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 coefficient 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 flood/void system, the microfill/drain system and from the main pump.
- (2) The control rods must be in the Safety Pattern, insuring that the reactor is shut down by at least $2\% \Delta k/k$.
- (3) The energized clutches required to maintain the "A" rods at the Safety Point also insure that these rods cannot be manually withdrawn; it would require a force of over 200 pounds to lift a rod against the restraint imposed by the energized clutch.
- (4) In order to insure that the clutches on the "B" rods are similarly energized, the rod drop switches must also be closed; otherwise these rods might only be resting on their down limit stops.
- (5) The source must be in, insuring that the nuclear instrument system and the audible monitor are on scale.
- (6) 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.
- (7) 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.

H. Superheater Manifolds Leak Testing

Initial testing of the superheater system for leaks, valve operation, proper orifice size, etc., will be carried out with natural uranium fuel rods in the superheater and no fuel in the remainder of the core. Subsequent testing may be carried out in a loaded dry core, using the micro-fill pump to supply the water, as follows:

1. The test must be authorized by the Facility Supervisor.
2. Partial Water Height and Superheater Flood/Void key-switches will be operated only by the Crew Chief.
3. Only one plenum may be tested at a time; the other three must be locked on void at all times.
4. The test requires the use of the "Critical" Power Bypass key, insuring that the dump valve is open, the circulating pump is off and the flow control valve is closed.
5. Control rods must be fully inserted into the core and are not to be moved during this test.
6. A plastic hose is connected to valve B (Figure 2); valve C is closed to prevent the water from going into the reactor tank via the microdrain line. The hose is then run to the first floor and inserted into the top of one of the "steam" tubes in the superheater manifold being tested.
7. The "Preselector" key switch for the plenum being tested is unlocked.
8. The microfill pump is energized, via the "Critical" Power Bypass key, the Partial Water Height key and the spring return microfill key. The filling of the section is observed visually and with the manometer.
9. Communication between cell and control room is maintained via the intercom and/or sound-powered phones and visually through the shielding window.
10. After testing for leaks, calibrating manometer, etc., the flood/void solenoids may be operated via the flood/void toggle switch and/or the "Preselector" key-switch. The "flood" solenoid may be tested by observing that the manifold drains and that the water trickles into the reactor vessel via the bottom port plate and then out through the dump valve (or microdrain line).
11. The section being tested must be observed to drain and be left locked in the voided position.
12. After the test is completed all valves and switches must be returned to their normal positions and be inspected by the Crew Chief.

V. EXPERIMENTAL PROGRAM

A. General Objectives

The critical experiments for the NUSU program will provide data for comparison with present nuclear theory and design techniques for a boiling water reactor with integral nuclear superheat. In order to provide as broad a basis for this comparison as possible, several types of superheater elements will be studied, e.g., steam cooled solid rod, single annular, double annular, etc. Various core geometries as characterized by the location of the superheater sections, i.e., internal, external or combined boiling and superheating, will be investigated. It was felt that the most expeditious means of delineating, as early as possible in the program, those areas in which theory and experiment are in disagreement was to start the program with fuel elements and geometries of a type for which extensive analytical studies had already been carried out, viz., the BONUS reactor core. This reactor consists of a central boiling region containing zirconium clad 1.85 w/o enriched UO_2 fuel rods surrounded by an external superheater. The superheater section is water moderated and contains stainless steel clad 3.5 w/o enriched UO_2 fuel rods; under normal operating conditions these fuel rods are cooled by steam flowing through concentric stainless steel flow tubes.

The initial NUSU experiments thus consist of a water moderated core with slightly enriched UO_2 in cylindrical fuel rods, steam flow tubes concentric with fuel rods, aluminum and stainless steel cladding, and dimensions and spacings generally quite similar to BONUS.

The objectives of the Phase B and C experiments of the NUSU program is to make a relatively brief survey of the nuclear properties of cores of this general type, including however, geometries with internal superheaters.

B. "Intermediate" Experiment (Phase B)

There are four general areas to be evaluated: the nuclear properties of a boiler without control rods*; the nuclear properties of the boiler and superheater without control rods; the nuclear properties of the boiler with control rods; and the nuclear properties of the boiler and superheater with control rods. The experiments are performed in slab type geometry. A constant boiler and superheater lattice spacing is to be used in all experiments unless unexpected discrepancies develop between theory and experiment; in this case a change of program may be effected. Figure 7 shows a detailed section of both the boiler and superheat regions. The shaded circles represent fuel rod locations in the intermediate experiment. The additional 1/4 inch diameter holes are provided for aluminum water-displacer rods.

Initially, criticality will be achieved for each core studied by withdrawal of control rods to a banked configuration. Using the partial

*This classification refers to a core with all movable (shim) control rods withdrawn.

water height technique, the movable control rods will then be completely withdrawn from the core by using reduced water height to compensate for the reactivity addition. The number of movable control rods for any core configuration will be such that the core is at least 2% subcritical with rods in the Safety Pattern and water above the core.

Figure 14 illustrates the core configurations of interest for the Intermediate Experiment. Movable rods are shown for illustrative purposes only. The actual number and position of these rods will be determined by rod worth calculations, supplemented by experimental data when possible, prior to assembly of each core. A discussion of these cores follows:

1. Boiler with No Control Rods (See Figure 14 A)

The core consists of three slabs of eight 32 rod boiler assemblies (6 x 6 with four center rods absent). The approximate cross section dimensions of the fuel slab are 13.1 by 35 inches. The critical water height with all movable rods withdrawn is calculated to be approximately 27 inches above the bottom of the fuel.

Aluminum rod water-displacers will be used to simulate voids. These simulated voids will reduce the reactivity by approximately 1.5% δk ; the critical water height will then be at approximately 32 inches. Criticality will also be measured after the four additional fuel rods are added to each assembly. The additional fuel is worth 0.65% δk . This will result in a critical height of about 25 inches.

2. Boiler and Superheater with No Control Rods (Figure 14, B-1, 2, 3)

The entire superheater region, similar to that on one side of the BONUS reactor, consists of an 3 x 32 array of fuel rods. The approximate cross sectional dimensions are 8.8 in. x 35 in. These experiments are to be performed first with one row of 32-rod boiler fuel assemblies on each side of the superheater (internal superheater), and then with the superheater region divided such that one-half is on each side of the two rows of boiler elements (split external superheater), and finally with the two rows of boiler elements adjacent to the full superheater region. The boiler and superheater together comprise a fuel slab whose cross sectional area is 17.5 in. x 35 in.

a. For the central superheater configuration, the critical water height is 38 in. with the superheater flooded; and 32 in. with the superheater voided. The change in reactivity introduced by removing water is 1.2% δk . This is the largest voiding effect which will be studied; if any of the flooding effects are positive they will be much smaller than this.

b. For the split superheater, the critical water height is 23 inches for both the flooded and voided superheater. There is essentially no change in reactivity upon removing water from the superheater.

c. No calculations have been performed as yet for the entire superheater on one side of the boiler. It is expected that the reactivity change upon water removal would be somewhere in between the above two cases or about $0.6\% \delta k$.

3. Boiler with Fixed Control Slabs (Figure 14, C-1, 2, 3) and
Boiler and Superheater with Fixed Control Slabs and Cruci-
forms (D-1, 2, 3, 4; E)

Complete calculations have not yet been performed for these cases. It is estimated for the small core configuration that a movable blade control rod at the center of the core could be worth as much as $3.5\% \delta k$; analysis is continuing for these cases. Prior to performance of these experiments, detailed data of rod worths and reactivity changes associated with the experiments will be available to make sure that maximum reactivity insertion limits are not exceeded.

The fixed control slabs illustrated in Figures 14-C, D extend the full width of the core (see Section III D). Figure 14-D shows the two control rod configurations to be studied for both configurations of the superheater region. Figure 14-E is similar to the BONUS reactor except that it contains only one quarter of the full BONUS superheater.

Typical measurements which will be performed during the course of the Intermediate experiment are as follows:

- a. Rod worths
- b. Radial and axial bucklings
- c. Differential axial worth of water in boiler
- d. Worth of water in the steam space of superheater tube
- e. "Void" simulation with aluminum water-displacer rods
- f. Temperature coefficients
7. Migration area

C. Clean Geometry (Phase C)

This experiment will concentrate on precise parameter or cell measurements (i.e., p, ζ, f) in order to shed light on those areas in which discrepancies between experiment and theory become apparent. The same general types of cores as in the Intermediate Geometry experiment, i.e., internal and external superheater regions, will be studied. The basic difference between this experiment and its predecessor is the use of a uniform pitch in both the boiler and superheater regions with the absence of water channels and holes. This will facilitate the analysis of the experiments. The experiment will use the same fuel rods, steam tubes and lower grid plates as in the Intermediate Experiment; the upper grid plates will require some slots for the shim and safety rods. The nuclear properties of these assemblies will be very similar to those described for the Intermediate Experiments.

D. Reactivity Coefficients

The analysis utilizes the three-group, two-dimensional IBM-704 PDQ code. The method used to determine the three-group constants, reactivity, reactivity coefficients, and control rod worth (as well as a comparison of these methods with cold critical experiments) is found in APPENDIX A.

The Physics analysis has so far been made for only a limited number of the intermediate core configurations (see previous section). Additional information may be derived from the physics analysis of the BONUS core*; some of the pertinent nuclear characteristics of this core are contained in Table 2.

TABLE 2

Nuclear Characteristics of the BONUS Core at 68°F

Reactivity (k_{eff})

All rods withdrawn, superheater flooded	1.157
All rods withdrawn, superheater voided	1.166

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

Total Reactor Reactivity Coefficients (Calculated for No Control Rods in Core)

Void Coefficient (δk /void)	-1.40×10^{-4}
Moderator temperature coefficient (δk /°F)	-0.25×10^{-4}
Doppler coefficient (δk /°F)	-0.10×10^{-4}

*BONUS Power Station, Preliminary Hazards Report. PRWRA-GNEC 2.

**BONUS slab rods - 2, 12" wide, 1/8" thick 2% Boron Stainless steel shim rods between boiler and each of the peripheral superheater groups.

A summary of reactivity coefficients for the NUSU cores is as follows:

1. Control Rod Withdrawal

The maximum worth of any rod for the configurations considered is estimated to be no more than $3.5\% \delta k$. If the maximum reactivity rate permitted is $4 \times 10^{-4} \delta k/\text{sec}$, the maximum rod speed can be as high as 16 in/min, since no more than one rod is withdrawn at a time (see Section IV B). If it were desired that all rods move simultaneously the rod speeds would be reduced accordingly. However, this problem is academic since the present ACE rod drive circuitry will not permit the simultaneous withdrawal of more than one control rod.

2. Water Level Control

For all experiments it is estimated that criticality cannot occur at water levels below 18 in. with all movable rods withdrawn. At 18 inches, the reactivity addition rate is calculated to be $100 \times 10^{-4} \delta k/\text{in.}$ Therefore, if the maximum reactivity insertion rate is to be limited to $\leq 4 \times 10^{-4} \delta k/\text{sec}$, the water addition rate should be $\leq 2.4 \text{ in/min}$ or $\leq 36 \text{ gal/min}$ in the reactor vessel with a full core. The maximum flow rate from the microfill pumping system is 20 gal/min (see Section VI C1).

3. Flooding and Voiding of the Superheater Steam Tubes

For the various core configurations to be studied, it has been estimated that the flooding or voiding of all coolant channels in the full superheater* controls no more than $1.2\% \delta k/k$. Since it is planned to flood or void only one superheater manifold at a time, the reactivity change is no more than $0.6\% \delta k$. Furthermore, since the orifice in the manifold (see Figure 10) prevents the covering or uncovering of the fuel in less than 30 seconds (Section VI C1), the maximum reactivity change is $4 \times 10^{-4} \delta k/\text{sec}$. These latter statements assume that in the course of covering the entire fuel length, the maximum reactivity rate is twice the average and that a single manifold may have twice the worth of the average.

*The term full superheater here refers to four manifold sections.

VI. ACCIDENTS AND HAZARDS

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 insure 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 co-efficient 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. 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 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., "A" rods at the Safety Point and "B" rods fully inserted.

c. $\delta k/sec \leq 0.04\% \delta k/sec$ for all reactivity addition rates (i.e., waterheight for criticality, rod withdrawal and superheater flooding and/or voiding).

2. Criticality will normally be obtained by withdrawing rods and not by raising water (see also Section IV E "Partial Water Height Experiments").

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. A minimum of two type "A" 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, or
- d. flooding of a single superheater manifold, or
- e. voiding of a single superheater manifold.

C. 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 initial 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 ACE 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 $0.04\% \beta/\text{sec}$. 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.

1. Causes

The three possible examples of an excursion of this type are cited: (i) uncontrolled rod withdrawal, (ii) uncontrolled water addition to the reactor vessel and (iii) uncontrolled water level change in the superheater.

a. Uncontrolled Rod Withdrawal

The failure of a rod drive control switch or relay could conceivably result in the uncontrolled withdrawal of a single control rod; this would normally lead to an automatic or manual scram. If, in addition, the crew and the scram circuitry both failed to take corrective action, the resulting power excursion would be terminated by the negative temperature coefficients of the reactor. This chain of events is considered to be highly improbable due to the succession of improbable events and circuit failures required to start it, to inactivate the scram circuitry, and to prevent a manual reactor scram. The uncontrolled with-

*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 II B and a ramp reactivity input of $0.04\% \beta/\text{sec}$. No matter what the initial power level, each of the scram circuits will trip when or well before the reactor reaches 10 kw.

drawal of two or more rods simultaneously is felt to be so highly improbable that it can be neglected here.

The maximum rod speed for each core configuration is pre-set by a suitable choice of gears (see Section II A2) on the basis of calculated and/or measured rod worths so as not to exceed the maximum allowable rate of reactivity addition, namely $0.04\% \delta k/\text{sec}$. For the $3.5\% \delta k$ maximum rod worth in the Phase B and C NUSU cores, and allowing a factor of 2 for peak/average differential worth, this corresponds to a withdrawal rate of 16 inches/minute for a 48 inch rod.

b. Uncontrolled Water Addition in the Reactor Vessel

The maximum flow rate for the main pumping system will be set at 50 gpm by a calibrated orifice in the vessel feed line. This flow rate limits the reactivity insertion by water addition in an unrodded core to a safe value under any plausible conditions. The maximum flow rate for the microfill pump system is determined by a padlocked valve in the vessel feed line and will be set at 20 gpm. The microfill pump extends the useful control range of the water fill system to values lower than those normally obtainable with the main pump; it also sets a definite upper limit to the water (reactivity) addition rate during partial water height experiments. The conditions for operation of the two pumps are mutually exclusive as discussed in Section IV D and only the microfill pump can be used during a partial water height experiment.

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 $100 \times 10^{-4} \delta k/\text{in}$. The only pumping system normally operative under reduced water height and rods above the Safety Pattern is the microfill pump which, at the maximum allowable pumping rate (20 gpm), can provide a maximum reactivity insertion rate of $0.022\% \delta k/\text{sec}$. If, after a succession of still more highly improbable events, the main pumping system is rendered operative under the above conditions the maximum reactivity insertion rate from it would be only $0.054\% \delta k/\text{sec}$. During the initial filling of the reactor vessel with the main pump at 50 gpm, the reactivity insertion rate would be less than $0.04\% \delta k/\text{sec}$ if the core were to become critical at any height above 21". The effect of the faster rate of rise of water in the tank due to the volume displaced by the fuel has been included in all of these calculations by assuming that all of the available NUSU fuel is inserted in the core.

c. Uncontrolled Superheater Flooding and Voiding

An accidental flooding or voiding of a superheater manifold and steam space could also lead to an uncontrolled reactivity insertion. The steam tubes extend above the top of the reactor vessel; therefore an accidental flooding cannot occur due to excessive water height in the vessel even if the overflow and the regulated drain line both failed at the same time.

A plausible accident can be described in terms of a break in the water line leading to a single superheater manifold. The rate of flooding or voiding will depend on the initial conditions within the superheater and the location of the break. In any event, the change of state will be confined to a single superheater manifold (see Figure 2).

The present manifold orifice (Figure 11) requires a minimum of 30 seconds for the water level to move a distance equal to the active fuel length (48 inches) in the superheater. This lower limit is independent of the location of the break in the plumbing.

The most severe flood/void reactivity change is calculated for the core configuration employing an internal superheater; the worth of water in the steam space is $1.2\% \delta k$ for all four superheater manifolds. If the reactivity addition on flooding or voiding of any single superheater manifold is conservatively assumed to be twice the average worth, flooding or voiding a single section could only introduce a maximum of $0.6\% \delta k$ - less than that required for prompt criticality. If the maximum incremental worth of water in the steam space is assumed to be twice the average value for a single section, then the net effect for flooding or voiding 48 inches of steam space in 30 seconds is $0.04\% \delta k/\text{sec}$ per manifold. The corresponding maximum value for an external superheater manifold is $\sim 0.008\% \delta k/\text{sec}$.

It should be emphasized at this point in the discussion that the present design is safe for these reactivity addition rates independent of whether they arise from flooding or voiding.

2. Results of Mild Excursions

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 Appendix B. The analysis assumes that the scram system is inoperative, thus the probability

that the power excursion would actually exceed ~10 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 4×10^{-4} δ k/sec.

TABLE 3

Results of a Mild Reactivity Excursion
(See Figure 15)

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

(1) Aluminum melting point 1220 $^{\circ}$ F

(2) UO₂ melting point ~ 5000 $^{\circ}$ F

D. Maximum Credible Accident

1. Causes

The most serious accident envisioned during phases B and C of the NUSU Critical Experiments is the rapid insertion of 1.5% excess reactivity in a core containing 3.2 tonnes of UO_2 . 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, see Section VI B) can be achieved by use of the access door bypass key (see Section IV G). The natural physical hindrances to be deliberately surmounted by a single individual in gaining access to a control rod, dismantling the rod hanger* and effecting a fast withdrawal of a 4 foot long object 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 movable rod in a small core configuration, whereas the results of the accident have been calculated for the largest Phase B core consisting of 64 boiler assemblies and four loaded superheater manifolds.

(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.

*When rods are in the safety pattern the magnetic clutches in the rod drive package are engaged; it is estimated that a force of over 200 pounds would be required to lift the rod against the restraining action of the clutch.

(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. 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.

b. Object Falling into Reactor Vessel

This type of accident would be most effective during a partial water height experiment when the reactor is critical at reduced water height and the control rods are withdrawn. Using a value of $1\frac{1}{2}$ δ k/in for the incremental worth of water at the minimum critical water height, the worth of an object (density > 1 gr/cc) is $0.43\frac{1}{2}$ δ k/k per cubic foot. Thus, the displacement of ≥ 3.5 cubic feet of water could lead to a Maximum Credible Accident. The probability of this occurring as a single event is negligible since the full complement of upper grid plates plus numerous cover plates will be used regardless of the core size (see Figure 6). The open area between edge of the reactor vessel and the grid and cover plates is of such size and shape that it would be nearly impossible for a $3\frac{1}{2}$ cubic foot object to gain access to the vessel. Unauthorized attempts to enter the cell during any experiments will automatically result in a rod and moderator dump scram. In addition, the likelihood of objects falling into the vessel from above is negligible because of daily inspection procedures.

2. Results of the Maximum Credible Accident

a. Nuclear Reaction

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

TABLE 4

Maximum Credible Accident

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

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 would depend to a great extent upon the integrity of the exposed fuel pellets. A very conservative estimate⁽¹⁾ of the fission products released is 5% of the total generated by the 1200 Mw-sec energy release, i.e., an amount equivalent to 60 Mw-sec.

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

b. Chemical Reaction (2)

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 $\sim 2200^{\circ}\text{F}$ 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 ($\sim 2800^{\circ}\text{F}$) 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

(1) Private Communication from W. P. Chernock, Metallurgy Section, Materials Development Lab., CEND.

(2) See Summary by R. O. Brittan, Reactor Containment, ANL-5948.

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. Hazards to Surrounding Area

A conservative estimate of the radiation dosage received by a person outside the Critical Facilities can be obtained by postulating the formation of a fission product cloud above the cell. The total quantity of fission products in the cloud is taken to be equal to the quantity of fission products released from the fuel. The methods of calculating the dose rates from the fission product cloud are described in numerous reports*.

YAEC-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 NUSU Critical Experiments since the basic assumptions inherent to the analysis are also applicable in this case. The basic assumptions are as follows:

- a. All fission products released in an accident are air borne in a cloud of 20 foot radius above the cell.
- b. 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.
- c. The cloud is assumed to be reflected giving a concentration double that of a normal Gaussian cloud.
- d. The center of the cloud moves in a straight line from the source through the point where the observer is standing.
- e. 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.

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

The radiation hazards fall into three separate categories:

- (i) External gamma radiation due to immersion in a fission products cloud.
- (ii) External gamma radiation due to deposition from fission products cloud.
- (iii) Internal radiation due to inhalation of radioactive isotopes from fission products cloud.

The immersion beta dose was mentioned in YAE-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 or 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 offsite dwelling is ~2000 feet from the Critical Facilities. Table 5 summarizes the maximum integrated radiation dose under adverse meteorological conditions (strong inversion) for 1200, 1900, and 5000 feet from the Critical Facilities.

TABLE 5

Summary of Radiation Hazard

Distance from Burst (feet)	Condition	Immersion	Fallout	Inhalation*, Total	
		Total Gamma Dose rem	3 hr. Gamma Dose rem	Dose to Lungs, rep. (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

*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 rep; Bone: (Sr-89) ~ 2.8×10^{-2} rep; Bone: (Sr-90) ~ 5.4×10^{-2} rep.

Therefore, a conservative estimate of the maximum body gamma dose that an offsite person could receive is approximately 9 rem. 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 rem. 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 5 indicate doses which are probably 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:

- (a) not serious even under the most improbable sequence of events which can plausibly be conceived,
- (b) no worse than those approved for other similar or even more densely populated areas,
- (c) no worse than those for which approval has been granted for previous operation of the ACE.

VII. FUEL STORAGE AND HANDLING

A. Buckling Calculations

An IBM-704 program for a four-group calculation of the critical (or material) buckling of a uniform homogeneous mixture of fuel, moderator and cladding was used to provide conservative values of material buckling and infinite multiplication factors over a range of water to uranium oxide volume ratios. The conservatism employed in the calculations may be attributed to a neglect of the neutron flux depression in and near the fuel rods; i.e., the disadvantage factors of the water and steel cladding relative to the fuel were taken as unity.

The conservative nature of the calculations is demonstrated by a comparison of calculated and experimental bucklings for three TRX loadings* of 1.311 w/o UO_2 . The calculated bucklings are 10 - 20% higher than the experimental values. For rods of fixed length, this corresponds to a conservatism of 10-20% in estimates of critical mass or critical number of rods.

*Fuel loadings and bucklings obtained from WAPD-176, "Kinetic and Buckling Measurements on Lattices of Slightly Enriched Uranium and UO_2 Rods in Light Water."

B. Fuel Element Storage Canisters

The fuel element storage canisters are 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. A fully loaded canister contains approximately 140 pounds of UO_2 and has a gross weight of approximately 200 pounds. A three-wheeled cart was designed to transport a single canister within the critical facility area and to provide support of the canister at a convenient angle when fuel is being transferred between canister and core.

The following data were used to determine an upper limit to the canister diameter (see Figure 17 also). The actual choice of size was based on consideration of: ease of handling, cost, and accessible storage area on the cell balcony.

TABLE 6

Enrichment w/o	Maximum k _{oo}	Maximum B_{m^2} (cm ⁻²)	Min. Critical Diameter* in.	Min. Critical Mass* (kg. UO_2)	(rods)
1.85	1.310	0.0082	15.4	363	290
3.5	1.445	0.0131	10.6	120	76

The maximum or minimum values referred to in the table correspond to the so-called condition of "optimum moderation." The critical mass and critical diameter correspond to fully reflected, water moderated assemblies of clad fuel rods with 48 inches of active fuel length. An 8 cm reflector savings was used both radially and axially.

The nuclear safety of the 4 inch diameter canister is readily apparent from the above table and Figure 17. In order to achieve optimum moderation within a canister the number of fuel rods must be reduced by greater than 50% of the maximum canister capacity. The following conservative safety rules have been adopted in the handling of full or partially filled fuel canisters, or equivalent numbers of rods, outside the fuel storage racks. Such canisters will be held securely and will be separated by a minimum distance of six feet; covered canisters will be assumed full until shown empty.

*Wet, fully reflected (8 cm) lattice.

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C. Storage Racks

The fuel element canisters will be stored vertically in a two-slab array on the balcony of the reactor cell (see Figures 1 and 18). The canister storage racks are approximately 19 feet long and provide room for 86 canisters (43 per row) spaced 5 inches on center within each row and 16 inches on center between each row. The effective multiplication factor for each row of racks, even if they contain all 3.5 w/o enriched rods and even with optimum moderation is 0.56 when wet and unreflected and 0.88 when wet and fully reflected.

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).

The nuclear safety of the fuel storage array was investigated using the calculations outlined in Section VII A. Limiting values for parameters relative to nuclear safety can be described in terms of the 3.5 w/o fuel since any configuration that is safe for this enrichment will also be safe for a lower enrichment. In addition, parameters evaluated at the optimum water to UO_2 ratio were assumed in all cases. The reflector savings for the fully reflected cases was taken to be 8 cm, independent of the H_2O/UO_2 ratio. The extrapolation length for the bare cases was assumed to be twice the epithermal diffusion coefficient.

Under the conditions of complete flooding of the storage facility, any interaction between the two parallel slabs, separated by 12 inches of water, is negligible*. In addition, the storage facility is located 12 feet above ground level, which, in turn, is well above recorded flood levels. (See CEND-105 for Topography of Site.)

The interaction between the two parallel slabs for the dry case can be described in terms of the basic interaction criteria put forth in report number K-1317**. "Individually safe units, if unreflected, are safe when $k \leq 0.65$ and the interaction solid angle is $< 48\%$ of 4π ." The interaction solid angle for 2 slabs $4' \times 19'$ separated by 12" is $< 48\%$ of 4π ; k_{eff} is surely less than 0.65 for a single dry slab since it was calculated above to be 0.56 for a single bare semi-

*K-1019 - Basic Critical Mass Information and Its Application to ORGDP Design and Operation - H. F. Henry, et al.

**K-1317 - Self Consistent Criteria for Evaluation of Neutron Interaction - See also K-1309, Figure 1.

infinite slab at the optimum water/ UO_2 ratio.

These figures are conservative for the following reasons:

1. Buckling calculations outlined in Section VII A are conservative.
2. A more realistic $\text{H}_2\text{O}/\text{UO}_2$ ratio, i.e., tighter lattice, for the 3.5 w/o fuel when wet and fully reflected would give a smaller k_{eff} .
3. All calculations assumed the presence of water in the lattice, therefore the actual k_{eff} for non-flooded conditions would be much smaller than 0.56.
4. The 3.5 w/o fuel comprises less than 20% of the total fuel by weight.

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D. Storage of UO₂ Pellets

The data in Table 7 is based on a set of conservative buckling calculations for an array of unclad UO₂ fuel pellets in water. These fuel pellets will be used to assemble the experimental fuel rods (see Section III C).

TABLE 7

Enrichment (w/o)	Min. Critical Mass (Kg UO ₂)	Equiv. No. of Rods	OPTIMUM	
			H ₂ O/UO ₂ Vol. Ratio	H/25 Atomic Ratio
1.85	165	132 -	3.5	550
3.5	40.7	26	6.0	550

The above results refer to a fully reflected spherical assembly* (8 cm reflector saving) at optimum H₂O/UO₂ volume ratio. The total number of fuel pellets available (expressed in the number of equivalent rods) is 11, 33, 22, and for the natural, 1.85 w/o, and 3.5 w/o enrichments, respectively.

On the basis of this data, the limit of 20 kg of fuel pellets of any enrichment to be stored in a single container provides adequate conservatism. Present plans on the storage of fuel pellets at the Critical Facilities calls for the use of cylindrical containers resembling fuel rods. A single container would be transferred from the fuel storage vault (located in the pre-assembly area) to the vented hood for the loading of pellets and foils in each experimental fuel rod.

*This assembly may be more aptly described as the intersection of a spherical surface with an array of unclad fuel rods.

APPENDIX A. PHYSICS CALCULATIONS*

The method of calculation is, first, to evaluate the lattice parameters and age-diffusion constants for the boiler, superheater, and reflector region. The parameters and constants are then reduced to the multigroup coefficients which, in turn, are inserted into three-group diffusion equations. These equations are solved numerically on digital computer machines by iteration techniques. The results of these calculations determine the reactivity, power distribution, reactivity coefficients, and control characteristics of the reactor.

A. Reactivity Calculations

A three-group method was used to calculate the reactivity.⁽¹⁾ Assuming that fissions occur only at thermal energies in U-235 and that fast fissions and epithermal absorptions occur only in U-238, the three-group equations are the following:

$$\begin{aligned}
 1) \quad & -D_1 \nabla^2 \phi_1 + \sum_{s1} \phi_1 = \epsilon \nu \sum_{f3} \phi_3 \\
 & -D_2 \nabla^2 \phi_2 + (\sum_{s1_2} + \sum_{a2}) \phi_2 = \sum_{s1} \phi_1 \\
 & -D_3 \nabla^2 \phi_3 + \sum_{a3} \phi_3 = \sum_{s1_2} \phi_2
 \end{aligned}$$

where D is the diffusion coefficient, $\sum_{s1}$ is the slowing down or removal cross section, $\sum_a$ is the absorption cross section, $\sum_f$ is the fission cross section, ϵ is the fast fission factor, ν is the number of neutrons released per fission, and the subscripts are designations of the neutron energy group.

The three-energy groups can be designated as the fast, the epithermal, and the thermal groups. The energy interval for the first group is from 10 Mev to 180 Kev; for the epithermal group, the energy interval is from 180 Kev to a thermal cutoff; for the thermal group, the energy interval is from the thermal cutoff to zero energy.

The solution of Equation No. 1, for a complete two dimensional reactor configuration, requires high speed digital computers. The details of these methods are described in WAPD-TM-70.⁽²⁾

(1) R. W. Deutsch, Computing 3-group Constants for Neutron Diffusion, Nucleonics, Vol. 15, No. 1 (1957).

(2) G. G. Bilodeau, W. R. Cadwell, J. P. Dorsey, J. G. Fairey, R. S. Varga, PDQ - An IBM-704 Code to Solve the Two-Dimensional Few-Group Diffusion Equations, WAPD-TM-70 (1957)

*This section was contributed by R. W. Deutsch, GNEC.

B. Determination of Multigroup Constants

1. Average Microscopic Thermal Cross Sections

In order to determine the thermal cross section, the microscopic cross sections must be averaged over the proper thermal neutron spectrum. The neutron spectrum is a function of reactor composition and lattice geometry and, in general, is different for each region. Unfortunately, very little information on heterogeneous systems, either of an experimental or theoretical nature, is currently available. Therefore, it is not possible to perform a precise neutron spectrum calculation. The procedure that has been adopted is to assume a "hardened" or shifted Maxwellian spectrum, T_n , such that:

$$T_n = T \left(1 + 0.75 \frac{\sum a}{\xi \sum s} \right)^2$$

where T is the physical temperature in electron volts and $\frac{\sum a}{\xi \sum s}$ is the ratio of absorption to slowing-down cross sections. The choice of this spectrum is based on theoretical predictions of the thermal spectrum of homogeneous systems. The shifted Maxwellian spectrum has a thermal cutoff of $4 T_n$. The microscopic cross sections are taken from BNL-325 second edition⁽¹⁾. The cross sections of the $1/v$ absorbers, at a neutron velocity of 2200 m/sec are, as follows:

	<u>σ (barns)</u>
H	0.332
Cr	3.1
Fe	2.53
Ni	4.8
Mn	13.2
Al	0.230
U-238	2.71

For the $1/v$ absorbers, numerical factors that are a function of T_n can be simply calculated to convert from the value at 2200 m/sec.

For U-235, a non- $1/v$ absorber, a numerical average of the absorption cross section has been calculated as a function of neutron temperature:

(1) D. J. Hughes and R. M. Schwartz, Neutron Cross Sections, BNL-325, Second Edition (1958)

<u>T(ev)</u>	<u>U-235</u>
0.025	633
0.040	484
0.060	387
0.080	338
0.100	303

For U-235, a value of 2.46 is used for ν ; the ratio of resonance-to-fission capture for thermal neutrons, is taken to be 0.184.

2. Disadvantage Factor and Microscopic Thermal Cross Section

The disadvantage factor is calculated, using diffusion theory in the moderator and a transport condition at the surface of the rod. In this formulation, a somewhat different model is used for calculating the boiler with aluminum cladding and the superheater with stainless steel cladding. For the aluminum cladding:

$$2) \quad \frac{\phi_m}{\phi_f} = \frac{3}{2} \frac{\sum R_o}{k \lambda} \left\{ kd + \frac{I_o (kR_1) K_1 (kR_2) + I_1 (kR_2) K_o (kR_1)}{I_1 (kR_2) K_1 (kR_1) - I_1 (kR_1) K_1 (kR_2)} - \frac{2 k R_1}{k^2 R_2 - k^2 R_1^2} \right\}$$

d is the reciprocal of the logarithmic derivative of the flux at the surface of the rod and is defined, as follows:

$$d = \Delta \left(\frac{R_o}{\lambda} \right) + \frac{4}{3} \lambda \left(\frac{1 - 2 f \sum R_o}{2 f \sum R_o} \right)$$

R_o = radius of fuel pellet;

R_1 = outer radius of fuel element cladding;

R_2 = outer radius of unit cylindricalized cell;

λ = neutron transport mean free path in the moderator;

k = inverse neutron thermal diffusion length;

$\sum$ = macroscopic absorption cross section of fuel;

f = self-shielding factor for a cylinder, defined in Reference 1;

$\Delta \left(\frac{R_o}{\lambda} \right)$ = function, defined in Reference 2.

(1) M. L. Storm, H. Hurwitz, G. M. Roe, Gamma Ray Absorption Distributions for Plane, Spherical and Cylindrical Geometries, KAPL-783, (1952).

(2) B. Davison, Physics Society of London Proceedings, Vol. 64, p. 881, (1951).

The same procedure is applied to the stainless steel cladding case, with the exception that the clad and fuel are treated as a uniform region. The disadvantage factor, as applied to the aluminum cladding, is taken as the same as for the water. For stainless clad, the disadvantage factor

is assumed to be $1 + \frac{1}{2} \left(\frac{\phi_m}{\phi_f} - 1 \right)$. For the voided superheater case, the flow tube is treated the same as the clad. For the flooded superheater, the flow tube has the same disadvantage factor as the water.

The macroscopic thermal cross section is, as follows:

$$3) \quad \Sigma = \frac{v_f \Sigma_f + \frac{\phi_m}{\phi_f} v_m \Sigma_m + \frac{\phi_{cl}}{\phi_f} v_{cl} \Sigma_{cl} + \frac{\phi_{st}}{\phi_f} v_{st} \Sigma_{st}}{(1 - v_m - v_{cl} - v_{st}) + \frac{\phi_m}{\phi_f} v_m + \frac{\phi_{cl}}{\phi_f} v_{cl} + \frac{\phi_{st}}{\phi_f} v_{st}}$$

v is the volume fraction of each material; $\frac{\phi_m}{\phi_f}$, $\frac{\phi_{cl}}{\phi_f}$, and $\frac{\phi_{st}}{\phi_f}$ are the disadvantage factors for the moderator, cladding, and structural material, respectively.

3. Epithermal Capture in U-238

The epithermal capture cross section for U-238 is defined, as follows:

$$4) \quad \Sigma_2 = v_f N_f \frac{\sigma_o (1 + \beta t)}{\Delta u}$$

where v_f is the volume fraction of the fuel, N_f is the number of atoms/cc, σ_o is the resonance integral, β is the coefficient that describes the Doppler broadening of the resonances, t is the temperature, and Δu is the lethargy interval of the epithermal group.

The resonance integral that is used in Equation No. 4 is

$$5) \quad \sigma_o = 5.25 + 26.6 \sqrt{\frac{S}{M}}$$

where $\frac{S}{M}$ is the surface-to-mass ratio of the fuel.

The above value is based on the measurement of Hellstrand⁽¹⁾ and includes 1.1 barn $1/v$ absorption.

(1) E. Hellstrand, Measurement of the Effective Resonance Integral in Uranium Metal and Oxide, Journal of Applied Physics, Vol. 28, p. 1493, (1957).

The variation of the effective resonance integral with temperature has been measured by Rodeback⁽¹⁾ and Davis⁽²⁾, and an average of their measurements is used:

$$\beta = 1.25 \times 10^{-4}/^{\circ}\text{C}$$

4. Fast Fission in U-238

The fast fission factor, ϵ , is calculated relative to the experimental measurements for UO_2 ⁽³⁾. It is assumed that ϵ , which is a measure of the number of fast fissions occurring in U-238 relative to that occurring in U-235, can be calculated as if the lattice were homogeneous. The calculated fast fission factor, ϵ , for light water lattice relative to an experimental measurement, ϵ' , is:

$$6) \quad \epsilon - 1 = (\epsilon' - 1) \frac{N_0}{N_0'} \frac{(\nu'_{\text{H}_2\text{O}} \Sigma'_{\text{H}_2\text{O}} + \nu'_{\text{U-238}} \Sigma'_{\text{U-238}} + \nu'_{\text{st}} \Sigma'_{\text{st}})}{(\nu_{\text{H}_2\text{O}} \Sigma_{\text{H}_2\text{O}} + \nu_{\text{U-238}} \Sigma_{\text{U-238}} + \nu_{\text{st}} \Sigma_{\text{st}})}$$

ν is the volume fraction of each material; N_0 is the number of U-238 atoms per cubic centimeter; Σ is the total slowing-down cross section; the subscript, st, refers to the structural and cladding material; the prime refers to the experimental measurement.

The microscopic cross sections which are used in Equation No. 6 are, as follows.

	<u>σ</u>
U-238	3.70
Aluminum	0.38
Stainless Steel	1.04
H ₂ O	5.20

- (1) G. W. Rodeback, Temperature Coefficients of Uranium and Thorium Resonance Integrals, NAA-SR-1641, (1956)
- (2) M. V. Davis, Resonance Absorption of Neutrons by Uranium Cylinders, Journal of Applied Physics, Vol. 2, p. 250, (1957).
- (3) D. Klein, et al, Measurements of Thermal Utilization, Resonance Escape Probability and Fast Effect on Water-Moderated, Slightly Enriched Uranium and UO_2 Lattices, Nuclear Science and Engineering, 3, p. 403, (1957).

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5. Diffusion Coefficient and Slowing-Down Cross Section

The diffusion coefficient, D , is found by averaging the measured cross section over the appropriate neutron flux in each group. Reference 1 describes the method and lists the cross sections.

The macroscopic slowing-down cross section is defined as the diffusion coefficient divided by the neutron age. The age is determined for the fast and epithermal groups by calculating the equivalent transport and slowing-down factors of the materials present relative to the pure moderator. The additions of these materials perturb the experimentally measured pure moderator age. The method and equivalent factors are described in Reference 1. In addition, the reactivity effect of the streaming of neutrons in the superheater pressure tubes is accounted for by Behren's method (1) of increasing the age and thermal diffusion length.

C. Evaluation of Control Rods

The effectiveness of the control rods is calculated by an absorption area method in which neutron captures occur in both the thermal and epithermal neutron energy group⁽²⁾. In this method, the transmission probability of thermal and epithermal neutrons in the control rod are calculated. The transmission probability is related to the logarithmic derivative of the flux at the boundary of the rod, and these results can be used directly in PDQ two-dimensional code. Also, the absorption area can be calculated for each individual rod and related to the fractional decrease to the slowing-down source for each group^(2,3), and the rod absorptions can be considered without treating the rods as discreet regions.

D. Determination of Reactivity Coefficients

Since in many cases, the NUSU criticals use two region cores, reactivity coefficients were calculated for both the boiler and superheater regions. A method of calculating the local reactivity change which results from temperature and void changes is to use perturbation theory. The method used is a first-order perturbation theory which has been developed by Brooks⁽⁴⁾. In this method, the three-group equations that are the adjoints of Equation No. 1 are solved by the PDQ code. Then, by simple mathematical manipulation of the real and adjoint fluxes, an expression for the reactivity coefficients can be determined⁽⁵⁾.

- (1) D. J. Behrens, The Effects of Holes in a Reactivity Material on the Passage of Neutrons, Physics Society of London Proceedings, 62A, p. 607, (1941)
- (2) R. W. Deutsch, Evaluation of a Bank of Slab-Type Control Rods, Nuclear Science and Engineering, 5, p. 150, (1959).
- (3) S. Pearlstein, Application of the Absorption Area Method of Three-Group Diffusion Theory Problems, Nuclear Science and Engineering, 4, p. 322, (1959).
- (4) Harvey Brooks, Perturbation Methods in Multigroup Calculations, KAPL-71, (1948).
- (5) H. E. Stone, Calculating Spatial Distribution of Reactivity Coefficients, Nucleonics, Vol. 15, No. 9, p. 187, (1957)

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E. Comparison of Analysis with Experiment

To indicate the accuracy of the calculational techniques being used for reactivity calculations, an analysis of the following three cold critical experiments has been made: The Westinghouse TRX Critical Experiment^(1,2), the Westinghouse Yankee Critical Experiment, and the Babcock and Wilcox N. S. Savannah Critical Experiment.

1. TRX Critical Experiments^(1,2)

The measurement of lattice parameters and critical bucklings were made for 0.383-inch UO_2 pellets clad with aluminum. The fuel had a U-235 enrichment of 1.3%. A comparison of the measurements with calculations for three water-to- UO_2 ratios is, as follows:

Resonance Escape Probability

<u>Water/UO_2</u>	<u>Calculation</u>	<u>Experiment</u>
1.426	0.799	0.775 ± 0.004
1.779	0.818	0.815 ± 0.004
2.396	0.862	0.850 ± 0.004

Fast Effect

<u>Water/UO_2</u>	<u>Calculation</u>	<u>Experiment</u>
1.426	1.045	1.046
1.779	1.039	1.042
2.396	1.032	1.034

- (1) D. Klein, et al, Measurements of Thermal Utilization, Resonance Escape Probability and Fast Effect on Water-Moderated, Slightly Enriched Uranium and UO_2 Lattices, Nuclear Science and Engineering, w, p. 403, (1957)
- (2) J. R. Brown, et al, Kinetic Buckling Measurements of Lattices of Slightly Enriched Uranium and UO_2 Rods in Light Water, WAPD-176, (1958)

Disadvantage Factor $\frac{(\text{flux in moderator})}{\text{flux in fuel}}$

<u>Water/UO₂</u>	<u>Calculation</u>	<u>Experiment</u>
1.426	1.068	1.10 $\pm$ 0.01
1.779	1.090	1.13 $\pm$ 0.01
2.396	1.114	1.13 $\pm$ 0.01

Age (cm²)

<u>Water/UO₂</u>	<u>Calculation</u>	<u>Experiment</u>
1.426	44.2	45.8
1.779	42.3	44.3
2.396	38.8	40.6

Based on the critical buckling (reactivity = 1.000), the calculated reactivity is, as follows:

<u>Water/UO₂</u>	<u>Calculated Reactivity</u> (Experimental Reactivity = 1.000)
1.426	0.987
1.779	0.994
2.396	1.000

2. Yankee Critical Experiments⁽¹⁾

Measurements have been made of critical configurations for close-packed lattices containing 0.300-inch diameter fuel clad in stainless steel jackets. The predicted values of reactivity, based on experimental bucklings and on experimental correction for control rod followers for three water-to-UO₂ volume ratios are, as follows:

<u>Water/UO₂</u>	<u>Calculated Reactivity</u> (Experimental Reactivity = 1.000)
1.05	0.983
1.41	0.993
1.86	0.992

(1) P. W. Davison, et al, Yankee Critical Experiments, YAE-94, (1959).

3. N. S. Savannah Experimental Criticals⁽¹⁾

UO₂ in the form of 0.500-inch-OD rods with stainless steel clad jackets have been arranged in critical arrays with water-to-UO₂ volume ratios of 1.54 and 1.15. The experiments were meant to mock up the actual core so that experiments included the effect of a water channel between boxes and box walls.

Five configurations are being analyzed. The three configurations with a water-to-fuel ratio of 1.54 consider the following: no internal can wall, water channel between aluminum box walls, and water channel between stainless steel box walls. The two configurations with a water-to-fuel ratio of 1.15 consider the following; water channels between aluminum box walls and water channels between stainless steel box walls.

Only the results for the 1.54 water-to-fuel ratio with no internal boxes or channels have been completed. The calculated reactivity for this critical configuration, using the experimental bucklings, is 0.991.

(1) R. M. Ball, et al, Critical Experiments for the N. S. Savannah Core, BAW-1131, (1958).

APPENDIX B*

Transient Analysis for the NUSU Critical Experiments

Calculation of power excursions for the NUSU critical experiments is based on the following main assumptions:

1. Reactor core is immersed in a water tank but there is no coolant flow.
2. Water temperature is initially at 68°F with superheater assemblies flooded.
3. During the excursion, thermal parameters are assumed constant, i.e., heat transfer coefficient, thermal conductivities, specific heats, etc. are assumed constant.
4. Five-group of delayed neutrons are used.
5. All the heat released during the excursion is stored in the fuel and water.
6. A steady state condition is assumed to exist prior to the excursion, i.e., the small power being generated (~1 watt) is dissipated continuously by radiation.

The following equations were used to describe the problem.

$$\frac{dn}{dt} = \frac{k_{ex}(1-\beta)-\beta}{l} n + \sum_{i=1}^5 \lambda_i C_i \quad (1)$$

$$\frac{dC_i}{dt} = \frac{(1+k_{ex})\beta_i}{l} n - \lambda_i C_i \quad (2)$$

$$k_{ex} = (1+k_{ex}) \left[D (T_f - T_{fo}) + E (T_w - T_{wo}) \right] + B t \quad (3)$$

$$D = \frac{1}{k_{eff}} \frac{d k_{eff}}{d T_f} \quad (4)$$

*This section was contributed by R. W. Deutsch, GNEC.

$$E = \frac{1}{k_{eff}} \frac{d k_{eff}}{d T_w} \quad (5)$$

$$W_f C_f \frac{dT_f}{dt} = H - A_h \bar{U} (T_f - T_w) \quad (6)$$

$$A_h \bar{U} (T_f - T_w) = C_w W_w \frac{dT_w}{dt} + H_o \quad (7)$$

Equations (1) and (2), modified for digital calculation, become:

$$\Delta n_j = \frac{\frac{\Delta t}{\ell} n_j \left[k_{exj} + \frac{\Delta k_j}{2} \left(1 - \sum_i \frac{\beta_i}{1 + \lambda_i \frac{\Delta t}{2}} \right) - \sum_i \frac{\left(\frac{dC_i}{dt} \right)_j \Delta t}{1 + \lambda_i \frac{\Delta t}{2}} \right]}{1 - \frac{\Delta t}{2\ell} \left[(k_{exj} + (2/3) \Delta k_j) \left(1 + \sum_i \frac{\beta_i}{1 + \lambda_i \frac{\Delta t}{2}} \right) - \sum_i \frac{\beta_i}{1 + \lambda_i \frac{\Delta t}{2}} \right]} \quad (8)$$

$$\Delta C_{ij} = \frac{\frac{\beta_i}{\ell} \left[n_j \Delta k_j + \Delta n_j (1 + k_{exj} + (2/3) \Delta k_j) \right] \frac{\Delta t}{2} + \left(\frac{dC_i}{dt} \right)_j \Delta t}{1 + \lambda_i \frac{\Delta t}{2}} \quad (9)$$

also,

$$\left(\frac{dC_i}{dt} \right)_{j=0} = \frac{k_{ex0} \beta_i}{\ell} \quad (10)$$

$$\left(\frac{dC_i}{dt} \right)_{j+1} = \frac{2 \Delta C_{ij}}{\Delta t} - \left(\frac{dC_i}{dt} \right)_j \quad (11)$$

$$\frac{dn}{ndt} = \frac{k_{exj}}{\ell} \frac{1}{n_j} \sum_i \left(\frac{dC_i}{dt} \right)_j \quad (12)$$

Equations (6) and (7) are manipulated for digital calculations to obtain,

$$\Delta T_{fj} = \left\{ \gamma' [(n_j - 1) + \phi \int_0^t (n_j - 1) dt] - \left\{ \beta' \gamma' e^{-\beta' t} \int_0^t e^{\beta' t} [(n_j - 1) + \phi \int_0^t (n_j - 1) dt] dt \right\} \right\} \Delta t \quad (13)$$

and

$$\Delta T_{wj} = \left\{ \phi (T_{fj} - T_{fo}) - \phi^2 e^{-\phi t} \int_0^t (T_{fj} - T_{fo}) e^{\phi t} dt \right\} \Delta t \quad (14)$$

In Equations (13) and (14),

$$\gamma' = \frac{H_o}{W_f C_f} \quad \theta = \frac{W_f C_f}{W_w C_w} \quad \beta' = \frac{A_h \bar{U}}{W_f C_f} (1 + \theta) \quad \phi = \frac{A_h \bar{U}}{C_w W_w} \quad (15)$$

$$T_{fj+1} = \bar{T}_{fj} + \Delta T_{fj} \quad (16)$$

$$k_{exj+1} = k_{exj} + \Delta k_j \quad (17)$$

$$n_{j+1} = n_j + \Delta n_j \quad (18)$$

To obtain the clad temperature, the following equation is easily derived.

$$T_{clad} = \frac{1}{2}(T_f \alpha_1 + T_w \alpha_2), \text{ where } \alpha_{1,2} = \frac{1 + \bar{U} \left\{ \frac{\left(\frac{k}{d}\right)_g}{\left(\frac{k}{d}\right)_g} - h_f \right\}}{\left(\frac{k}{d}\right)_g \cdot h_f} \quad (19)$$

This set of equations was programmed for an IBM-610 digital computer and the power and temperature excursion determined for different values of ramp rates of reactivity. These are reported in the main body of the report.

* $(k/d)_g$ is the ratio of thermal conductivity to thickness for the 2.5 mil radial gas layer between the fuel and clad. The results of Table 4 are based on a uniform helium filled gap rather than a uniform air gap. The latter model would have resulted in a maximum temperature of $\sim 4400^\circ\text{F}$ and $\sim 750^\circ\text{F}$ for the fuel and clad, respectively. It is felt that the latter model would be unrealistic and non-conservative since it neglects the possible existence of heat shunts in the form of oxide fuel in contact with the clad.

DATA

Evaluation of the constants at 68°F for the NUSU critical experiments gives:

<u>i</u>	<u>λ_i</u>	<u>β_i</u>
1	0.0258	0.0016700
2	0.1120	0.0013545
3	0.3050	0.0027943
4	1.1480	0.0009366
5	3.0660	<u>0.0003356</u>
		= 0.007097

$$\ell = 49 \times 10^{-6} \text{ sec}$$

$$D = 1.0 \times 10^{-5} \Delta k/k/^{\circ}\text{F}$$

$$E = 2.48 \times 10^{-5} \Delta k/k/^{\circ}\text{F}$$

$$\theta = 0.3005$$

$$\gamma' = 15.2 \times 10^{-7} \text{ }^{\circ}\text{F/sec for } H_0 = 1 \text{ watt}$$

$$\phi = 0.0644 \text{ sec}^{-1}$$

$$\beta' = 0.279 \text{ sec}^{-1}$$

$$T_{wo} = 68^{\circ}\text{F}$$

$$T_{fo} = 68 + 7.1 \times 10^{-6} \text{ }^{\circ}\text{F for } H_0 = 1 \text{ watt}$$

$$n_0 = 1$$

$$k_{ex_0} = 0$$

SYMBOLS

- A_h = heat transfer surface
- B = rate of reactivity addition
- C_w = specific heat for water
- C_f = specific heat for fuel
- C_i = delayed neutron pressure concentration
- D = Doppler coefficient of reactivity
- E = temperature coefficient of reactivity
- H = actual reactor power
- H_f = water film heat transfer coefficient
- k_{eff} = effective multiplication constant
- k_{ex} = $k_{eff} - 1$
- l = mean neutron lifetime
- n = normalized neutron power
- $\bar{T}_f$ = fuel average temperature
- T_w = water average temperature
- T_c = clad surface average temperature
- t = time
- $\bar{U}$ = over-all heat transfer coefficient used in connection with average fuel element temperature
- W_f = fuel mass
- w_w = water mass
- λ_i = delayed neutron decay constant for i th group
- ϕ = inverse of water thermal time constant
- γ' = inverse of fuel thermal time constant
- θ = ratio of fuel thermal capacity to water thermal capacity

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FIG. 1
ACE FACILITY
SECTION THROUGH CELL

FIG. 2
ACE FLOW DIAGRAM

SYMBOLS

- GLOBE VALVE
- GLOBE VALVE SOLENOID OPERATED
- HOSE CONNECTION
- ORIFICE
- GLOBE VALVE MOTOR OPERATED
- PUMP
- GLOBE VALVE AIR OPERATED

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FIG. 3

ACE ROD DRIVE PACKAGE

CHANNEL

NOTE:

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

080

FIG. 4
NUCLEAR INSTRUMENTATION SCHEMATIC

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.

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FIG. 5
ROD DROP AND WATER DUMP SCRAM

NUSU REACTOR TANK

082

FIG. 6
CUTAWAY VIEW OF REACTOR TANK

083

FIG. 7
BOILER AND SUPERHEATER FUEL ROD GEOMETRY

084

FIG. 8
GRID ASSEMBLY
LOWER STRUCTURE

085

FIG. 9
GRID ASSEMBLY
UPPER STRUCTURE

086

FIG. 10
SUPERHEATER MANIFOLD SECTION

FIG. 11
STANDARD FUEL ROD ASSEMBLY

087

COVER PLATE WINDOW SLOT

FUEL PELLETS COVER PLATE

SECTION A-A

FIG. 12
EXPERIMENTAL FUEL ROD ASSEMBLY

088

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.

FIG. 13
BLOCK DIAGRAM OF CONTROL SYSTEM

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

091

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

092

CRITICAL NUMBER OF FUEL RODS AS A
FUNCTION OF THE CANISTER DIAMETER
(WET AND FULLY REFLECTED LATTICE)

093

FIG. 17
NUMBER OF 3.5 W/O FUEL
RODS VS CYLINDER DIAMETER

094

FIG. 18
TOP VIEW OF ACE FACILITY, CELL 2