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HAZARD SUMMARY REPORT ON THE EXPERIMENTAL
BOILING WATER REACTOR (EBWR)

by

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PREFACE

A Hazard Summary Report for the EBWR was originally issued in two parts: J. M. West, et al., "Hazard Summary Report on the Boiling Experimental Reactor (BER)," ANL-WHZ-390 (January, 1955) and J. M. West, "Supplement to Hazard Summary Report on the Boiling Experimental Reactor (BER)," Supplement to ANL-WHZ-390 (October, 1956). The latter reported modifications in the design of EBWR, and the effects which such changes might have on the safe operation of the reactor. Both of these reports were issued to a strictly limited distribution in accordance with the then-existing policy.

As a result of the successful operation of EBWR and the publication of a detailed plant description,¹ many requests for information relating to the safety of boiling water reactors have been received. Accordingly, and in fulfillment of the policy to co-operate with the increasing numbers of industrial participants in the Atomic Energy Commission's Civilian Application Program, a new report, which represents a combination of the previous hazard summaries, has been prepared. This report does not incorporate changes in the facility or in the procedure of operation which may have been adopted on the basis of experience in operating the reactor.

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¹ "The EBWR," ANL-5607 (May, 1957); obtainable from the Superintendent of Documents, U.S. Government Printing Office, Washington 25, D. C., for \$2.25.

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I. INTRODUCTION

In 1954 Argonne National Laboratory proposed to build a 20-megawatt Experimental Boiling Water Reactor (EBWR) power plant as part of the Atomic Energy Commission's five year program for the development of power reactors. On the premise that this type of reactor had attractive inherent safety characteristics, it was suggested in a preliminary hazard evaluation report¹ that this reactor should be constructed at the DuPage site of the Laboratory. After reviewing this preliminary evaluation, the Advisory Committee on Reactor Safeguards concluded that "the operation of this reactor at the Argonne Site should not present undue hazards if adequate containment is provided. It was understood that this problem will be presented for further review at such time as the detailed design of the reactor and the structure providing containment is completed."

At the time of the preliminary safeguard report the design features of the EBWR and building were tentative. Since then the Argonne staff, working in collaboration with the engineering firm of Sargent and Lundy, Chicago, developed the design to a point where further changes affecting safety in a major way were unlikely. A hazards evaluation report based on this design was, therefore, submitted for review prior to making major contractual arrangements for actual construction of EBWR, which started operations in late 1956.

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¹ ANL-5261, "Preliminary Hazard Summary Report on the Boiling Experimental Reactor (BER)" by J. M. West, G. A. Anderson, J. R. Dietrich, J. M. Harrer, A. S. Jameson, and S. Untermyer (April 1954).

II. SUMMARY

The EBWR is intended to be a demonstration of a direct steam cycle, natural circulation boiling reactor power plant on a scale (20 megawatts heat) which can be extrapolated to sizes of economic interest for central station power plant application. Experience with the Borax-I and II reactors provided the experimental basis for the design of the EBWR.

The EBWR core is basically a slightly enriched uranium lattice which is cooled and moderated by H_2O . The fuel element is a uranium alloy plate clad with Zircaloy-II. The plates are manufactured in two thicknesses and in two enrichments. One size is 54 in. long, $3\frac{5}{8}$ in. wide, and 0.212 in. thick. The other size has the same length and width, but is 0.280 in. thick. The two enrichments are natural uranium and 1.44% enriched uranium.

The fuel plates are arranged in bundles or assemblies containing six parallel plates each, with 0.42 in. water gaps between plates. Some assemblies contain natural uranium while others contain uranium enriched in U^{235} .

An aluminum alloy may be substituted for Zircaloy-II as the cladding material for EBWR fuel plates if tests show adequate corrosion resistance at 417F (300 psig) or higher. In this case, the meat for enriched plates would be aluminum-uranium alloy. From the inherent safety standpoint, aluminum would be at least as good as Zircaloy-II. The much higher (15 times) thermal conductivity of the aluminum more than compensates for its lower melting point.

The EBWR core is centered horizontally in a 7-ft ID pressure vessel, the annular space between the core and pressure vessel wall serving as a flow downcomer for the natural circulation cooling process. The water in this downcomer is slightly subcooled as a result of introduction of cold condensate from the power plant condenser near the top of the downcomer. This subcooling prevents net boiling as the water moves upward through the cooling channels of the core until about one-third of the core has been traversed. The reduced density compared with the downcomer, as a result of net steam generation from that point upward, provides the driving head for the natural circulation process.

The reactor pressure vessel is capable of operation at 200 psi in excess of the anticipated operating pressure of 600 psig. A large steam volume above the water line acts as a surge volume to enhance the inherent safety mechanism of steam formation. The steam volume also acts as a moisture separator which is expected to reduce the contaminants (water and radioactive impurities) in the steam by a factor of 10^4 per unit weight relative to the water in the reactor.

Nine cross-shaped control rods pass through vertical guide channels in the core. Five are fabricated from hafnium and Zircaloy-II, and four are made from 2% boron-Type 304 stainless steel and Zircaloy-II. The mechanical drive mechanisms for these rods are located underneath the reactor. The design is such that the maximum rate of addition of reactivity is limited to 0.01% k per second. Various scram signals produce rapid insertion of these rods under the influence of gravity augmented by springs, to produce a reduction in k of at least 12.5% in 0.5 sec. When the reactor is at pressure, the increased accelerating force provided by this pressure accomplishes the insertion of the rods in even less time.

A secondary non-inherent safety mechanism in the form of a pressurized concentrated boric acid solution can reduce the reactivity by at least 12.5% k in about 20 seconds. This injection can be initiated either manually or automatically. The automatic injection results when any four control rods do not reach their innermost position within two seconds after a scram signal.

With regard to inherent safety, the EBWR has a much stronger metal temperature coefficient, but a considerably weaker steam formation shutdown mechanism, than the Borax reactors. The stronger metal temperature coefficient is due largely to the "Doppler Effect" associated with the U^{238} in the fuel. Another contribution arises from the higher melting point of the fuel and the higher ratio of fuel to water in the EBWR. The thermal expansion of the fuel plates displaces a significant volume of water during a power excursion. The poorer steam formation shutdown feature is due to the low thermal conductivity of the fuel and the zirconium cladding in the EBWR core compared with the aluminum in Borax, and to the reduced heat transfer area per unit volume in the EBWR core. Calculations which are described in considerable detail in this report indicate that the metal temperature coefficient will be able to prevent melting of any fuel following the sudden addition of up to 1.1% k (0.017-sec period) to the cold reactor. For an excursion starting at operating temperature (489°F) the metal temperature coefficient could control about 1.0% k.

With the performances of the Borax-I and II reactors as a starting point, estimates have been made of the amount of reactivity which can be controlled by steam formation in the EBWR. For the cold case this automatic feature alone would protect against periods of 0.057 sec (0.84% k) or longer. At 212°F this effect increases to 1.1% k. At the operating temperature (489°F) the effect would be considerably higher. The increased effectiveness of the inherent steam formation safety mechanism with temperature is the basis for a design requirement that control rods cannot be withdrawn from the core until the reactor water has been heated to 325°F by means of auxiliary heaters.

Although the metal temperature coefficient and steam formation shutdown mechanisms certainly complement each other, a quantitative evaluation of the amount of additional reactivity which can be controlled by the combination cannot be made without more information about steam formation in rapid transients. Since the individual effects were calculated on pessimistic premises, the claim that the combined mechanisms will control at least 1.1% k is clearly very conservative.

The average power density of 25 kw per liter of coolant in EBWR is conservative relative to the performance (38 kw/liter) obtained in Borax-II. The amount of reactivity controlled by steam bubbles is also considerably less than in Borax-II when operating at 300 psig without instability. The Borax data indicate that stability improves with operating pressure, a fact which adds confidence in the stability of EBWR operating at 600 psig.

The maximum heat flux in EBWR is only about 155,000 Btu/hr ft². This is believed to be below the burnout point by at least a factor of 4 to 5, based on laboratory tests with electrically heated plates. The behavior of a boiling reactor at very high heat fluxes and power densities has not been demonstrated, however, and the possibility of unstable operation, sometimes referred to as "chugging," is recognized. If chugging does not occur, a very large amount of excess k could be added slowly to EBWR without any danger of burnout. The slow addition of reactivity would merely increase the power level slowly, the steam bubble concentration at any time being that required to compensate for the reactivity added. At 6 times the normal power level the average steam content would be more than 50%. The reactivity controlled by this steam would be at least 10% k.

Although chugging could, and probably would, eventually result from slow addition of a very large amount of reactivity, experiments with the Borax reactors have never produced chugs with periods anywhere near as short as the theoretical minimum period corresponding to complete collapse of all steam bubbles. For example, when the theoretical minimum period was about 0.005 sec, the shortest period ever observed in chugging was about 0.020 sec. It also appears to be true that violent chugs do not occur without forewarning. Minor instability can be detected prior to chugging. The first chug is never as violent as subsequent ones. For any credible circumstance other than one involving sabotage of all control circuits, it appears to be relatively easy to avoid short period chugging.

The EBWR is surrounded by a gas-tight steel building having a volume of about 400,000 ft³. This steel shell is designed to withstand an internal pressure of 15 psig. The basement of the building, in which most of the energy would be dissipated in case of a major pressure excursion in the reactor, is completely lined with reinforced concrete of 2-ft minimum thickness. A 1-ft thick lining of reinforced concrete protects the above-grade portion of the steel building against missiles from the turbine or other miscellaneous sources.

The control room and other service rooms are housed in a conventional masonry structure adjacent to the steel building and connected thereto by a double-door air lock. The doors of the airlock are mechanically linked to prevent opening one door unless the other is closed and sealed.

After the turbine is placed on its turning gear, the reactor and its associated power plant can be operated entirely from the control room. It will not be necessary for operators to enter the gas-tight building except perhaps once per eight-hour shift for routine inspections.

A 15,000-gal reservoir of water is suspended from the dome of the building for emergency cooling. By discharging through the building spray system, this water could quickly reduce the pressure resulting from a blow down of the pressurized water due to a rupture of the pressure vessel or piping, thus markedly reducing leakage from the gas-tight building. The spray system would also wash suspended radioactivity from the building atmosphere.

The gas-tight building could easily withstand the internal pressure resulting from blowdown of the pressurized water in the reactor. It could also contain the products resulting from explosive chemical reaction of 10% of all metal in the reactor core with water. Rapid reaction of all metal in the core with water followed by explosive burning of the hydrogen resulting from the reaction would rupture the building. This is considered to be a tolerable risk, since the probability of its occurrence is the product of three individually small probabilities: (1) that the reactor core can be made to melt, (2) that all of the molten metal will react explosively with the water, and (3) that the hydrogen will explode upon being expelled to the building atmosphere.

As a convenient reference, the major design and operating features of the EBWR facility are summarized in Table I which follows.

TABLE I

Plant Parameters

<u>General</u>	<u>Type</u>	<u>Direct cycle boiling reactor</u>
Heat output		20,000 kw
Gross electrical output		4,600 kw
Operating pressure		600 psig
Operating temperature		489 °F
Coolant		H ₂ O
Moderator		H ₂ O
Fuel		Natural and 1.44% enriched uranium

Core

Active diameter	4 ft
Active height	4 ft
Total uranium content	6.1 tons
Average U ²³⁵ content	1.20%
Total U ²³⁵ content	66 kg
Average water/uranium volume ratio	3.64
Structural material	Zircaloy-II

Fuel Assemblies

Total number in core	112
Cross-sectional dimensions	3.75 x 3.75 in.
Number of plates per assembly	6
Side plates	0.060-in. thick Zircaloy-II
Cladding	0.020-in. thick Zircaloy-II
End fittings	stainless steel
Total thin-plate natural U assemblies	18 *
Total thick-plate natural U assemblies	18 *
Total thin-plate enriched (1.44% U ²³⁵) assemblies	38 *
Total thick-plate enriched (1.44% U ²³⁵) assemblies	38 *
Composition of fuel "meat"	93.5% U, 5% Zr, 1.5% Nb (by weight)
Thickness of meat in thin plates	0.170 in.
Thickness of meat in thick plates	0.233 in.
Width of water channels between thin plates	0.428 in.
Width of water channels between thick plates	0.360 in.
Power generation in average natural thin plate fuel assembly	0.1226 mw
Power generation in average natural thick plate fuel assembly	0.1645 mw

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*The initial loading of the EBWR was actually 56 enriched thin-plate assemblies, 50 enriched thick-plate assemblies, 2 natural thin-plate assemblies, and 6 natural thick-plate assemblies.

Fuel Assemblies (Cont'd.)

Power generation in average enriched thin plate fuel assembly	0.1887 mw
Power generation in average enriched thick plate fuel assembly	0.2014 mw

Nuclear Data

Average thermal flux in fuel	1.0×10^{13}
Maximum/average radial flux ratio	1.58
Maximum/average axial flux ratio	1.49
Control rod channel peaking factor	1.30
Power generation in thick enriched plate relative to average plate	1.12
Neutron lifetime	6.3×10^{-5} sec
Conversion ratio	0.74
Reactivity changes $\frac{\Delta k_{eff}}{k_{eff}}$	
Temperature, 68°F to 489°F	1.65%
Operating Voids (10% bubbles)	1.10%
Xenon + Samarium (cold)	2.63%
Total reactivity to control (including 1% for xenon override)	6.38%
Maximum strength of control rods	14%
Maximum Δk due to adding enriched fuel assembly	0.5%

Heat Transfer and Fluid Flow

Average power density in core coolant	25 kw/liter
Maximum power density in core coolant	62 kw/liter
Steam flow (at 20 mw)	60,000 lb/hr
Average steam voids in heated channels	10%

Heat Transfer and Fluid Flow (Cont'd.)

Average steam voids in total moderator	8%
Average exit steam quality	1.6%
Lb of water circulated per lb of steam	63
Subcooling at core inlet	5.2°F
Non-boiling height of core	40%
Average coolant inlet velocity	2.5 ft/sec
Average heat flux	45,500 Btu/hr ft ²
Maximum heat flux	155,000 Btu/hr ft ²
Estimated burnout heat flux	750,000 Btu/hr ft ²
Average surface temperature of fuel plates	497°F
Average fuel centerline temperature - enriched, thick plates	540°F
Average fuel centerline temperature - enriched, thin plates	529°F
Average fuel centerline temperature - natural, thick plates	526°F
Average fuel centerline temperature - natural, thin plates	513°F

Control Rods

Total number	9
Spacing	12.75 in. x 12.75 in.
Shape	cruciform, 10 in. x 10 in.
Penetration of absorber into core	45 in.
Scram time (unpressurized)	0.5 sec.
Maximum withdrawal rate	(0.4 in./sec) (0.01% k/sec)
Hafnium rods number	5
thickness	0.125 in.
Boron - stainless rods number	4
thickness	0.250 in.
boron content	2% by weight
Combined strength of rods	> 14% k
Strength of center rod	~2.3% k
Strength of innermost 5 rods	~10% k

Pressure Vessel

Diameter, inside	7 ft
Height	23 ft
Working pressure	600 psig
Design pressure	800 psig
Thickness of cylindrical portion	2 $\frac{3}{8}$ in.
Material	SA 212 Grade B boiler plate
Cladding	0.1 in. Type 304 stainless steel
Thermal shield	1 in. 18-8 stainless steel containing 1% boron
Relief valve settings	610, 725, and 750 psig
Total weight	60 tons
Weight of contained water	10 tons

Power Plant

Condenser pressure	2 $\frac{1}{2}$ in. Hg
Flow rate to cooling tower	13,650 gpm, max
Feedwater flow rate	120 gpm
Feedwater temperature	110°F
Generator output voltage	4,160 volts

Steel Building

Diameter	80 ft
Height	119 ft
Net volume	~400,000 ft ³
Design pressure (internal)	15 psig
Design pressure (external)	0.5 psig
Maximum leakage rate at 15 psig	1,000 ft ³ /day
Type steel	ASTM A-201 Grade B Firebox Quality
Sides of tank	5/8 in. thick cylindrical
Bottom of tank	5/8 in. thick elliptical
Top of tank	3/8 in. thick hemispherical
Lining of tank below grade	2 ft (minimum) concrete
Lining of tank above grade	1 ft concrete
Capacity of water reservoir	15,000 gal
Flow rate of sprinkler system	1,000 gpm
Ventilation air flow rate	3,000 gpm
Personnel access	air lock doors
Freight access	bolted and gasketed door

III. DESCRIPTION OF THE EBWR FACILITY

A. Site

The proposed site of the EBWR Facility is located near the center of the total exclusion area of Argonne National Laboratory. The site plan, Figure 1, shows the proposed location with respect to other buildings at Argonne, while Figure 2 shows the principal population centers within a 35-mile radius of Argonne. It is 8/10 mile from this site to the nearest boundary of the Argonne Reservation to the south. Because of the sanitary canals, the uninhabited area extends about one mile beyond the southern boundary of the ANL reservation. In the west, north, and east directions the boundary is approximately $1\frac{1}{2}$ miles from the EBWR site. The only communities within a five-mile radius of the reactor site are Lemont (2500), three miles to the southwest, and the southernmost part of Downer's Grove (15000), five miles to the north. The population distribution within a 10-mile radius of the site is presented in Table II.

Table II

POPULATION DISTRIBUTION

Du Page Site as center

Region	Population (1950 census)**	Area (sq. mi.)	Density (per sq. mi.)
Within $1\frac{1}{2}$ -mile radius*	3,500	6.9	507
$1\frac{1}{2}$ to 5-mile radius	13,000	71.6	182
Total within 5-mile radius	16,000	78.5	204
5 to 10-mile radius	84,000	235.7	357
Total within 10-mile radius	100,000	314.2	319

*During normal working hours.

**The population density is increasing rapidly in this area. A rate of 10% per year would not be surprising.

Data on meteorological, hydrological and seismological conditions at the site are included in Appendices B and F.

Figure 2

TERRITORY SURROUNDING REACTOR SITE

(1) RADII SHOWN FROM
DUPAGE SITE (solid line) AND
PALOS PARK SITE (dotted line)

(2) NEARBY POPULATED AREAS, WAR
PLANTS, MAJOR ROADS AND RAILROADS
ARE INDICATED.

(3) DISTANCE FROM REACTOR IS
INDICATED BY CIRCLES AT FIVE MILE
INTERVALS.

SCALE OF MILES

INDUSTRIAL ORGANIZATIONS

1. GENERAL MOTORS (ELECTROCHEMICAL DIVISION)
2. REYNOLDS METALS CO. (ALUMINUM PLANT)
3. CORN PRODUCTS REFINING CO.
4. UNIVERSAL OIL PRODUCTS.
5. CHICAGO SCREW CO.
6. JOHNSON & JOHNSON.
7. VISCOSI CORPORATION
8. HINDEALE SANITARIUM
9. ALUMINUM CO. OF AMERICA
10. INTERNATIONAL HARVESTER COMPANY
11. ARMOUR & COMPANY
12. REYNOLDS METALS CO.
13. GLOBE OIL CO.
14. TEXAS OIL COMPANY
15. CARBIDE ILLINOIS STEEL CO.
16. UNITED STATES STEEL CO.

LEGEND
AIR PORT
RESIDENTIAL & COMM.
REGIONS OF MAXIMUM
POSSIBLE DAMAGE

RAILROADS AS INDICATED — RIVERS
MAJOR HIGHWAYS
FOREST PRESERVE

5000

20 MILE R.

B. Reactor

1. Mechanical Design

The reactor plant is shown in perspective in Figure 3A. The heart of the reactor is the heat-producing core assembly which is located in the lower portion of the Reactor Pressure Vessel as shown in Figures 3B and 3C. The control rods are actuated from below the vessel by drive mechanisms located in the Sub-Reactor Room. Reactor auxiliary systems, such as water purification and shutdown cooling, are on the lower level outside the primary shield.

Fig. 3A

PERSPECTIVE OF
ENTIRE PLANT

FIG. 3B

PERSPECTIVE OF REACTOR CORE

a. Core:

The core, shown in Figures 4A and 4B, consists of vertically stacked, closely packed fuel assemblies, supported at the bottom by a lower grid plate and guided at the top by a metal guide structure. A fuel assembly can be placed in each of the 4-inch square locations shown. The control rods, guided in metal channels, are pushed up out of the core by control mechanisms located in the Sub-Reactor Room.

Fig. 3C
CUTAWAY PICTORIAL OF EBWR REACTOR AND COMPONENTS

FIG. 4A

The bottom grid is a six-inch* thick Type 304 stainless steel plate installed at the bottom of the pressure vessel to support the weight of the core assembly and to fix the position of the fuel assemblies and control rods. The grid support members are designed to provide the necessary flexibility for thermal and elastic variations. Cooling water holes are provided to reduce temperature rise due to gamma and neutron heating. This grid and the guide structure are removable from the pressure vessel in the event that a revised core arrangement is desired.

Each fuel assembly contains six fuel plates (see Figure 5). Zirconium side plate extensions and stainless steel end fittings make the over-all length of the fuel assembly $77\frac{1}{2}$ in. The fuel assemblies are of two types: "natural," using a natural uranium alloy; and "spiked," using an enriched uranium alloy. The plates for the "natural" assemblies contain a heat-treated ternary alloy of 93.5 weight % natural uranium ($0.72\% \text{ U}^{235}$), $1\frac{1}{2}\%$ niobium, and 5% zirconium, clad with 0.020 in. of Zircaloy-II. The plates for the "spiked" elements contain zirconium with $1.44\% \text{ U}^{235}$, clad with 0.010 in. of Zircaloy-II. The Zircaloy-II side plates are designed to be thick enough to permit welding to the fuel plates, yet weak enough to stretch (creep) if the fuel plates elongate under irradiation. The weakness is achieved by perforating the side plates.

The $3\frac{3}{4}$ -in. square fuel assemblies are centered in their 4-in. square lattice positions by lower stainless steel sockets fitting into the core support plate and by leaf springs in the sides of the top stainless steel fittings. Nine fuel assemblies occupy the 12-in. square grid cells between control rod guide channels.

Since no completely reliable data are available to indicate the proper water-to-uranium volume ratio required in the core to achieve the desired dependence of reactivity on steam content, two different thicknesses of fuel plates are provided. The thin plate (0.174 in. U + 0.040 in. Zircaloy-II clad) assemblies give a water-to-uranium volume ratio of 4.49. The thick plate (0.239 in. U + 0.040 in. Zircaloy-II clad) assemblies give a water-to-uranium ratio of 2.79. Calculations indicate that the reactivity in a 4-ft diameter core having an equal number of thick and thin plate assemblies will decrease very slightly as steam is formed at room temperature. At higher temperatures the decrease in reactivity is of greater magnitude. Increasing the proportion of thick elements makes the void (steam) coefficient of reactivity more negative. The proportions of thick and thin plate assemblies required to give a slightly negative void coefficient at room temperature will be determined experimentally in the initial loading.

The fuel element design just described, in which the cladding and side plates are of Zircaloy-II, is subject to change if tests now under way show that the corrosion of aluminum is tolerable under

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*In the final design this was made $5\frac{1}{4}$ in. thick.

FIG. 5

EBWR FUEL ASSEMBLY

operating conditions. The operating pressure might even be lowered to 300 psig if aluminum or aluminum alloys have sufficient corrosion resistance at the saturation temperature (417°F) corresponding to this pressure. The process for cladding uranium plates with aluminum appears to be much cheaper than any known method for zirconium-clad plates. From the standpoint of inherent safety in excursions, aluminum cladding would be at least as safe as zirconium. The thermal conductivity of aluminum is about 15 times that of Zircaloy-II. The meat of "spiked" elements would also be high conductivity uranium-aluminum alloy. The higher thermal conductivity of aluminum would more than offset the disadvantage of its lower melting point.

The design of the core permits flexibility of loading. There are a total of 148 fuel positions which, when filled, produce a core 5 ft in diameter. With 112 fuel positions filled, the core diameter is 4 ft. The initial loading as calculated is 112 assemblies of which 36 are "natural" and 76 "spiked." Locations outside the 4-ft diameter will be loaded with dummy assemblies of aluminum-nickel alloy. The bottom pilots of the "spiked" assemblies are fitted with keys which permit loading only in certain fuel positions. No more than five "spiked" assemblies can be loaded in any nine-assembly grid cell. This will prevent the possibility of arranging an uncontrollable group of "spiked" assemblies.

The amount of U^{235} required for operating a 4-ft diameter core at 20 mw will be determined experimentally prior to startup. In both plate thicknesses, fuel assemblies containing natural uranium as well as assemblies containing uranium enriched to 1.44% U^{235} are provided. By changing the proportions of these two types, criticality can be achieved in the 4-ft diameter core.

Uniform lateral support for the top ends of the fuel assemblies is provided by the stainless steel end fittings with spring leaves on all sides, thus preventing vibration of the fuel assemblies. These spring leaves, in conjunction with the double-taper seats at the lower ends, provide a degree of freedom of lateral movement of the fuel assemblies in case of fuel plate swelling or fuel assembly distortion. A total of $3/8$ in. distortion in two directions can be accommodated by this means. No provision is made for holding the fuel assemblies down (they are free to expand or contract lengthwise) as no forces are envisioned which could dislodge them from place. When less than a full loading is used, dummy fuel assemblies are provided to fill the vacant holes.

It is to be noted that the flux can be peaked or flattened by putting enriched or natural elements, respectively, in the center of the core. By substituting enriched elements for natural elements, the reactivity can be increased, in case the required enrichment has been underestimated. Likewise by putting natural assemblies exclusively in the center of the reactor, reactivity reduction and power flattening can be achieved simultaneously.

To direct flow under pump pressure through the fuel elements in the core, a cylindrical shroud will be placed around the core and core support plate to form a plenum below the support plate and serve as a flow divider between core and downcomer. This shroud does not seal perfectly, but a small amount of leakage is permissible in an experiment of this sort. The entire core and support structure can be removed and replaced with one more suitable for use with D_2O as coolant-moderator.

Control of the reactor is effected by nine cross-type control rods which can be withdrawn individually or inserted either individually or in groups. All can be scrambled at once. The five most

centrally located (Figure 4B) control crosses are made of hafnium, 0.125 in. thick and 10 in. wide. The four corner crosses are made of 0.250-in. thick Type 304 stainless steel, containing 2% boron by weight, and also 10 in. wide. These two types of rods are expected to be of approximately equal effectiveness for neutron absorption. Hafnium rods are far more expensive than those of the boron-stainless steel type. The nine rods are estimated to have a combined strength of at least 14% k. The total amount of k to be controlled is expected to be about 6.4%, distributed as follows:

Operating voids (10% of coolant channels)	1.10%
Temperature and density effects from room temperature to operating temperature (488°F)	1.65%
Xenon and samarium (at room temperature)	2.63%
Allowance for shutdown xenon, etc.	1.00%
Total	6.38%

When the reactor is operating, the control rods are held in the shroud structure above the core (see Figure 3B). The upper portion of this shroud structure also serves to position the upper ends of the fuel assemblies, as well as providing additional riser height above the fuel assemblies to enhance natural circulation.

The control rods are guided in a fuel and control rod guide structure consisting of a sheet metal assembly 94 in. high mounted on the bottom grid. It is made of 1/8-in. zirconium and stainless steel plates spaced with zirconium and stainless steel bars. The assembly provides nine cross-shaped channels for the control rods and a number of cells for the guidance of the upper ends of the fuel assemblies. The lower portion, in the region of the active core, is zirconium; the upper part is stainless steel. Available data on radiation damage indicates no distortion of this structure need be expected for at least 10 years. The upper portion also acts as a shroud directing the steam/water mixture upward thereby enhancing natural circulation by its chimney effect, aiding the separation of steam from the water, and reducing the amount of steam in the outer annulus which otherwise would be condensed by the feedwater.

b. Control Drive Mechanisms:

The design of the control drive mechanisms, located outside the reactor shield and below the reactor, is shown in Figure 6A. The mechanism incorporates a screw shaft and screw nut for translation of the rotating drive motion into the linear motion of the control rod. A triggering device in the form of a roller latch and solenoid connects the control rod shaft with the screw nut. The solenoid, upon interruption of its electrical circuit, releases the latch and causes the rod to scram.

CONTROL DRIVE MECHANISM

FIG.-6A

SECTION A-A

Only one rod can be withdrawn from the core at a time. The maximum rate of addition of k is 0.01% per sec.

A spring, always in the cocked position when the rods are operable, provides initial acceleration when the rods are released from the drive nut. Weight of the rod is sufficient for the rod to drop into the core in not over $1/2$ second. At plant operating conditions pressure serves as an additional force tending to drive the rods into the reactor. A restricted section in the bottom of the vessel makes it impossible for the rods to drop beyond the core. All rods scram simultaneously.

The seal design used in the mechanism is of the breakdown labyrinth type. This type of seal breaks down the pressure to the atmosphere. The leakage from the seal is collected and returned to the feedwater line. The labyrinth breakdown rings fit on the shaft with definite clearance. The friction of this type of seal is negligible and does not impair the scrambling operation.

The drive for the mechanism will be a standard gear-motor with magnetic brake. Change gears are used to vary the speed as required for the different control rods and in order to limit the maximum rate of addition of Δk to 10^{-4} per second.

The following design criteria have been used to prepare one mechanism for testing.

Seal - Breakdown bushing types. Extensive tests have been conducted previously on this type of seal and are reported in ANL-5220.

Shaft - Armco 17-4 PH, heat treated to maximum hardness and chrome plated.

Motion - 0.1 in. per sec minimum to 0.4 in. per sec maximum.
 10^{-4} Δk /sec maximum (stronger rods move slowly to maintain this maximum).

Position - within 0.020 in.
 Indicator

Travel - 48 in.

Scram Speed - Travel of 42 in. must occur in $1/2$ sec, or a minimum acceleration of $7/8$ g. Latter part of stroke is slower due to action of dashpot.

The facility used to test the mechanism is shown in Figure 6B. Some 5,000 cycles have been made (and over 500 full stroke scrams) without any measurable or visible signs of wear on the seal or shaft. The test conditions were for the most part at room temperature and pressures up to 600 psig. Temperatures up to 490°F have been run. Since the temperature of the reactor thimbles will be held below 200°F, the application of heat is not essential to the test, except for the determination of temperature gradient in the thimble.

c. Reactor Pressure Vessel:

The Reactor Pressure Vessel, illustrated in Figure 7A, is designed in accordance with the applicable section of the ASME Boiler and Pressure Vessel Code (1952 Revision) for the following conditions:

Design Pressure - cylinder and lower head	800 psig
Design Pressure - upper head	800 psig
Working Pressure	600 psig
Maximum Metal Temperature	650°F
Diameter	84 in. ID
Height	25 ft
Closure diameter	64 in. ID
Vessel Material	SA212, Gr. B
Clad Material	AISI Type 347 or 304 Stainless Steel

Within the pressure vessel and beneath the closure is a 1-ft thick stainless steel shield plug. The bolted closure is a 9-in. thick flat steel plate. A double gasket is used, with a leakoff between gaskets which can be connected to the condenser to recover leakage.

The pressure vessel is fabricated of SA-212 Grade B fine-grained carbon steel and lined with 0.10 in. Type 304 stainless steel. The operating pressure is 600 psig. Design pressure is 800 psig. The vessel has been hydrostatically tested at 1200 psig.

FIG.-6B
CONTROL ROD TEST FACILITY

FIG. 7A
EBWR PRESSURE VESSEL

The pressure vessel is supported at its upper end by lugs resting on "C"-shaped compression springs which in turn rest upon a Carilloy T-1 steel framework attached to columns which extend downward to steel "roots" embedded in the concrete floor (Figure 7B). The springs and framework are designed to absorb dynamic loadings which might result from accidental dropping of heavy objects, such as the vessel cover plate on top of the pressure vessel.

The shield plug, penetrated by control rod drive nozzles beneath the reactor, consists principally of dense concrete containing steel punchings as coarse aggregate and finely crushed magnetite ore as fine aggregate. Holes for the nozzles are stepped to prevent radiation streaming. Layers of lead and ferroboron plaster* are used in critical areas.

The number and type of connections to the vessel are summarized below:

Feedwater	3 in. dia.
Lower startup heating and shutdown cooling	4 in.
Liquid level (2)	1½ in.
Poison injection	3 in.
Steam	6 in.
Control rod thimbles (9)	4 in.

Additional pipes attached to the pressure vessel extend through the reactor shield to allow future connections for operation with forced circulation cooling. Four 6-in. diameter inlet pipes pass through the bottom of the shield plug just outside the pattern of nine control rod drive nozzles. Two 12-in. diameter outlet pipes are attached to the vertical wall of the pressure vessel. These pipes make bends in the shielding to eliminate streaming of radiation and emerge in the sub-reactor room. All forced circulation pipes are capped off with blind flanges. Piping, pumps, and valves for future operation with forced circulation are being ordered.

All nozzle connections, control rod thimbles excepted, are to project 12 in. from the pressure vessel wall and are to be at least schedule 80 seamless stainless steel welded into the walls. Thermal sleeves for the vessel connections are to be provided wherever large thermal gradients exist between the vessel wall and the connections. Thimble flanges are to conform with ASA standards for the 600-lb class, with additional provisions for the containment of a radioactive liquid.

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*A plaster containing ferroboron powder substituted for a portion of the sand used as fine aggregate.

Fig. 7B

EBWR PRESSURE VESSEL HOLD-DOWN STRUCTURE

In addition to the core and core-supporting structure, the vessel contains various other internals. The feedwater enters the reactor above the core through a 3-in nozzle and flows into a ring header located in the upper section of the downcomer. A poison solution injection ring, similar to the feedwater injection ring, is located above the normal water level of the vessel.

The nine control rod thimbles extend below the lower vessel head into the sub-reactor room housing the control rod actuating shafts. Cold water from the Reactor Feed Pump discharge is used to cool the thimbles, maintaining a maximum temperature of 200°F. A plate inside the lower pressure vessel head, together with a baffle arrangement around each control rod shaft, acts as a thermal sleeve and also prevents accumulation of sediment in the thimbles.

Since the design of the Reactor Pressure Vessel involves certain conditions not normally encountered in the design of conventional equipment, a brief discussion of these follows.

(1) Applicable Design Code

The Reactor Pressure Vessel has been designated as a "fired" pressure vessel and will be designed and fabricated in accordance with applicable sections of the ASME Boiler and Pressure Vessel Code.

(2) Thermal Heating and Stresses

Absorption of neutron and gamma radiation in the pressure vessel wall causes heating which in turn produces an appreciable steady state thermal stress between the inside and outside walls of the vessel. The thermal stress produced is a direct function of the coefficient of expansion, the modulus of elasticity, and the temperature difference.

A 1-in. thick Type 304 stainless steel thermal shield inside the pressure vessel keeps thermal stresses in the vessel wall at low levels. The stainless steel contains 1% by weight of boron to reduce the thermal neutron flux (and secondary gamma-ray heat generation) in future operation when D₂O is substituted for H₂O. The pressure vessel behind the thermal shield is adequately cooled by natural convection circulation of a water-steam mixture through the 0.5-in. annular gap between the thermal shield and the vessel wall.

A thin sheet of stainless steel inside the upper part of the pressure vessel serves as a shock shield to prevent cold water from the shutdown cooling spray nozzles from striking the hot pressure vessel wall. The pressure vessel behind the shock shield is cooled by steam flowing through the louvers and upward through the annular space.

(3) Irradiation Damage

The fast neutron flux at the inner vessel wall is estimated to be 1.6×10^9 neutrons/cm² sec and the outer wall 1.5×10^8 . Thus the total integrated fast flux at the inner wall for operation equivalent to 20 years at full power is 10^{18} neutrons/cm² and the total integrated thermal flux is 3×10^{19} neutrons/cm².

The choice of material for construction of the vessel is limited by the general lack of data on irradiation damage to steels. The material selected, SA212 Gr. B, is a high-grade carbon steel boiler plate for which data on irradiation damage are available up to a total integrated fast flux of 5×10^{19} neutrons/cm². These data are presented in ANL-4792 and indicate a small increase in the yield and ultimate strength and reduction in ductility and impact strength; however, the experiments were run at 140°F while the reactor vessel operates at 489°F, where irradiation damage will be even smaller than results indicate.

A test of a welded specimen at a fast nvt of 4×10^{19} and 140°F had the following changes in physical properties:

	Yield Strength, psi	Ultimate Tensile Strength, psi	Elongation, %
Unirradiated	43,500	62,500	14.0
Irradiated	71,000	75,000	6.8

Another 140°F test with tensile specimens made from non-welded SA212 gave results as follows:

	Yield Strength, psi	Ultimate Tensile Strength, psi	Elongation, %
Unirradiated	42,000	74,000	19.2
Irradiated	70,000	82,000	14.5

It may be expected that the EBWR vessel will, over a period of years, gradually assume the properties of a very high-strength, low-ductility alloy steel, but no difficulties are anticipated as a result of this change.

(4) Radioactivation of Materials

The bombardment of the pressure vessel walls and internals by neutrons results in an intense level of radioactivity. As a result, direct maintenance or repair of the vessel is impossible. To minimize the corrosion of the vessel, and carryover of corrosion products to the external system, the vessel is lined with corrosion resistant steel, AISI Type 304 or 347, and all internal surfaces are either of stainless steel or zirconium.

(5) Pressure Transients

The Code stipulates that a vessel be protected with safety valves of such capacity that the maximum rate of energy release can be handled. Since a boiling reactor has no theoretical maximum (instantaneous) rate of energy release, but has shown itself to be inherently self-limiting in the event of large sudden increases of reactivity, the overpressure protection consisting of three safety valves discussed in Section III, C, is considered adequate.

(6) Metal-Water Reaction

With the aid of explosion experts at Armour Research Institute, a rather elaborate system was evolved to prevent puncturing the gas-tight steel building in case of an explosive metal-water reaction in the core of the reactor. It was arbitrarily assumed that 25% of the metal in the core reacted with water instantaneously. The tremendous forces developed would rupture the portion of the pressure vessel below the water line, but the shock transmitted through the steam space would be very much reduced. The design adopted should prevent fragments from being blown upward into the building. The pressure would relieve itself by reducing the lower shield to rubble and blowing fragments downward into the basement of the building, where protection to the steel gas-tight building is afforded by a very thick liner of concrete. More details concerning the study made by Armour Research Foundation are given in Appendix G.

Upward movement of the pressure vessel following an explosion in the core is prevented by a 6-in. thick steel plate restrained by three enormous, horizontal H beams (Figure 7B). These H beams are pinned by steel wedges to the six large Carillo T-1 steel columns whose bases are firmly anchored in the very thick concrete forming the basement floor of the building. Laminated blast shields, 21" thick, consisting of alternate layers of steel and compressible wood are interposed between the reactor and the steel columns to protect the columns from the horizontal shock waves. Calculations indicate that, even for the pessimistic explosion conditions assumed, the gas-tight steel building should not be penetrated by fragments.

(7) Accessibility for Maintenance

The vessel design features a large diameter opening to permit the removal of the entire lower grid if desired. A rotary indexing plug arrangement is also provided for loading and unloading any desired fuel assembly without disturbing a particular loading. Calculations indicate that for the loading and unloading procedures contemplated the radiation from the vessel interior is sufficiently attenuated through the 18-in. thick steel indexing plug to permit safe operations. By filling the reactor vessel with water, the indexing plug can be removed to provide direct visual observation of the entire core.

d. Vessel Insulation:

The reactor is provided with thermal insulation consisting of 3 in. of steel wool on all surfaces. Thus, the heat loss from the vessel is limited to 12.5 kw or 0.06% of full power. Protection against corrosion of the steel wool and other exposed metals is provided by maintaining a dry gas atmosphere in this sealed space.

e. Primary Shield:

As mentioned, insulation for the pressure vessel is in the form of 3-in. thick stainless steel wool. Following this, and separated therefrom in a radial direction by an air gap, is a 0.75-in. thick steel tank which serves as an inner form for the shielding. Outside this tank is a 3-in. thick layer of lead in which water-cooled copper tubing is embedded. This is followed by a 0.5-in. thick layer of ferroboron plaster* and about eight feet of biological shield consisting of ordinary concrete and magnetite concrete in varying proportions depending upon the radiation intensity.

The shielding above the reactor consists of a rotating plug within the top head, the head itself, the armor plate, and a removable top shield. The shielding below the reactor consists of the 1/2 in. thick steel tank, 3 in. of lead and 5 ft of heavy concrete. The bottom shield is penetrated by the nine control rod thimbles and forms the ceiling of the Sub-Reactor Room.

Additional shielding at the top of the reactor vessel is provided in the form of a water tank, 12 in. in vertical thickness and 11 ft in diameter. The 6-in. thick explosion load distribution plate forms the top of this tank, and upward forces are transmitted from the pressure vessel cover through the tank by means of a steel ring of 3-in. radial thickness. Provision is made for cooling the water in the tank.

The copper cooling coils located in the lead thermal shield remove the heat transferred from the pressure vessel and the heat generated in the shield by the neutron and gamma attenuation. Under full power conditions, 12.5 kw will be transferred from the pressure vessel and 18 kw will result from the gamma heating. Approximately 98% of the 18 kw will be generated in the lead. The cooling coils form a part of a closed circuit which includes two pumps and filters in parallel and a heat exchanger. Circulating water from the cooling tower will be used as the cooling medium in the heat exchanger. A duplicate set of cooling coils will be placed in the shield for use in case one of the coils develops a leak or becomes plugged. Remote low-flow alarms are provided for each of these coils.

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*A plaster containing ferroboron powder substituted for a portion of the sand used as fine aggregate.

f. Fuel Handling and Storage:

New fuel elements are not radioactive and can be lowered into the reactor vessel directly by means of an unloading rod. When all loading operations are completed the shielding plug is inserted in its hole, the reactor vessel cover installed, and the thermal insulation and top shielding cap set into place.

A positive stop is fastened to the unloading rod to prevent fuel from suddenly dropping into the core during loading operations. In case a fuel element is accidentally dropped, it can drop only to the top of the control rod guides. The stop must be removed at this point before fuel can be inserted into the core. The stop and unloading rod are designed to absorb the total energy of the fuel assembly on a free fall.

Irradiated fuel is removed from the reactor by means of a lead shield transfer coffin on a carriage (Figures 3C and 8). The carriage is mounted on rails embedded in the floor and can be moved horizontally from the reactor to the fuel storage pit. When removing spent fuel elements, the reactor is shut down and cooled to 120°F. After the H beams, top shield tank, and pressure vessel cover are removed, a thick steel plug carrier assembly containing bearings and gears for rotation is placed on top of the vessel. The inner stainless steel shield plug, which contains an eccentric plug to match the rotating eccentric plug in the plug carrier, is attached to and supported by the plug carrier. The two plugs are then rotated to position the discharge opening in the two plugs over the fuel assembly to be removed. The coffin is located over the discharge opening, a telescoping fuel grapple is lowered and engaged, and the assembly is raised by a winch into the coffin. The coffin is moved to a position over the water-filled fuel storage pit (Figure 9) and the fuel assembly is lowered through a shielded chute into the pit, where it is placed in a boron-stainless steel storage rack.

The well is a shielded concrete pit adjacent to the reactor shield. The water in the fuel storage well provides adequate shielding above the fuel elements. The bottom of the well is equipped with a small transfer cart and a rack for storing spent fuel elements. The transfer cart is moved back and forth on tracks by means of an endless cable and crank. The rack is a light stainless steel frame with sheet cadmium facings to prevent criticality. The well has water inlet and drain connections which may be used to circulate the water through filters or ion exchangers if this proves necessary.

Used fuel elements stored in the well are cooled by thermal convection currents until the gamma heating diminishes to a level where the fuel elements can be shipped to a chemical processing plant. Meanwhile the inactive ends of the fuel elements can be cut off and other operations performed in the bottom of the well by direct underwater manipulation.

FIG.-8
FUEL UNLOADING COFFIN

VERTICAL CROSS SECTION OF
PRESSURE VESSEL & SHIELD FOR EBWR

Fig. 9

g. Startup Heating:

The reactor vessel and its contents must be heated to the startup temperature (325°F) before the control rods can be withdrawn. The heating is done by boiling reactor water in the tubes of the Startup Heater, using laboratory steam on the shell side. The flow in the Startup System is by natural circulation since the Startup Heater is located below the reactor water level. The Startup Heater has 84.6 ft² of heat transfer area, which is sufficient to heat the vessel and its contents from 70°F to 325°F in about three hours using 190 psia steam (378°F). When the reactor is in operation, the laboratory supply steam and condensate return lines to the cooling tower are isolated from the startup heater to eliminate any possibility of radioactive contamination outside the gas-tight building in case of a rupture of or leak in the heating unit. For such a casualty the leakage is direct to the building.

h. Shutdown Cooling:-

The decay heat which must be removed from the reactor after shutdown drops from about 6% of operating power immediately after shutdown to about 1% after 7 hours. During a normal shutdown the reactor is cooled by bleeding steam to the main condenser until the reactor temperature has been lowered to about 220°F. The reactor feed pump and the circulating water pump are in operation during this time. Further cooling is done using secondary coolers in the reactor cleanup loop. These coolers are designed to hold the reactor water temperature at or below 120°F with the decay heat at 1% of operating power. In event of a shutdown caused by a loss of electrical power, the Emergency Shutdown cooler which is located in the steam separator is used to remove the decay heat. Water flowing by gravity from the 15,000-gal water storage tank is boiled inside the tubes, condensing reactor steam on the shell side. The condensate returns to the reactor by gravity flow. The steam-water mixture is returned to the overhead tank to heat up the 15,000 gal of water; circulation is established by the difference in density in the water supply and steam-water discharge lines. The main steam valve is closed when this cooler is in use so that the reactor water level will not fall. The cooler is designed to remove 1,000 kw (5% of operating power) when the steam pressure is at 600 psig.

A system (see Figure 24) is provided for automatic injection of water into the reactor if the pressure and water level are both below predetermined limits. A water line from the laboratory main leads to a tank which contains a large quantity of coarse boric acid crystals in a perforated basket. The pipe then continues to the spray ring in the steam dome of the reactor. Both the piping and the boric acid tank are protected against damage from shock and missiles. Two solenoid valves, one on either side of the boric acid tank, must be opened to start the injection of boric water. Both valves are automatically opened by a combination of low-water level (6 in. above the top of the core) signal and low reactor pressure. When

both valves are open the water flows through the tank where it dissolves a small amount of boric acid and then to the reactor. This system assures that the reactor core will always be cooled with water when the pressure is low. This situation could arise as a result of a rupture of the pressure system. In this case, the reactor would be scrambled automatically by rapid decrease in pressure or low water level. While the reactor is blowing down to the condenser the steam passing by the fuel plates will prevent dangerous overheating. When the blowdown has progressed to a point where the reactor pressure and water level have reached their trip valves the injection of borated water will begin.

i. Boron Injection:

Eight of the control rods (excluding the center one) are grouped into four pairs. The two members of each pair are functionally identical. If at least one rod from each pair does not go in following a scram signal, automatic injection of a concentrated solution of boric acid begins. At the discretion of the operator, the concentrated boric acid solution can also be injected into the reactor vessel through the boric acid spray ring.

The boric acid solution is held in a tank (located inside the reinforced concrete missile shield on the main floor) at 1600 psig with sufficient air above it so that 160 gal of solution can be injected into the reactor vessel within 20 sec when the operating pressure is 800 psig. The volume of concentrated boric acid solution (51.5 gm H_3BO_3 /liter) required to overcome $12\frac{1}{4}\%$ reactivity in the EBWR is 139 gal.

A small air compressor, permanently connected to the boric acid injection tank, is used for pressurization. If the pressure falls below 1600 psig the operator is notified of this fact by an alarm signal in the control room.

j. Reactor Water Purification System:

The reactor water is continuously cleaned up by a side-stream purification system. Water (at 489°F) is removed from the reactor at a rate of 10 gpm, cooled to 250°F by a regenerative heat exchanger, further cooled to 110°F by a secondary cooler, and then flows through a prefilter. This filter is a fine cotton cartridge type and, in addition to removing particulate matter as small as 2 microns, is expected to partially remove colloidal iron and aluminum. The filtered water flows through a mixed-bed ion exchanger which removes essentially all ionic and most colloidal impurities. A 2-micron cotton after-filter prevents carryover of resin into the reactor. The demineralized water from the resin bed is pumped back into the reactor feed-water line through the regenerative heat exchanger where its temperature is raised to 300°F.

2. Instrumentation and Control

a. Control Rod Drives:

Each of the nine rod drives is operated by means of a reversible electric motor to give a rod speed, in terms of reactivity, of $10^{-4} \Delta k/\text{sec}$. One of the nine, probably the center one, will be selected as a regulating rod. A separate set of switches controls this rod. An additional set of switches controls the remaining eight rods in conjunction with a rod selector switch which prevents the simultaneous operation of any two or more rods to increase reactivity. It is possible to drive eight of the nine rods in under power simultaneously to decrease reactivity.

Each drive is provided with an electromagnet which holds in a latching device attaching the control rod extension to the drive linkage. Upon release of the latch, the rod is given initial acceleration by means of a spring to aid gravity and reactor pressure in returning the rod to the full-in or minimum k position in the reactor. A travel of 42 in. occurs in not more than $1/2$ sec, which corresponds to a net constant acceleration of $7/8$ g. Total scram time when scrambling from any intermediate rod position can be determined using a constant acceleration of $7/8$ g, and the time required to reach any specified intermediate point in the travel during a scram is no more than that determined using a constant acceleration of $7/8$ g. A timer is provided to measure scram time from any initial position to the full-in position for any rod selected. Each rod is timed during a reactor interlock startup check to determine whether or not it is sticking. The rod latch release used for this test releases only the rod selected, thus permitting individual rod-drop reactivity calibrations. A scram operator is provided which releases all the rods into the reactor. Either of these operators cause all nine drives to run to the full-in position, driving in any rods that may have hung up and not dropped into the full-in position. Interlocks operated by each rod at its full-in position are arranged to prevent the regaining of rod drive control power after a scram unless all rods are at the full-in position.

b. Rod Position Measurement, Indication and Limit Switching:

Since rod position measurement is by synchro with the transmitter geared to the rod drive, the position indicated is that of the drive. However, the rod cannot be farther out of the core than the drive position, and, if the rod is unlatched, it goes to the full-in position. A limit switch used with the rod latching device operates an indicating light to show that the rod is picked up and latched to the drive. Another limit switch operated by the rod at its full-in limit operates an indicating light to show that the rod is in, and is also used for rod scram timing and in the start-up interlock sequence which prevents rods being withdrawn after a scram unless all rods have dropped to the full-in position. Additional limit switches

provide for stopping the drive motors at the in and out limits of travel and will operate an indicating light when the drive is at the out limit. An indicating light is operated by the rod selector switch to indicate which rod drive can be operated. A number of warning lights throughout the plant area indicates that the rod drive control power is on, and a one-minute alarm bell sounds each time the control power bus is energized, providing plant-wide warning that the reactor is operating or can be operated.

c. Safety and Alarm Circuitry:

Table III is a compilation of the signals that cause loss of control power with resultant insertion of all rods. The signals which cause alarm only are listed in Table IV.

A scram by-pass selector switch will be used for test purposes in conjunction with the high flux trips and other trips requiring checking with the reactor operating. It may be used only when a key switch provided for the purpose is turned on.

In addition to the safety or scram circuits listed in Table III, a group of start-up circuits is used to enforce proper basic reactor start-up procedure involving instruments and valves which do not appear on the annunciator. The start-up circuits are:

1. Flux instrument shunt set for most sensitive scale.
2. All rods at maximum shutdown position.
3. Main steam stop valve open.

d. Power and Period Measurements:

Power and period measurements are made using nine instrument channels which include those used for start-up. Two BF₃ proportional counter channels are used to start from source level, covering four decades of flux. These channels are equipped with scalars and mechanical registers as well as counting rate meters. The next higher level channel uses an electrometer instrument with compensated ion chamber, covering five decades of flux, overlapping the proportional counter region two decades. The intermediate power range channel, using a microammeter instrument with compensated ion chamber, overlaps the electrometer chain, and covers four decades of flux. The power range channel covers three decades by means of a galvanometer and Ayrton shunt switch which are supplied with current by an uncompensated ion chamber. The galvanometer shunt includes a set of interlock contacts on the most sensitive scale position to be used in the reactor start-up circuits; it is necessary to set the shunt to this position before it is possible to start the reactor. Three additional high-flux channels using uncompensated ion chambers incorporate adjustable level safety trip circuits to scram

Table III
SAFETY AND ALARM SIGNALS

Rod Insertion Signals

1. Plant area radioactivity high
2. Reactor flux high - No. 1* electronic relay
3. Reactor flux high - No. 2* electronic relay
4. Reactor flux high - No. 3* electronic relay
5. Reactor pressure high
6. Reactor pressure rate of change above set value
7. Reactor water temperature below 325°F
8. Condenser pressure high
9. Reactor water level high or low
10. 700-psig relief valve open
11. 725-psig safety valve open
12. 750-psig safety valve open
13. Reactor period under 5 seconds**
14. Reactor flux high - No. 1 magnetic sensing instrument
15. Reactor flux high - No. 2 magnetic sensing instrument
16. Reactor flux high - No. 3 magnetic sensing instrument
17. Spare
18. Spare
19. Spare
20. Spare

---

*Two of these electronic trips must operate in coincidence to produce a scram.

**This will be used only for start-up with a key switch provided to lock it out of operation.

Table IV

ALARM SIGNALSAlarms

1. Selected temperatures - Recorder No. 1
2. Selected temperatures - Recorder No. 2
3. Steam line valve closed
4. Reactor steam pressure 3% above or below the control point
5. Condenser circulating water differential pressure above or below normal
6. Feedwater pump off
7. Reactor water temperature entering ion-exchanger high
8. Resistivity of reactor water entering ion-exchanger low
9. Emergency boron injection system level or pressure low
10. Shield cooling water flow low
11. Shield cooling water temperature high
12. Ventilation exhaust duct activity high
13. Reactor water level outside normal range
14. Cooling tower basin temperature low
15. Cooling tower basin level low
16. Radioactivity in condenser exhaust air high
17. Radioactivity in water purification circuit high
18. Pressure drop across feedwater filters high
19. Feedwater temperature high
20. to 35. Spares

the reactor when the flux exceeds normal. Each channel contains a magnetic trip of the meter relay type as well as an electronic trip. Operation of any one of the magnetic trips causes a scram. However, the electronic trips are arranged to require simultaneous operation of the trips in at least two of the chains in order to produce a scram. This coincidence requirement is introduced to reduce the probability of false scrams. The magnetic trips are present to increase reliability. The ninth channel contains a logarithmic amplifier operated from a compensated ion chamber, providing level indication over seven decades, and is equipped with differentiation to provide period indication and a safety trip signal. As a result of the boiling process, which causes the reactor flux seen by the detectors to fluctuate, period safety trip signals are considerably less reliable here than on other types of reactors. Therefore, provision is made for a key switch to lock out this trip during steady-state operation, at the discretion of the supervisor.

Figure 10 presents an over-all picture of the coverage and overlapping of the instrument channels. The scale is given in terms of reactor heat generation power expressed in watts. A flux attenuation of four decades was used between the reactor core and the instrument positions. The normal design sensitivity of the flux detectors is 10^{-14} amp per $n/(cm^2)(sec)$.

The flux detectors for the nine channels will be located in six horizontal instrument holes provided in the reactor shield. The axis of each of these holes is tangent to the reactor pressure vessel exterior. The flux detectors' sensitivities are established by adjusting their location within the instrument hole. The proportional counters are provided with a means of insertion and withdrawal so that they may be withdrawn after startup to prevent damage by exposure to high fluxes.

e. Temperature Interlocks:

An important reactor start-up requirement is that the reactor water be heated to 325 F by external heaters before the control rods can be withdrawn. The inherent reactor safety resulting from bubble control of reactivity is increased by this preheating process, since the reactor void sensitivity is much greater at the higher temperatures. The time lag between power output and bubble formation is essentially eliminated. To insure fail-safe operation of this circuit, the power to operate the contacts is derived from the heat in the reactor water directly by means of a thermocouple placed in the start-up heating line leaving the bottom of the reactor at the point where this pipe emerges from the reactor shield. The thermocouple operates a pyrometer featuring an indicating meter-type relay with magnetic low limit contacts which require manual electrical reset once they close. If the reactor water temperature falls below the set limit, the contacts close and either prevent a startup or, if the reactor is operating, cause a scram. Once this happens it is necessary to raise the temperature above the set point and then press the reset button to clear the interlock

INSTRUMENT CHANNELS

1. COUNTER NO. 1
2. COUNTER NO. 2
3. ELECTROMETER
4. MICRO-MICRO-AMMETER
5. GALVANOMETER
6. PERIOD METER
7. SAFETY TRIP NO. 1
8. SAFETY TRIP NO. 2
9. SAFETY TRIP NO. 3

FIG. 10
FLUX INSTRUMENTATION

relay. A second relay is used to monitor the interlock relay power supply and provide fail-safe operation. A duplicate pyrometer complete with thermocouple will be provided for increased reliability.

f. Pressure and Rate of Change of Pressure Interlocks:

The main steam pressure regulating system is expected to control steam pressure to within about plus or minus one per cent of 600 psi under normal operating conditions. A steam pressure-measuring element with electrical transmitter is installed in the main steam line as close to the reactor shield as practicable. The electrical signal from this transmitter operates a recorder in the control room. The recorder is equipped with alarm contacts which respond when the pressure deviates by some preset amount such as plus or minus 3 per cent from the set pressure, operating an alarm to warn the operator. If the pressure continues to build up after exceeding the high alarm limit and the operator is unable to restore operation of the pressure-regulating system, a pressure gauge equipped with mercury switch scrams the control rods into the reactor when the pressure reaches 625 psig. This gauge is located near the pressure-measuring element described above. If, for some reason, this pressure switch does not function to produce a scram, the pressure will continue to rise until it reaches 700 psig which is the set point of the pressure relief valve. The relief dump valve opens, quenching the reactor by blowdown until the pressure drops to a preset pressure such as 500 psig at which point the relief valve becomes a back pressure regulator and maintains this pressure. A set of interlock contacts on the dump valve insert the control rods into the reactor if the valve opens as described above. If for any reason the dump valve fails to operate, two safety relief valves set at 725 and 750 psig open, operating interlock contacts to insert the control rods if they have not already been inserted.

The rate of change of steam pressure must not be too rapid because of the positive feedback on reactivity. An instrument to measure rate of change is provided to operate interlock contacts, dropping the rods into the reactor if the rate of change exceeds a set value. The primary element, which will be mounted adjacent to the 625 psig pressure switch described above, is a bourdon tube element coupled to a potentiometer supplied with a regulated dc voltage across its fixed terminals. An R-C derivative circuit is connected between its arm and the arm of a second potentiometer whose fixed terminals are supplied with the same dc voltage. The output of the derivative circuit is connected to a sensitive meter-type relay with magnetic locking contacts which must be reset after each closing. These contacts actuate an interlock relay which causes the rods to be inserted into the reactor. The second potentiometer is to balance out disturbances from the dc voltage supply. An additional relay monitors the voltage supply and gives fail-safe operation.

g. Reactor Water Level Measurement and Control:

The level of water in the reactor is measured in a separate leg extending outside the reactor shield. A differential pressure cell measures the level continuously.

Safety interlocks to shut down the reactor if the water level is too high or too low are provided in the form of float switches. Industrial television is used to view a sight glass from the control room, backing up the level measurement by the differential pressure cell which operates the feedwater flow controller. The differential pressure cell ignores the changes in the water level that result from temperature change during warmup so that the feedwater controller does not overflow the reactor when the temperature is low. A recorder connected to the differential pressure cell contains alarm contacts to warn of impending trouble before the float switches cause shutdown.

The feedwater can be manually controlled by a motor-operated throttling valve in the reactor feedwater line. Alternatively a control system incorporating the components necessary for a three-element feedwater controller, steam flow, feedwater flow and reactor level can be used for automatic control of feedwater flow. An automatic valve separate from the motor operated valve is operated by the feedwater controller. The position of both valves is indicated in the control room.

h. Temperatures:

All temperatures of auxiliaries closely allied to reactor operation, such as the reactor water ion exchange system, shield cooling system, etc., are measured by thermocouples and brought to suitable temperature indicators or recorders.

3. Physics*

Two-group calculations have been performed on the UNIVAC in order to determine the critical loading and reactivity changes with temperature, density, and steam voids. The effect on flux distribution and critical mass of non-uniform distribution of steam voids in the operating reactor has been determined.

Thermal disadvantage factors within fuel elements have been based on the spherical harmonics P-3 approximation to transport theory.

---

*For the most part the contents of this section are the calculations based on the use of "spiked" fuel of $2\frac{1}{2}\%$ enrichment in U^{235} . All fuel plates were of uniform thickness, there being 71 natural uranium assemblies and 42 enriched uranium assemblies. The change to the design of fuel plate described in Section III. B. 1. introduced only minor changes in the nuclear characteristics of the core.

In addition inter-cell disadvantage factors between the natural and enriched fuel elements have been calculated from diffusion theory.

Table V.

Quantities pertinent to the physics design are listed in

Table V

THERMAL DISADVANTAGE FACTORS

Material	Volume Fraction in Cell		Relative Thermal Flux	
	Natural Elements	Enriched Elements	Natural	Enriched
U ²³⁵	0.00175	0.00175	1	1.153
U ²³⁸	0.2412	0.0	1	--
H ₂ O	0.6074	0.6074	1.091	1.212
Zr	0.1409	0.3908	1.036	1.161
Nb	0.00875	0.0	1	--

Resonance escape probabilities were computed from the formula $P = \exp \left[- \left(\frac{1}{f_r} - 1 \right)^{-1} \right]$, where f_r is the "resonance utilization."

The resonance absorption cross section of U²³⁸ was obtained from

$$\int \left(\frac{\sigma U^{238}}{E} \right)_{\text{eff}} dE = 7.99 \left[(1 + \beta \Delta T) + \frac{3.14}{0.1 + \frac{M}{S(1 - c)(1 + 2 \gamma \Delta T)}} \right]$$

The above was obtained by correcting the resonance integral of U²³⁸,

$$\int \left(\frac{\sigma U^{238}}{E} \right)_{\text{eff}} dE = 7.5 \left[1 + \frac{3.4}{0.1 + \frac{M}{S}} \right] \text{ barns,}$$

for the effects of the zirconium alloyed with the U²³⁸, the Doppler effect ($\beta = 1.4 \times 10^{-4}/C$), metal expansion ($\gamma = 2.84 \times 10^{-5}/C$), and resonance shielding. For flat plates the resonance shielding factor is $C = 2E_3 (t/\lambda)$, where t/λ is the plate separation in mean free paths.

The fast fission effect ϵ was based on calculations for parallel plates. These calculations compare favorably with the results obtained at BNL¹ for the H₂O/uranium ratios of interest.

¹G. Price, "Fast Effect Measurements," BNL-Log No. C-7762.

The calculation of τ was based on the experimental values obtained at BNL² for water-uranium lattices, corrections being made for the zirconium and niobium concentrations and for voids.

The calculated parameters for the four-foot diameter, four-foot height uniformly "spiked" reactor are summarized in Table VI. The reactor was assumed to be at 600 psig, 489°F in the hot and operating condition and at 68°F in the cold condition.

Table VI
CALCULATED PARAMETERS FOR THE EBWR CORE

Constant	Operating 14.2% Steam	"Hot" Without Boiling	"Cold" With Xe + Sm	"Cold" Without Xe + Sm
ϵ	1.0402	1.0362	1.0308	1.0308
η	1.5458	1.5471	1.5752	1.5752
f	0.8264	0.8101	0.7824	0.8065
p	0.8364	0.8583	0.8876	0.8876
L^2	8.886	7.710	3.411	3.516
τ	61.37	50.65	37.69	37.69
k_{∞}	1.1113	1.1146	1.1276	1.1624
k_{eff}	1.000	1.0167	1.0515	1.0839

The uniform mixture of 42 enriched fuel elements and 71 natural fuel elements would provide sufficient reactivity for operation with 14.2% average steam void and equilibrium concentrations of xenon and samarium. Allowing for at least 1/2 hour of shutdown xenon override the total loading of the enriched elements is 17.8 kg.

At 20-mw operation the average fuel flux is 1.6×10^{13} $n/(cm^2)(sec)$ with the power derived as follows:

²H. J. Kouts, J. Chernick, I. Kaplan, "Exponential Experiments on Light Water Moderated, 1% U²³⁵ Lattices," BNL-Log No. C-6687.

37% from the enriched elements

56% from U^{235} fission in natural elements

7% from U^{238} fast fission in natural elements*

Figures 11 and 12 show the calculated radial and axial flux distributions. Radially the steam voids were assumed to be proportional to the thermal flux. Axially the steam voids were assumed to be proportional to the integral of the thermal flux from the point at which net boiling begins. The average steam voids in the core were taken to be 11.2%. More recent heat transfer data indicates that the average void may be about 17.5%. Hence, while the flux plots are not quantitatively correct, they do indicate the effect on flux of the non-uniform void distribution.

a. Reactivity:

The reactivity has been calculated as a function of water density for both the cold (68°F) and hot (489°F) reactor. The results are shown in Figure 13. Temperatures, both neutron and metal, have been assumed constant as a function of void fraction in the water, except in the transition from the cold to hot reactor, with the associated change in metal temperature from 68°F to 489°F and the associated change in water density from 1 to 0.8 gm/cc.

Corresponding reactivity versus void curves for the reactor loaded to its full capacity (5 ft diameter) are shown in Figure 14. It is to be noted that the effect of a given reduction in density is reduced when the diameter of the core is increased from 4 ft to 5 ft.

Figure 15 indicates how the reactivity change with voids varies as the size of the zone containing enriched elements is reduced. The over-all diameter of the core was kept at 4 ft. With the proposed spiked loading, it is possible to achieve increased sensitivity to steam formation

*With the use of fuel assemblies as described in Section III. B., the power distribution for operation at 20 mw was calculated to be as follows:

	18 thin natural uranium assemblies	18 thick natural uranium assemblies	38 thin enriched uranium assemblies	38 thick enriched uranium assemblies
% of total power	11.04	14.80	35.88	38.27
Megawatts	2.208	2.96	7.176	7.654
Megawatts per fuel assembly	0.1226	0.1645	0.1887	0.2014

4' D. UNIFORM SPIKING

FIG. 11
RADIAL FLUX DISTRIBUTION

UNIFORM SPIKING

FIG. 12
AXIAL FLUX DISTRIBUTION

REFERENCE REACTOR 4' DIA. X 4' HEIGHT
UNIFORM SPIKING

FIG. 13
REACTIVITY VS DENSITY

FULLY LOADED REACTOR 5' DIA.
UNIFORM SPIKING

FIG. 14
REACTIVITY VS VOIDS

FIG. 15
REACTIVITY VS VOIDS FOR VARIOUS
DIAMETERS OF ENRICHED ZONE

by concentrating the spikes near the center of the core. This gain in sensitivity is obtained at the expense of a higher maximum to average power ratio however.

Reactivity changes associated with reactor operation are summarized in Table VII.

Table VII

REACTIVITY CHANGES

	Reactivity Change, %
Temperature effects, cold to hot Xe + Sm, equilibrium 17.5% operating steam voids 1/2-hr transient Xe override	3.48 3.36 2.23 0.23
Total reactivity to control	<u>9.30%</u>

b. Neutron Lifetime:

The thermal neutron lifetime was calculated from the approximate formula

$$\ell^* = \frac{1}{v\bar{\Sigma}_a(1 + L^2B^2)}$$

For the cold reactor with neutron temperature of 358°K, an average neutron velocity $v = 2.4 \times 10^5$ cm/sec; $\bar{\Sigma}_a = 0.06932$; $1 + L^2B^2 = 1.006$; the calculated neutron lifetime ℓ^* is 6×10^{-5} sec. This lifetime was used in the calculation of the asymptotic period versus reactivity curve shown in Figure 16.

c. Control Rods:

The worth of control rods was based upon calculations using the Nordheim and Scalettar³ method with the fast flux boundary conditions changed so that no fast absorption was permitted. This then gave the worth of thermally "black" rods. Calculations were then made which compared the effectiveness of hafnium (which is both a thermal and resonance neutron absorber) to cadmium. It was found that a 1/4-in. plate (50% Zr, 50% Hf) was 18% more efficient than a similar thickness of cadmium in capturing neutrons in a spectrum consisting of a Maxwellian joined

³ L. W. Nordheim, R. Scalettar, "Theory of Pile Control Rods,"
MDDC-42.

FIG. 16
ASYMPTOTIC PERIOD (τ) VS
EXCESS REACTIVITY (k_{ex})

with a "1/E tail." Combining the results of these calculations it was found that the insertion of nine cross-shaped control rods with an 8-1/2 in. span, a blade thickness of 1/4 in. (50% Zr, 50% Hf) would cause a 12.5% reduction in reactivity in the cold reactor.

The central cross inserted alone was worth 1.9% reactivity while an outermost rod alone was worth 0.47% reactivity.

The effect of burnup of the hafnium rods has been calculated. Exposure to the average operating flux for five years would produce a reduction in effectiveness of 18%.

d. Maximum Concentration of Enriched Elements:

The nine cross-shaped control rods divide the central portion of the reactor core into cells containing nine fuel elements. Surrounding such a cell with an absorber which captures all the thermal neutrons, but allows free passage of fast neutrons between the cell and an infinite medium of $k_{\infty} = 1$, yields a maximum k_{∞} for the material within the cell. Calculations on the above basis indicate that no more than five enriched fuel elements can be allowed in any cell even though the cell is surrounded by control rods.

e. Addition of Enriched Fuel Element:

The effect of replacing a natural element by an enriched element has been calculated on the UNIVAC. Since the natural element is more of a poison than is water, the result of this calculation should be an upper limit on the effect of the insertion of an enriched element into a water hole.

The effect of adding an enriched element is calculated to be no more than 1.16% reactivity in the center of the reactor and 0.49% reactivity in the average fuel position.

f. Effect of Metal Temperature on Reactivity:

Differentiating the resonance escape probability with respect to metal temperature yields

$$\frac{1}{p} \frac{\partial p}{\partial T} \approx \left\{ \frac{7.99}{\int \left(\frac{U^{238}}{\sigma_a} \frac{dE}{E} \right)_{\text{eff}}} \beta + \left[1 - \frac{7.99}{\int \left(\frac{U^{238}}{\sigma_a} \frac{dE}{E} \right)_{\text{eff}}} \right] 2 \gamma \right\} \ln p,$$

where the first terms is the Doppler effect and the second the effect of metal expansion on the surface to mass ratio.

The Reactor Handbook lists both measured and calculated values of the effect of metal temperature on the volume resonance integral of natural uranium. From the data of Creutz, Mitchell, Dancoff, Table 1.6.4 of the Handbook, and Chernick, BNL-152, one obtains values of $\frac{1}{A} \frac{\partial A}{\partial T}$, where A is the volume resonance integral. These values yield

$$\beta = \frac{1}{A} \frac{\partial A}{\partial T} = (1.4 \pm 0.3) 10^{-4}/^{\circ}\text{C} \quad ,$$

where the data are averaged and the limits indicate the maximum spread in the data.

For the cold reactor with $p = 0.8876$

$$\frac{\Delta k}{k} = \frac{1}{p} \frac{\partial p}{\partial T} = - (0.99 \pm 0.21) \times 10^{-5}/^{\circ}\text{C} \quad \text{due to Doppler effect.}$$

$$\frac{\Delta k}{k} = \frac{1}{p} \frac{\partial p}{\partial T} = - 0.27 \times 10^{-5}/^{\circ}\text{C} \quad \text{due to metal expansion with}$$

$$\gamma = 2.84 \times 10^{-5}/^{\circ}\text{C}.$$

Additional effects of metal expansion are the change in $\text{H}_2\text{O}/\text{uranium}$ ratio and the elongation of fuel plates.

Metal temperature coefficients of reactivity are summarized in Table VIII:

Table VIII

METAL COEFFICIENTS OF REACTIVITY

Change Due To:	Reactivity Change $\frac{\Delta k_{\text{eff}}}{k_{\text{eff}}}$
Doppler effect	$-0.99 \times 10^{-5}/^{\circ}\text{C}$
Metal expansion:	
Surface to mass ratio	$-0.27 \times 10^{-5}/^{\circ}\text{C}$
Elongation of plates	$+0.11 \times 10^{-5}/^{\circ}\text{C}$
$\text{H}_2\text{O}/\text{uranium}$ ratio	$-0.05 \times 10^{-5}/^{\circ}\text{C}$
Total metal temperature coefficient	$-1.20 \times 10^{-5}/^{\circ}\text{C}$

g. Xenon and Samarium Poisoning:

The change in reactivity (Figure 17) due to buildup of xenon and samarium after initial startup of the reactor was calculated as a function of time on the basis of an average flux $2.15 \times 10^{13} \text{ n/(cm}^2\text{)(sec)}$. This particular flux was chosen as probably more representative of the statistically averaged equilibrium value for xenon (and therefore for xenon and samarium). The equation for $\Delta k_{\text{eff}}/k_{\text{eff}}$ is:

$$\frac{\Delta k_{\text{eff}}(t)}{k_{\text{eff}}} = - \frac{\text{total fissions}}{\text{thermal fissions}} \left(\frac{\bar{U}_{\text{as}}^{235}}{\bar{\Sigma}_{\text{as}}^{\text{Total}}} \right) \left(\frac{1}{1 + L^2 B^2} \right) \frac{\bar{\Sigma}_{\text{as}}^{\text{Xe}} + \text{Sm}(t)}{\bar{\Sigma}_{\text{as}}^{U^{235}}}$$

For the calculation of the equilibrium xenon for full power operation and of the xenon buildup after complete shutdown from this equilibrium operation, a bare sphere with the same buckling as the operating reactor was chosen and the two-group perturbation formula for parasitic absorption was applied, the core being taken as a concentric subsphere equal in volume to the EBWR.

$$\frac{\Delta k_{\text{eff}}}{k_{\text{eff}}} = - \frac{\int_{\text{core}} \left[\frac{\bar{\Sigma}_{\text{as}}^{\text{Xe}} + \text{Sm}}{\bar{\Sigma}_{\text{as}}^{U^{235}}} \times \frac{\bar{U}_{\text{as}}^{235}}{\bar{\Sigma}_{\text{as}}^{\text{Total}}} \right] \phi_s(r) \phi_s^*(r) r^2 dr}{\frac{k_{\infty}}{p} \int_{\text{core}} \bar{\Sigma}_{\text{as}}^{\text{Total}} \phi_s(r) \phi_f^*(r) r^2 dr} \times \frac{\text{total fissions}}{\text{thermal fissions}};$$

the results are presented in Figure 18.

h. Long-Term Irradiation:

The time variation in reactivity due to the burnup of U^{235} , the formation and burnup of the Pu isotopes, and the formation of long-lived fission products (with Xe^{135} and Sm^{149} assumed constant at equilibrium values) has been calculated for the first 18 months at full power operation (Figure 19). With an initial conversion ratio of 0.69 and an assumed stable fission product cross section of 100 b per fission, the reactivity

$$\frac{\Delta k_{\text{eff}}}{k_{\text{eff}}} = \frac{\Delta \bar{\eta}}{\bar{\eta}_0}$$

risks to a maximum of +0.6% after 4 months of operation. At 8-1/2 months the reactivity returns to its initial value, after which it continues to decrease indefinitely (no reloading assumed).

FIG. 17
REACTIVITY LOSS DUE TO X_{e135} AND S_{m149}
VS TIME AT FULL POWER OPERATION

FIG. 18
 REACTIVITY LOSS DUE TO $X_{e^{135}}$ AND $S_{m^{149}}$
 VS TIME AFTER SHUTDOWN FROM FULL POWER

FIG. 19
VARIATION IN REACTIVITY
WITH TIME

INITIAL CONVERSION RATIO = 0.69

4. Heat Transfer

a. Natural Circulation Mechanism:

The greatest problem connected with use of four different fuel plates is the large variation in heat generation per unit volume of coolant. Not only is the heat generation greater in the thick plates but the associated volume of coolant is smaller. The water channels between thin plates are 0.428 in. wide as compared with 0.360 in. for the thick plates. The average is 0.394 in. The ratio of average to minimum is 1.09. This factor, together with the ratio of 1.13 between the heat generation in a thick enriched assembly and an average assembly, gives a combined effect of $(1.09)(1.13) = 1.23$ as the power density in the water channels of a thick enriched assembly relative to an average assembly in the same location. The average power density throughout the core is 25 kw per liter of coolant at 20-mw operation. The maximum power density in a thick enriched fuel assembly adjacent to the central control rod channel is 62 kw/liter. This is in the neighborhood of the maximum power density in the channels of the Borax-III reactor.

Removal of heat from the core is accomplished by natural circulation boiling. Water in the core is heated and evaporated, producing a relatively low density mixture of steam and water. The water in the annulus around the core, denoted as the downcomer, does not contain steam voids and is at full water density. The difference in density between the downcomer and core will cause circulation of water upward through the core. Electrically heated experiments at atmospheric pressure with geometries similar to EBWR indicate that power densities of at least 100 kw/liter of water can be removed by natural circulation boiling. At higher pressures the power density can be substantially increased. This mechanism is stable at low power densities with a tendency toward oscillating flow only as burnout approaches.

b. Pertinent Variables:

The driving pressure for flow created by the density difference between downcomer and core is equal to the pressure drops created by the flow. The most significant pressure drops are channel friction pressure drop, pressure drop due to the acceleration of the flow in the core as steam is formed, and entrance and exit losses from the boiling channel.

At an operating pressure of 600 psig and with condensate cooled to 110°F before being returned to the reactor, 35% of the reactor heat is required to raise the condensate to its boiling temperature (489°F). This water returns to the reactor through the feedwater distribution ring located in the downcomer at a height just slightly above the top of the core.

Absence of steam bubbles in the downcomer is thereby assured. The slightly subcooled water flows downward through the downcomer and then upward through the fuel elements of the core. Because of the subcooling, net boiling does not begin until the water has traversed 40% of the core. (This is 40% rather than 35% because of the cosine distribution of heat generation.) Steam then accumulates gradually to reach a value of about 28% by volume at the outlet of an average tube.

To better interpret the mechanism of natural circulation boiling, various quantities or parameters have been established, those of major significance include:

- (1) Quality = ratio of steam weight flow to total weight flow at any point in the channel.
- (2) Void Fraction = fractional volume of channel filled with steam.
- (3) Slip Ratio = ratio of steam velocity to water velocity.
- (4) Power Density = kilowatts per liter of water volume.

c. Experimental Results:*

A rational calculation of the system requires a knowledge of two-phase friction factors and losses, some of which have not been adequately determined at the present. For this reason the majority of the design conditions for the reactor have been based on laboratory experiments designed to approximate reactor conditions and geometries as closely as possible.

Although there have been a great number of experiments run to increase the general understanding and knowledge of the boiling phenomenon, only the results of one recent series of experiments matches reactor conditions closely enough and has measured the variables extensively enough to be included in this report.

This test was run at about 600 psia. The test section of interest consisted of plates, defining a series of water channels 7/16 in. x 4 in. in cross section and 4 ft long. Power density was varied from 20 to 50 kw/l.

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*A more complete and extended summary of the experimental studies is given in Appendix B of "The EBWR," ANL-5607 (May, 1957), available from the Supt. of Documents, U. S. Govt. Printing Office, Washington 25, D.C., at \$2.25 per copy.

Channel density was measured by means of a thulium oxide monoenergetic gamma source, and a scintillation crystal and photo multiplier tube arrangement which could traverse the entire channel length. A 2-ft riser was placed above the heated section. The pertinent results of these experiments relative to EBWR are shown in Figures 20, 21, 22 and 23.

The relationship between average reactor voids and reactor power level is shown in Figure 20. In determining the average radial void fraction, it has been assumed that the average reactor channel behaves as the average of the sum of all reactor channels. This assumption is reasonably correct since the variation in inlet velocity with power densities in the range of interest to EBWR appears to be small, though present flow data are not adequate to draw definite conclusions.

The relationship of average voids in a single EBWR channel to the power density in the channel with the reactor at a power level of 20 megawatts is shown in Figure 21. This curve is also based on the assumption that the average channel behaves as the average of the sum of all channels.

The void distribution along the average channel for a reactor power of 20 megawatts is given in Figure 22. The average exit quality and recirculation ratio of the reactor as a function of power density are given in Figure 23.

d. Operating Characteristics:

Table IX summarizes the important data relative to the operation of EBWR. All void and flow data are based on experiments which have a uniform axial flux.

e. Burnout Considerations:

In general, burnout in a system such as the EBWR can occur as a result of two distinct phenomena. Burnout can occur in heated sections with or without two-phase flow if heat fluxes are high enough to produce vapor layers over a sufficiently large area of the heated surface. Experiments indicate that this type of burnout would require heat fluxes in excess of 750,000 Btu/(hr)(sq ft) at velocities and qualities in the range of interest to the EBWR. Burnout can also occur in heated sections at lower heat fluxes if there is insufficient liquid to maintain reasonable heat transfer coefficients from surface to bulk liquid. Experiments have indicated that heat transfer coefficients maintain their high values up to steam qualities of at least 50% and then begin to drop. Other previous work has usually produced the result that the "dry pipe" type of burnout occurs at qualities near 80%.

FIG. 20
AVERAGE REACTOR VOIDS VS
REACTOR POWER DENSITY

FIG. 21
CHANNEL VOIDS VS CHANNEL POWER
DENSITY WITH REACTOR AT 20 MW

FIG. 22
VOID DISTRIBUTION IN AVERAGE
CHANNEL AT 20 mw REACTOR POWER

FIG. 23
AVERAGE EXIT QUALITY AND RECIRCULATION RATIO
VS AVERAGE POWER DENSITY OF EBWR

Table IX
EBWR HEAT TRANSFER DATA

*1. Design power of reactor	20 mw
**2. Average power density	25 kw per liter of coolant between fuel plates
3. Operating pressure	600 psig
4. Operating temperature	489F
5. Steam flow at 20 mw	60,000 lb/hr
6. Average voids in heated channels	20%
7. Average voids in total moderator	17.5%
8. Maximum to average radial flux	1.57
9. Maximum to average axial flux	1.49
10. Average heat flux in core	45,000 Btu/hr/sq ft
11. Maximum heat flux	155,000 Btu/hr/sq ft
12. Fuel centerline temperature, average	518F
maximum	561F
13. Temperature at fuel clad interface,	
average	504F
maximum	529F
14. Temperature at clad surface, average	495F
maximum	508F
15. Average exit steam quality	2.56%
16. Average recirculation rate	39 lb H ₂ O/lb steam
17. Inlet sub-cooling	8F

*All subsequent data based on 20-mw operation.

**Borax II ran 38 kw/liter at 300 psig.

There is only one experiment giving data on burnout which is applicable to EBWR. This experiment was run at atmospheric pressure with a channeled section having channel widths of $1/4$ in. and $1/2$ in. and a length of 2 ft. During the course of the experiment, burnout occurred due to a "dry pipe" condition. The burnout heat flux was 200,000 Btu/(hr)(sq ft) for the $1/2$ -in. channel, which is much less than that required to produce burnout by the vapor layer (film boiling) mechanism. Furthermore, it is felt that this type of burnout occurred due to unstable flow conditions, since the extrapolated exit quality of the system was well below the 80% limit mentioned above.

The extrapolation from the experiment to EBWR conditions results in at least a three-fold increase in burnout power density over the burnout power density of the experiment, since the void heat capacity in EBWR is 28 times as great, the slip ratio is $1/5$ that of the experiment and the channel height effect would conservatively reduce the value by 2. The resulting safety factor against burnout of the hottest EBWR channel, then, is about 6, and the estimated flux at burnout is about 750,000 Btu/(hr)(sq ft).

f. Miscellaneous Geometric Effects:

The data given in this discussion are based on the experiment described previously. This experiment, while a fair approximation of the EBWR core, has a slightly different geometry than the reactor. The major differences are channel width, 0.4375 in. against 0.417 in., and height and equivalent diameter of the riser or chimney above the active section. The effect of channel width should be negligible; however, the riser geometry can have a significant effect. It has been found that risers in general are beneficial and have their maximum benefit when channeled in the same manner as the core in order to maintain low slip ratios and hence greater voids within them. The fact that the riser in the EBWR is 3- $1/2$ ft rather than the 2 ft in the experiment should tend to reduce the average voids in the core by as much as 10 to 15%. Some of this benefit may be lost, however, since losses in the flow over the top of the core may be greater than those occurring in the experiment; and also some steam carryover may take place in the downcomer of the reactor, reducing the sub-cooling at the reactor inlet.

C. Power Plant

1. Steam System

a. General:

The major components of the EBWR steam plant are listed below:

Turbine-Generator
 Steam Bypass System
 Overpressure Relief System
 Condenser
 Feedwater System
 Circulating Water System
 Feedwater Makeup System
 Air Drying and Fluid Recovery System

These are shown schematically in Figure 24.

FIG. 24

EBWR FLOW DIAGRAM

b. Turbine-Generator:

The turbine-generator is a 5,000-kw standard condensing unit, using conventional materials of construction, designed for throttle conditions of 560 psig saturated and an exhaust pressure of 2-1/2 in. Hg abs.

The turbine and condenser (as well as other components in contact with the primary fluid) have some special design features to prevent loss or contamination of heavy water when the future conversion from D₂O to H₂O is made. Figure 25 shows special seals on the turbine. On the labyrinth shaft seals, dry air at a positive pressure of 6 in. of water is supplied between the outboard and center seals. The space between the inboard and center seals is maintained at a vacuum of 6 in. of water. The dry air at positive pressure acts as a buffer to prevent outleakage of D₂O beyond the center seal as well as inleakage of building air containing H₂O as moisture. The D₂O vapor-air mixture is drawn off at low pressure and piped to a refrigerating and chemical drying system where the D₂O is recovered. Leaking D₂O is also collected from a low pressure leakoff within the horizontal joint of the turbine casing. Makeup air from the atmosphere is stripped of its H₂O content before going to the seals.

FIG. 25
AIR DRYING AND
FLUID RECOVERY SCHEMATIC

In addition to these seals, the turbine is equipped with light-gage welded housings covering all flanged-and-bolted joints. A small recess cut circumferentially into one of the horizontal joint flange faces is evacuated by the recovery system to collect leakage. The interior of the housing around the flange is held at +4 to +6 in. of water as in the case of the shaft seals.

The turbine is designed and instrumented to be remotely operated from the control room once the initial manipulations are made and the unit placed on turning gear.

c. Main Steam System:

Steam leaving the reactor first passes through the Steam Dryer-Emergency Cooler tank (Figure 24) where excess moisture is stripped out in corrugated plate dryers. This tank also contains cooling coils which can dissipate core shutdown heat even when the main condenser is valved off and power is lost to all circulating pumps. Cooling is accomplished by water flowing down from an overhead water tank mounted on the underside of the dome of the steel building. Reactor steam is condensed on the outside of the tubes and the condensate flows by gravity back to the reactor. Steam is generated within the tubes, and the lower density of the steam-water mixture sets up a natural circulation flow from the overhead water tank. This cooling source is turned on automatically by a power failure of more than one minute.

Beyond the Steam Dryer-Emergency Cooler the main steam line divides. One line goes to the turbine and the pressure-regulating bypass valve. The other is attached to the back pressure-regulating valve, and two safety valves. All of these valves discharge excess steam through a desuperheater to the condenser.

It should be noted that although the condenser is designed to handle only 20 mw of heat output from the reactor at a pressure of $2\frac{1}{2}$ in. of mercury, a much higher heat load could be accommodated by letting the pressure go up slightly. A rupture diaphragm in the condenser wall will open at a pressure of 15 psig. The only effect of a decrease in vacuum is a loss in cycle efficiency of the turbine generator.

The piping, valves, and fittings for the 6-in. main steam system are carbon steel, ASME specification SA-53, schedule 80, insulated with 85% magnesia block. Welded connections are used throughout, except for occasional flanged joints to facilitate cleaning sections of the line. Pressure taps, flowmeter connections, etc., are provided as necessary.

Steam is exhausted from the pressure vessel through a 6-in. diameter collection ring in the upper part of the pressure vessel. Most of the moisture and particulate matter settle out of the steam as it moves upward at low (0.4 ft/sec) velocity through the upper steam dome portion of the pressure vessel from the core to the collection ring. This gravitational settling is expected to reduce the long-lived radioactive content of the steam by a factor of about 10^4 (as determined by laboratory tests) relative to the water in the reactor.

The vessel which houses the separating devices also contains cooling coils which are used for emergency cooling of the reactor as described heretofore. Water separated from the steam or condensed by the cooling coils runs back into the reactor by gravity flow.

In addition to the main line to the turbine, this system has two 6-in. branch lines which are tied in shortly beyond the emergency cooler pressure vessel. One branch connects to the steam bypass valves and the other branch to the relief and safety valves and to a 3/4 in. auxiliary steam line for the condenser air ejector and the gland steam supply. The branch to the safety valves is drained as necessary to assure that water will not collect behind the valves.

With the exception of the few special valves, all valves in this system are of the conventional globe or gate design with welded or pressure sealed bonnets slightly modified for deep packing glands, lantern rings, and bleed-off openings. Teflon or teflon-impregnated asbestos packing is used in all valves. Although packing will be installed above and below the lantern rings initially, the bleed-off openings will not be used unless operation dictates the necessity for a collection or alarm system. Past experience indicates that packing glands hold leakage to a very low value under normal circumstances and only a careful maintenance program should be necessary.

The safety and relief valves are of an enclosed design with packed lifting gear to eliminate leakage to or from the atmosphere under all conditions, whereas the stem seals for the pressure control valves are tied into the air drying and vapor recovery system to assure minimum working fluid out-leakage or air in-leakage at this point.

d. Steam Bypass System:

During normal plant operation, the Steam Bypass System maintains the reactor steam pressure at any set point from 100 psig to 650 psig over a flow range of 3,000 to 82,500 lb/hr for load changes not exceeding 1% per second. This control is obtained primarily by the use of a hydraulically operated pressure-regulating valve which bypasses high-pressure steam directly to the main condenser as necessary to hold a

constant steam flow from the reactor for any given reactor control position. In case of a loss of oil pressure the valve automatically closes. It is necessary to readjust the reactor controls manually after load changes in order to minimize steam waste.

The bypass pressure-regulating valve maintains the steam pressure in the line to the turbine at essentially 600 psig regardless of the steam demand of the turbine. When the electric load decreases and the turbine inlet valves close proportionately, the rejected steam would cause the pressure to increase in the steam line and in the reactor itself. Since the volume of steam corresponding to a given amount of contained energy decreases with pressure, this would automatically increase the power output of the reactor. The bypass valve minimizes this positive feedback effect by sensing both the change in pressure and the change in position of the turbine inlet valves. Excess steam is bypassed to the condenser to keep the pressure practically constant. Normally about 5% of the reactor steam is bypassed in this manner, but this can be varied at will. This provides a cushion of 5% for a power demand increase, and 100% for a decrease in power demand. Since the EBWR electrical output is such a small portion of the capacity of the utility system to which it is connected, changes in load demand should be minor except in the very rare case of failure of a power line.

The pressure control valve is arranged for automatic control under normal conditions; local and remote manual control are also possible. Transfers from one mode of control to another are possible either remotely or locally without flow surges.

e. Overpressure Relief System:

To supplement steam pressure system safety, one special relief valve and two conventional pop safety valves are installed. These valves are mounted adjacent to the main condenser, discharging directly to the condenser, with their inlet nozzles connected to the reactor outlet piping by an unobstructed 6-in. header.

The back pressure-relief valve is of the steam-powered type with adjustable pressure range. Its pressure setting for beginning of opening is variable, but will probably be about 610 psi. At this setting, the valve would open by an amount proportional to how much the pressure exceeds 610 psi. At 675 psi the valve would be fully open and could pass about 200,000 lb/hr of steam. At 20 mw the reactor generates only about 60,000 lb/hr.

The two safety valves are standard ASME code approved valves, set to pop at 725 psig and 750 psig, respectively, and reseal at pressures about 30 psig lower. Each valve has a capacity of approximately

65,000 lb/hr at its popping pressure and saturated temperature. Spring enclosures and packed lifting levers are used to assure leak tightness to the building whether the valve is open or closed.

Reactor safety rods are automatically scrambled when any of the relief valves open. The cause of the excess pressure would be determined and corrected before operation was resumed.

Since the relief and safety valves discharge to the condenser, the condenser is designed to accommodate this influx of relatively high pressure, high temperature steam. In addition, the condenser is equipped with a 20-psig rupture disc which protects the equipment in the event of circulating water stoppage while the safety valves are open. Rupture of this disc will result in discharge of steam directly to the building.

f. Condenser:

The condenser is a 5,300 sq ft single-pass unit with the following special features:

Leakage of the circulating water around tube joints into the reactor system is prevented by providing a double tube sheet with tubes rolled into each sheet and an evacuated space between the sheets to trap and bypass leakage.

The use of aluminum in place of conventional copper bearing alloys greatly reduces the radioactivity level in the reactor circuits and also in the steam circuits. Thus the potential hazard of maintenance of radioactive equipment is markedly reduced.

Tubes of the single-pass condenser are tilted slightly to promote natural circulation to the basin of the cooling tower in case power to the circulating pumps is lost.

Since a deaerating feedwater heater is not included in the design, the condensate will be deaerated in the condenser. Steam from the bypass system introduced below the tubes assures good deaeration at light loads, thus reducing corrosion of the system.

It should be noted that although the condenser is designed to handle only 20 mw of heat output from the reactor at a pressure of $2\frac{1}{2}$ in. of mercury, a much higher heat load could be accommodated by letting the pressure go up slightly. A rupture diaphragm in the condenser wall will open at a pressure of 15 psig. The only effect of a decrease in vacuum is a loss in cycle efficiency of the turbine generator.

g. Feedwater System:

The feedwater system consists of the following equipment:

Condensate piping from condenser to feed pumps.
Reactor feed pumps (2).
Reactor feed piping from pumps to reactor.
Feedwater full-flow filters.
Feedwater regulators.

The feed piping and other components are of stainless steel to minimize corrosion and are designed for a maximum operating pressure of 1,000 psig.

The elevations of the condenser hotwell and the reactor are such that the water in the system can be allowed to seek its own level while maintaining the proper depth of water in the pressure vessel. Reactor water level is maintained by gravity flow and will be held, even with loss of power to feedwater pumps.

The functions of the condensate and feedwater pumps in a normal plant are combined in a single pump in EBWR to reduce the number of sources of fluid leakage. Two feedwater pumps are installed, one to serve as a spare. Each pump has a capacity of 180 gpm against a pressure of 735 psig. The feedwater flow rate at the 20-mw power level will be 120 gpm.

Feedwater passing through a heat exchanger serves as a coolant for the condenser air ejectors. At reactor power levels corresponding to feedwater flows less than 70 gpm, a bypass valve opens to achieve recirculation of coolant to the condenser. The valve is controlled to make the total flow through the air ejector coolers at least 70 gpm.

The rate of flow of feedwater to the reactor is governed by a throttling valve with a 3-element control. Signals for the controller come from the reactor steam flow, water level in reactor, and feedwater flow to the reactor. The system attempts to make the return rate of feedwater equal to the exhaust rate of steam from the reactor, thereby maintaining a constant water level in the reactor. A correction is automatically applied to gradually restore the water level when it departs from the set value.

Before the feedwater enters the reactor it passes through disposable cotton fiber filters to minimize the entrance of particulate matter into the pressure vessel. When the pressure drop through one filter chamber becomes excessive the flow can be diverted through the spare filter.

A small sidestream of feedwater is introduced into the lower portions of the control rod thimbles beneath the reactor. This cold water replenishes the leakage through the seals and keeps the close fitting parts at low and essentially constant temperature. Without this cooling, differential expansion of the various materials would make the sealing problem difficult.

h. Air Drying and Fluid Recovery System:

As a guidepost for possible heavy water installations, an extensive air drying and fluid recovery system is used to minimize the theoretical contamination and loss of the EBWR working fluid. This system is shown schematically in Figure 25.

In this system, dry air is blown to a small gland at each critical leakage point from which the dry air at +4 in. H₂O pressure leaks through seals both to the atmosphere and to a gland under 2 in. H₂O suction. Since this latter chamber is adjacent to the gland to be sealed, it also collects all the working fluid leakage. This mixture is conveyed together with wet gases from the condenser aftercooler to the vent condensers. Liquids drain from these condensers to a sump for pumping to the main condenser or for further draining to the retention tanks while the remaining gases are drawn to a chemical dryer for recycling.

Since the system has a closed cycle, theoretically the only leakage of working fluid to the atmosphere is that in the "dry" air at the various individual seals. Moisture contamination occurs only from incomplete drying of the makeup air and from leakage in the external circuits which enters this system with the steam and condensate.

The air drying and fluid recovery system design limits the moisture content of the makeup air to a maximum of 0.5 lb per day and limits the moisture content of the dry air to the glands and seals to 1 lb per day. Since not all of the moisture will leak to the atmosphere at the seals, these design criteria assure that light water contamination and working fluid loss from this system will be less than the specified 0.5 lb and 1 lb per day respectively.

i. Circulating Water System:

The circulating water system consisting of the cooling tower, circulating pumps and piping is designed to dissipate the heat load of the main condenser plus the auxiliary coolers. The tower is rated at about 14,000 gpm cooled from 96°F to 85°F with a wet bulb temperature of 73°F. The system is not subject to radioactive contamination from the main condenser since the water pressure is higher than the steam pressure. Of the auxiliary coolers, only the Reactor Water Purification System Secondary cooler could introduce radioactive material into the circulating water system.

j. Feedwater Makeup System:

A separate ion exchange circuit is used for continuous purification of reactor water. By pumping 10 gpm of water through filters and a mixed resin ion exchange column, it is expected that total impurities in the reactor water can be maintained below 1 ppm. A spare circuit can be used when the filters or resins become saturated and require replacement. This pure water, together with the decontamination factor of 10^4 between the water and steam phase in the reactor, are depended upon to keep the long-lived radioactivity in external equipment at low levels.

Laboratory water is demineralized in a mixed bed ion exchange column at a maximum rate of 10 gpm. This water is used as reactor water makeup, shield flooding for the reactor, and shield cooling system makeup. A 200-gal storage tank in the upper section of the Power Plant Area provides for day-to-day storage.

k. Retention Tanks:

Two 3,000-gal storage tanks are located in the basement of the Power Plant Area to collect water from the various plant drains. In accordance with the waste disposal procedures at Argonne, the water will be monitored before disposal and, if not radioactive, will be dumped to the sewer. If radioactive, the water will be transferred to Building 310, Waste Disposal Facility, for treatment.

2. Electrical System

a. General:

The principal design features of the electrical system for the EBWR plant are shown on Figure 27.

The Argonne National Laboratory power distribution system consists of two lines operating at 13 kv, together with switch-gear and step-down transformers as required. The two 13-kv lines, designated the North bus and the South bus, are electrically separated and are fed from two electrically separate transmission lines of the connected utility.

The EBWR generator feeds over 4,000 kw of power into the North bus via Facility 544, a substation located 3,600 ft from the EBWR plant.

Three sources of full auxiliary power capacity are provided. A 4160V/440V station service transformer connected to the generator bus feeds auxiliary power either from the EBWR generator or from the North bus. In addition, a 13 kv/440V transformer connected to the South bus automatically picks up the auxiliary load if the North bus is de-energized (see Figure 26).

FIG. 26

ARGONNE POWER DISTRIBUTION SYSTEM

LEGEND

- 1 AMMETER
- 2 VOLTMETER
- 3 WATTMETER
- 4 HANDBY
- 5 FREQUENCY METER
- 6 WATTMETER
- 7 WATTMETER
- 8 WATTMETER
- 9 WATTMETER
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- 100 WATTMETER

FIG. 27

EBWR SINGLE LINE KEY ELECTRICAL DIAGRAM

As shown in Figure 27, certain essential loads are fed from the 440-volt bus through control center No. 2. In case of loss of both full capacity feeders mentioned above, power for control center No. 2 is supplied by a 60-kw diesel generator which is started automatically.

b. Generator and Switchgear:

The generator bus voltage is 4.16 kv which is the same as the main EBWR bus voltage. The generator is a standard air-cooled machine and has a rated capacity of 5,000 kw at 0.8 power factor. It has a directly coupled 125-volt exciter complete with automatic voltage regulator. The 4.16-kv switchgear has drawout-type air circuit breakers, rated at 1,200 and 600 amperes, which have an interrupting capacity of 150,000 kva. Current transformers and potential transformers necessary for metering and relaying are included in this gear.

c. Instrumentation and Control:

The generator and electrical system instrumentation and controls are located on the control panels in the Control Area. All controls and indications essential to the operation of the plant are provided; recordings of some quantities are made. The protective relaying equipment for the generator and main power transformer, also located on the control panels, includes differential relaying protection for the generator and main power transformer in case of internal faults, negative sequence overcurrent relay for generator field protection against prolonged 13-kv system faults, generator neutral relay for external 4.16-kv ground faults and overcurrent and ground relays for the main power transformer.

All motors are of the full voltage starting type. Where possible, open frames that feature drip protection are used. Where it is necessary to protect the motor windings, such as for the turbine oil pump and the cooling tower fans, totally enclosed motors are specified. Overload protection for motors and other auxiliary equipment is provided by the thermal elements of either the circuit breaker or the magnetic switch. Short circuit protection is provided by the instantaneous features of the air circuit breakers.

d. Lighting:

The lighting load is supplied by the 100 kva 480/120 volt distribution transformer. The normal lighting load is fed from the 120-208 volt bus. About 10% of the total lighting is fed from a separate section of the 120-208 volt bus which is normally fed from the distribution transformer, but which is automatically transferred to the 125-volt station battery in case the normal supply is lost.

e. Battery Power:

A 125-volt battery with an 8-hour rating of 280 ampere hours is provided for operation of the reactor control board, all switchgear, and circuits associated with emergency valves. This provides a reliable source of power for all essential operations. The diesel generator is started from this battery and certain selected lighting is connected to the battery to provide lighting during outages of the main power system.

D. Building

1. General

General offices, laboratories, and the plant control room are in a conventional masonry building, referred to in this report as the Control Area, which is connected to the gas-tight Power Plant Areas as shown on Figure 28.

The welded steel power plant building is designed for the following conditions:

1. Maximum internal pressure - 15 psig.
2. Maximum negative pressure - 0.5 psi below barometric. A vacuum breaker, described later, is used to prevent greater negative pressure resulting from barometric changes.
3. Maximum temperature differential from erection conditions - 200F.
4. Factor of safety for maximum internal pressure - 4.5.
5. Total building leakage - less than 1,000 cu ft per day at 15 psig.
6. Building volume - sufficient to accommodate flashing of all high-pressure water.

The Power Plant Area building recommended by Sargent & Lundy, the architect-engineer, is shown in Figures 29 and 30. All radioactive components of the plant are housed in a gas-tight steel building. The entire plant can be operated remotely from a control room in the adjacent masonry service building. Occupancy of the gas-tight building will be limited to infrequent general inspections (perhaps once every 8-hour shift) while the reactor is operating. For major repairs the plant will be shut down.

The gas-tight steel building is 80 ft in diameter and 119 ft high. The steel used conforms with the ASTM A-201 Grade B Firebox quality specifications and was produced in accordance with the ASTM A-300 specifications with V-notch Charpy test at -50° F. The latter insures against embrittlement at low temperatures. The thickness varies from 5/8 in. on the ellipsoidal bottom and the cylindrical sides to 3/8 in. on the hemispherical

FIG. 28

GENERAL ARRANGEMENT -- MAIN FLOOR -- EBWR FACILITY

GENERAL CROSS SECTION "AA" - EBWR FACILITY

FIG. 29

FIG. 30

top. This structure will withstand an internal pressure of 15 psig with normal factors of safety. Vacuum breakers are used to prevent external pressures greater than 0.25 psig.

Critical visual inspections were made as the welding of the steel building progressed. In addition, spot-test radiographs were made of approximately 10% of the total length of the welded seams. A large proportion of radiographs were made at weld junctures where improper fusion is most likely to occur. Indicated faults were traced to their vanishing points and repairs made, followed by new radiographs.

Upon completion of the steel building it was pressurized to 18.75 psig as a strength test. The pressure was then reduced to 15 psig and measurements of internal temperature and pressure made over a period of about eight days. No leaks were found. Results indicated that the leakage, if any, was less than the 500 ft³ per day as specified for this structure.

In construction work carried out subsequent to the strength and leakage tests on the steel building a number of openings were made in the shell. A large temporary opening was cut for bringing equipment into the shell. A 5 ft x 10 ft gasketed freight door was installed. The main personnel airlock and a small emergency escape airlock have been installed. Openings were made for pipes carrying water, steam, ventilating air, etc. Another test will be performed prior to startup of the plant to determine whether any leaks exist. If so they will be repaired. The permissible leakage for the completed plant is specified as 1,000 ft³/day at 15 psig. The total free volume in the shell is about 400,000 ft³.

Roughly one-half of the steel building is below grade (Figure 3A). The turbine generator is mounted on the main floor at grade level. The top of the reactor is slightly below this floor. The condenser and cooling tower water circulating pumps are located on the first basement floor. The feedwater pumps and the Air Drying and Fluid Recovery System are mounted on the next floor down. The ion exchange system, retention tanks, and shield cooling equipment are installed on the third (lowest) floor.

To provide a level floor above the elliptical bottom of the steel shell, concrete was poured to a maximum depth of about 8 ft at the center. The depth of concrete beneath the center of the reactor is about 7 ft. This provides very substantial protection for the steel shell against missiles which might be ejected downward from a reactor explosion. The concrete also acts as an anchoring medium for the six vertical columns which restrain the three huge horizontal steel H beams above the reactor.

The entire steel shell below ground level is lined with concrete having a minimum thickness of 2 ft. Above grade the concrete liner and ceiling have a thickness of 1 ft.

The steel shell is ventilated by 3,000 cfm of air drawn in through the side of the building and exhausted through a short stack near the top of the dome. Quickly closing, solenoid-operated butterfly valves in the inlet and outlet ducts close automatically under gravitational forces when a radiation monitor located near the outlet duct indicates high radioactivity.

The control room is located in the service building adjacent to the steel shell. Electric cables which pass through the shell are sealed in U-shaped steel pans welded to the steel shell and filled with a special cement to prevent leakage.

Switchgear and motor control centers are located in the basement of the service building. A diesel generator, which starts automatically upon loss of other power, supplies power to critical items of equipment in the plant. In addition to the power generated by EBWR itself, incoming power is available from two separate branch lines of the Laboratory electrical distribution system. The main power transformers are mounted on the concrete pad near the service building.

Corrosion protection of the steel building below grade is in the form of coatings of bituminous enamel, both inside and outside the shell. Above grade the shell is insulated with 3 in. of Foamglass impaled over steel studs welded to the shell. Asphaltic mastic, Glasfab cloth and an additional coat of mastic are applied over the Foamglass. The surface is finished with aluminum paint.

2. Missile Protection

The design of the EBWR Power Plant Area Building is predicated upon maintaining the gas-tight integrity of the steel building shell, thus containing any radioactive material released during an accident.

Precautions have been taken to prevent puncture by fragments. A concrete liner extending from the bottom of the vessel to above the building crane is provided to prevent penetration of the gas-tight steel shell by any missiles resulting from an explosion within the building. A reinforced concrete ceiling across the entire building above the building crane completes the missile protection system. Further, all equipment normally operating under pressure is located under the 2-ft thick concrete main floor or behind barriers, except for the turbine which is located on the main floor. The location of all auxiliary and emergency equipment and the layout of the interconnecting piping takes advantage of the blast protection afforded by all massive concrete structures within the building, thus insuring a supply of emergency cooling water to the core even in the event of an explosion.

The building design is more than ample to withstand the pressure which would be built up if all the water in the reactor vessel were flashed into the building. The release of the stored energy of the water in the reactor vessel is sufficient to vaporize only one third the water into steam. The vaporization of all the water would have to be the result of a large additional heat release, such as caused by a metal-water reaction. Tests to date (described in Appendix C) indicate that such a reaction is possible only under special conditions. As discussed elsewhere, it is believed that such a reaction could be the result only of a sabotage effort in which large excess reactivity was suddenly added to the core. In maintaining the building integrity, due to pressure buildup from such a reaction, the limiting factor is likely to be the prevention of an out-sized hydrogen explosion.

A metal-water reaction involving 25% of the core material would mean an energy release of 8×10^8 cal. A design evaluation of the reactor pressure vessel and surrounding structure has been undertaken by Armour Research Foundation to determine whether, with the present design, such an explosion can be contained (see Appendix G). As described previously, this design is based on maintaining the integrity of the top structure and directing the force downward into the control rod room and the foundation where weakened areas are provided to release the pressure.

The top of the pressure vessel is designed to be 50% stronger than the rest of the vessel and is backed by a 2-in. plate of Carilloy T-1 armor plate. However, of probably greater advantage is the steam phase above the reactor core. The shock wave from an explosion is of much lesser intensity in the steam than in the water surrounding the core. The steam may also act as somewhat of a cushion against water thrown upward by the explosion.

The exterior of the steel gas-tight building is covered above ground with 2 in. of insulation, preventing embrittlement of the steel due to low ambient temperatures. During operation of the reactor, the steel in the building shell will never be in the temperature range where the impact strength of the steel is reduced due to embrittlement.

3. Spray System

A 15,000-gal water storage tank is suspended from the dome of the steel shell. A building spray system, consisting of spray rings located above the main floor and in the basement, is supplied from this tank at the initial rate of 1,000 gpm until the reserve has been used up and is then supplied from the Laboratory Water system at the rate of 100 gpm. In the event of a rupture of the pressure system releasing steam into the building, the spray system is turned on manually, either from the control room or from the Power Plant Area.

If the water in the reactor were flashed to steam from 600 psig into the building at an ambient temperature of 90F, the resultant temperature and pressure would be 178F and 9.8 psig as shown in Figure 31. The spray system would reduce the temperature and pressure to 125F and 3.2 psig within 15 minutes. In the calculation of the cooling rate due to the spraying, it was assumed that the additional heat energy contributed by the reactor vessel was equivalent to the energy absorbed by the exposed surfaces in the interior of the Power Plant Area and that additional energy equal to twice that required to melt the core was somehow added. The condensing capacity of the exposed surface in the Power Plant Area has been conservatively estimated to be 3,400 lb of steam out of the 7,500 lb which is flashed.

In the event of a vessel rupture accompanied by a failure of one or more fuel plates, the washing effect of the sprays would be helpful in reducing the radioactive contamination of the atmosphere in the building. Thus, radioactivity in any ensuing leakage from the gas-tight building would consist mainly of gaseous fission products.

4. Air Lock

Personnel access to the Power Plant Area is through a double-door airlock that is designed to maintain the gas-tight feature of this building.

Doors having clear openings of 3 ft by 7 ft are mounted in the ends of a cylindrical shell 10 ft in diameter and 7 ft long. When closed, the doors are pressed against sealing gaskets by mechanical latching mechanisms. The latching mechanism for each door is geared to a valve which equalizes the pressures on either side of the bulkhead before unlatching.

A pressure-equalizing valve is incorporated in each door of the airlock. As the handwheel is turned, the valve opens to equalize pressures before the door is unlatched. Loss of radioactivity from the airlock chamber will occur during operation of the outer door mechanism. The internal volume of the airlock is about 550 ft³. Thus, when the building is pressurized to 15 psig, as much as 0.14% of the radioactive air could be lost through the airlock during each exit. If the outer door were closed after the exit (this is necessary before the inner door of the airlock can be opened again) this loss would be reduced to 0.07%.

The doors are mechanically interlocked so only one door can be opened at any one time. Manual operation is planned for the airlock doors. Lights in the Control Room will show if a door is open or closed. The entire unit is designed to withstand 15 psig pressure with a minimum

FIG. 31
BUILDING PRESSURE AND TEMPERATURE
AFTER REACTOR VESSEL RUPTURE

of air leakage. The doors will seal on each closure, with the mechanical interlock so adjusted that a door cannot be opened until the other has been closed and properly seated against its gasket.

In case of malfunction of the main airlock doors, emergency escape for personnel is provided in the form of a small auxiliary airlock. This airlock is too small for normal use. Doors are sealed with gaskets, and interlocking mechanisms are similar to the larger airlock. Normally this airlock can be used only as an exit. To gain access to the plant, a key must be used on a padlock located within the airlock cylinder between the doors.

5. Connections Through Gas-Tight Enclosure

All electrical cable and pipes leading from the Power Plant building are sealed for the 15 psig internal pressure. Various schemes for sealing these cables are currently being tested at Argonne.

Building ventilation is limited to a 3,000 cfm fresh air and exhaust system in order to use small ducts and dampers. Exhaust air leaves the plant near the top of the steel shell through a duct designed for two seconds holdup time. Weight-operated, quickly closing, solenoid-controlled butterfly valves are located in inlet and outlet ducts. These valves close upon detection of activity in the plant or upon power failure. Closing time is less than one second.

Heat loss of the power equipment into the air in the building is removed by a recirculating building air system consisting of four unit air conditioners, two on the ground floor and two in the basement. The plant circulating water system provides cooling for the air conditioning condensers. Building heating, when required, is accomplished with steam coils in these same units. Steam will be supplied from the Laboratory steam plant.

6. Facilities for Viewing Power Plant Area

The use of a thick concrete liner for the gas-tight shell precludes the use of viewing windows in the wall of the building. To provide for viewing of the Power Plant Area from the Control Room, a television monitor, located in the Control Room, can be switched to one of several cameras located in the Power Plant Area.

IV. OPERATING PROCEDURE

A. Preliminary

After the core grid structure is in place and control rods and drives are installed, control rod mechanisms are tested for smooth operation and time of scram to insure 1/2-sec insertion. Several hundred operations will be performed at this point. Instruments, interlocks, and alarms are given an operational check. The high and low-pressure boron injection systems are operationally checked and at standby for use during initial loading if the need arises.

1. A strong antimony-beryllium source, irradiated in another reactor, is to be used for fuel loading and startup testing. This is installed in a dummy element which will be positioned near the center so that the fuel will be adjacent to the source in initial loading. Source strength will eventually be maintained for subsequent startups by irradiation in the EBWR itself.
2. Check counting rate due to source with BF₃ counters located inside the pressure vessel and inside of temporarily installed tubes next to the core grid opposite the source. These counters will register neutron counts in the control room and will remain in this position until a cold critical mass has been established.
3. Install an ion chamber inside the pressure vessel. This ion chamber feeds a sensitive electrometer or micro-micrometer to indicate multiplication or approach to critical.

B Initial Loading of Fuel

Since no critical experiment will have been run prior to this loading, a careful approach to a critical loading will be carried out.

1. A reference minimum critical mass in kilograms will have been computed to be used as a guide in loading.
2. Starting at the space next to the source the fuel elements will be installed in a direction toward the counters.
3. After each incremental loading, rods will be pulled one at a time to allow a measurement of unpoisoned multiplication. To do this the temperature interlock must be by-passed for cold testing. Multiplication will be plotted against loading as a means of predicting the approximate critical loading prior to actually loading that many elements into the reactor.
4. When a cold critical mass has been obtained, the counters are to be moved individually to the permanent operating position in the shield holes.

5. A reference run to critical will then be made to establish the base counting rate for the counters. Ion chambers located in the shield will also be checked against the ion chamber inside the pressure shell to see that instrumentation is set up to read the core neutrons. The ion chamber and tubes can then be removed from inside the pressure vessel.

C. Preparing for Operation

The probable loading is expected to result in a critical reactor with eight rods all out and one rod inserted to some extent since a rod at the center more than compensates for an enriched fuel element. Thus, the first critical loading, cold, should be obtained with the central rod at some point in the core. The objective of the following procedure is to calibrate rods and determine loading for power operation.

1. Adjust water to the normal operating point. Insure that all neutron instruments in the shield are operating and in place.
2. Take periods on the center rod to obtain reactivity per inch at various points, while heating the reactor water with startup heaters to obtain different critical rod positions.
3. Determine the worth of all rods and the temperature coefficient by installing natural or enriched elements singly at various locations around the core lattice to get combinations of rod positions and determine the approximate reactivity worth of the element in each location.
4. Establish a desirable loading conforming to excess reactivity requirements for normal operation and rod worth. Install top shield and close pressure vessel.
5. Further calibration of rods and temperature coefficient measurement can now be performed.

D. Normal Startup

In normal startup procedure, the first step is to clear the parameters for which abnormal conditions will be indicated on the operating console annunciator. Subsequent steps follow:

1. Allow the water levels in condenser hotwell and reactor to equalize and verify that these levels are in their normal ranges. The reactor control system is so designed that if the water level drops 6 in below normal, the reactor control rods will automatically scram.
2. Start water pumps for auxiliary cooling circuits.

3. Start the air drying and fluid recovery system. Check humidity indicator and blower discharge pressure.
4. Start one shield cooling pump. Check flow indicators for flow to individual cooling coils. Water temperature in shield should not exceed 120F.
5. Start one condenser circulating water pump.
6. Close feedwater control valve at reactor feed pump discharge. Open the feed pump suction valves. Start one feed pump.
7. Raise the reactor water temperature to 325F by turning on the building steam to the startup heater.
8. Start pressure regulator system oil pump.
9. Open main steam line stop valve and open valves controlling the air ejectors.
10. Manually set the pressure control system to bypass steam to the desuperheater at 100 psig.
11. Check to insure that all annunciator scram points are clear and properly adjusted.
12. Engage turning gear manually. Check warning lights at local panel or in control room. Start turning gear motor.
13. Start withdrawing rods to bring reactor critical and turn off start-up heater. Control reactor feedwater manually. Bring reactor power up gradually, gradually increasing pressure to 600 psig.
14. Reset turbine trip valve mechanism and open valve.
15. Build up vacuum to between 16 and 18 in. Hg before rolling turbine.
16. Open governor valves by backing off load limit screw until turbine speed reaches 400 to 500 rpm.
17. Build up vacuum to the highest possible value, adjusting the load limit meanwhile to maintain a speed of 400 to 500 rpm.
18. Hold at 500 rpm for 15 min or until heat redistribution has taken place as indicated by the leveling off of the exhaust casing temperature. Accelerate uniformly to rated speed during the next 15 min.

19. Field can be energized at any speed above 3000 rpm.
20. Open cooling water valves to the generator air coolers.
21. When reactor pressure reaches 600 psig, control pressure manually with by-pass control valve until 5% by-pass is attained. Put automatic pressure control system into operation.
22. Start automatic feedwater controller.
23. Raise reactor power level to 20 megawatts.
24. Synchronize turbine speed and then recheck operation of automatic pressure regulation system.
25. Put 10% load on generator observing coarse regulator valve to insure that it is closing properly and that pressure is being maintained. Hold this load until the exhaust end has cooled down and then load machine as required.

E. Normal Operation

Normal operation of the EBWR is quite simple and hazard-free. The plant can be operated automatically, responding to changes in system load, or manually at whatever load the operator selects.

During automatic operation of the plant, the turbine-generator responds to changes in system load, as do all other turbines in the system, by very small changes in rotating speed. These small changes are used to initiate a change in setting of the governor valves in such a manner as to bring the system back to synchronous speed (3600 rpm). Thus a decrease in load causes the system to speed up, and this increase in speed causes the governor valves to close slightly, reducing the speed.

It should be pointed out that all generators in the system rotate at exactly the same speed, i.e., they are tied together synchronously.

It is not possible for any change of load in the system to be reflected on the turbine-generator except in proportion to its rating compared to the generating capacity of all the machines in the system that are in operation.

When the governor valves of the turbine close slightly, the steam flow through them is reduced. Since the reactor output has not been reduced, the pressure tends to rise. Until such time as it has been adequately determined that the reactor itself will automatically respond to changes in load, changes in pressure will be prevented by the operation of the pressure control system to hold reactor pressure to 600 psig.

Emergency operating conditions are discussed in Section VI, Enumeration and Evaluation of Possible Hazards.

F. Normal Shutdown

1. Remove load from generator gradually, disconnect the generator from the line and allow steam to by-pass to condenser.
2. Trip the turbine trip valve either locally or from control room. Do not reset until next startup.
3. Insert the rods either slowly or rapidly (scram).
4. Maintain water level in reactor by either manual or automatic control of feedwater flow control valve. Gradually reduce pressure by by-passing steam to the condenser.
5. After turbine-generator rotor comes to rest, engage turning gear and start turning gear motor.
6. Start reactor shutdown cooling pumps.
7. Secure air drying and fluid recovery system after turbine and condenser have reached ambient temperature.
8. Shut off turning gear and auxiliary oil pump after 8 hr operation or as soon as unit temperature drops to 120F as noted on the exhaust casing thermometer.

V. INHERENT SAFETY OF EBWR AGAINST REACTIVITY ADDITIONS

By the inherent safety of the reactor is meant its ability to withstand the effects of accidental reactivity additions without responding in such a way as to endanger its surroundings, and to withstand these effects either indefinitely or until the probability that shutdown will be accomplished by external means (safety circuits, manual scram, etc.) is for all practical purposes unity. The EBWR has a higher degree of inherent safety than is usually attainable in power reactors, principally because of the large and relatively quick-acting negative power coefficient of reactivity which results from the formation of steam in the reactor core. In the following discussion the limits of inherent safety of the reactor will be defined as closely as possible. Since a number of unknowns enter the analysis it is not possible to define the limits with great precision. In most cases of uncertainty, however, it has been possible to consider limiting cases which are conservative (see Appendix E). It is therefore believed that, although the estimates may be unduly pessimistic, there is little probability that they are optimistic.

In the following discussion the limiting safe accident is assumed to be that which causes the hottest point in the hottest fuel element just to reach the melting temperature. Such an accident would, of course, damage the reactor itself, but would hardly lead to external consequences.

Reactivity additions of two different types are to be considered: those which are made slowly enough that equilibrium temperature distributions are maintained within the reactor, and those made rapidly enough to generate transient temperature distributions. Since it is possible for slow additions, if they persist long enough, to lead to conditions similar to those produced by rapid additions, the latter case will be considered first.

A. Effect of Rapid Reactivity Increases

Detailed estimates for this case have been made in Appendix E on the basis of experiments with the Borax I and Borax II reactors. The results, which depend on the initial condition of the reactor, may be summarized as follows:

1. Reactor Initially at Room Temperature

For this case the most effective shutdown mechanism is the metal temperature coefficient of reactivity. This will protect the reactor against sudden additions up to 1.1% k_{ex} (0.017-sec period). If there were no metal coefficient, steam formation alone would protect against periods as short as 0.057 sec (0.84% k_{ex}). The effects of smaller additions of reactivity can be read from Figure 47.

2. Unpressurized Reactor, Initially at Saturation Temperature

Again the metal temperature coefficient is most effective for shutdown and will handle about 1.1% k_{ex} . Shutdown by steam is almost as effective, however, with a limiting safe period of 0.025 sec (0.99% k_{ex}). The effects of smaller additions of reactivity can be read from Figure 45.

3. Pressurized Reactor

At pressures in the 150-300 psig range, steam formation will give protection against the effects of periods as short as 0.017 sec (1.1% k_{ex}). This behavior, or better, is expected to hold up to pressures of 600 psig, although no tests have been made above 300 psig. In any case, the effectiveness of steam can hardly fall below that for the saturated atmospheric pressure case. The effectiveness of the metal temperature coefficient is somewhat less at elevated pressure because of the higher initial fuel plate temperature. At 600 psig the metal temperature will protect against additions of 1.0% k_{ex} or slightly more.

To summarize, the reactor is safe against sudden reactivity additions up to at least 1.1% k_{ex} under all conditions, and both steam formation and the metal temperature coefficient are of comparable effectiveness when the initial reactor temperature is above the atmospheric saturation value. In the subcooled reactor, however, steam formation is estimated to be relatively less effective. It is for this reason that the preheat circuit, and the interlocks which prevent control rod actuation until preheating is accomplished, have been provided for the EBWR.

The present understanding of the process of steam formation is not sufficiently detailed to permit firm estimates of the combined effects of steam formation and metal heating. The processes are certainly complementary, however, and the net effect must be appreciably better than the component effects.

The above estimates are based on the effect of the power excursion that follows immediately after the introduction of the excess reactivity. If the excess reactivity is not too large, this first power excursion will settle down into steady boiling operation at a safe power level. If the excess reactivity is larger than a certain threshold value, the initial excursion will be followed by a succession of milder power excursions as the water alternately is ejected from the reactor by steam formation and flows back in. This "chugging" operation occurred in the Borax reactors only if the addition of excess reactivity was greater than 0.95% k_{eff} . It may be favored in the EBWR by the greater heat storage in the fuel plates, but in any case would hardly be likely to occur for reactivity additions less than β . For larger sudden reactivity additions "chugging" may occur, and it is conceivable that plate burnout could occur during the "chugging" process if a few of the hottest fuel plates became steam bound and could not cool between

"chugs." Consequently, complete safety against local plate melting can be claimed for the larger reactivity additions only if externally actuated "scram" acts before the second "chug." The time available for action of this scram will be from 1.5 to 2 sec after reactor power reaches a high value. The signal available for actuating the scram is of course extremely large.

The inherent safety of the reactor, by the figures quoted in the foregoing summary, is just about sufficient to handle safely the sudden addition of an enriched fuel element,* in the most effective position, to the critical reactor (worth less than 1.16% k_{eff} , see Section III-B-3-e). It cannot be shown that the sudden removal of the most effective control rod from the critical core could be tolerated (1.9% k_{eff}). Such removal could not, of course, be effected by the normal rod mechanism, which is quite slow. Indeed, the sudden removal (in a fraction of a sec) of the heavy rod, at a time when the reactor is subcritical by less than 0.8% k_{eff} is an event of very low probability.

The addition of a fuel element and the removal of a control rod are the only obvious sources of sudden large reactivity increases, except those which arise from certain unstable operating conditions. The latter are discussed in the next section.

B. Effect of Large Slow Reactivity Addition to Operating Reactor

In normal full power operation the reactor core will contain an average 17% of steam by volume in the coolant water. This steam represents a negative reactivity of about 2.9% k_{eff} . If reactivity is increased, the reactor will maintain criticality by the formation of a greater steam volume, and the power output will increase accordingly. The power cannot increase much faster than linearly with steam volume, however, whereas steam volume will increase at a rate less than linear with increasing reactivity (Figure 13). The net result is that power output increases at a rate which is less than linear with increasing reactivity. Thus the tripling of the reactivity held by steam (by the slow addition of 4.4% k_{eff}) would increase the power output by less than a factor of 3. Such an increase in power would not be dangerous from the standpoint of fuel plate burnout if the reactor maintained stable operation.

There is a limit to the amount of reactivity which can be compensated by steam without causing instability of operation. This limit, experimentally, increases with operating pressure, and for the Borax reactor lies somewhere above 3.1% k_{eff} at an operating pressure of 300 psig. At lower pressures, the limit for the Borax reactor is in the neighborhood of 3% k_{eff} . When instability occurs, the reactor power goes through a series

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* This refers to a fuel element enriched to the extent of 2½% U^{235} , as originally designed, and not to the enrichment of 1.44% U^{235} , as actually employed.

of minor oscillations for a few seconds (gives warning), and then begins a series of excursions (chugs) similar to those which follow the introduction of sudden reactivity increases. For this case, the reactor behavior is similar to that described in Section A above. The minimum periods observed in Borax for such chugs were in the 0.02 to 0.03 sec range. The EBWR, at operating temperature, will be safe against transients of such period.

To summarize, if reactivity is added slowly to the operating reactor, the only effect will be a safe increase in operating power until something more than 1% k_{eff} has been added above the normal operating reactivity. If reactivity addition is continued long enough, unstable operation will result, which, after some warning in the form of preliminary oscillations, will develop into power excursions of large, but probably safe, amplitude.

C. Effect of Slow Addition of Reactivity to Reactor Initially Subcritical or in Very Low Power Operation

If the reactor is initially subcritical or critical at a very low power (order of watts), the slow addition of reactivity, as by the steady withdrawal of control rods by their actuating mechanisms, can lead to a relatively fast reactor period before the power rises high enough to bring the inherent limiting processes into play. The subcooled case will be discussed, even though it is prevented by preheaters and interlocks, because it is the most severe.

The nominal maximum rate at which reactivity can be added to the EBWR by the control rod drives is 0.01% k_{eff} per second. A numerical calculation has been made of the variation of reactor power with time as reactivity is added at a uniform rate of 0.04% k_{eff} per second (four times the maximum EBWR rate) to the reactor, which is initially at a very low power. The high rate of addition was chosen for the calculation to demonstrate that safety does not depend on very accurate specification of drive speeds and very accurate predictions of control rod worth. If the reactor power is initially 1 watt, by the end of about 19 sec the addition of 0.04% k_{eff} per second will have caused an increase of reactor power sufficient to heat the fuel plates (initially cold) to the boiling temperature of water. At this time the total reactivity addition is just slightly more than β , and the reactor period is 0.12 sec. This period is well within the safe limit for the EBWR. For the actual rate of possible reactivity addition to the EBWR, the period would be even longer. After the initial power excursion the reactor should go into steady boiling operation - at first with subcooled water and later, as the reactor water heats up, with net steam production. After this stage is reached, the situation is essentially like that discussed in Section B above. It would be expected that a total of at least 3% or 4% k_{eff} could be added before instability occurred.

VI. ENUMERATION AND EVALUATION OF POSSIBLE HAZARDS

A list of events which the authors believe should be evaluated from the safety standpoint appears below. Each of these events is in turn discussed in the following pages.

- A. Shutdown Cooling with Loss of Water
- B. Power Failure
- C. Pump Failure
- D. Instrument Failure
- E. Safety Rod Failure
- F. Change in Power Demand
- G. Too Much or Too Little Water in Reactor
- H. Water Leaks From the External System
- I. Fuel Cladding Failure
- J. Startup Accident
- K. Melting of Fuel Plates
- L. Excessive Pressure
- M. Sudden Closure of Steam Valve
- N. Failure of Steam System
- O. Sudden Introduction of Cold Water
- P. Improper Charging of Fuel
- Q. Earthquake
- R. Fire
- S. Sabotage with Fissionable Material or Explosives
- T. Sabotage by Pressurizing Reactor Tank with Gas
- U. Explosive Chemical Reaction Between Fuel Elements and Water

A. Shutdown Cooling with Loss of Water

The removal of the decay heat from the fuel elements of the EBWR following a regular shutdown is by boiling of the coolant. The heat contained in the steam generated is in turn then rejected to the condenser. After the decay heat has dropped to 1% of full rated power (7 hr), the secondary cooler in the Reactor Water Purification System is capable of removing the heat at a rate sufficient to keep the temperature of the reactor water below 120F.

Under abnormal conditions which result in loss of water from the reactor vessel, cooling of the fuel is provided in several ways, depending upon the conditions which exist.

1. A major rupture of the reactor vessel that results essentially in the confinement of air within a waterless reactor vessel will cause fuel elements to melt (refer to Appendix A). To guard against this contingency, the EBWR vessel is provided with a means of cooling even in the event of such an accident. Water from the 15,000-gal overhead storage tank or from the Laboratory Water System can be admitted to the vessel through the Boric Acid Spray Ring via the Boric Acid Crystal Tank. This is done automatically when both the water level and pressure in the reactor drop below the set limits. It can also be done manually from either the Control Room or the Power Plant Area. The shutdown heat of the core can be dissipated by evaporation of the water as it trickles over the fuel plates.

2. A major rupture of the vessel that results in the loss of water from the vessel but is of such a nature that air from the building can flow through the core will result in the fuel plates reaching a maximum temperature of 1650F (900C), at which temperature the fuel elements will not melt. However, the zirconium will oxidize, exposing the hot uranium, which can then burn, releasing fission products to the atmosphere. If operable, water cooling from the Boric Acid Spray Ring will prevent this high temperature; if not operable, due to damage to the spray ring from this reaction, the building must contain the fission products.

3. A minor rupture of the vessel could result in the eventual loss of all water from the vessel. If the rupture were above the top of the core, the flashing operation would prevent overheating of the fuel elements until the pressure had dropped to a point where water injection would automatically begin and the core could be kept flooded by water from the Laboratory main. If the rupture occurred below the core, the operator would reduce pressure in the reactor as rapidly as possible by opening the relief valves and dumping steam to the condenser, thereby permitting low pressure injection of water from the overhead tank or Laboratory water main through the Boric Acid Spray Ring.

B. Power Failure

Loss of power to the solenoid holding the safety rods would release these rods, and they would be fully inserted to essentially their maximum shutdown position in less than 0.5 sec (fail-safe operation). If four control rods do not reach their full-in positions within two seconds, the Boric Acid Injection System automatically starts. Water from the overhead tank to the emergency cooler in the steam separator is automatically turned on by a power failure for more than one minute.

In case of loss of power from both incoming utility lines and subsequent scrambling of the reactor and tripping of the turbine-generator, the pressure-regulating valves in the main steam system by-pass steam to the main condenser, preventing reactor pressure from rising above 600 psig. Continued operation of the pressure-regulating system is assured by providing an oil accumulator. Since the circulating water pumps are without power the flow of cooling water to the condenser is greatly reduced. The circulating water piping is such that there will be considerable natural circulation. The natural circulation is aided by the automatic opening of a valve which by-passes water from the condenser outlet directly to the storage basin of the cooling tower, since the height of the cooling tower risers would otherwise prevent circulation.

The ventilation openings in the Power Plant Area building shell are automatically closed on power failure.

C. Pump Failure

The most probable reason for failure of pumps is loss of power. The effect of failure of individual pumps is discussed separately.

Failure of one of the two circulating pumps would lead to a slight increase in back pressure in the condenser, with a correspondingly higher temperature of water in the hotwell. Introduction of this hotter water into the reactor by the feedwater pump would result in more steam per unit of power and cause the power level to drop.

Failure of both pumps would lead to complete loss of vacuum in the condenser. The reactor scrams when the back pressure rises to 20 in. Hg abs. In the event this circuit fails to function, the operator would be notified by a separate high back-pressure alarm, or a loss of differential across the condenser, and would manually shut down the reactor.

If the feedwater pump should fail, the introduction of feedwater would be discontinued and the reactor would automatically reduce power level. Water would gradually be evaporated from the reactor to accumulate in the condenser hotwell. Alarms would notify the operator of this sequence of events. If these were unheeded, the level of water in the reactor would

eventually drop to the point where the low level trip would insert the control rods and shut the reactor down. If any control rod did not reach its full-in position within two seconds, the high pressure boron injection would be automatically initiated.

In case of failure of the auxiliary cooling water pumps, cooling water to the building ventilation system, vapor recovery and reactor purification system secondary coolers is lost. The loss of cooling for the purification system (ion exchanger circuit) would automatically shut off the pump in this circuit. The operators would shut down the reactor manually and no hazard would exist during this period.

In case of failure of a shield cooling pump, an alarm in the control room would notify the operator of low flow in this circuit, whereupon he would turn on the spare pump. If the operator did not turn on the spare pump, the primary shield would be heated to approximately 450F. It is possible, although not likely, that some crumbling of the concrete in the primary shield would occur if the concrete were maintained at 450F for a long period. However, the results would not be hazardous.

D. Instrument Failure

Instrument circuits will be designed to fail safe. Instruments which are capable of inserting the rods upon receipt of the proper signal will also insert the rods when the instruments themselves are not functioning properly. Similarly, alarm circuits will be designed to give alarms when malfunctioning of the circuits develops.

The pressure-control system fails open, flashing water in the reactor due to decreasing pressure. This also automatically reduces power level in the reactor. The motorized feature of these valves provides a means of setting the maximum opening to which the valves can be driven by springs. This can be used to limit the blow down rate in case of control system failure. If the rate of pressure change of the reactor were greater than a predetermined value, the reactor would be scrambled; otherwise the reactor would be shut down manually.

E. Safety Rod Failure

The possibility of malfunction of a single rod at some time can never be ruled out entirely. Since there will be nine safety rods whose combined strength at operating conditions is greater than the maximum amount of k they will ever be required to control, failure of one, or even two, safety rods does not create a dangerous situation. Any safety rod which does not function properly will be repaired or replaced. The reactor will never be operated with less than the number of safety rods required to control all available excess k, plus two additional rods. Failure of rods to reduce reactivity is backed up by the borated water injection.

It is to be noted that to accomplish shutdown from normal operation, only enough rods need to be inserted to compensate for the amount of reactivity held in steam voids (2.3%). If, because of open valves, the ambient temperature decreases, the additional reactivity compensation is easily achieved by driving rods in under power or injecting boron.

F. Change in Power Demand

Since the demand for electricity changes in a manner which is not entirely predictable, the generator output must be varied accordingly. Changes in power demand at a rate of more than 1% per second are not expected except for outages of a transmission line. The latter type of event must be reckoned with.

An increase in load of more than that corresponding to the amount of steam being by-passed (normally about 5%) will completely close the pressure-regulating valves. Unless the reactor power level is adjusted by the operators, the turbine governor valve opening increases, thereby bringing about a gradual reduction in pressure in the reactor. If the rate of pressure change exceeds a predetermined amount, the reactor is automatically scrammed. Otherwise the operator either adjusts the control rod position to that corresponding to the new power demand or shuts down the plant manually.

A decrease in load causes the pressure to rise, opening the pressure-control valves. If the rate of pressure change is greater than a predetermined rate or if the pressure increases to 625 psig, the reactor scrams automatically. If the pressure increases to 700 psig, the relief valve is opened, dumping steam to the condenser. Further increases in pressure cause opening of the two safety valves which are set to open at 725 and 750 psig. Opening of any of these three relief valves scrams the reactor.

G. Too Much or Too Little Water in Reactor

Safety rods cannot be withdrawn if the water level in the reactor is too low or too high. Likewise, the rods will be scrammed if the water level ever gets outside these limits during operation.

If the high-level safety trip should not work or be by-passed, and if the tank were filled entirely with water, the reactor would be less safe under fast excursion conditions because expulsion of water from the core would be difficult, just as it is for conventional water-cooled reactors. It is not sufficient to just fill the reactor completely with water. Some way must also be found to add a large amount of k to the reactor suddenly. This would require rapid insertion of fissionable material or rapid removal of poison. Insertion of one enriched fuel element would add 1.2% k or less. The mechanical design is such that a fuel element cannot be added suddenly. This amount of reactivity could be controlled by the metal temperature coefficient.

A large amount of k could also be added by withdrawing the safety rod suddenly. Before this can be done the rod must be disconnected from its drive mechanism. A lifting device for attachment to the top of the rod would be required. The rod weighs about 150 lb and is therefore not easy to remove quickly. Essentially wholehearted collaboration of the entire operating staff over a considerable length of time would be required to accomplish this series of operations.

If the safety rods could be withdrawn from the reactor while it is cold and clean and with the water level well below the top of the core, and if water were added to the reactor at the feedwater pump rate of 120 gpm, the total available k of 9.6% could be added over a period of about 4 min. The maximum rate of addition of k might be as much as 0.08% per second. If this rate of addition of k were maintained, the reactor would reach a period of about 0.06 sec by the time the fuel plates were heated to saturation temperature. This period can be handled by the mechanism of self-limitation discussed in Section V. Nevertheless, safety features are incorporated in the design to prevent the occurrence of this situation; these include the releasing of rods for low water level and a water temperature below 325F.

H. Water Leaks From the External System

The dominant radioactivity in the steam circuit will be due to fast neutron irradiation of O^{16} to form N^{16} . Since the molecular density of steam is very low compared with pressurized water, the amount of N^{16} will also be correspondingly low. The short (8 sec) half-life of the N^{16} prevents accumulation of this activity in the vicinity of a leak. Even if all of the water in the system should flash down to atmospheric pressure the integrated exposure from the N^{16} is not near lethal for operating personnel in the building.

Development of faults in condenser tubes would permit circulating water to leak into the condenser. An increase in electrical conductivity of condensate will be the first indication of major leakage. Water level indicators would eventually show an increase in water inventory. Upon noting evidence of a leak the operator would isolate that half of the condenser where the leakage was occurring by closing butterfly valves in the inlet and outlet circulating water lines. The water boxes and tubes on the leaking side would then be drained to a tank in the basement of the Power Plant Area. The condenser would still be able to condense the turbine exhaust and/or by-passed steam, although at a slightly higher back pressure. The operator manually opens the valve admitting spray water into the by-pass desuperheater to prevent damage to the condenser due to high thermal gradients.

I. Cladding Failure

Zircaloy-II is quite resistant to corrosion at the 508F maximum fuel surface temperature in this reactor. However, in the fabrication of a very large number of plates it cannot be guaranteed that all welds are perfect nor that cladding will be completely free of defects. The occasional cladding failure would be followed by rapid reaction of the uranium with water, causing swelling and discharge of oxide particles to the coolant stream if the uranium were not alloyed with an inhibitor. Prior to irradiation, the 5% zirconium, 1-1/2% niobium alloy fuel proposed for this reactor corrodes at a very slow rate in water at 488F. A limited number of tests indicate, however, that the corrosion resistance is markedly reduced as the fuel is irradiated in a reactor. Development work aimed at improving this alloy or discovering a better alternate is being vigorously pursued at Argonne.

Since the possibility exists that a cladding failure may occur after a long irradiation period, conservatism demands an assumption that the corrosion rate of the fuel will be high. The bulk of the radioactivity will remain in the reactor or be removed by the reactor water purification system. The decontamination factor for non-gaseous radioactivity getting into the external system is at least 10^4 .

J. Startup Accident

The low neutron background in the freshly loaded reactor could be dangerous because of lack of knowledge of the multiplication or period. As is customary in startups of reactors and in critical experiments, a strong neutron source will be used in the initial loading and startup of the EBWR. A permanent antimony-beryllium neutron source previously irradiated to saturation and maintained at full strength by operation of the EBWR will be built into the core.

A pile period meter, set to scram safety rods at periods less than 5 sec, will be used in all reactor startups. It may not be possible to use a period meter at full power because of the short period variations of the neutron flux.

Safety rods cannot be withdrawn unless the flux-monitoring instruments are on their most sensitive scales and the water temperature is 325F or above. The rate of addition of k by rod withdrawal is limited to 0.01% per sec. With this rate of addition of reactivity the water will begin boiling as soon as the fuel plates reach a temperature just slightly higher than the boiling point of the water. The negative steam and temperature coefficient will automatically compensate the reactivity added by rod withdrawal. If no xenon were present the power level could go much higher than the nominal value of 20 mw. Since the burnout heat flux is at least six times the normal heat flux, this does not appear to be a dangerous situation. The burnout heat flux corresponds to a steam volume percentage of more than 50%.

K. Melting of Fuel Plates

Any clad fuel elements can be made to melt if an arbitrarily large amount of k is added suddenly. The Borax-I reactor was able to shut itself down with periods as short as 5 msec. An excursion with a period of 2.6 msec destroyed the core. In that reactor, the shutdown mechanism was almost entirely one of steam formation, with a minor contribution from expansion of the fuel elements. The proposed reactor is at a disadvantage relative to Borax I in that it must form much more steam per unit area of heat transfer surface for control of a given amount of excess k . This steam must be formed by transferring heat through zirconium cladding, which is a very poor thermal conductor relative to the aluminum cladding on the Borax fuel elements. The EBWR reactor has other important shutdown features, however. The metal temperature coefficient (Doppler effect and metal expansion) together with delayed neutrons is estimated to be able to compensate for 1.1% k added suddenly.

If the pressure vessel should rupture as the result of a short period excursion, at least 30% of the water could be lost from the reactor as the stored energy flashed to steam. A considerable quantity of water would be expected to be blown out also. The resulting pressure increase would be considerably less (Figure 31) than the 15 psi which the building is designed to withstand. Any fuel blown out would be expected to cool sufficiently by radiation, conduction to objects with which the element is in contact, and air convection to keep it below the melting point. The building spray system, if operating, would aid in cooling.

Possible radiation dosages resulting from leakage of fission products and plutonium from the gas-tight building are given in Appendix E.

L. Excessive Pressure

Since an increase in pressure in a boiling reactor can lead to a still further increase in pressure, mechanisms for limiting the pressure are particularly important. The safety rods are scrambled when the pressure exceeds 625 psig. A pressure relief valve opens at 700 psig. Two safety valves, one set at 725 psig and one set at 750 psig, further protect the reactor by releasing steam to the condenser through the desuperheater. Opening of these valves will reduce the power level of the reactor while the flashing is occurring. The safety rods are scrambled by the opening of any one of these valves, thereby shutting down the reactor. The safety rods are backed up by the boric acid systems, as pointed out previously. Injection of the boric acid would be completed in about 20 sec, thereby guaranteeing permanent shutdown before the temporary relief afforded by the flashing operation has ended.

In a fast power excursion which is terminated by inherent safety characteristics at a fuel temperature just short of the melting point, the total energy stored in the hot fuel would be about 500,000 Btu. If it is pessimistically assumed that this heat is dissipated instantaneously in the water, the pressure rise would be less than 200 psi. In this situation nothing would happen other than the opening of the relief valves and scrambling of the control rods.

M. Sudden Closure of Steam Valve

The steam system design is such that there will be no valves other than the turbine trip-throttle valve which can close suddenly in the main steam line and cause a major change in steam flow from the reactor. The closure of the turbine trip-throttle valve scrams the reactor. The various valves discussed in Section VI, F, merely change the proportions of steam to the turbine and condenser. The pressure-regulating system will have a dominant time constant no greater than 3 sec with a 3-second time constant and, for the case of sudden closure of the trip-throttle valve on the turbine, the pressure transient may go as high as 10% above the operating pressure, thereby scrambling the rods via the high pressure trip. Even without a pressure control system, the pressure increases at the rate of only 30 psi per second or less, which could be compensated by rod insertion. However, the reactor will be scrambled if the rate of change of pressure exceeds a predetermined rate, less than 20 psi per second.

N. Failure of Steam System

A failure of the main steam system would consist of one of three types of failures:

1. Rupture of the steam piping or a valve upstream of any control mechanisms.
2. Rupture of the steam piping or valves downstream of the control mechanisms or the turbine or condenser casing.
3. Internal mechanical failure of turbine valves or control mechanisms.

The rupture of the main steam piping upstream of the valves controlling the rate of flow of steam would permit the steam to escape into the gas-tight building. In case of a wholesale failure, the rate of steam flow would be so great as to shut the reactor down instantaneously due to flashing. The reactor safety rods would be scrambled when either the general level of radioactivity in the building rose above a predetermined level, when the water level in the reactor dropped, or when the water temperature dropped to 325F if the operators had not already inserted the rods.

The effect on the reactor of a failure of the condenser or turbine casing or other elements downstream of the control mechanisms would be quite similar to that discussed in the preceding paragraph. In addition, however, the recent disastrous failure of the shaft of the No. 4 Ridgeland steam turbine in the Chicago area, with considerable consequential damage to the building and to adjacent equipment, compels an evaluation of such an occurrence in the EBWR plant. It is held that the occurrence of this type of accident is highly improbable for the EBWR turbine. The much smaller diameter shaft to be used on the EBWR turbine is much easier to forge and test than the one which failed. Every means available to the turbine manufacturer will be used to insure a defect-free shaft from the most careful selection of the forging to the use of gamma-ray, Magnaflux, Zyglol and other non-destructive tests in the manufacturer's shop.

In the highly unlikely event that the EBWR turbine shaft should fail in the same manner as Ridgeland No. 4, the reinforced concrete lining of the steel shell should prevent puncturing of the shell by flying fragments. The failure of the turbine could either trip the throttle valve or completely open the governor valves, depending on the exact sequence and nature of the failure. Tripping the throttle valve would shut off steam to the turbine, opening the pressure regulating by-pass valves. The reactor would be scrambled by either radioactivity in the building, loss of water from the reactor, loss of vacuum in the condenser, or manually by the operators. The opening of the governor valves would increase the flow of steam from the reactor and, in addition to the scram signals just mentioned, would shut the reactor down temporarily by flashing in the core. The reactor, after being scrambled, must be cooled by using one of the cooling systems previously described, since the condenser would be inoperative.

O. Sudden Introduction of Cold Water

Since the reactor has a negative temperature coefficient, introduction of cold water increases the reactivity. This could happen, for example, if the feedwater pump were not turned on until the reactor had been raised to its operating temperature. However, the rate of change of reactivity is quite slow since only about 4% of the total water in the reactor is added by the feedwater pump per minute. Perhaps the most dangerous situation of this type would be if the main steam line is closed and the reactor brought up to temperature before the feedwater pump is turned on. The reactor power level would merely rise to a point where the increase in temperature and steam formation nullified the increase in k due to addition of cold water. The temperature at which this occurred would be somewhat less than the temperature existing when the pump was turned on. That the excursion is slow can be deduced from the fact that cooling all of the water from 488F to 70F increases the reactivity only 3.5%. Since a total change of water would take 25 min, this means that k changes at a rate about one fifth as fast as the rod-controlled k is introduced.

P. Improper Charging of Fuel

The effect of adding an enriched fuel assembly* is an increase in reactivity of 0.5% k for the average position and less than 1.16% k for the center position. If the reactor is just barely subcritical when the fuel is inserted, an excursion of 50 msec or of a longer period would be initiated.

Low reactor water temperature would have tripped all rods before the vessel lid was removed, making the reactor very subcritical. As an extra precaution the fuel elements are prevented from being accidentally dropped by a stop attached to the handling rod, as described in Section III.B. If this stop will hold the fuel just above the control rod guides. The fuel element is allowed to move the remaining distance by a speed-limiting hand-operated winch.

Careful monitoring and accounting procedures will insure that the U²³⁵ content of any enriched fuel assembly is not materially greater than the design value.

Q. Earthquake

Earthquakes in the Chicago area have been infrequent and mild. The most serious consequences of a severe earthquake, if it should occur, would be to rupture the vessel or piping and jam the safety rods. Concentrated boric acid is automatically injected into the reactor if any 4 control rods have not reached their minimum reactivity positions within two seconds after a scram signal.

In case of rupture of the reactor vessel, enough water would be lost from the reactor, suddenly or gradually depending on circumstances, to shut the reactor down. Shutdown cooling would be accomplished through the boric acid spray ring as previously described. The piping for this cooling system is designed to prevent damage from tremors.

The concentration of fuel in the core of the reactor could not be changed by earthquake since the fuel elements are bound closely together. Thus, there should be no danger of an increase in reactivity.

R. Fire

The reactor and its building will be essentially fireproof. Hazardous amounts of combustible materials will not be allowed in the building.

S. Sabotage with Fissionable Material or Explosives

As mentioned earlier, the fuel in any heterogeneous reactor can be made to melt by the sudden addition of an arbitrarily great amount of excess

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*2 1/2% U²³⁵

reactivity. This can be done by injecting fissionable material or withdrawing poison. The proposed reactor should certainly handle 1.1% k without mishap and has a good chance of handling considerably more than this without melting the fuel. It does not appear possible to add this much k without resorting to a sequence of operations requiring collaboration from the entire operating staff.

A reactor can, of course, be badly damaged by explosives. For example, the pressure vessel could be ruptured, leading to loss of water. The water lines which are depended on for emergency shutdown cooling could be broken by proper placement of explosive charges, or be disconnected. The gas-tight features of the building could also be destroyed. Although this series of events is technically feasible, it is not likely to be accomplished by only one saboteur.

T. Sabotage by Pressurizing Reactor Tank with Gas

By pressurizing the reactor with air or other gas prior to startup, the saboteur could raise the boiling point of the water and inhibit the inherent safety mechanism of steam formation. A large number of gas cylinders would be required to raise the pressure to a point where any hazard is involved. To do any damage, the saboteur must also introduce a large amount of k suddenly. The introduction of gas would merely be a way to make the excursion resulting from sudden introduction of k more serious, but would not be a hazard in itself.

U. Explosive Chemical Reaction Between Fuel Elements and Water

If the fuel elements could be made to melt in the presence of water, there is a possibility of a violent reaction between the molten metal and water. The only way in which fuel elements can be made to melt in the presence of water is believed to be by the addition of large amounts of reactivity in a very short time. For all the fuel elements, the clad is the material of highest melting point, and it is, therefore, conceivable that the meat of the fuel element can be above its melting point and still not be in contact with the water because of the unmelted sheath interposed.

For the purpose of estimating the consequences, it is customary to assume that the fuel elements are somehow made to melt. The possibility of a chemical reaction between the molten metal and water must then be reckoned with. As explained in Appendix C, the conditions under which zirconium and uranium react explosively with water have not been accurately defined. Experiments by the Aerojet Corporation produced violent reactions when molten zirconium was dispersed as fine particles in water under the influence of a blasting cap. When molten zirconium was merely poured into water, no reaction was observed, however. No dispersing agent comparable to a blasting cap is present in the reactor core. Stored fission gases represent the only mechanism available for subdividing the fuel.

The rate of heat generation in the central fuel of the EBWR reactor is about eight times that at the corners of the core and 2.3 times the average for the core. Thus, in an uncontrolled power excursion, the center of the core would melt first. If this portion of molten metal should react explosively with water the core would be blown apart before the outer regions reached their melting point. To get complete reaction between molten metal and water, these reactants must be kept in contact. Maintenance of contact is inconsistent with the assumption of an explosive reaction.

The EBWR facility is designed to withstand the effects of chemical reaction of at least 10% of the core metal. The energy release from such a reaction would be 3×10^8 calories or 1×10^9 ft lb. If such an energy release occurred suddenly the shock wave would probably rupture the reactor pressure vessel below the water line. The pressure would thereupon proceed to relieve itself through the weak members at the sides (lower) and bottom of the reactor. The bottom shield plug is designed to blow out at 300 psig. The loose shielding sections on the sides would be pushed out by even smaller pressures. The 2-ft thick (minimum) reinforced concrete lining for the basement of the gas-tight steel building should prevent damage to the steel shell.

The shock wave reaching the top of the reactor pressure vessel will be attenuated as it passes through the steam dome. The top of the pressure vessel is also 50% stronger than the bottom and is anchored to steel in the shielding, so the shock wave would not be expected to either rupture or displace the top of the pressure vessel. The large mass of water which would be propelled upward by the force of the explosion would arrive a short time later, however, and the kinetic energy of this water might be a significant fraction of the total energy of the chemical reaction.

The slug of water as it moved upward would compress the steam in the dome of the reactor. The steam would act as a cushion and would distribute the pressure uniformly over the top of the reactor vessel. If this pressure increased to 7500 psig the bolts holding the lid of the pressure vessel would break. In doing so, these bolts would absorb about 5×10^6 ft lb of energy. The next resistance to the upward momentum of the water and the vessel lid would be the 2-in. thick armor plate which is restrained at its periphery by a ring bolted down with thirty 6-in. diameter bolts, 140 in. long. These bolts will absorb about 1.7×10^8 ft lb of energy before breaking. The distortion of the armor plate prior to breaking of the bolts will also absorb about 1.7×10^8 ft lb. Thus, before the armor plate is broken or displaced, a total of about 3.5×10^8 ft lb of energy will have been absorbed in the three places cited. This is 3.5% of the energy from a complete chemical reaction or 35% of the energy from a 10% chemical reaction. Since most of the energy will be absorbed in distorting and rupturing the lower portion of the pressure vessel and in blowing components downward and outward, it is clear that the top of the reactor is adequately protected against chemical reaction of at least 10% of the core metal.

It will be noted that a woven steel cable explosion mat over the top of the reactor will stop any miscellaneous fragments, such as bolt heads, which might be directed upward. The 1-ft thick reinforced concrete lining and ceiling above the main floor also provide substantial protection against missiles from the reactor, the turbine, or any other internal source.

Expulsion of the contents of the reactor, following a reaction of 10% of the core metal with water, would raise the temperature and pressure of the building to 170F and 10 psig or less. If the hydrogen resulting from the reaction burned explosively upon becoming mixed with the building atmosphere, the upper limits for these values would become 187F and 12 psig. The steel shell would be able to contain these products without any difficulty.

If all of the metal in the reactor core should react violently with water and expel the reactor contents into the building, the equilibrium temperature and pressure would be 378F and 31 psig, respectively. If the hydrogen should burn explosively these peak values would increase to 1525F and 98 psig. Under this most pessimistic set of assumptions the gas-tight building would be expected to rupture and expel its radioactive contents into the atmosphere.

In Appendix D estimates have been made of the hazard due to airborne radioactivity resulting from an accident in which all fission products are expelled as a result of rupture of the building. The hot low-density radioactive mass would be expected to rise about one mile into the air and to drift downwind. Most of the fallout of particulate radioactivity would be expected to occur before the cloud had traveled 5 miles. The ground level activity in the fallout area near the EBWR site would undoubtedly be high. The integrated exposure of individuals to radioactivity from the overhead cloud itself would be quite tolerable, however.

Another estimate of airborne radioactivity hazard was made (Appendix D) for the case where the integrity of the gas-tight building was not violated. A leakage rate of 100 ft³/d (overage) was assumed.

APPENDIX A

SHUTDOWN COOLING

I. GENERAL

In the event of a sudden shutdown of the EBWR, accompanied by the loss of water in the reactor core, the only means for removing the decay heat from the fuel plates would be by natural circulation cooling of air or air-water vapor through the core.

Tests were made to determine the corrosion resistance of zirconium clad uranium core plates in hot air and steam. Tests run at a temperature of 1470F in moist air resulted in clad failure beginning after three hours and the complete exposing of the uranium core by four hours.

A further test run on a piece of Zircaloy-II in dry air, which duplicated the calculated temperature-time curve for the cooling of natural uranium elements in 80F ambient air for a period of six hours, indicates that the clad should remain reasonably secure for this period of time.

Figure 32 shows a summary of the results of natural circulation tests made to determine the steady-state relationship between maximum plate temperature and power density of shut-down heating for plate spacings of $3/4$ and $1/2$ in. and with a length of 4 ft. All tests were run in room temperature ambient air except one series that was run in a 210F air-water vapor mixture. Figure 33 shows the results of approximate transient calculations that were made to determine the time-temperature relationship of fuel plates with natural circulation vapor cooling. These calculations were based on the data shown in Figure 32 and the estimated decay heat curve of the core.

Figure 34 shows the relationship of heat flux at the vessel wall and decay heat with time, assuming that all of the heat has to be dissipated through the vessel wall. Because of a very low convective heat transfer coefficient, of the order of magnitude of 1 Btu/(hr) (ft²) (F) for air or steam, it can be seen from Figure 34 that excessively high film drops would occur if all the heat had to be transferred by convection to the vessel wall.

It can be seen from Figure 33 that, on the basis of present zirconium corrosion data, if a 210F ambient atmosphere could somehow be maintained, an improvement in clad material corrosion resistance would be necessary for the natural uranium plates to cool satisfactorily.

However, all of these above mentioned conditions or modifications are dependent upon the maintenance of an ambient temperature of 210F or less. Since no cooling method is known for maintaining the ambient

4 ft CHANNEL LENGTH, 4 1/2 in. WIDTH

FIG. 32
NATURAL CIRCULATION COOLING TEST RESULTS

FIG. 33
TEMPERATURE VS TIME AFTER SHUTDOWN FOR NATURAL
URANIUM FUEL ELEMENTS IN PEAK FLUX CHANNEL

FIG. 34
 REACTOR CHANNEL DECAY HEAT AND VESSEL SURFACE
 HEAT FLUX (Q/A) VS TIME (ASSUMING ALL CORE
 DECAY HEAT UNIFORMLY ABSORBED OVER INNER
 SURFACE OF REACTOR VESSEL)

temperature within an enclosed, waterless reactor vessel to less than about 1000F, shutdown cooling gains brought about by power density reduction and increased clad corrosion resistance are of relatively little consequence.

II. COMPARISON OF REACTOR AND TEST CONDITIONS

A. Effect of Longitudinal Heating Distribution

Calculations indicate that there is not much difference in the net circulating head for natural convection of air or steam in a channel with a uniform heat flux along its length as compared to a channel that has the same total heat output but distributed along its length in a symmetrical cosine pattern. Both steam and air would be expected to have a slightly higher temperature rise under natural circulation cooling (probably less than 5%) with reactor cosine distribution heating than with uniform heating.

B. Effect of Exit Losses

Exit cooling losses are thought to be higher in the channel arrangements tested than in the actual reactor since there would not be as great a percentage area enlargement going from all of the reactor channels to the plenum as there is in going from the few test channels to the open room. Because of this, it is thought that there will be a slightly smaller air temperature rise across the core in the reactor (about 5%) than was achieved in the tests. However, this cooling difference is small and there is some question that it would be even as much as 5%.

III. CONCLUSIONS

The results of this work indicate that the reactor core will not be cooled adequately during shutdown when there is a loss of water from the core area. Means for adding water to the core are necessary to be sure that there will not be any fission product release due to fuel plate melting or clad failure. In general, the reason for this conclusion is based on tests and calculations which show that it is necessary to have an ambient temperature of not more than 80F in order for the core to have a chance of being vapor cooled by natural circulation without having any fuel plate clad failures. This 80F ambient temperature cannot be obtained unless there is a major rupture of the reactor vessel exposing the core to room temperature air. External cooling of the wall of a sealed reactor vessel by water or other means will not keep the ambient temperature this low. If the reactor vessel wall were 60F, this alone would not keep the ambient temperature below 1000F during shutdown. It is more likely to go above 2000F.

The following is a summary of different shutdown conditions associated with loss of water which could occur:

A. A reactor failure that results essentially in the confinement of air within a waterless reactor vessel will cause fuel elements to melt, even if the vessel walls are cooled and maintained at 60F by some means. This is because the shutdown heat production of the core cannot be transferred to the inner surface of the reactor vessel by air unless the air temperature is more than 2000F above the temperature of the vessel wall, even after the first two hours following shutdown. Radiation cooling of the fuel elements by the vessel wall is limited to those elements at the outer radius of the core. The high flux elements near the center are hidden from radiation cooling except for a small section at their ends.

Water vapor in the air can reduce ambient temperatures some because of the additional heat that can be handled by radiation directly from the moist vapor to the vessel walls, should the vessel be externally cooled. However, even with a steam atmosphere and a vessel wall temperature of 60F, the maximum ambient temperature would probably not be lower than approximately 1000F, which would result in fuel plate melting anyway.

B. A failure that results in the loss of part of the core water and exposure of a major part of the fuel elements above the water could result in the melting of portions of the fuel elements and the splitting open of the zirconium clad of some others. Cooling of the vessel wall could make conditions even worse. This is because of what little cooling the exposed elements get; practically all is due to the steam formed by the vaporization of the water in the lower half of the core. This vaporization would be reduced because of the cooling of this water by the vessel wall.

C. If a failure occurred in which an air or steam ambient temperature of about 210F was somehow maintained in the reactor vessel, it is expected that some of the fuel plate cladding of the natural uranium plates would fail, exposing uranium. There is insufficient test information available at the present time on zirconium corrosion to indicate whether or not the cladding of the enriched fuel plates would also be expected to fail. It should be pointed out that it would be necessary to somehow have some water or vapor makeup to maintain a 210F atmosphere. A failure that could result in this condition is considered less likely than the type of failures mentioned in items A and B.

D. Fuel plate cooling in the event of a failure that results in the core being exposed to 80F ambient air cannot be predicted at this time because of insufficient information on the corrosion resistance of zirconium. However, just on the basis of present corrosion test data, it appears probable that all elements will cool without any melting or clad failure in air. However, it is felt that a failure that would be severe enough to expose the core to 80F ambient air could very likely result in sufficient distortion of some core cooling passages that adequate natural circulation cooling would be prevented anyway and fuel plate melting or clad failure would result.

APPENDIX B

HYDROLOGY AND SEISMOLOGYI. SOIL AND HYDROLOGYA. Surface Characteristics and Drainage

Topographically, the DuPage site of Argonne National Laboratory is a rolling and partly rugged, partly forested plateau rising quite sheer from the DesPlaines River. The site is located on soil of a composition differing from the locality in general. Approximately 80% of the site is covered with yellow-gray silt loam (upland timber soil) and most of the remaining surface is exposed limestone, glacial river bottom, and river silt soils. This moraine overlies a Niagaran limestone formation to a depth varying from zero to 160 ft. Less than 4% of the soil is of the typical DuPage County composition, which is brown and black silt loams. There is very little precipitation penetration because, due to the type of soil, almost all rainfall is surface drained.

Bounding the DuPage site (elevation plus 750 ft) on the south is the DesPlaines River (elevation plus 590 ft) which flows southwesterly. Several small creeks drain the surface water of this site into the river. The DesPlaines River, the Sanitary and Ship Canal, the Illinois and Michigan Canal, and the Sag Channel have all been interconnected by the time they reach Joliet, which lies about 14 miles southwest of the DuPage site.

The water flow in the DesPlaines River just prior to Joliet varies between 10 and 100 cu ft per sec depending upon the season. When the flow rate exceeds 50 cfs, the excess is diverted to the Sanitary Canal, at the point just south of Lemont, before reaching Joliet. The flow rate of the Sanitary Canal for 1944 and 1945 averaged about 3400 cfs; of this about 1600 cgs is water which has been diverted from Lake Michigan, and the balance is made up of domestic and industrial discharge from this district. Both the river and the canal flow approximately 20 miles per day. Any surface drainage, therefore, from the DuPage site would be diluted first by the river, bordering the site, and then further diluted by the canal within 14 miles, normally within 17 hours, unless such drainage occurred during a wet period when it would then be diluted more extensively and sooner.

Canal water is used prior to and in Joliet for some industrial uses, such as hydro-electric generators and condensers, as well as for irrigation at the state prison. After leaving Joliet, the DesPlaines River carries these waters southwest to a point approximately 14 miles further where the river is joined by the Kankakee River forming the Illinois River. The first use made of this water for drinking purposes is at Alton, Illinois,

which is about 275 miles from the DuPage site and 10 miles downstream from the point where the Illinois River merges with the Mississippi River. Both Alton and East St. Louis treat about five million gallons per day of this Mississippi River water.

B. Sub-surface Water Conditions

The geology of the DuPage site including the sub-surface water conditions is described in a report entitled "Geology and Ground-Water Hydrology of the Argonne National Laboratory Area, DuPage County, Illinois" by Allen, Drescher and Foley, consultants to the U. S. Atomic Energy Commission. The following statement concerning any possible hazard due to water-borne activity has been prepared by Dr. F. C. Foley, who is the senior author of the above-mentioned report.

Ground water in the Laboratory area is obtained from two principal aquifers, the Niagara dolomite and deeply buried Ordovician and Cambrian sandstones. The two shallow supply wells in the Laboratory area obtain water from the Niagara dolomite while the deep well is cased and sealed through the Niagara and gets its water from the deep sandstones. Only the Niagara dolomite needs to be considered in connection with possible contamination from an accidental spill of radioactive liquid, as the deep aquifers are separated from the surface by hundreds of feet of rock formations of which the Maquoketa shale is particularly impermeable.

The uppermost formation at the Laboratory site is glacial till of Late Wisconsin age, which overlies the Niagara dolomite. It covers the whole area except along the bluff on the valley of the DesPlaines River, Chicago Sanitary and Ship Canal and along Mill Creek below Bluff Road. The till has a maximum thickness of about 180 ft. It consists of clay and silty clay containing fragments of dolomite and other consolidated rock, and a few thin lenses of sand.

The glacial till is relatively impermeable and water moves through it very slowly. No experimental work has been done at the Laboratory site to determine the exact rate of movement, but such work is planned. At the site of the Argonne Research Reactor CP-5, the glacial till is about 110 ft thick. It is estimated that about one year would be required for water introduced near the surface to penetrate to the top of the Niagara dolomite. All the time, and even if it did, it would be diluted with normal ground water descending with it. Once it entered the Niagara it would move through open fissures down the hydraulic gradient, which is generally southeastward and not toward the present supply wells. The closest well in use is the meteorology station well. It might seep gradually into the DesPlaines River over an extended period of time. In view of the great thickness of relatively impermeable glacial till at the reactor site, and the direction of the hydraulic

gradient in the Niagara dolomite, it is very doubtful that radioactive liquids would enter the Laboratory supply well or private wells. If contaminated liquid could move toward a supply well, it is probable that such liquids would be extremely diluted by natural ground water.

II. SEISMOLOGY

Realistic predictions regarding earthquakes or earthshocks, their frequency and severity, can only be based upon the seismic history of the area concerned. Apparently, new trouble areas seldom occur and this is particularly true of the United States. Insofar as the Argonne sites are concerned, it is expected that no building of sound construction on firm ground will suffer any damage as a result of the infrequent and small earthquakes which may occur anywhere within 150 miles from the sites. It is conceded that earthquakes occurring as far as 500 miles away may disturb sensitive laboratory apparatus.

APPENDIX C

METAL-WATER REACTIONS

I. GENERAL

Specific tests on explosive metal-water reactions with aluminum, uranium, and zirconium (and some of their alloys) have been carried out at the following installations:

*Appendix C
Ad 135-138*

1. zirconium (Ref. 1).

onium (Zircaloy)

onium (Ref. 3).

Co.); zirconium (by
Ref. 4).

n (Ref. 6).

sylvania Electric Prod-
nsequence and have
e industrial experience
t Alcoa (Ref. 7), but
-Sharon Plant in Niles,

; tests differ greatly;
each experiment. It
temperatures, etc.,

nium, and zirconium
denser discharge. At
a result of the following
i.

NSIC 691 A

(However, only in the case of aluminum was a run made with lower energy release, resulting in a dud.) Higher energy releases resulted, in general, in finer particle subdivision and more complete reaction.

B. Aerojet-General Corp.

Molten metal (1/2 lb) was poured into water. No reaction resulted with aluminum, zirconium or Zircaloy. Detonation of a No. 6 blasting cap (with an energy of 200 cal) in the water when the molten metal hit it resulted in violent explosions for both zirconium and Zircaloy but not aluminum. It should be noted that the tests were made at temperatures only slightly above the melting points, which are very different for zirconium and aluminum.

Induction heating of zirconium in water (presumably to melting, but at an unknown rate) resulted in no reaction.

C. Mine Safety Appliances Company

No reaction occurred when zirconium or aluminum was melted in an argon-water vapor atmosphere or when melted and dropped into a large excess of tap or synthetic sea water (15 g metal into 150 ml). When the amount of water was decreased to 10-15 ml, a mild, incomplete reaction (fizzing, stopped by the evaporation of the water) followed, resulting in hydrogen evolution. Somewhat more hydrogen was given off in the case of zirconium than aluminum.

D. MTR

No reaction resulted when uranium-aluminum alloy and uranium-zirconium alloy (50 and 200 mg with up to 12 w/o uranium) in water-filled autoclaves were lowered into the operating reactor. No details on the aluminum tests are known here, but Westinghouse personnel are confident that none of the zirconium samples approached melting. (They have seen the samples on all but their most severe test and Argonne calculations indicate that the sample temperature reached about 1500C, versus about 1820C melting point.) Equilibrium temperature is reached in 5 to 10 sec and maximum heating rate is about 0.14 cal/mg/sec.

E. duPont

Two-pound samples of molten aluminum at 900 to 1000C dropped into 20 lb of water in a relatively confined space resulted in no reaction (in 17 tries). Five per cent uranium-aluminum alloy did not react either (in 28 tries) unless KCl flux had been used in melting. Only one experiment was made with pure uranium and no reaction resulted.

F. Argonne National Laboratory

Fine sprays of molten aluminum at 740C and 1000C injected into water resulted in no reaction.

G. Alcoa

Under certain conditions, violent explosions occurred when molten aluminum was poured into water. Apparently a solid stream of metal is required; a grid which breaks up the stream is effective in preventing explosions.

H. Mallory-Sharon

A leak of water into a crucible containing molten titanium resulted in a violent reaction, causing equipment rupture. Released hydrogen exploded in the air, resulting in the serious damage and fatalities.

III. DISCUSSION

Based on the experimental results listed above, there appear to be at least three important variables:

1. Rate of energy input to metal.
2. Particle size.
3. Ratio of metal to water.

The first two appear to be interrelated in that higher energy inputs result in finer particle sizes.

It should be noted that in the NAA work, the energy input for both aluminum and uranium was not sufficient to heat the entire sample to melting. (It is reasonable to suppose that zirconium would have reacted with a lower energy input than was tried.) Assuming no heat loss, final uranium and aluminum temperatures were about 850C and 300C, respectively, compared to their melting points of 1130C and 660C; yet explosions resulted. On the other hand, Westinghouse samples of zirconium, brought reasonably close to melting at a much slower rate (0.14 cal/sec/mg versus about 400 cal/sec/mg) did not react. Somewhere in that vast range is the critical rate for reaction, undoubtedly dependent on temperature. The NAA experiments do not correspond to any conceivable reactor period but on the roughest type of approximation, the Westinghouse result represents the EBWR on about a period of 1.5 sec. The Westinghouse results will of course be much more significant when melting is reached.

If the material is in the molten state, only a small, rapid energy release (a blasting cap whose energy would not raise the metal temperature measurably) is required to set off the reaction in zirconium, although it is not sufficient for aluminum.

Although available reports are not too clear on this point, the titanium explosion was probably a case of not much water in a lot of metal. This would be in line with the MSA results which indicate a lot of water is less likely to cause a reaction than a little. For zirconium, therefore, the pattern is rather clear cut, i.e., it is more reactive at the extremes, such as fine particles in a lot of water, and massive molten metal mixed with comparatively less water.

For aluminum, the pattern is less clear. It can be exploded when finely subdivided, as shown by NAA, but Argonne and Alcoa work indicate that thick, molten streams are more reactive than fine sprays.

REFERENCES FOR APPENDIX C

1. NAA-SR-197, October 27, 1952.
2. Memo, L. W. Fromm to J. M. West: "Metal-Water Reaction Experiments at Aerojet-General Corporation," December 13, 1954.
3. MSA-PR-25, December 20, 1954.
4. WAPD-P-513, (unknown).
5. REA(SPP)-51, November 29, 1954.
6. ANL-4503, October 1, 1950, pp 6-8
ANL-4549, December 29, 1950, pp 5-6 and 10-23.
7. Russell, A. S., via NAA-SR-197.
8. Bureau of Mines, "Report of Explosion Disaster, Titanium Melting Furnace, Mallory-Sharon Titanium Corporation; Niles, Ohio, June 11, 1954."

APPENDIX D*

EVALUATION OF RADIOACTIVE HAZARDS IN
THE SURROUNDING AREA

I. INTRODUCTION

In this section the hazards in the surrounding area are evaluated, based on a number of assumptions representing relatively unfavorable conditions. It is assumed that: (1) metal-water reactions occur with complete burnup of the uranium and zirconium cladding, (2) the reactor pressure vessel and building are ruptured at a time when nearly all of the water has reacted with the metals, (3) the resulting oxides of uranium and zirconium, including the fission products, are finely divided, (4) fifty percent of the fission products are expelled from the building, and (5) Sutton's formulae are valid for determining diffusion of the cloud formed from the material leaving the building.

As indicated earlier, the volume of the building is 400,000 cu. ft. After complete burnup occurs, the major products, including building air, are as listed in Table X.

Table X

PRODUCTS OF COMPLETE BURNUP OF CORE

Reactor Product	No. of Moles	Mass in Grams
N ₂	366,000	10.2 x 10 ⁶
O ₂	63,450	2.0 x 10 ⁶
H ₂ O	489,000	8.83 x 10 ⁶
UO ₂	16,150	4.37 x 10 ⁶
ZrO ₂	33,750	3.08 x 10 ⁶
	Total	28.5 x 10 ⁶

If the uranium and zirconium react with water and the resultant hydrogen burns in the building atmosphere, 9.5×10^9 cal are released. The resulting temperature and pressure in the building will be 852°K and 86.5 psig, respectively.

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*This appendix is a revision based upon more recent knowledge.

If the building is ruptured after the reaction is nearly complete, the gases will expand adiabatically to atmospheric pressure. By the use of Poisson's equation

$$\frac{T_1}{T_2} = \left(\frac{P_1}{P_2} \right)^{\left(\frac{C_p - C_v}{C_p} \right)},$$

one obtains a value of 510°K for the resulting temperature. T_1 , T_2 , P_1 and P_2 are corresponding temperatures and pressures, and C_p and C_v are the specific heats at constant pressure and volume, respectively. The radius of a sphere which will contain these gases is calculated to be 21.5 meters and may be considered the initial dimension of the cloud. The density of the cloud material, the mass divided by the volume, would be 720 g/m³. At a temperature of 0°C and a pressure of 29.92 in. of mercury, the density of dry air is 1293 g/m³.

Since this sphere has a density of substantially less than that of the surrounding atmosphere, it will rise. The rise will be very rapid at first and will slow down as it approaches its terminal height.

II. DOSAGES FROM SPHERICAL RADIOACTIVE CLOUD

A. Cloud Growth and Rise

Sutton's formula¹ relating the distribution of matter or energy from an instantaneous point source due to atmospheric diffusion was used to calculate the rate of cloud growth. This formula may be written as:

$$X(r) = \frac{Q}{\pi^{\frac{3}{2}} (r_0)^3} \exp. \left\{ -\frac{r^2}{r_0^2} \right\} \quad (1)$$

where

$X(r)$ = concentration of matter or energy in units per m³

Q = number of units of matter or energy in cloud

r = distance from center of cloud in meters

r_0 = distance from center of cloud in meters where concentration

is $1/e$ that at the center; $r_0 = C(r_V + d)^{1 - \frac{n}{2}}$

d = distance traveled from the source $\frac{n}{2}$

C = a diffusion parameter in units of (meters) $\frac{n}{2}$

n = a dimensionless diffusion parameter

r_V = distance from actual to virtual source in meters.

In the case of a reactor incident, a cloud released from a building would have a finite volume at time t equal to zero. Since the formula assumes that the volume at the source is zero, it is proper to assume a virtual source, r_V meters from the actual source. The quantity r_V , therefore, represents the distance the cloud would have to travel before its volume equaled that of the actual cloud at time zero.

In equation 1 it is assumed that the material is contained in a spherical volume and that the concentration along any diameter is Gaussian, with the maximum at the center.

The radius of the cloud, R , is arbitrarily assumed to be the distance from the center to the point where the concentration equals one tenth that at the center. From (1) it follows that the concentration, X_C , at the center of the cloud may be expressed as:

$$X_C = \frac{Q}{\pi^{\frac{3}{2}} r_0^3} ;$$

therefore

$$\frac{1}{10} X_C = X_C \exp. \left(-\frac{R^2}{r_0^2} \right)$$

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¹O. G. Sutton, "Micrometeorology," McGraw-Hill Book Company, Inc., New York (1953).

Dividing both sides by X_c and taking logarithms, it follows that:

$$R = (\ln 10)^{\frac{1}{2}} r_0 \text{ or } 1.52 r_0.$$

Values for the cloud radius as a function of horizontal distance from the actual source are given in Table XI.

The value of d as used above consists of two components: (1) that due to horizontal travel, and (2) that due to the vertical rise. If the cloud rises to a height h at an average vertical speed of w , then the resultant distance is $\left\{ h^2 + \left(\frac{h}{w} u \right)^2 \right\}^{\frac{1}{2}}$, where u is the average wind speed and w is the average rate of vertical rise.

The total distance traversed, d , is therefore $\left\{ h^2 + \left(\frac{h}{w} u \right)^2 \right\}^{\frac{1}{2}} + u(t - t_1)$, where $t_1 = \frac{h}{w}$, the time required for the cloud to reach the height h , and t is the time since the formation of the cloud, provided $t - t_1 > 0$. In the following calculations $u = 1$ meter/sec and $w = 7.5$ meters/sec; therefore

$$d = 0.88h + ut.$$

Estimates of the height of rise were made from another formula developed by Sutton.² The formula used is:

$$h = \left[\frac{2(3m + 2p)Q}{9C_p \rho \pi^{\frac{3}{2}} C^3 a} \right] \frac{1}{P + \frac{3}{2}m} \quad (2)$$

where

$$p = 1$$

$m = 2 - n$; n is a dimensionless diffusion parameter

Q = quantity of heat injected into the cloud in calories

C_p = specific heat at constant pressure of the cloud in calories/degreeC/gram

ρ = density of the cloud in grams/m³

C = diffusion parameter in (meters) ^{$\frac{n}{2}$}

a = rate of change of potential temperature with height.

In this analysis as well as throughout the discussion of the instantaneous cloud, $n = 0.5$, $C = 0.45$, and $C_p = 0.24$. The expression (2) therefore reduces to

$$h = 2.15 \left[\frac{Q}{\rho a} \right]^{0.308}$$

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²O. G. Sutton, "Note on Entrainment and the Maximum Height of an Atomic Cloud" by Lester Machta, Bull. Am. Meteorol. Soc., 31(6): 217-218 (1950).

The density was taken as 720 grams per cubic meter and the energy as 9.5×10^9 calories. If α is assumed to be $3.5 \times 10^{-3}^\circ\text{C}$ per meter, the value for a standard atmosphere, the height of rise is calculated to be 1880 meters. For a strong inversion of 5°C per 50 meters, such as frequently occurs at the Laboratory, the height of rise becomes 720 meters.

Since there is uncertainty concerning both the amount of energy imparted to the cloud and the effect of particulates in retarding its rise, calculations were performed for rise heights of 100, 300, and 500 meters and also for no rise at all.

B. Dosages at Ground Level from Overhead Cloud

1. Beta Dosages

The beta dose received by the body of a man standing on the ground from a spherical cloud whose center passes overhead may be calculated from the expression:

$$\frac{\text{Total Integrated Beta Dosage in Rads}}{\frac{\rho_0}{\rho}} = (0.5)(0.64) \times \frac{1 \text{ rad}}{8.1 \times 10^{10} \text{ Mev/m}^3} \frac{\rho_0}{\rho} \int_0^\infty X_\beta dt, \quad (3)$$

where $\frac{\rho_0}{\rho}$ is the ratio of air density at sea level to that at Argonne. Since the reactor site is at about 700 feet above sea level, this ratio may be considered as unity.

$$X_\beta = \frac{2 Q_\beta \exp. \left\{ -\frac{r^2}{r_0^2} \right\}}{\pi^{\frac{3}{2}} r_0^3}$$

The symbol r_0 was previously defined and

$$r = \sqrt{(x - ut)^2 + y^2 + (z - h)^2}$$

A Cartesian coordinate system is used with the origin at the reactor; x denotes the downwind distance, y is the crosswind distance, and z is in the vertical direction. Q_β represents the beta source strength in Mev/sec:

$$Q_\beta = 2.3 \times 10^{14} P t^{-0.21},$$

where P is the previous steady power in kilowatts.

The use of the expression (3) assumes that the cloud concentration is nearly uniform over the range of the beta radiation (roughly 1 to 10 meters). If it is assumed that the cloud neither changes in size nor that the fission products decay in the interval of passage past an individual

Table XI
RADIUS OF RADIOACTIVE CLOUD IN METERS AS A
FUNCTION OF DISTANCE FROM REACTOR

Distance from Reactor, meters	Cloud Height, meters			
	Ground Level	100	300	500
300	70	79	97	112
500	91	99	114	134
800	119	126	144	161
1600	185	193	208	240
3200	301	315	321	331
6400	499	506	512	523
12800	827	836	844	853

Table XII

TOTAL INTEGRATED BETA DOSAGES IN RADS
FROM RADIOACTIVE CLOUD
(50% of Fission Products Released)

Distance from Reactor, meters	Cloud Height, meters			
	Ground Level	100	300	500
300	830	17	<1	<1
500	436	32	<1	<1
800	235	49	<1	<1
1600	83	41	<1	<1
3200	27	20	3	<1
6400	9	7	4	1
12800	3	3	2	1

standing at the ground, equation (3) may be integrated with respect to time which is taken as $t = \frac{x}{u}$, provided $r^2 = (x - ut_2)^2 + y^2 + (z - h)^2$. The symbol t_2 represents the time taken for the center of the cloud to reach the observer.

Letting $y = 0$ and $z = 0$, and assuming that ut_2 is large, it follows that

$$\int_0^\infty X_\beta dt = \frac{2 Q_\beta \exp\left\{-\frac{h^2}{r_0^2}\right\}}{\pi u r_0^2} \quad (4)$$

Assuming that 50% of the fission products are released to the atmosphere, equation (3) becomes

$$\frac{\text{Total Integrated Beta Dosage in Rads}}{= \frac{(0.5)(0.64)(2.3) \times 10^{18} \times 2 \times t^{-0.21}}{8.1 \times 10^{10} \pi u r_0^2} \exp\left\{-\frac{h^2}{r_0^2}\right\}} \quad (5)$$

or

$$\frac{5.8 \times 10^6 \times t^{-0.21}}{r_0^2} \exp\left\{-\frac{h^2}{r_0^2}\right\} \quad (6)$$

The factor 0.5 in equation (3) arises from the fact that the human body is a nearly perfect absorber of beta radiation and therefore the flux per unit area at the skin surface over most of the body is one half that in air. A further reduction of radiation flux is caused by the earth's surface. At the ground the flux is reduced to $1/2$ that in free air. For a standing man, 1.8 meters tall, J. J. Taylor³ has computed the factor to be 0.64. This factor is included in equation (3).

Table XII shows the values of beta dosages for the three cloud heights 100, 300, and 500 meters and also for a cloud which hugs the ground. It is evident from this table that the beta dosages received from clouds 100 meters and higher for distances beyond one mile are relatively small. The beta dosages for a cloud whose center is higher than 500 meters would be 1 rad or less. Only if the cloud should stay near the ground do the dosages become appreciable and these occur only within a mile of the reactor incident. This area of large dosages is on Laboratory grounds. If the cloud fails to rise, it is likely that appreciably less than 50% of the fission products would be released to the atmosphere, with the result that the beta dosages would be substantially less than indicated.

2. Gamma Dosages

Total integrated gamma dosages in roentgens at a point at the ground were calculated through the use of a nomogram developed by J. Z. Holland.⁴ It was assumed that 50% of the fission products escaped to the atmosphere.

³Meteorology and Atomic Energy, Government Printing Office (July 1955)

⁴Ibid., p. 108.

The total integrated gamma dose is defined as the dose received at a point on the ground due to the gamma radiation from the spherical cloud as its center approaches, passes overhead, and moves away. In constructing the nomogram Holland considered multiple scattering buildup, attenuation due to scattering and absorption, and radioactive decay according to the $t^{-0.21}$ law. Table XIII shows the calculated gamma dosages at distances of 300 to 12,800 meters (roughly 1000 feet to 8 miles) from the reactor.

It is apparent that, if the cloud rises to 100 meters or higher, the dose at the ground is rather small, 20 r or less. Serious dosages result if the cloud does not rise. However, if this should happen, then it is probable that the intensity of the explosion would be less than postulated and appreciably less than 50% of the fission products would be released. In this case smaller dosages than shown in Table XIII would occur. For clouds exceeding 500 meters in height, the gamma dosage would be 2 r or less.

C. Dosages during Fumigation

When a nocturnal inversion is destroyed by surface heating shortly after sunrise, pronounced vertical mixing takes place, with the result that relatively high concentrations of the contaminant from either a cloud puff or plume may be brought to the ground. This process is called fumigation.

A formula for calculating the concentrations (X) during the fumigation process underneath the center line of a plume or the time integral of concentrations to be expected underneath the cloud as its center approaches, passes over-head, and recedes was given by Holland.⁵ His expression may be written as:

$$X = \frac{Q}{\pi^{\frac{1}{2}} u H r_0} \quad (7)$$

In the derivation he integrates the concentration with respect to height and mixes the contaminant uniformly throughout a layer of thickness H.

This formula actually oversimplifies this process and gives a pessimistic result, since H is really a function of time, increasing as more and more of the inversion is destroyed. However, the calculations of beta dosages shown in Table XIV represent the upper limit which may be expected. In the calculations for this table it was assumed that 50% of the fission products were released to the atmosphere and the dosages are given in rads. The height of the cloud center was taken as H.

Since the wind speed was taken as 1 meter per second, dosages for wind speeds other than 1 meter per second may be determined by

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⁵Meteorology and Atomic Energy, Government Printing Office, (July 1955).

Table XIII

TOTAL INTEGRATED GAMMA DOSAGES IN
ROENTGENS FROM RADIOACTIVE CLOUD
(50% of Fission Products Released)

Distance from Reactor, meters	Cloud Height, meters			
	Ground Level	100	300	500
300	400	15	4	< 1
500	300	15	4	< 1
800	180	20	4	< 1
1600	70	20	6	< 1
3200	34	20	8	1
6400	13	10	6	2
12800	5	4	5	1

Table XIV

TOTAL INTEGRATED DOSE OF BETA RADIATION IN
RADS FOR FUMIGATION CASE ASSUMING
50% FISSION PRODUCT RELEASE

Distance from Reactor, meters	Cloud Height, meters		
	100	300	500
300	294	81	41
500	211	61	31
800	150	44	24
1600	85	26	14
3200	45	15	8.6
6400	28	7.9	4.6
12800	13	4.3	2.5

dividing the values in Table XIV by the actual wind speed. In this expression the symbols have the same meaning as in the previous sections. Q is taken to be equal to $2.3 \times 10^{18} \times t^{-0.21}$ Mev/sec.

In the case of a ground source, the values for beta dosages such as given in Table XII give the upper limit for the dosages which would occur during the fumigation case, since more diffusion would take place during the fumigation process.

An estimate of the gamma dosages from the cloud moving on the ground can be obtained from the gamma dosage values given for $h = 0$ in Table XIII. Again these values represent the upper limit of dosage because of the increased diffusion rate in the fumigation case.

Examination of Table XIII shows that relatively high dosages would occur in the first half mile but become rather small in the region beyond the Laboratory boundaries.

D. Dosages Arising from Ground Deposition due to Rainout

If a reactor incident should occur during rain or should rain begin shortly afterward, fission products would be deposited on the ground. Some of the deposited material would sink into the soil or be deposited over a vegetative cover, so that its effectiveness in producing radiation above ground would be diminished. However, in these calculations it is assumed that none of the material actually penetrates the ground nor does vegetation act as a shield to reduce the dosages.

A method for calculating the deposition of contaminants from a cloud was developed by A. C. Chamberlain.⁶ In his work he assumed that the amount of material deposited is a function of the quantity present or

$$Q = Q_0 \exp\left(-\frac{\Lambda x}{u}\right) \quad (8)$$

where

Q_0 = the strength of the source at the time deposition begins

Q = the strength of the source at some other time

Λ = an elimination constant which may be expressed as percentage removed per sec

x = distance from the reactor

u = the mean wind speed.

Sutton's formula for an integrated instantaneous source modified for the effect of rainout may be expressed as:

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⁶A. C. Chamberlain, Aspects of Travel and Deposition of Aerosol and Vapour Clouds, A.E.R.E. HP/R 1261, Harwell, Berks (1955).

$$X_{(x,z)} = \frac{2 Q_0}{u \pi (r_0)^2} \exp\left(-\frac{\Delta x}{u}\right) \exp\left[-\frac{(z-h)^2}{r_0^2}\right]. \quad (9)$$

The amount of cloud $w(x)$ deposited per second per m^2 at a point over which the center of the cloud passes may be expressed as

$$w(x) = \Delta \int_0^\infty X(x,z) dz. \quad (10)$$

Substituting (9) in (10) and integrating with respect to z , there is obtained

$$w(x) = \frac{\Delta e^{-\frac{x}{u}}}{\pi^2 u r_0} Q_0. \quad (11)$$

This expression is a maximum for $\Delta = \frac{u}{x}$. Substitution of $\Delta = \frac{u}{x}$ in (11) yields

$$w(x)_{\max} = \frac{Q_0}{e \pi^2 r_0 x}. \quad (12)$$

Table XV shows the beta dose rates received at a point 1 meter above the ground from rainout at optimum scavenging rate. These values were calculated from the expression:

$$\begin{aligned} \text{Dose Rate of Beta} \\ \text{Radiation due to Rainout} &= \frac{Q_0 E_\beta}{e \pi^2 r_0 x}, \end{aligned}$$

where E_β was taken equal to 1.25×10^{-11} rad/sec/Mev/ m^2 /sec and Q_0 was equal to $2.3 \times 10^{18} t^{-0.21}$ Mev/sec. Relatively high values are shown in this table. Although no account is taken for shielding by clothes, vegetation, or ground penetration, nonetheless this calculation indicates that serious dosages would be received by anyone standing over contaminated ground for even 15 minutes.

The dose rate of gamma radiation was calculated from the expression:

$$\begin{aligned} \text{Dose Rate of Gamma} \\ \text{Radiation due to Rainout} &= \frac{Q_0 E_\gamma}{e \pi^2 r_0 x}, \end{aligned}$$

where E_γ was taken equal to 1.51×10^{-13} r/sec/Mev/ m^2 /sec. The other symbols were defined previously.

The calculations shown in Table XVI indicate the dose rate of gamma radiation at a point 1 meter above the ground. It is evident from these calculations that a serious hazard would exist if a person stayed out in the rain or was located for an appreciable time over an area where contamination of this magnitude occurred.

Table XV

BETA DOSE RATE (rads/sec) AT 1 METER ABOVE GROUND DUE
TO DEPOSITION OF RADIOACTIVE MATERIAL AS A RESULT
OF RAINOUT AT OPTIMUM SCAVENGING RATE

Distance from Reactor, meters	Cloud Height, meters			
	Ground Level	100	300	500
300	132	116	95	82
500	55	50	44	37
800	23	22	19	18
1600	6.5	6.4	5.8	5.1
3200	1.7	1.7	1.6	1.6
6400	0.45	0.45	0.44	0.42
12800	0.12	0.12	0.12	0.12

Table XVI

GAMMA DOSE RATE (r/sec) AT 1 METER ABOVE GROUND DUE TO
DEPOSITION OF RADIOACTIVE MATERIAL AS A RESULT OF
RAINOUT AT OPTIMUM SCAVENGING RATE

Distance from Reactor, meters	Cloud Height, meters			
	Ground Level	100	300	500
300	1.6	1.4	1.2	9.8×10^{-1}
500	6.6×10^{-1}	6.0×10^{-1}	5.3×10^{-1}	4.5×10^{-1}
800	2.8×10^{-1}	2.7×10^{-1}	2.4×10^{-1}	2.1×10^{-1}
1600	7.9×10^{-2}	7.6×10^{-2}	7.0×10^{-2}	6.1×10^{-2}
3200	2.1×10^{-2}	2.0×10^{-2}	2.0×10^{-2}	1.9×10^{-2}
6400	5.5×10^{-3}	5.4×10^{-3}	5.3×10^{-3}	5.1×10^{-3}
12800	1.4×10^{-3}	1.4×10^{-3}	1.4×10^{-3}	1.4×10^{-3}

III. SLOW DISCHARGE FROM THE REACTOR BUILDING - EXTERNAL DOSAGE

If an incident occurred in which the reactor vessel but not the building was ruptured, and the temperature and pressure within the building rose appreciably, air containing fission products would begin to leak out slowly to the atmosphere. In this event the sprinkler system would become operative and act to reduce the temperature and pressure. Nevertheless, some air containing fission products would leak out. Therefore a computation of the external dose rate in the downwind direction from the reactor building is presented.

The following assumptions were made in carrying out the calculations:

1. The fission products were uniformly mixed in the building air.
2. Fission products produced in the power excursion are neglected, since it is assumed that the reactor has been operating for a relatively long period and that the long-lived fission products would have a dominant effect.
3. The beta and gamma source strengths are equal (this is only a rough approximation) and may be expressed as

$$Q = 2.3 \times 10^{14} P t^{-0.21} \text{ Mev/sec,}$$

where

Q = the source strength in Mev/sec

P = the previous steady power in kilowatts. In this case it is 20000 kw.

t = time in seconds

4. The building air leaks out at a constant rate of 100 cu ft/day.

Since the building air is contained in a volume of 400,000 cu ft, the beta or gamma source strength of the material emitted is:

$$\begin{aligned} & \frac{100 \text{ cu ft}}{\text{day}} \times \frac{1}{400,000 \text{ cu ft}} \times 2.3 \times 10^{14} \times 20,000 \text{ kw} \times t^{-0.21} \text{ Mev/sec.} \\ & \quad \frac{3600 \text{ sec}}{\text{hr}} \times \frac{24 \text{ hr}}{\text{day}} \\ & = 1.33 \times 10^{10} t^{-0.21} \text{ Mev/sec}^2 \\ & = 1.33 \times 10^{10} \text{ Mev/sec}^2 \times t^{-0.21} \times \frac{\text{disintegration}}{0.7 \text{ Mev}} \times \frac{1 \text{ curie-sec}}{3.7 \times 10^{10} \text{ disintegration}} \\ & = 0.50 t^{-0.21} \text{ curies/sec} \\ & = 1.81 t^{-0.21} \text{ kilocuries/hour} \end{aligned}$$

A. External Dose Rates from the Plume

In calculating the dose rates from the plume the work of Sutton was used. The following diffusion parameter values used in the equations were assumed in making the calculations: $n = 0.5$, $C_y = 0.1$, $C_z = 0.06$, $C = \sqrt{C_y C_z} = 0.077$. The wind speed, u , was taken as 2 meters/sec.

1. External Beta Dose Rates

The beta dose rate $\frac{dD_\beta}{dt}$ may be calculated from the expression

$$\frac{dD_\beta}{dt} = (0.5)(0.64) \times \frac{1 \text{ rad}}{8.06 \times 10^{10} \text{ Mev/m}^3} \times \frac{2 Q_\beta}{\pi u C_y C_z x^{2-n} \text{ rad/sec}}, \quad (13)$$

where Q_β = the beta source strength of the leak in Mev/sec² = $(2.3 \times 10^{14} \text{ P t}^{-0.21})$ (2.89×10^{-9}). The factor 2.89×10^{-9} represents the fraction per second of the total quantity of fission products leaking out of the building. Also x is the distance from the reactor building in meters. The other symbols have been defined previously. Substituting the assumed values for the diffusion parameters and other variables, it follows that:

$$\frac{dD_\beta}{dt} = \frac{1.40 \text{ t}^{-0.21}}{x^{1.5}}. \quad (14)$$

The total dose at a fixed distance from the reactor building over a time interval Δt may be obtained by integrating equation (14) thus:

$$D_\beta = \frac{1.40}{x^{1.5}} \int_t^{t+\Delta t} t^{-0.21} dt = \frac{1.40}{(.79) x^{1.5}} [(t + \Delta t)^{0.79} - (t)^{0.79}] \quad (15)$$

Table XVII shows the beta dose rates for distances up to about 8 miles from the building and for times after the incident ranging up to seven days.

2. External Gamma Dose Rates

The nomogram developed by J. Z. Holland which is based on Sutton's equation was used to calculate the gamma dose rates. Computations (see Table XVIII) were made for various distances from the reactor building up to about eight miles and for release times beginning immediately after the incident to seven days later. The leak from the building was considered as a ground source.

3. External Dose Rates during Fumigation Conditions

The values for beta and gamma external dose rates shown in Tables XVII and XVIII represent the upper limit of the dose rate which may be expected during fumigation conditions. Since the calculations in these tables

Table XVII

BETA DOSE RATE (millirads/hr) TO MAN STANDING ON GROUND
FROM 100-CU FT/DAY LEAK

Distance from Reactor, meters	Time After Incident						
	Immediate	1 hour	6 hours	1 day	2 days	4 days	7 days
300	684	344	240	178	154	133	118
500	286	160	110	83	72	62	55
800	128	78	55	41	35	31	27
1600	39	27	19	14	13	11	9.6
3200	12	9.0	6.8	5.1	4.4	3.8	3.4
6400	3.7	3.1	2.3	1.8	1.6	1.4	1.2
12800	1.1	1.0	0.81	0.63	0.55	0.48	0.42

Table XVIII

GAMMA DOSE RATE (mr/hr) TO POINT ON GROUND FROM
100-CU FT/DAY LEAK FOR 0.7-MEV GAMMA

Distance from Reactor, meters		Time After Incident						
		Immediate	1 hour	6 hours	1 day	2 days	4 days	7 days
300	35	18	12		9.1	8.0	6.9	6.1
500	21	12	8.4		6.3	5.5	4.6	4.2
800	14	8.3	5.7		4.3	3.8	3.2	2.9
1600	6.9	4.9	3.4		2.7	2.2	2.0	1.7
3200	3.5	2.7	2.0		1.5	1.3	1.2	0.99
6400	1.7	1.4	1.1		0.83	0.73	0.62	0.56
12800	0.81	0.73	0.59		0.46	0.41	0.35	0.31

are for stable conditions and are for a ground source, the additional turbulence present during fumigation conditions would tend to increase diffusion and thereby reduce the dose rate.

4. External Dose Rates due to Rainout

The expression used for calculating deposition on the ground due to rainout at the optimum scavenging rate may be written as:

$$w = \frac{Q_0}{e \pi^{\frac{1}{2}} C_y x^{\frac{n}{2}-\frac{n}{2}}} \quad (16)$$

The symbols have been defined previously. The dose rate for beta or gamma radiation at one meter above the ground after deposition due to rainout for T seconds at a given distance x from the reactor along the centerline of the plume may be expressed as:

$$\frac{dD_{\beta}(\text{rainout})}{dt} = \frac{Q_{\beta} E_{\beta} T}{e \pi^{\frac{1}{2}} C_y x^{\frac{n}{2}-\frac{n}{2}}} \quad (17)$$

for beta radiation and

$$\frac{dD_{\gamma}(\text{rainout})}{dt} = \frac{Q_{\gamma} E_{\gamma} T}{e \pi^{\frac{1}{2}} C_y X^{\frac{n}{2}-\frac{n}{2}}}$$

for gamma radiation. It has been assumed that $Q_{\gamma} = 2.3 \times 10^{14} \text{ P t}^{-0.21}$ (2.89×10^{-9}) = Q_{β} .

In the above expression t represents the time in seconds from the incident.

Tables XIX and XX show the beta and gamma dose rates calculated from formulae (17) and (18) after a 3-hour accumulation of deposited material on the ground, up to about 8 miles from the reactor and for emissions immediately, 1 day, and 7 days after the incident. These tables indicate, as in the case of rainout from the puff case, that serious dosages may be obtained by long exposure to the deposited material from rainout.

Table XIX

BETA DOSE RATE (rads/hr) AFTER THREE-HOUR
RAIN ACCUMULATION
(BUILDING LEAK RATE - 100 CU FT/DAY)

Distance from Reactor, meters	Time After Incident		
	Immediate	1 day	7 days
300	91	56	37
500	36	22	16
800	16	10	6.8
1600	4.6	3.0	2.0
3200	1.3	0.91	0.60
6400	0.40	0.27	0.18
12800	0.12	0.077	0.053

Table XX

GAMMA DOSE RATE (r/hr) AFTER THREE-HOUR
RAIN ACCUMULATION
(BUILDING LEAK RATE - 100 CU FT/DAY)

Distance from Reactor, meters	Time After Incident		
	Immediate	1 day	7 days
300	1.1	6.7×10^{-1}	4.5×10^{-1}
500	4.3×10^{-1}	2.7×10^{-1}	1.9×10^{-1}
800	1.9×10^{-1}	1.2×10^{-1}	8.2×10^{-2}
1600	5.6×10^{-2}	3.6×10^{-2}	2.4×10^{-2}
3200	1.6×10^{-2}	1.1×10^{-2}	7.2×10^{-2}
6400	4.8×10^{-3}	3.2×10^{-3}	2.2×10^{-3}
12800	1.4×10^{-3}	9.3×10^{-4}	6.4×10^{-4}

IV. INHALATION CASES

Some examples of calculated internal radiation dosage are given in this section for certain assumed conditions. The two cases treated are (1) inhalation during passage of a radioactive cloud brought down to ground level by break up of a temperature inversion, and (2) inhalation for one hour in a streamer of radioactivity originating as a steady leak from a gas-tight building.

The first set of bases for the examples pertains to the operation of the reactor in order to be explicit about the reservoir of radioactivity. It is assumed that the reactor has been operated at a 20,000-kw heat rate for 70 per cent of the time until one per cent of the uranium has fissioned; this means a time of 4000 days. The choice of such a high burnup is for the purpose of investigating the possible hazard of long-lived fission products and plutonium isotopes compared with relatively short-lived fission products. The reactor is assumed to have the equivalent of 5.5 tons of uranium with an average enrichment of 1.2 per cent. The initial conversion ratio is taken as 0.74. The approximate amounts of plutonium isotopes and the activities due to strontium and iodine-131 are listed in Table XXI.⁷

Table XXI

SELECTED RADIOACTIVITIES IN EBWR AT 1% BURNUP

	Amount	Half Life	Emission
Pu ²³⁹	22,000 grams	24,400 years	α 5.16 Mev
Pu ²⁴⁰	6,400 grams	6,600 years	α 5.10 Mev
Pu ²⁴¹	940 grams	14 years	$\left\{ \begin{array}{l} \beta \ 99 \pm 0.02 \text{ Mev} \\ \alpha \sim 10^{-3} \% \end{array} \right.$
Am ²⁴¹	160 grams	470 years	α 5.45 Mev
Sr ⁸⁹	7.5×10^5 curies	53 days	β 0.55 Mev
Sr ⁹⁰	170 grams	25 years	1.00 Mev
I ¹³¹	4.5×10^5 curies	8 days	β 0.22 Mev

The bases for estimating the dosage to critical organs after breathing radioactive air are given in Table XXII. (An RBE of 20 is used for alpha emitters for both lung and bone.)

⁷J. C. Carter and J. M. West, "Reactivity as a Function of Irradiation Time in Thermal Reactors," ANL-5186 (December 1953).

Table XXII
CONSTANTS FOR CALCULATING INHALATION DOSAGES⁸

Element	Critical Organ	Organ Mass, (gm)	Effective Half Life (days)	Fraction Reaching Critical Organ
Americium and Plutonium Pu^{239} and Pu^{240} Pu^{241} Am^{241} Sr^{89} $\text{Sr}^{90} + \text{Y}^{90}$ I^{131}	Lung	1000	120*	0.12
	Bone	7000	4.3×10^4	10^{-5}^{**}
	Bone	7000	4.6×10^3	10^{-5}^{**}
	Bone	7000	885	10^{-5}^{**}
	Bone	7000	52	0.22
	Bone	7000	2.7×10^3	0.22
	Thyroid	20	7.7	0.15

*The effective half-life is taken equal to the measured biological half-life for transport of radium sulfate from the lung.⁹

**The basic assumption is that plutonium oxide is made slightly soluble by gastric juices after reaching the gastrointestinal system by ciliary action. Thus the fraction retained by the lung, 0.12, is gradually delivered to the GI tract, where 1.4×10^{-4} reaches the blood stream and 0.7 of this amount reaches the bone, giving an overall fraction based on rate of inhalation of 10^{-5} . The two values, 1.4×10^{-4} and 0.7, are taken from Reference 8 for soluble Pu.

A. Fumigation Case - Inhalation Examples

The meteorological parameters and coefficients are the same as given previously under "External Dosage Due to Fumigation," Sect. III. A-3. As before, the dilution at a distance, D, from the reactor has been corrected for virtual source term and drift during rise of cloud. The concentration at D per unit source strength is TID/Q, given by

$$\frac{\text{TID}}{Q} = \frac{1}{C_{uh} X^2} \frac{\text{sec}}{M^3} \cdot$$

⁸"Maximum Permissible Amounts of Radioisotopes in the Human Body and Maximum Permissible Concentrations in Air and Water," NBS Handbook 52, (March 20, 1953).

⁹L. D. Marinelli, et al, "Transport of Radium Sulfate from the Lung and Its Elimination from the Human Body following Single Accidental Exposures," Radiology 61 (No. 6), 903 (1953).

The internal dose to a particular organ over a period of 30 years as a result of the observer being in the path of the cloud during its passage is as follows:

$$\text{Dose in rems} = \left(\frac{\text{TID}}{Q} \right) (S) \left(\frac{1}{3600} \right) \left(\frac{f}{100} \right) \frac{\text{RBE}}{M} \int_0^t \frac{dt}{e^{T_m t}}$$

where

T_m = effective mean life

t = 30 years

f = fraction retained in critical organ

M = mass of organ, grams

S = source strength (ergs/sec).

Further factors considered were

1. The rem is based on the rad, 100 ergs/gm.
2. The breathing rate is 1 cubic meter/hr.
3. The source includes 50% of the reactor contents given in Table XXI.
4. The velocity of radioactive particle deposition is essentially zero.

Computed values are given in Table XXIII.

Table XXIII

INHALATION DOSAGES CALCULATED FOR FUMIGATION CASE

Isotope	Mixing Height (meters)	Distance from Source (meters)			
		300	800	3200	12800
Pu ²³⁹ +Pu ²⁴⁰ Am ²⁴¹	100	4,000	2,700	1,100	400
	500	600	400	200	80
	100	800	500	240	80
	500	160	80	40	16
Sr ⁸⁹ Sr ⁹⁰ - Y ⁹⁰	100	700	470	200	70
	500	100	70	40	10
	100	7,200	4,800	2,000	720
	500	1,100	720	390	160
I ¹³¹	100	6,000	4,100	1,700	600
	500	900	600	300	120

The bone dosage for plutonium isotopes is calculated to be 3 rem over 30 years for the values of mixing height equal 100 meters and distance equal 300 meters.

B. Steady-State Leakage of Radioactivity from EBWR Building - Inhalation Examples.

1. Average Air Escape Rate from the Building.

It is specified that the leakage rate at 15 psi differential be no more than 1,000 cu ft/day. Since a building spray system is available to reduce the differential readily to that due to barometric fluctuations, the leakage is based on an average differential of 1% of the 15 psi. If the leakage points produced streamline flow, the leakage rate would be proportionately smaller. For turbulent leakage the flow would change approximately as the square root of the pressure, so that a leakage rate of 100 cu ft/day would result. This latter value is the rate chosen.

2. Dilution Factors

The dilution factors for various distances along the wind axis are taken from page 21 of ANL-WHZ-299 as furnished by H. R. Newton and H. Moses. The stable case is chosen.

3. Dosages Calculated

In addition to the pertinent values and comments of Tables XXI and XXII, the following were considered:

- a. Values of dosages are for isotopes from 1 kg of uranium mixed in 400,000 cu ft of building air. No settling or scrubbing action taken into account.
- b. Observer resides at the center of the streamer for one hour.

Computed values are given in Table XXIV.

The blood dosage by inhalation, assuming that 25% of the beta energy of mixed fission products is absorbed by the blood with an elimination half-life of 24 hours, is 0.34 rad at 25 meters from the source beginning one hour after the end of long reactor operation. The actual fraction of fission products contributing to blood dosage is not known.

Only a few of the possible critical organs have been considered. The chemical nature of the fission products and particle size distribution are two factors which make analysis uncertain even if the manner of uptake and disposal were known.

Table XXIV
INHALATION DOSAGES CALCULATED FOR
STEADY-STATE LEAKAGE CASE

Isotope	Distance (meters)			X/Q (Units/cc for source of 1 unit/sec)
	25	200	1,600	
	16,000			
	4.1×10^{-7}	1.8×10^{-8}	7.8×10^{-10}	2.5×10^{-11}
Bone Dosages Calculated (rems or rads)				
Sr^{89} $\text{Sr}^{90} - \text{Y}^{90}$ Pu^{239} Pu^{240} $\text{Pu}^{241} - \text{Am}^{241}$	1.1×10^{-2}	5×10^{-4}	2.1×10^{-5}	7×10^{-7}
	0.11	5×10^{-3}	2.1×10^{-4}	7×10^{-6}
	2.2×10^{-5}	9.5×10^{-7}		
	2.2×10^{-5}	9.5×10^{-7}		
	6.8×10^{-6}	3×10^{-7}		
Lung Dosages Calculated (rems)				
Pu^{239} and Pu^{240} Am^{241}	6.2×10^{-2}	2.7×10^{-3}	1.2×10^{-4}	3.8×10^{-6}
	2.5×10^{-2}	1.1×10^{-3}	4.8×10^{-5}	1.5×10^{-6}
Thyroid Dosages Calculated (rads)				
I^{131}	9.3×10^{-2}	4.1×10^{-3}	1.8×10^{-4}	5.7×10^{-6}

4. Discussion Regarding Dosage Due to Steady State Leakage.

There are several intervening phenomena and circumstances which markedly lower the amounts of isotopes which actually reach critical organs as would be obtained by multiplying the values in Table XXIV by the number of kg in the reactor at 1% burnup, viz., 5,500. One can postulate efficiencies (1) for the escape of the isotopes from the reactor into the air of the reactor room, (2) for the ability of the spray system to wash the air, (3) for the confinement of the air of the building to permit settling of the particles, and (4) for the ability of the olfactory system of the receiver to limit particle uptake.

Based on the premise of limited solubility of Pu isotopes, the bone dosage due to $\text{Sr}^{90} - \text{Y}^{90}$ shows the largest value, as seen from Table XXIV. The judgment as to the relative effects of the dosages shown is still a problem, since no account is taken of the variation in location within the bone between Sr and Pu. It is also seen that higher isotope

dosage is not inconsequential compared to values for Pu^{239} and Pu^{240} . One conclusion is that reactor operation with recycled plutonium will result in accumulation of relatively short life alpha emitters which might give the controlling dosages. The conclusion for purposes of this report is that the degree of building gas tightness specified reduces the calculated inhalation dosage from the magnitude associated with lethal values to the magnitude of trivial insults. There should be no inference that the reactor will be operated to 1% burnup of fuel.

From additional information available since NBS 52 was published it appears that: a) the RBE for α emitters is probably closer to 5 than 20; b) the long-term retention of Sr is probably smaller than indicated above; and c) the uptake of Pu from the intestinal tract is probably higher.

APPENDIX E

INHERENT SAFETY OF THE REACTOR

It is characteristic of most nuclear reactors that, in the range of high power operation, reactivity is a function of power output. This dependency arises from the changes in temperature and density of the reactor materials which accompany power changes. In the case of the boiling reactor it is feasible so to design the reactor that these effects automatically limit the power to a safe value even if rather large reactivity increases are introduced by extraneous means. The peculiar adaptability of the boiling reactor concept to this design principle stems from two basic characteristics: first, the reactor, in the power operating range, has an unusually high and relatively fast acting negative power coefficient of reactivity; second, the normal operating power density, as limited by steam content of the reactor, is far below that which can produce failure of the reactor.

The EBWR has been designed to take advantage of the inherent safety characteristic of the boiling reactor principle. While it is probably impossible to design a practical power reactor which is safe against arbitrarily large increases in reactivity, the EBWR does achieve a degree of inherent safety far greater than that of conventional power reactor types. The remainder of this section will be devoted to a quantitative description of this characteristic, and to estimates of its limitations. The estimates will be based to a large extent on results of the Borax experiments,^{1,2} some of which have not been previously published.

A. General Types of Accident

Ordinarily, two general types of reactor accident are considered as sources of externally produced reactivity increases - those in which relatively large reactivity increases are introduced rapidly enough to cause the reactor power to rise on such a short period that the temperature distributions within the reactor differ widely from their equilibrium values, and those in which reactivity is introduced more slowly and the temperature distributions in the reactor approximate the equilibrium values. The "start-up accident" is typical of the former type, while the latter type might be caused by a faulty control system, or by error of the operator. It will be shown in a later section that the EBWR cannot reach a dangerous power level in steady (equilibrium) operation. Hence, it is necessary, at the start, to discuss the behavior of the reactor in the case of a power variation sufficiently rapid to cause a transient temperature distribution in the fuel plates.

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¹ANL-5211, "Transient and Steady-State Characteristics of a Boiling Reactor: The Borax Experiments, 1953" (Feb. 1954)

²ANL-5323, "Experimental Investigation of the Self-Limitation of Power during Reactivity Transients in a Subcooled Water-Moderated Reactor: Borax-I Experiments, 1954"

B. Temperature Distributions in the Reactor

In normal operation the water temperature within the reactor is very nearly uniform. The surface of each fuel plate is hotter, by a few degrees, than the adjacent water, but this temperature difference which is associated with the transfer of heat from the plates to the water is not an important magnitude so long as boiling is underway. Within each fuel plate there is a temperature distribution which depends upon the rate of heat generation in the plate. For the steady case (rate of heat transfer to water equal to rate of heat production in plate) the temperature distribution within the meat of the plates is parabolic and proportional to the rate of heat production. Thus the temperature difference between plate surface and any internal plane parallel to the surface, and at distance x from the surface, can be expressed by a relation

$$\Delta T = q f(x) ,$$

where q is the rate of generation of heat per unit surface area, or its equivalent, the heat flux through the plate surface into water. Curve (a) of Figure 35 shows this distribution function, $f(x)$, for the natural uranium plates of the EBWR. Since the central value of $\Delta T/q$ is 5.3×10^{-4} , the plates may be characterized by an effective conductance = $\frac{1}{5.3 \times 10^{-4}} = 1890$ Btu/ft² hr F which relates the central temperature linearly to the heat flux. Thus, if a plate is operating at a flux of 10^5 Btu/hr ft² (the maximum value at the design power of EBWR) the center of the plate will be at a temperature 53F above that of the surface. Quite evidently the EBWR could operate at many times its rated power without experiencing large temperature differentials in the fuel plates.

If reactor power is varying so rapidly that the rate of heat loss to the water is not equal to the rate of heat generation, the temperature distribution can be drastically modified. A significant case is that in which the reactor power is increasing exponentially. Figure 36 shows temperature distributions for exponential power increases of period 0.02, 0.04, and 0.08 sec, calculated under the following conditions: (a) entire plate initially at constant temperature equal to the water temperature; (b) negligible temperature drop between plate surface and water; (c) exponential power increase has been underway for several periods. For this type of power transient the temperature (measured from the initial value as zero) at each point in the plate rises exponentially with a period equal to that of the power increase. It is evident from the curves that for periods in this range there is no important heat loss from the center of the plate. That is to say, the cooling effect of heat transfer to the water is negligible at the center of the plate, even when the heat transfer coefficient at the plate surface is effectively infinite. Consequently, if the maximum plate temperature is accepted as the criterion of dangerous reactor failure, this temperature can be computed for hypothetical short-period accidents simply as the ratio of the total energy production per unit volume in the fueled part of the plate to the

volume heat capacity of the plate materials. The heat transfer to the water is then important only insofar as it produces steam to reduce reactivity and terminate the power excursion. In the latter connection it is instructive to examine the ratio of plate temperature to heat flux into the water. Such a curve has been plotted for the 0.04-sec exponential period as curve (b) of Figure 35. Note that the central value of $\Delta T/q$ (4.65×10^{-4} degrees F/Btu ft² hr) is slightly below that for the steady-state curve. Alternatively, it may be said that the effective thermal conductance which relates surface heat flux to central temperature is slightly higher for the 0.04-sec exponential transient than for the steady state. Obviously, the value of this effective conductance will vary with period of the exponential. It has been computed for a number of periods and is plotted in Figure 37. Note that, although the details of the temperature distribution vary markedly (Figures 35,36), the value of the effective conductance remains within 15% of the steady state value (1890 Btu/ft² hr F) for all periods down to about 0.015 sec.

The foregoing considerations lead to a number of conclusions which will simplify greatly the further consideration of reactor transients. They are enumerated below.

1. For periods of 40 msec or shorter the error is small if, in computing reactivity effects which depend on fuel plate temperature only (thermal expansion, Doppler broadening), the effective plate temperature is assumed equal to the center temperature.

2. Those fuel plates will melt first for which the ratio $\frac{\text{Volume heat capacity} \times \text{melting temperature}}{\text{Power density}}$ is highest. The values of these quantities for the natural uranium and enriched uranium plates are summarized in the following table:

ρC	Volume heat capacity (average, room temp. to melting), Btu/ft ³ °F	P Relative Power Density	T_m Melting Point, °F	$\frac{P}{\rho C}$	$\frac{P}{\rho C T_m}$
Enriched plates	36.5	1.13	3350	0.031	9.2×10^{-6}
Natural plates	45.6	1.00	2067	0.022	10.6×10^{-6}

Since $P/\rho C$ is higher for the enriched plates, these plates will, in a power excursion, become hotter than the natural uranium plates. The natural plates will melt first, however, because of their lower melting

point. Consequently, it is conservative in estimating the shutdown of the reactor by steam formation to assume that all plates have the same thermal characteristics as the natural uranium plates.

3. Anticipating the results of the Borax experiments (discussed later) that: (a) shutdown of the reactor by steam is accomplished in a few reactor periods after the temperature rise of the plates becomes high enough to form steam in appreciable quantity, and (b) the total quantity of heat transferred to the water to effect shutdown is small compared to that stored in the fuel plates, the conclusion is drawn that, whatever the temperature distribution within the plates during the power transient, the effective conductance relating the surface heat flux to the central plate temperature will not differ greatly from those values computed above (Figure 37). A constant value of 2000 Btu/ft² hr °F will be assumed.

C. Summary of Borax Results

An estimate of the degree to which the power of the EBWR is self-limited by the formation of steam will be derived from experimental measurements on the Borax reactors. Since the thermal and nuclear characteristics of the EBWR are different from those of either of the Borax reactors, it will be necessary to modify the Borax results in order to make them applicable. In the following section the Borax results will be summarized, and their application to the EBWR will be discussed in a later section. The results from Borax-I have been published in ANL-5211¹ and ANL-5323.² Those which are useful for the EBWR analysis will be repeated here. In addition, results from Borax-II which have not been previously published will also be presented. So far as its transient behavior is concerned, the significant differences in Borax-II are that its fuel plates, which are identical except for uranium loading to the Borax-I plates, were separated by water channels 1/4 in. thick rather than 1/8 in. thick as in Borax-I; the uranium density in Borax-II is lower in the ratio 0.66 and the critical volume of the reactor core is nearly twice that of Borax-I. These characteristics are summarized in the following table.

	<u>Borax-I</u>	<u>Borax-II</u>
Number of fuel elements in core (volume per fuel element identical for the two reactors)	28	46
Number of fuel plates per element	18	10
Width of water channel between plates (in.)	0.125	0.25
U ²³⁵ loading per fuel plate (grams)	7.7	9.3*
U ²³⁵ loading per element (grams)	140	93*

* a few of the fuel elements (about 25%) were of higher loading (157 grams per element) to adjust the steam coefficient of reactivity to the desired value. These elements were at the periphery of the core.

Because of these differences in fuel element composition Borax-II has roughly twice as much water associated with each fuel plate as did Borax-I. That is to say, for the same volume rate of steam production, the average Borax-II plate produced steam at roughly twice the rate of the average Borax-I plate. Further, because of the greater fuel dilution and larger size of Borax-II, the steam coefficient of reactivity was considerably smaller. The calculated value for the coefficient was about half that for Borax-I ($-0.10\% k_{eff}$ for Borax-II vs. $-0.27\% k_{eff}$ for Borax-I per percent of water replaced by steam). Experiments indicated that the difference may have been considerably larger than calculated, as it was necessary to replace some of the standard Borax-II fuel elements by more heavily loaded elements in order to avoid a steam coefficient which was positive for small values of the steam content. The combined result of these two differences was that the average fuel plate of Borax-II had to produce at least 5.4 times as much steam as the average Borax-I plate to effect a given decrease in reactivity.

Experiments were made in the Borax reactors by making sudden additions of excess reactivity to the reactor which had previously been operating at a very low power level, of the order one watt. The power of the reactor was then allowed to increase without external interference until the reactor shut itself down by the formation of steam.

The general characteristics of Borax behavior during such a power excursion are illustrated by Figure 38 which is a record of reactor power and fuel plate temperature as a function of time during the excursion. The figure also includes a record of the temperature of one of the fuel plates which had been thermally insulated from coolant water. The temperature of this fuel plate, except for the effect of a small residual heat loss, is proportional to the total energy produced in the plate. The experiment was made with the reactor water initially at a temperature of 82F and the reactor at atmospheric pressure. Initially an excess reactivity of about 1.1% was applied to the reactor by the sudden ejection of a control rod. Ejection of the rod had been completed and the reactor power had reached a stable exponential period of 0.021 sec before the time arbitrarily selected as zero in Figure 38. Reactor power and temperature rise of the fuel plate continued to increase exponentially until the fuel plate temperature had exceeded the saturation temperature by about 50F. It is quite evident that at this point rapid steam formation began, and the temperature of the normal (bare) fuel plate broke sharply away from the temperature curve of the insulated fuel plate. At about this same time the curve of reactor power departed from exponential, and some 10 msec later the power reached its maximum value. After the maximum, the power dropped quite rapidly as the formation of more steam reduced reactivity below the critical point. At the point of minimum power, considerable water had been forced out of the reactor by the steam. In general, since the fuel plates were at this time relatively cool, the water would flow back into the core in a time of the order of one second. The result of this return of water (not included in Figure 40) was to

return reactivity to a critical or supercritical value, and the reactor would, in some cases, go through a succession of further power excursions qualitatively similar to the initial one, but of energy lower by a factor of 3 to 10. Under other conditions the power, after going through its initial minimum value, would increase rather gradually to a steady value, in the range 1 to 10 mw, and continue at this value until the reactivity was reduced by external means. In general, the behavior in which a succession of power excursions occurred resulted when the initial excess reactivity application was large, and the steady boiling operation resulted when small initial excess reactivities were used. The steady boiling operation, however, was favored by the subcooled condition of the reactor water, and this type of behavior persisted in Borax for initial reactivity injections as large as 1.3%, whereas, if the reactor water was initially saturated, excess reactivity of as much as about 1.1% would cause the successive power excursions. This latter type of behavior has generally been referred to as chugging. This terminology will be used in the remainder of this report.

Of the quantities measured in these experiments, the total nuclear energy liberated in the reactor during the initial power excursion is considered to be most useful for predicting the performance of the EBWR. Since the area available for heat transfer in the EBWR is much different from that which existed in the Borax reactors, the energies will be expressed as total energy liberated per unit surface area of fuel plate at the position of maximum thermal neutron flux in the reactor.

Figure 39 shows the total energy production for the Borax-I and II reactors for the case in which the initial temperature of the reactor water was atmospheric saturation temperature. The measurements were made for various values of applied excess reactivity. In the figure the energy values are plotted as a function of the reciprocal of the asymptotic reactor period produced by these reactivities.

If the reactor water is not at saturation temperature but at some lower temperature when the excess reactivity is applied, a considerably greater quantity of energy is liberated in the resulting power excursion. This increase is due partly to the simple circumstance that enough energy must be provided to heat the fuel plates to the saturation temperature before the shutdown mechanism can come into play, and partly to the inherent characteristics of the process of reactor shutdown by steam formation in cold water. In Figure 40 the energy liberation has been plotted as a function of reciprocal reactor period for such a subcooled condition. In this case, however, the energy required to heat the fuel plate from its initial temperature to saturation temperature has been subtracted, and the net energy used in the steam formation process has been plotted. The lower section of the Borax-II curve has been drawn as a dashed line. In this region there was no definite minimum value for the reactor power after the initial peak, but the reactor went into a condition of steady boiling

operation at a safe but appreciable power level. It is therefore not really possible to define the energy liberated during the power excursion. It can be said, however, that it lies somewhere below the dashed curve of Figure 40.

In both Figures 39 and 40 the energy values for the Borax-II reactor are considerably higher than those for Borax-I, the ratio at a given period being in the neighborhood of 2.5. This behavior can only be attributed to the lower steam coefficient of reactivity and the wider plate spacing of Borax II. There have been no experiments which separate the effects of the two variables, plate spacing and steam coefficient of reactivity. It would seem quite reasonable to assume that the two variables entered in much the same way, and that the energy transferred to the water during a power excursion by each unit area of fuel plate surface depends simply upon the quantity of steam per unit fuel plate surface necessary to compensate the initial excess reactivity. For purposes of extrapolating the results to the EBWR a much more pessimistic assumption has been made. It has been assumed that the energy depends entirely on plate spacing and that the effect of change in steam coefficient of reactivity is negligible. On the basis of this assumption the two sets of Borax measurements have been extrapolated linearly from the Borax-II plate spacing (0.25 in.) to that of EBWR (0.4 in.). The resultant curves are shown in Figures 39 and 40. These curves are interpreted to indicate the energy transferred to the water during power excursions in a Borax-type reactor with plate spacing 0.4 in. Further analysis will be made (next section) to apply the curves to plates with the thermal characteristics of the EBWR plates.

The self-limiting performance of the Borax-II reactor was investigated at elevated pressure as well as at atmospheric pressure. Since the reactor was pressurized by steam which evaporated from the reactor water, power excursions in the Borax reactor (and in the EBWR also) can occur at elevated pressures only when the temperature of the reactor water is at the saturation value for that pressure. Although it might be expected that, since the volume heat of formation of steam increases almost linearly with pressure, the energy required to terminate a power excursion of given period would increase rapidly with reactor pressure, the experiments showed that other factors are actually the controlling ones, and that excursions at elevated pressures are considerably less energetic.* The experimental results are shown in Figure 41. The results at 150 psig and at 300 psig are practically identical. The energies fall below those at atmospheric pressure by more than a factor of 2. During the experiments represented by Figure 41, the maximum increase in steam pressure in the reactor tank, as a result of a power excursion, was about 5 psi.

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* Theoretical analysis of the Borax results has not yet reached the point where a theoretical discussion of the pressure effect is justified. It is probable, however, that the increase of $(dP/dT)_{\text{sat}}$ with saturation pressure is one of the important factors.

Figure 42 shows, as a function of reciprocal period, the maximum temperature reached by the surface of the hottest fuel plate during the power excursion. These curves cannot be extrapolated to the EBWR, but they are reproduced here simply as further proof of the fact that the power excursions under pressurized conditions are considerably milder than those at atmospheric pressure. The temperature measurements at 300 psig are not exactly comparable with those at atmospheric pressure and 150 psig because the thermocouples used for the lower pressure measurements failed at 300 psig, and a different type of thermocouple on a different fuel plate was used. This latter fuel plate was placed in a position symmetrical to that of the original plate in the reactor, but there may have been some asymmetry in the neutron flux.

A few measurements made at pressures between atmospheric and 150 psig indicated that the decrease in energy between these two pressures was gradual. Since there was no appreciable change between 150 and 300 psig it might be argued that the energies might become higher again for pressures above 300 psig. It hardly seems reasonable, however, that at the EBWR operating pressure of 600 psig the energy curve would have increased enough to lie above that for atmospheric pressure. The self-protection performance of the EBWR at elevated pressure can therefore hardly be worse than such performance under saturated conditions at atmospheric pressure, and probably is much better.

D. Limitation of Transients by Steam in the EBWR

In the preceding section curves were derived from the Borax results which give, as a function of reactor period, the upper limit of the energy which must be transferred to the water to terminate power excursions in a reactor having the nuclear characteristics and the plate spacing of the EBWR. It remains to determine the maximum plate temperatures which would be associated with this energy transfer in the EBWR. Power excursions which occur with the reactor water at saturation temperature will be considered first.

It has been shown in Section B that the heat flux at the fuel plate surface is related to the maximum fuel plate temperature by an effective thermal conductance which does not vary greatly with the period of the transient so long as the period is not shorter than 0.015 sec. The value of the effective conductance (h) is about $2000 \text{ Btu}/(\text{ft}^2)(\text{hr})(\text{F})$. Consequently, since the Borax experiments indicate that in the EBWR the temperature drop between the fuel plate surface and the water will be small compared to the temperature drop within the fuel plate, the energy transferred per unit fuel plate area to the water, up to any time t after the start of the excursion, will be just

$$E(t) = h \int_0^t \Delta T_m(\theta) d\theta, \quad (1)$$

where ΔT_m is the (spatial) maximum temperature difference in the fuel plate (i.e., difference between center and surface temperature) and θ is simply an integration variable. Further, since the loss of heat from the center of the fuel plate is small during a power excursion, we have, very nearly

$$\Delta T_m(t) = \frac{1}{\rho C} \int_0^t p(\Psi) d\Psi, \quad (2)$$

where p is the power density in the meat of the fuel plate, ρC is the volume capacity of the meat, and Ψ is another integration variable.

If the variation in p with time during the shutdown in the EBWR were known, or if a relation between $p(t)$ and $E(t)$ valid for the EBWR were known, T_m could be computed from this relation and equations (1) and (2). Since the relation is not known, an argument is resorted to which at least sets an upper limit to the plate temperature. Figures 39 and 40 give curves of the energy required for steam shutdown if the fuel plates are like those of the Borax reactors, i.e., if they are highly conducting and have relatively low heat capacity. Once boiling has begun with such plates, the heat transferred to the water, up to a given time t , is nearly equal to the total energy produced, up to that time, in the fuel plate whereas with the EBWR plates the energy transferred to the water is proportional to the time integral of the energy formed in the plates. It must be true that the faster heat is transferred to the water, i.e., the faster steam is formed, the sooner the reactor becomes subcritical and the less the total release of energy in the fuel plates. Hence, it is argued that the energy transfer to the water necessary for shutdown of the EBWR will be no greater than that shown in Figures 39 and 40, provided the rate of transfer is never less than the rate of energy production (power) in the Borax reactors. It now becomes necessary to examine the time dependence of power in the Borax reactors.

It has been pointed out previously (ANL-5211¹) that if the Borax power records are plotted non-dimensionally (time measured in periods; power measured as a fraction of the peak value) their shape is roughly constant over a wide range of periods. Figure 43 is such a curve for one of the Borax-II power excursions. Curve (a) is reactor power, curve (b) is the integral of the power (i.e., total energy production), and curve (c) is the integral of curve (b). Curve (d) is an exponential, normalized to coincide with curves (a), (b), and (c) during their early exponential portions. If curve (a) is assumed, for the moment, to describe the power density variation in the EBWR fuel plates, then curve (b), if multiplied by $\tau/\rho C$, could give the fuel plate center temperature, and if multiplied by the factor $h\tau/\rho C$ would give the heat flux into the water as a function of time. Curve (c), if multiplied by $h\tau^2/\rho C$, would give the total energy transferred into the water per unit fuel plate area. It is then argued that, if the EBWR power is assumed to vary like that of a Borax-type reactor (curve a), but to be at all times greater in the ratio $\rho C/h\tau$, then the heat flux to the water will be at all times as great as or greater than that for the Borax case, and the

resulting maximum fuel plate temperature will be an upper limit to that which may be actually expected in an EBWR plate. The value which results from this procedure for the ratio of maximum EBWR fuel plate temperature rise to total energy per unit fuel plate area for a Borax-type reactor (as obtained from the upper curve of Figure 39) is just:

$$\frac{\Delta T_{\max}}{E_{\text{total}}} = \frac{1}{h\tau} \times \frac{\text{ordinate of curve b (Fig. 44)}}{\text{ordinate of curve c (Fig. 44)}},$$

where the ordinates of curves b and c are read at the time when the shutdown process is effectively complete. Unfortunately, this time cannot be identified very precisely - it certainly lies somewhere between $t/\tau = 6$ and $t/\tau = 7$, but between these two points the ratio $\frac{\text{ordinate b}}{\text{ordinate c}}$ varies from 0.95 to 0.65. Certainly the upper limit for $\Delta T_{\max}$ is obtained if a value of 1.0 is used for the ratio. The use of this value is equivalent to the assumption that reactor power rises exponentially until an energy equal to the total energy of a Borax-type excursion is transferred to the water, then falls instantaneously to zero. In Figure 44 the maximum fuel plate temperature rise resulting from power excursions of period τ has been computed on this assumption and has been plotted as a function of $1/\tau$. To recapitulate, the maximum temperature rise was taken as

$$\Delta T_{\max} = \frac{EW}{h\tau}$$

where EW is given by the top curve of Figure 39.

At elevated pressures, where the energy required to produce shutdown is experimentally lower, the maximum fuel plate temperature rises will be proportionately less. For example, at 300 psig (and at 150 psig) the excursion energy is lower by about a factor 2.7. The ΔT values would then lie below those of Figure 44 by the factor 2.7 and a power excursion of period as short as about 20 msec could be tolerated without reaching the fuel plate melting temperature.

In the case of power excursions which occur when the temperature of the reactor water is below the saturation point (subcooled condition), the situation varies from that discussed above in that the surface temperature of the fuel plate must be raised to the saturation point, by heat generation in the plate, before steam formation can begin. If the effect of the metal temperature coefficient of reactivity is neglected, the power increase during this initial heating period will be exponential, and if the period is short the heat loss from the plate surface may be expected to be by solid conduction in the water only. The computed temperature distributions in the plate for two different exponential periods (0.02 and 0.04 sec) are shown in Figure 45. The ratio $\Delta T(\text{center})/\Delta T(\text{surface})$ is about 3 for the period 0.04 sec and will, of course, be smaller for all longer periods.

For the moment it will be assumed that this necessity for heating the plate surface to saturation temperature has negligible effect on the total energy of the power excursion. The curve in Figure 46 shows the resulting maximum plate temperatures, computed as for the case of saturated water, but using the extrapolated Borax data for subcooled excursions (top curve, Figure 40). The minimum period which can be tolerated without raising the maximum plate temperature to the melting point is about 0.07 sec. For this period the ratio $\Delta T(\text{center}) / \Delta T(\text{surface})$ before boiling begins is only about 2. If the excursion started with the water temperature 150 degrees below saturation, the center temperature would rise only about 300F by the time the surface temperature reached saturation. Since this temperature rise is small compared to that required to melt the plates, it is concluded that the effect of the initially low temperature of the plates themselves is indeed negligible in the practical EBWR case, and the important quantity in determining the severity of the power excursion is the water temperature. Figure 46 gives, then, the predicted values of the upper limit of plate temperature in subcooled power excursions.

Up to this point the extra contribution of the enriched fuel plates to shutdown by steam has been neglected in order to simplify the discussion. Since these plates heat faster than the natural uranium plates and have thinner clad they are considerably more effective in transferring heat to the water. Computation of temperature distributions in these plates during exponential power excursions show that the effective conductance is about $3000 \text{ Btu/ft}^2 \text{ hr } ^\circ\text{F}$ over the range of periods of interest. The corresponding figure for the natural uranium plates was $2000 \text{ Btu/ft}^2 \text{ hr } ^\circ\text{F}$. Further, because of the lower heat capacity, the temperature of the enriched plates rises 1.25 times as fast as that of the natural plates. The net result of these effects is that a typical enriched plate will supply heat to the water at a rate that is higher than that of a natural plate by the factor

$$\frac{3000 \times 1.25}{2000} = 1.88. \text{ Since one-third of the fuel plates are enriched, the}$$

extra fraction of heat supplied to the water by the enriched plates above that computed on the assumption that all the plates are natural uranium is $1/3 (1.88 - 1) = 0.29$. Thus, at any given time in the transient 29% more heat has been delivered to the water than has heretofore been calculated. It is evidently consistent with the picture of the steam shutdown process which has been used up to now to add this heat to that previously calculated. The effect is to lower the calculated fuel plate temperature rise by 29%. The resulting values are plotted in Figures 44 and 46.

E. Behavior after the Initial Power Excursion

The supercritical reactor in the closed reactor tank cannot, of course, be made permanently subcritical by either the steam coefficient or the metal coefficient of reactivity. In the EBWR safety circuits are provided to shut the reactor down by control rod insertion, and, indeed, the action of these circuits will preclude that of the inherent shutdown processes in all but the very short period accidents. In these latter cases the inherent processes produce a temporary shutdown which lasts long enough for even the

most rugged and insensitive shutdown circuits to become effective. Some attention should be given, however, to the reactor behavior in the improbable event that all these shutdown circuits fail to operate.

Experience with the Borax reactors has shown that, after the water is ejected from the reactor core to terminate the initial power excursion, it flows back in about one second, and the reactor either settles down to steady boiling operation or goes through a series of further power excursions, of frequency about one per sec, and of amplitude considerably less than that of the initial one. It is to be expected that if the EBWR does undergo a series of excursions, their frequency of occurrence will be considerably less than those in the Borax case because of both the greater length and the longer cooling time of the EBWR fuel plates. The thermal time constant of the EBWR plates (time for temperature to drop by the factor e when there is no heat generation) is about 0.5 sec even if the plates are assumed to be in contact with an infinite heat sink. It would, therefore, be expected that the time between consecutive power peaks would be lengthened to at least 1.5 or 2 sec. In any case, the subsequent power excursions cannot be more energetic than the first, and the average reactor power over a series of such excursions cannot be many times greater than the normal operating power. Plate burnout should, therefore, not occur unless in the unlikely event that a few of the hottest plates remain steam bound, while sufficient water flows into cooler parts of the reactor core to restore criticality and allow the power to rise to a high level.

F. The Metal Temperature Coefficient

Heating of the fuel plates alone causes a decrease in reactivity in magnitude $(1.2 \pm 0.2) \times 10^{-5}$ per degree Centigrade (Section III-B-3). This coefficient, which results from Doppler broadening of the U^{238} resonances and thermal expansion of the fuel plates, was calculated on the assumption of uniform temperature for all fuel plates. If the coefficient is to be referred to the hottest plate in the actual reactor power distribution, it must be reduced by a factor 1.60 for the effect of flux distribution. The coefficient then becomes $(0.752 \pm 0.12) \times 10^{-5}$ per degree Centigrade on the hottest fuel plate. With this coefficient, an increase from room temperature to melting point in the temperature of the hottest point on the hottest plate would result in a reactivity decrease of 0.82 ± 0.13 per cent k_{eff} .

The importance of the metal temperature coefficient lies in the fact that it is effectively instantaneous in action. In a short-period reactor runaway it will slow down the rate of power increase and thus make available more time for complete shutdown by steam formation. It is instructive, however, to examine the behavior of the reactor under the assumed condition that the fuel plates are insulated from the surrounding water and no steam is formed. Figure 47 shows the calculated course of a power excursion under such conditions. The excursion is assumed to be initiated by the instantaneous introduction of 1.1% excess reactivity (0.34% prompt excess reactivity) at a very low initial value of reactor power. The figure

shows that the power reaches its maximum value when the plate temperature rise has reached a value of about 465C. In the case of Figure 38 the total reactivity removed by plate temperature is somewhat more than the total (1.88%) originally added. For cases where considerably less than this amount of reactivity is added initially the first power excursion will fail to remove all of the delayed excess reactivity. In such cases the power, after the first rapid decrease, will remain at a low but appreciable level, supported by multiplication of the delayed neutrons, and the plate temperature will slowly drift up to, and beyond, the value which removes all of the excess reactivity. This latter portion of the transient is considered unimportant, however, since in the practical case it will be prevented by the formation of steam. Figure 48 shows, as a function of excess reactivity, the maximum temperature reached by the fuel plate during the rapid part of the transient.

Using the metal temperature coefficient of 0.85×10^{-5} k_{eff} per degree C for the EBWR, the maximum excess reactivity that can be handled by metal coefficient alone, without melting of fuel plates, would be 1.2% k_{eff} (or maybe 1.1%). The combined effect of metal temperature and steam shutdown will be discussed in a later section. At this temperature the metal coefficient has reduced the initial excess reactivity by $465 \times 0.75 \times 10^{-5} \times 100 = 0.35\%$ k_{eff} , i.e., it has removed the prompt excess reactivity. This result is, of course, to be expected on theoretical grounds, so long as the initial excess reactivity is well above prompt critical. Figure 47 shows further that the temperature continues to rise at a relatively rapid rate until about twice the prompt excess reactivity has been removed by temperature coefficient. A number of similar calculations (not all applicable to the EBWR) have established that this behavior is characteristic of shutdown by metal temperature coefficient so long as the initial excess reactivity is well above β . Naturally, if $2(k_{ex} - \beta)$ is less than k_{ex} (as for the case of Figure 47) the plate temperature will continue to drift slowly upward until the product of temperature and metal coefficient is equal to $-k_{ex}$. This part of the temperature rise is so slow, however, that it will certainly be prevented in a boiling reactor by steam shutdown. It will therefore be ignored.

It is worth pointing out that, except for a shift in zero of the time scale, the curve (Figure 47) of fuel plate temperature rise is independent of the heat capacity of the fuel plates - increasing heat capacity simply shifts the curve to the right. Further, the maximum plate temperature reached is inversely proportional to the magnitude of the metal coefficient of reactivity. Consequently, the curve of metal temperature in Figure 47 is a quite general one, which, if the zero of the time scale is allowed to shift, and the ordinate scale is properly adjusted, can handle all values of fuel plate heat capacity and metal temperature coefficient of reactivity. If curves like those of Figure 47 are plotted in time units of t/τ , where τ is the stable period of reactor power rise before plate temperature has risen enough to affect reactivity, their shapes change only by a relatively small amount for different values of τ . Figure 48 shows a replot of the power

curve of Figure 47 on this basis and, for comparison, the power curve for the case in which the initial excess reactivity was 1.88% k_{eff} . Both curves have been normalized to the same arbitrary maximum value.

In consideration of the relationships discussed above, it can be assumed with confidence that the maximum fuel plate temperature reached during the time of rapid temperature rise in a super-prompt-critical excursion controlled by metal coefficient alone will be very nearly proportional to the prompt excess reactivity initially introduced. Figure 49 is a curve based on this assumption and drawn to pass through the points determined by the two calculations of Figure 48. In determining these points, the end of the "time of rapid temperature rise" was arbitrarily defined as the time when the logarithmic time derivative of temperature rise fell to one-tenth of its initial value.

The melting point of uranium is marked on Figure 49 for the case in which the reactor is initially at room temperature. If the reactor is initially at saturation temperature (atmospheric pressure) the melting point line would be about 70° lower on Figure 49. In either case, the maximum excess reactivity which can be tolerated without melting fuel plates is between 1.1% and 1.2% k_{eff} .

Since the metal temperature coefficient causes the period of a power excursion to begin to lengthen as soon as the fuel plate temperature rises appreciably, it is to be expected that this effect would lengthen the time available for the formation of steam, and thus assist in shutdown of the reactor by steam even in those cases where the excess reactivity is too large to be handled safely by metal temperature coefficient alone. Unfortunately, the knowledge of the details of the steam shutdown process is not presently sufficient to allow a firm estimate of the combined effects of steam and metal coefficient. It can only be said that the reactor is safe against the maximum excess reactivity which can be handled by whichever process is the most effective in a given situation, and that the action of the other process must be regarded simply as an additional safety factor.

THE CURVES ARE NORMALIZED TO GIVE ΔT ABOVE
SURFACE TEMPERATURE PER UNIT HEAT FLUX THROUGH
THE OUTSIDE SURFACE OF THE CLADDING.

FIG. 35
TEMPERATURE DISTRIBUTIONS IN
NATURAL URANIUM FUEL PLATE

THE QUANTITY PLOTTED IS $T / \frac{E}{\rho C}$ WHERE E IS THE TOTAL ENERGY GENERATED PER UNIT VOLUME OF FUEL - i.e. THE QUANTITY PLOTTED IS THE RATIO OF ACTUAL TEMPERATURE TO THE TEMPERATURE THE FUEL WOULD REACH IF THERE WERE NO HEAT LOSS.

FIG. 36

TEMPERATURE DISTRIBUTIONS IN FUEL PLATE FOR VARIOUS EXPONENTIAL PERIODS OF POWER RISE, ASSUMING NEGLIGIBLE TEMPERATURE DROP FROM PLATE SURFACE TO WATER, AND INITIAL PLATE TEMPERATURE = INITIAL WATER TEMPERATURE = 0

EFFECTIVE CONDUCTANCE = HEAT FLUX THROUGH
 PLATE SURFACE PER UNIT TEMPERATURE DIFFERENCE
 BETWEEN CENTER AND SURFACE OF FUEL PLATE

FIG. 37
 COMPUTED VARIATION OF EFFECTIVE FUEL
 PLATE CONDUCTANCE WITH RECIPROCAL
 PERIOD FOR EXPONENTIAL POWER EXCURSIONS

PERIOD = 0.021 sec
INITIAL TEMPERATURE = 82 F

FIG. 38
POWER AND FUEL PLATE TEMPERATURE
RISE DURING EXCURSION

REACTOR WATER INITIALLY AT SATURATION TEMPERATURE
FOR ATMOSPHERIC PRESSURE

BORAX I CURVE FROM ANL-5211, WITH CORRECTED
POWER CALIBRATION

FIG. 39
TOTAL ENERGY RELEASED DURING BORAX - TYPE
POWER EXCURSIONS, AS A FUNCTION OF THE
RECIPROCAL OF THE REACTOR PERIOD WHICH
EXISTED BEFORE SHUTDOWN BEGAN

REACTOR WATER INITIALLY AT ROOM TEMPERATURE
(~80 F)

BORAX I CURVE FROM ANL-5323

DASHED LINES INDICATE UPPER LIMIT ONLY

FIG. 40
TOTAL ENERGY RELEASED DURING BORAX -
TYPE POWER EXCURSIONS, AS A FUNCTION
OF THE RECIPROCAL OF THE REACTOR PERIOD
WHICH EXISTED BEFORE SHUTDOWN BEGAN

FIG. 41
TOTAL ENERGY RELEASED DURING BORAX II
POWER EXCURSIONS: COMPARISON OF NON-
PRESSURIZED AND PRESSURIZED CASES

FIG. 42
 MAXIMUM FUEL PLATE TEMPERATURE RISE DURING
 BORAX II POWER EXCURSIONS: COMPARISON
 OF UNPRESSURIZED AND PRESSURIZED CASES

FIG. 43
TYPICAL POWER VARIATION DURING
SHUTDOWN BY STEAM IN BORAX II

REACTOR INITIALLY AT ATMOSPHERIC PRESSURE
AND SATURATION TEMPERATURE

FIG. 44
ESTIMATED UPPER LIMIT OF MAXIMUM FUEL PLATE
TEMPERATURE RISE IN POWER EXCURSIONS, AS A
FUNCTION OF RECIPROCAL PERIOD OF EXCURSION

FIG. 45
TEMPERATURE DISTRIBUTION IN FUEL
PLATE DURING EXPONENTIAL POWER RISE

FIG. 46
ESTIMATED UPPER LIMIT OF MAXIMUM FUEL PLATE
TEMPERATURE RISE IN POWER EXCURSIONS, AS A
FUNCTION OF RECIPROCAL PERIOD OF EXCURSION

FIG. 47
 SHUTDOWN BY METAL TEMPERATURE COEFFICIENT:
 REACTOR POWER AND FUEL PLATE
 TEMPERATURE RISE FOR 1.10% k_{ex}

FIG. 48
NON-DIMENSIONAL PLOT OF POWER VARIATION
DURING SHUTDOWN BY METAL
TEMPERATURE COEFFICIENT

FIG. 49
SHUTDOWN BY METAL TEMPERATURE COEFFICIENT:
MAXIMUM FAST TEMPERATURE RISE OF HOTTEST
PLATE VS APPLIED EXCESS PROMPT REACTIVITY

APPENDIX F*

CLIMATOLOGY (METEOROLOGY GROUP)

Annual climatological summaries based on measurements taken at the DuPage site of the Argonne National Laboratory have been prepared and published as reports ANL-4538, ANL-4793, ANL-4928, and ANL-5256. These reports cover the period beginning July 1949 and ending June 1953. As a supplement to the Argonne data, monthly weather summaries prepared by the United States Weather Bureau are included as appendices to these annual climatological reports.

I. Wind Measurements

Wind measurements taken by both the U. S. Weather Bureau and the Argonne National Laboratory are presented in this report. Although the wind measurements made at Argonne are more representative of this site, the Weather Bureau data are included since they represent the best available long series of measurements for this area. Upper air wind data obtained by the U. S. Weather Bureau are also presented here. No upper air measurements obtained with the aid of balloons are available for the Argonne site.

Surface wind roses based on 10 years of Weather Bureau data gathered at the Chicago Midway Airport are shown in Figure 50. Figures 51 and 52 are upper air wind roses for this area.

Several wind studies based on data included in the four above-mentioned ANL climatological reports and on data for an additional year have been made and are presented in Tables XXV - XXVIII. Table XXV consists of a joint frequency distribution of wind speed and direction for the 5-year period July 1949 to June 1954. A similar study for a single year is shown in Table XXVI. However, in this table the frequencies are given on a percentage basis.

To facilitate hazard evaluation, information on how long a given wind direction may persist is desirable. An investigation of the number of times the wind from a particular direction persisted for a given number of hours was made and the results are given in Table XXVII. Again these data are based on the 5-year period from July 1949 through June 1954. A similar study based on data from July 1949 through June 1950 is shown in Table XXVIII.

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*Appendix F has been somewhat revised from the original presentation.

SURFACE WIND ROSES
 CHICAGO MIDWAY AIRPORT, CHICAGO, ILLINOIS
 BASED ON 85,133 OBSERVATIONS COPIED FROM AIRWAY METEOROLOGICAL ATLAS FOR THE UNITED STATES.
 U. S. WEATHER BUREAU

Figure 50

NOTE: THESE DATA WERE TAKEN FROM THE AIRWAY
METEOROLOGICAL ATLAS FOR THE UNITED STATES
U.S. WEATHER BUREAU

CHICAGO MIDWAY AIRPORT, CHICAGO, ILLINOIS

UPPER AIR WIND ROSES

Figure 51

Measured from circumference of circle

% SCALE

SEPTEMBER - OCTOBER - NOVEMBER

5000 METERS
(205 OBSERVATIONS)

3000 METERS
(1309 OBSERVATIONS)

1000 METERS
(3620 OBSERVATIONS)

500 METERS
(3985 OBSERVATIONS)

JUNE - JULY - AUGUST

5000 METERS
(417 OBSERVATIONS)

3000 METERS
(2233 OBSERVATIONS)

1000 METERS
(4036 OBSERVATIONS)

500 METERS
(4145 OBSERVATIONS)

Indicated heights are above sea level.

LEGEND

NOTE: THESE DATA WERE TAKEN FROM THE AIRWAY
METEOROLOGICAL ATLAS FOR THE UNITED STATES
U.S. WEATHER BUREAU
CHICAGO MIDWAY AIRPORT, CHICAGO, ILLINOIS

UPPER AIR WIND ROSES

Indicated heights are above sea level.

Table XXV
Joint Frequency Distribution of Wind Direction and Wind Speed
July 1949 - June 1954

LEVEL	DIRECTION (16 POINTS)	WIND SPEED (MPH)											TOTAL	PER CENT
		CALM	1-3	4-7	8-12	13-18	19-24	25-31	32-38	39-46	> 46	MISSING		
150 FOOT	CALM	258	8											
	NNE	2	82	272	892	696	212	50	1			3	269	0.6
	NE		51	207	644	510	118	30	1			2	2209	5.0
												2	1563	3.6
	ENE	2	51	229	789	528	178	40	7			8	1832	4.2
	E	2	90	440	1290	516	120	33	2			4	2497	5.7
	ESE	1	65	247	681	365	54	3				4	1420	3.2
	SE	1	58	256	645	286	57	11	1			3	1318	3.0
	SSE		81	346	831	400	151	25	7			2	1843	4.2
	S	4	76	423	1452	1213	460	139	27	8		4	3806	8.7
150 FOOT	SSW		64	330	1193	1390	705	134	16	4		9	3845	8.8
	SW	1	61	265	960	1294	574	131	7	6		10	3309	7.6
	WSW	2	60	260	918	991	423	169	37	10		10	2880	6.6
	W		86	369	1245	1528	629	319	107	15	1	6	4305	9.8
	WNW	1	68	304	893	1142	422	154	20			9	3013	6.9
	NW	1	52	209	701	884	281	64	2	1		12	2207	5.0
	NNW		82	281	806	966	304	70	22	1		18	2550	5.8
	N	5	71	339	998	945	280	63	5			8	2714	6.2
	VARIABLE	13	134	93	20	6	5						271	0.6
	MISSING	8	61	241	774	379	65	5	3	2		435	1973	4.5
150 FOOT	TOTAL	301	1301	5111	15732	14039	5038	1440	265	47	1	549	43824	
	PER CENT	0.7	3.0	11.7	35.9	32.0	11.5	3.3	0.6	0.1		1.3		
	CALM	1115	73	972	789	231	38					1	1189	2.7
	NNE	8	454					1				21	2512	5.7
	NE	9	329	648	548	158	20					13	1726	3.9
	ENE	9	344	719	632	215	40	4				11	1974	4.5
	E	5	503	1047	568	210	22					7	2362	5.4
	ESE	5	431	521	345	72	1					6	1381	3.2
	SE	3	392	586	338	91	3					5	1416	3.2
150 FOOT	SSE	1	420	653	521	162	15	2				3	1777	4.1
	S	2	322	967	1134	564	99	17	1			6	3112	7.1
	SSW	2	358	1329	1382	723	69	12	1			9	3885	8.9
	SW	5	458	1208	1237	471	60	9				4	3452	7.9
	WSW	2	389	1055	902	444	131	25	1			2	2951	6.7
	W	2	422	1564	1380	774	282	54				7	4485	10.2
	WNW	2	219	971	911	467	123	14				3	2710	6.2
	NW	1	206	749	769	357	49	1					2152	4.9
	NNW	2	261	783	858	525	85	17				5	2536	5.8
	N	9	402	1036	924	397	50	7				12	2837	6.5
	VARIABLE	7	264	147	12							3	433	1.0
150 FOOT	MISSING	18	79	228	204	61	6	1				337	934	2.1
	TOTAL	1207	6326	15183	13473	5922	1093	164	3			453	43824	
	PER CENT	2.8	14.4	34.6	30.7	13.5	2.5	0.4				1.0		

Table XXVI
WIND DIRECTION AND SPEED - PER CENT OF TOTAL NUMBER OF HOURLY
OCCURRENCES FOR EACH DIRECTION AND SPEED GROUP
(19-Foot Level - July 1949 to June, 1950)
Wind Speed in Miles Per Hour

	1	1-3	4-7	8-12	13-18	19-24	25-31	32-38	Miss- ing	Total %
CALM	2.8	1.1	2.0	1.5	.3				.2	2.8
NNE			1.6	1.2	.3				.1	5.0
NE		.6	1.7	1.4	.7				.1	4.0
ENE		.4	2.1	1.4	.6	.1			.1	4.4
E		.8	.9	.8	.2				.1	4.9
ESE		.6							.1	2.5
SE		.6	1.2	.9	.3					3.0
SSE		.5	1.3	1.3	.3	.1				3.5
S		.6	2.5	3.1	2.0	.4	.1			8.8
SSW		.5	2.7	2.5	.9	.1				6.7
SW		.6	3.0	3.4	1.2	.2				8.5
WSW		.7	2.4	1.7	.9	.5	.1			6.2
W		.7	3.1	3.0	2.0	.7	.3		.1	9.9
WNW		.2	1.8	1.9	1.1	.6	.1			5.8
NW		.3	1.6	1.9	1.0	.2			.1	5.0
NNW		.3	1.6	1.9	1.4	.2			.1	5.4
N		1.1	2.2	1.3	.5					5.2
VARIABLE		.9	.5	.1						1.6
MISSING		.9	2.2	1.8	.6	.1				6.8
TOTAL %	2.8	11.4	34.4	31.1	14.3	3.2	.6		1.3	100.0

Table XXVII
Persistence of Wind Direction - 150-Foot Level
July 1949 - June 1954

HOURS	WIND DIRECTION (8 POINTS)										TOTAL FRE- QUENCY	TOTAL HOURS	
	CALM	NE	E	SE	S	SW	W	NW	N	VARI- ABLE			MISSING
1	163	511	428	422	552	653	668	586	560	168	41	4752	4752
2	29	220	193	216	240	277	299	282	243	21	8	2028	4056
3	7	128	107	134	137	182	188	199	129	10	5	1226	3678
4	3	87	90	85	106	130	120	101	87	5	4	818	3272
5		44	60	42	62	192	173	64	59	1	4	501	2505
6	1	33	34	42	74	64	66	44	32	1	5	396	2376
7		27	32	18	42	49	47	45	35		4	299	2093
8		24	14	19	36	41	40	35	23		4	232	1856
9	1	28	21	24	29	30	30	24	19		4	210	1890
10		14	22	12	27	40	26	14	12			167	1670
11		9	11	10	21	31	20	16	18		1	137	1507
12		5	8	6	14	29	18	12	10			102	1224
13		9	6	1	15	8	13	10	12			74	962
14		4	8	6	7	9	19	12	10		4	79	1106
15		6	5	3	11	10	12	4	10		1	62	930
16		7	4	1	8	12	8	9	3		1	52	832
17		7	4		6	9	9	3	1		1	40	680
18		2	6	1	10	8	5	9	2		1	44	792
19		5	3	1	9	5	7	5	2		1	38	722
20		2	4	1	8	3	4	4	2		2	28	560
21		2	1	1	4	6	5	2	1			24	504
22			2		2	4	1	1	4		1	14	308
23		1	1		1	1	2	1	2			10	230
24		1	1	1	3	2	3	1	2			14	336
25			1		1	3	4	1	2			11	275
26			1		1	3	3	1	3			11	286
27			1		2	2	4	1			1	10	270
28				1	2	3	2	2	2			12	336
29			1	1	3	1	1	1			1	7	203
30					3	1	2	1				9	270
31						2	2	1	2			5	155
32					2	1	1	1				4	128
33		1			1	1	1	1	2			5	165
34							1					2	68
35			1		1	1		1	1			3	105
36					1	2						4	144
37						1	1				1	1	37
38					1			2				3	114
39			1								1	3	117
40												2	80
41							1	1			1	3	123
42		1					1		1			3	126
43						1					1	2	86
44						1						1	44
45					1	1		1				2	92
46												2	47
47					1							1	144
48					1	2					1	1	49
49												1	51
50						1	1					2	108
51												1	58
52							1				1	1	61
53											1	1	63
54											1	1	72
55											1	1	302
56											1	1	380
57											1	1	424
58											1	1	438
59											1	1	482
60											1	1	526
61											1	1	570
62											1	1	614
63											1	1	658
64											1	1	702
65											1	1	746
66											1	1	790
67											1	1	834
68											1	1	878
69											1	1	922
70											1	1	966
71											1	1	1010
72											1	1	1054
73											1	1	1098
74											1	1	1142
75											1	1	1186
76											1	1	1230
77											1	1	1274
78											1	1	1318
79											1	1	1362
80											1	1	1406
81											1	1	1450
82											1	1	1494
83											1	1	1538
84											1	1	1582
85											1	1	1626
86											1	1	1670
87											1	1	1714
88											1	1	1758
89											1	1	1802
90											1	1	1846
91											1	1	1890
92											1	1	1934
93											1	1	1978
94											1	1	2022
95											1	1	2066
96											1	1	2110
97											1	1	2154
98											1	1	2198
99											1	1	2242
100											1	1	2286

Table XXVIII

PER CENT OF TIME DURING ONE YEAR THAT WIND FROM A SPECIFIC DIRECTION PERSISTED,
DISTRIBUTED ACCORDING TO WIND DIRECTION AND DURATION OF PERSISTENCE

(150-Foot Level, July 1949-June 1950)

Hours	Calm %	NE %	E %	SE %	S %	SW %	W %	NW %	N %	Variable %	Total %
1 hour or less	.26	1.97	2.05	1.89	2.73	3.38	3.23	2.74	2.24	.66	21.16
Longer than 1	.10	5.47	5.13	3.71	10.63	11.76	11.45	7.68	5.14	.36	61.59
Longer than 2	.02	4.31	3.94	2.63	8.83	9.58	9.37	5.87	3.90	.18	48.78
Longer than 3	.01	3.53	3.10	1.90	7.55	7.99	7.83	4.67	2.99	.07	39.79
Longer than 4	0	2.95	2.46	1.40	6.47	6.69	6.61	3.83	2.35	.02	32.92
Longer than 5	0	2.53	2.03	1.03	6.38	5.63	5.63	3.21	1.85	.01	27.62
Longer than 6	0	2.21	1.68	.72	5.56	4.82	4.76	2.68	1.46	0	23.22
Longer than 7	0	1.92	1.39	.50	4.86	4.14	4.10	2.22	1.12	0	19.59
Longer than 8	0	1.67	1.16	.29	4.26	3.56	3.54	1.88	.90	0	16.60
Longer than 9	0	1.48	.98	.14	3.75	3.05	3.08	1.62	.73	0	14.16
Longer than 10	0	1.32	.82	.07	3.33	2.64	2.70	1.39	.59	0	12.20
Longer than 11	0	1.17	.67	.02	2.95	2.33	2.35	1.17	.48	0	10.49
Longer than 12	0	1.04	.54	0	2.62	2.10	2.02	.96	.37	0	9.01
Longer than 13	0	.91	.43	0	2.32	1.89	1.75	.77	.28	0	7.71
Longer than 14	0	.81	.34	0	2.07	1.70	1.51	.61	.21	0	6.60
Longer than 15	0	.71	.25	0	1.84	1.51	1.33	.48	.16	0	5.63
Longer than 16	0	.62	.17	0	1.66	1.33	1.18	.35	.14	0	4.80
Longer than 17	0	.53	.10	0	1.51	1.20	1.05	.26	.12	0	4.13
Longer than 18	0	.46	.04	0	1.37	1.07	.95	.20	.10	0	3.56
Longer than 19	0	.40	.02	0	1.26	.97	.86	.15	.08	0	3.10
Longer than 20	0	.37	.01	0	1.19	.87	.79	.12	.06	0	2.76

II. Frequency of Turbulent and Stable Conditions

Several studies to determine the relative frequency of turbulence or periods of rapid diffusion have been made. One of these studies, based on data for the year July 1949 - June 1950, uses the wind data only, i.e., wind speed ratio is used as the criterion of turbulence. The wind speed ratio is obtained from readings at the 150-foot and 19-foot levels on the Meteorology Tower. This study is based on wind data only since temperature lapse rate data are unavailable for this period.

For the period July 1951 - June 1954 both wind data and temperature lapse rate data are available. A number of studies combining these parameters are shown below.

Wind ratios as a measure of turbulence have been used very successfully by the British Chemical Warfare Service. Also it is by means of the wind ratio expressed by the parameter n that Sutton varies the index of turbulence in his diffusion formula (see Section IV below). (Ministry of Supply-Rearmament Records of Research and Development, No. 9,400.)

The following relationship between wind ratio and n as presented by Sutton is a satisfactory first approximation:

$$\frac{u}{u_0} = \left(\frac{Z}{Z_0} \right)^{\frac{n}{2-n}}$$

$$u = \text{Wind speed at upper level}$$

$$u_0 = \text{Wind speed at lower level}$$

$$Z = \text{Height of upper level}$$

$$Z_0 = \text{Height of lower level}$$

The British found that the wind speed ratio was related to three meteorological quantities: wind speed, change of temperature with height, and surface roughness. They also found that it is a useful indicator for the degree of turbulent diffusion of gases (or particulate matter of less than 10μ) in the atmosphere. This is to be expected since the distribution of wind speed with height is a result of the transfer of momentum, and it is generally assumed that this transfer is accomplished by the same turbulent eddies which achieve the diffusion of gases (or particulate matter of less than 10μ) in the lower layers of the atmosphere.

The calculations of wind ratios at the DuPage site are made under different surface roughness conditions and over a different height interval than the British measurements, which were made at heights of 2 meters and 1 meter above the ground. Measurements of surface roughness have not been made at the DuPage site. However, a spot check of the data indicates that the contribution of the surface roughness to the ratio of wind speed at 150 feet to 19 feet is not as large as the contribution found by the British. This is, perhaps, because of the larger and more elevated height interval used at the DuPage site.

A comparison of the values of n computed from the Argonne wind ratios with the values of n obtained by the British, using time of day and wind speed to determine roughly the amount of turbulence present, indicates that the relationship between n and turbulence determined experimentally by the British can be applied to the Argonne data. The ranges of n adapted for the various stability limits differ slightly from those of the British and are shown in Table XXIX.

Table XXIX

NUMBER OF OCCURRENCES OF RATIO OF WIND SPEED AT 150 FEET
TO WIND SPEED AT 19 FEET (R) FOR THREE GENERAL CLASSIFICATIONS OF TURBULENCE
GROUPED ACCORDING TO WIND SPEED AT 19-FOOT LEVEL (JULY 1949 - JUNE 1950)

	Wind speed for 19-foot level in miles/hr												Total	%
	n	R	0	1-2	3-7	8-12	13-18	19-24	25-31	32-38				
Turbulent	-1.000-	.273	0.50-1.39	14	61	498	940	695	189	44	1	2442	27.9	
Neutral	.279 -	.340	1.40-1.53	0	27	318	592	344	78	16	0	1375	15.7	
Non-Turbulent	.344 - 1.000		1.54-8.00	103	446	2575	1105	198	6	0	0	4433	50.6	
Missing Data													5.8	
Total													100%	

Theoretically n should not be negative. The few cases of R less than 1 appear to be due to local terrain effects.

Since the biggest possibility of error lies in the boundaries of the neutral zone, between 0.300 and 0.335 according to the British, this neutral zone was expanded from 0.279 to 0.340 to include any doubtful cases of stable or unstable turbulence in the neutral or change-over zone. The use of only three categories for the whole range of conditions of turbulence is somewhat misleading. However, it is impossible to make a detailed breakdown without a more thorough investigation into the quantitative variation of wind ratio with both wind speed and the vertical distribution of temperature. This breakdown is sufficient for the type of calculations which can be made in this report.

In Table XXIX, the cases where the wind speed at the 19-foot level was 0 had to be considered in a different manner, since the wind ratio would be infinite. Here the breakdown was accomplished by assuming stable conditions during the hours of darkness and unstable conditions with

the hours of sunshine. Research since this breakdown was completed indicates that in some cases with a wind ratio of 1.54 - 1.69 the accompanying wind speed may create enough mechanical turbulence to place these cases in the unstable category. However, the possible error cannot exceed 5 or 6 per cent of the data and is on the side of safety.

A summary of the frequency of occurrence of atmospheric stability based on measurements taken at the DuPage site for the period July 1951 - June 1954 is presented in Table XXX. The temperature differences were obtained by thermopiles which were aspirated and shielded from solar radiation. One thermopile was mounted about 5 feet above the ground and the other on a tower at 144 feet.

Table XXX

FREQUENCY DISTRIBUTION OF TEMPERATURE DIFFERENCES IN HEIGHT INTERVAL OF 42 METERS

July 1951 - June 1954

	Temperature Diff. °C (T_{top} minus T_{bottom}) Over Height Interval of 42 meters	No. of Cases July 1951 - June 1954
Unstable 26.3%	≤ -2.0 -1.9 to -1.5 -1.4 to -1.0 -0.9 to -0.5	10 224 1615 4615
Near Neutral 28.3%	-0.4 to 0.0	6957
Stable 45.4%	0.1 to 0.5 0.6 to 1.0 1.1 to 1.5 1.6 to 2.0 2.1 to 2.5 2.6 to 3.0 >3.0	3597 1942 1329 1098 793 541 1868

As a further aid in determining how often periods of rapid and slow diffusion may prevail the studies of Tables XXXI - XXXIV are presented. In these studies the data were separated into two periods. The first period, 0800 - 1800, was chosen as representing the period of working hours or the period during which the population density onsite is high. The second period, 1900 - 0700, is considered the off-hour period.

In Tables XXXI and XXXII are shown the relationship between lapse rate and wind speed and in Tables XXXIII and XXXIV the relationship between lapse rate and wind ratio. It is evident from these tables that the parameters depend upon each other in a complicated fashion. However, it is clearly shown by these investigations that atmospheric conditions favoring rapid diffusion are likely to be present during working hours while conditions favoring slow diffusion are considerably more prevalent during off hours.

Data on the frequency with which stability conditions occur during each month and during the 6-hour periods 0100-0600, 0700-1200, 1300-1800, and 1900-2400 are presented in Tables XXXV through XXXVIII. This information is based on hourly observations of the temperature difference between the 144 foot level on the Meteorology Tower and 5.5 feet above ground during the period from July 1951 - June 1954.

III. Precipitation, Stability, and Wind

Based on 75 years of Weather Bureau records the average monthly totals of precipitation range from 1.88 in. in February to 3.61 in. in May. During the spring and summer months the average monthly totals of precipitation are 1-1/2 to 2 times as large as those in the fall and winter months. A summary of precipitation and other climatological data prepared by the U. S. Weather Bureau is presented in Table XXXIX.

Data showing the frequency of occurrence of combinations of wind speed and wind direction during precipitation are shown in Tables XL and XLI. The information in Table XL is based on measurements by the U. S. Weather Bureau taken during the 8-year period, January 1934 - December 1941 at the Chicago Midway Airport. This material was taken from a study by Col. Benjamin G. Holzman of the USAF. Argonne data for the 5-year period, July 1949 - June 1954 are presented in Table XLI.

It is interesting to note that there were 8198 hourly observations of precipitation or precipitation on 11.4% of the measurements, in the 8-year period of the Weather Bureau data. Precipitation occurred during only 8.3% of the observations during the period of the Argonne data. This difference may not be this large since the Argonne Weather Station is manned only 8 hours per day 5 days a week. Precipitation measurements at Argonne are made by a modified weighing type Bendix-Friez precipitation gauge. It is likely that some hours during which only a trace occurred were not detected by the precipitation gauge for periods when Argonne observers were off duty.

Table XXXI
 JOINT FREQUENCY DISTRIBUTION OF VERTICAL TEMPERATURE DIFFERENCE VERSUS
 19-FOOT WIND SPEED DURING THE HOURS 1900 THROUGH 0700
 FOR THE PERIOD JULY 1951-JUNE 1953

Temperature Difference $T_{144} - T_{5.5}$ (°C)	19-foot Wind Speed (mph)										Total
	0-4	5-9	10-14	15-19	20-24	25-29	30-34	35-39	≥40	Missing	
≤2.0	1	1	4	1	1	2	16	5			4
-1.9 to -1.5											7
-1.4 to -1.0		1			2						267
-0.9 to -0.5	34	107	71	31		5	3				2273
-0.4 to 0.0	270	881	764	251	87	14	4			2	1790
0.1 to 0.5	314	812	506	125	23	5	2			3	1057
0.6 to 1.0	328	553	159	9	4	1				3	727
1.1 to 1.5	351	347	24	5							646
1.6 to 2.0	367	261	17							1	471
2.1 to 2.5	318	146	3		1					3	431
2.6 to 3.0	314	113	2		1					1	1155
>3.0	899	255								1	675
Missing	276	257	110	23	8	1					9503
Total	3472	3734	1660	445	143	26	9			14	

JOINT FREQUENCY DISTRIBUTION OF VERTICAL TEMPERATURE DIFFERENCE VERSUS
19-FOOT WIND SPEED DURING THE HOURS 0800 THROUGH 1800
FOR THE PERIOD JULY 1951-JUNE 1953

Temperature Difference $T_{14} - T_{2.5}$ (°C)	19-foot Wind Speed (mph)										Total
	0-4	5-9	10-14	15-19	20-24	25-29	30-34	35-39	≥40	Missing	
≤2.0	6	32	57	30	5	1					4
-1.9 to -1.5											130
-1.4 to -1.0	113	252	255	68	20	2					710
-0.9 to -0.5	437	913	809	236	61	4					2461
-0.4 to 0.0	266	890	876	339	113	25				3	2512
0.1 to 0.5	113	322	335	87	27	5					889
0.6 to 1.0	97	166	102	34	6	2					407
1.1 to 1.5	70	79	26	5	1						181
1.6 to 2.0	37	51	9								97
2.1 to 2.5	33	14	9	4							60
2.6 to 3.0	17	7		3							27
>3.0	26	6	1	1							34
Missing	124	206	133	50	15					1	529
Total	1339	2938	2614	858	249	38				5	8041

JOINT FREQUENCY DISTRIBUTION OF VERTICAL TEMPERATURE DIFFERENCE VERSUS
150/19 FOOT WIND SPEED RATIO DURING THE HOURS 0800 THROUGH 1800
FOR THE PERIOD JULY 1951-JUNE 1953

Table XXXIII

Temperature Difference $T_{144} - T_{5.5}$ (°C)	Wind Speed Ratio (U_{150}/U_{19})									
	<1.0	1.0 - 1.2	1.3 - 1.5	1.6 - 1.8	1.9 - 2.1	2.2 - 2.4	2.5 - 2.9	≥3.0	Missing	Total
≤2.0	1	64	51	3	11	2			7	4
-1.9 to -1.5	5	254	292	31					99	130
-1.4 to -1.0	21	680	1144	172	25	6	4	1	396	710
-0.9 to -0.5	33	456	1324	364	84	17	10	8	215	2461
-0.4 to 0.0	34	128	408	194	40	20	8	8	81	889
0.1 to 0.5	2	35	121	92	46	25	9	20	58	407
0.6 to 1.0	1	8	32	24	29	21	16	15	36	181
1.1 to 1.5			5	20	14	14	14	13	17	97
1.6 to 2.0		1	12	7	3	3	4	13	17	60
2.1 to 2.5			3	1	3	1	4	8	7	27
2.6 to 3.0						2	4	9	16	34
>3.0										
Missing	8	83	187	85	20	14	5	12	115	529
Total	104	1710	3585	993	275	125	78	107	1064	8041

Table XXXIV

JOINT FREQUENCY DISTRIBUTION OF VERTICAL TEMPERATURE DIFFERENCE VERSUS
150/19 FOOT WIND SPEED RATIO DURING THE HOURS 1900 THROUGH 0700
FOR THE PERIOD JULY 1951-JUNE 1953

Temperature Difference $T_{144} - T_{5.5}$ (°C)	Wind Speed Ratio (U_{150}/U_{19})										Total
	<1.0	1.0 - 1.2	1.3 - 1.5	1.6 - 1.8	1.9 - 2.1	2.2 - 2.4	2.5 - 2.9	≥ 3.0	Missing	Total	
≤ 2.0											
-1.9 to -1.5	1	1	1	1	2	1		1		4	
-1.4 to -1.0	2	48	130	48	9	2				267	
-0.9 to -0.5		215	1060	533	139	56	21	29	30	2273	
-0.4 to 0.0	4	47	533	535	224	126	67	44	212	1790	
0.1 to 0.5	2	8	90	219	249	138	77	96	180	1057	
0.6 to 1.0		2	23	83	135	100	89	106	189	727	
1.1 to 1.5				40	71	112	104	113	198	646	
1.6 to 2.0				10	41	60	73	115	166	471	
2.1 to 2.5				15	28	43	58	121	165	431	
2.6 to 3.0				27	42	86	132	260	598	1155	
>3.0											
Missing	1	27	109	116	89	47	37	61	188	675	
Total	7	355	1967	1627	1029	772	658	946	2142	9503	

Table XXV
Frequency Distribution of Temperature Difference
Between 144-Foot and 5.5-Foot Level
For Hours 0100 - 0600
July 1951 - June 1954

TEMPERATURE DIFFERENCE °C 144-FT. MINUS 5.5-FT. LEVEL	JULY	AUG	SEPT	OCT	NOV	DEC	JAN	FEB	MAR	APR	MAY	JUNE	TOTAL
	MONTH												
-1.4-- -1.0	3	1	6	7	21	63	18	21	22	24	9	4	5
-0.9-- -0.5	85	69	68	104	163	192	205	212	179	188	116	93	1674
0.1--0.5	96	90	92	66	119	90	132	97	128	77	90	98	1175
0.6--1.0	62	58	45	53	65	43	54	46	45	41	61	80	653
1.1--1.5	27	50	36	51	43	17	33	19	49	28	49	58	460
1.6--2.0	39	52	40	43	38	5	25	30	35	24	31	64	426
2.1--2.5	36	36	46	36	25	6	16	15	25	19	26	40	326
2.6--3.0	29	33	41	28	11	4	9	12	10	11	26	18	232
> 3.0	114	157	143	154	52	7	24	45	26	58	84	57	921
MISSING	67	12	23	16		131	42	13	39	70	66	26	505
TOTAL	558	558	540	558	540	558	558	510	558	540	558	540	6576

Frequency Distribution of Temperature Difference Between
144-Foot and 5.5-Foot Level
For Hours 0700 - 1200
July 1951 - June 1954

Table XXXVI

TEMPERATURE DIFFERENCE °C 144-FT. MINUS 5.5-FT. LEVEL	MONTH												TOTAL
	JULY	AUG	SEPT	OCT	NOV	DEC	JAN	FEB	MAR	APR	MAY	JUNE	
<-2.0	10	6	2		1	1		11	1	27	37	1	2
-.19--.15	133	89	79	93	64	35	17	88	88	127	153	86	1052
-.14--.10	198	267	247	240	216	178	139	160	193	210	232	292	2572
-.09--.05	115	148	142	136	167	131	235	177	151	106	94	111	1713
0.1-0.5	31	23	31	29	39	40	69	31	44	6	11	36	390
0.6-1.0	5	10	9	14	17	29	24	8	15	2	1	7	141
1.1-1.5	3	4	4	4	17	4	11	7	4	7	1	1	67
1.6-2.0	1	1	1	8	5	2	5	4	2		1		30
2.1-2.5		1	3	8	4	2	5	7	3				33
2.6-3.0			1	1			4						6
> 3.0			1	15	9	2	11	4	2				44
MISSING	62	9	20	10	1	134	38	13	28	45	39	6	405
TOTAL	558	558	540	558	540	558	558	510	558	540	558	540	6576

Table XXXVII
Frequency Distribution of Temperature Difference Between
144-Foot and 5.5-Foot Level
For Hours 1300 - 1800
July 1951 - June 1954

TEMPERATURE DIFFERENCE °C 144-FT. MINUS 5.5-FT. LEVEL	MONTH												TOTAL
	JULY	AUG	SEPT	OCT	NOV	DEC	JAN	FEB	MAR	APR	MAY	JUNE	
<-2.0	5	8	3		1				1	6			8
-1.9--1.5									12	30	15	2	97
-1.4--1.0	54	41	48	50	34	28	7	56	53	80	67	38	556
-0.9--0.5	141	150	136	139	126	135	89	160	150	160	171	151	1708
-0.4-0.0	137	181	128	134	188	155	259	166	203	148	158	173	2030
0.1-0.5	90	107	99	85	81	61	88	55	63	49	67	87	932
0.6-1.0	50	33	39	52	38	21	35	23	21	14	26	45	397
1.1-1.5	19	21	26	28	27	5	15	8	7	9	6	30	201
1.6-2.0	4	6	17	20	14	10	11	8	5	4	1	9	109
2.1-2.5	1	1	13	14	12	3	7	3	4	2		4	64
2.6-3.0		1	4	10	8	4	2	1				1	31
> 3.0			1	10	7	2	4	2			1		47
MISSING	57	8	17	6	4	134	41	16	21	46	46		396
TOTAL	558	558	540	558	540	558	558	510	558	540	558	540	6576

Frequency Distribution of Temperature Difference Between
144-Foot and 5.5-Foot Level
For Hours 1900 - 2400
July 1951 - June 1954

Table XXXVIII

TEMPERATURE DIFFERENCE °C 144-FT. MINUS 5.5-FT. LEVEL	MONTH												TOTAL
	JULY	AUG	SEPT	OCT	NOV	DEC	JAN	FEB	MAR	APR	MAY	JUNE	
-1.9--1.5					1								6
-1.4--1.0										1			2
-0.9--0.5		1		1	19	46	15	9	22	18	5		136
-0.4--0.0	39	40	44	102	157	209	208	214	192	156	108	71	1540
0.1--0.5	77	65	77	73	119	66	134	107	94	102	91	95	1100
0.6--1.0	61	65	56	70	59	47	65	43	72	56	61	96	751
1.1--1.5	63	70	50	44	51	22	33	34	62	35	61	76	601
1.6--2.0	60	62	63	55	44	15	19	37	38	28	54	58	533
2.1--2.5	43	49	56	51	25	9	16	15	15	16	40	35	370
2.6--3.0	34	43	37	22	21	12	10	9	14	15	31	24	272
> 3.0	127	151	145	134	44	6	14	28	20	59	51	77	856
MISSING	54	12	12	6		126	44	14	29	48	56	8	409
TOTAL	558	558	540	558	540	558	558	510	558	540	558	540	6576

Table XXXIX

METEOROLOGICAL DATA

(OFFICIAL RECORDINGS FROM THE CHICAGO WEATHER BUREAU)

NO. OF YEARS RECORDED	WIND AVERAGE HOURLY VELOCITY	PREVAIL- ING DIREC- TION*	TEMPERATURE		PRECIPITATION		
			AVERAGE DAILY MAX.	AVERAGE DAILY MIN.	AVERAGE	AVERAGE RAIN- FALL	AVERAGE SNOW- FALL
74	30	74	73	73	75	75	61
MONTH							
JANUARY	12	SW	31.8	17.9	24.9	1.92	9.2
FEBRUARY	12	W	33.7	20.0	26.9	1.88	8.5
MARCH	13	SW	43.0	29.2	36.1	2.66	6.0
APRIL	12	NE	54.4	39.6	47.1	2.80	1.0
MAY	11	NE	65.4	49.6	57.5	3.61	0.1
JUNE	10	NE	75.2	59.7	67.5	3.52	0.
JULY	9	SW	89.6	66.0	73.3	3.13	0.
AUGUST	9	NE	78.7	65.3	72.0	3.20	0.
SEPTEMBER	10	SW	72.5	58.3	65.3	3.21	0.
OCTOBER	11	SW	61.1	46.9	53.9	2.58	0.1
NOVEMBER	12	SW	47.0	33.8	40.2	2.37	1.8
DECEMBER	12	SW	35.6	23.4	29.2	1.93	6.7
ANNUAL AVERAGE	11	SW	56.6	42.5	49.5	32.81	33.4

* PREVAILING DIRECTION = DIRECTION FROM WHICH WIND BLOWS

The direction of the prevailing winds expressed in percentage of yearly time is recorded as follows:

North.....	6 %	Southeast....	10 %	West.....	13 %
Northeast....	17 %	South.....	13 %	Northwest...	13 %
East.....	9 %	Southwest.....	19 %	Calm.....	0 %

The average weather conditions for this area follow:

Sunshine: Total hours per year.....2,645

Percent of year..... 58

Average number of days per year:

Clear.....	117
Partly cloudy.....	120
Cloudy.....	128
Rain (over .01)...	124
Snow.....	59
Thunder storms.	38
Dense Fog.....	10

Table XL

**SURFACE WIND SUMMARY - FREQUENCY OF DIRECTION
BY VELOCITY GROUPS DURING PRECIPITATION PERIODS
CHICAGO AIRPORT, JANUARY 1934 - DECEMBER 1941**

DATA OBTAINED FROM A STUDY BY
(COL. BENJAMIN G. HOLZMAN, USAF)

ANNUAL

DIR/VEL	CALM	1-3	4-12	13-24	25-31	32-46	47 & OVER	TOTAL OBS.	4 MPH & OVER %	TOTAL OBS.	ALL OBS. %
N	-	29	400	446	36	4	-	885	10.7	913	11.1
NE	-	40	588	374	12	3	-	978	11.8	1018	12.4
E	-	67	691	310	2	-	-	1002	12.1	1070	13.0
SE	-	33	607	182	2	-	-	792	9.6	825	10.0
S	-	57	734	420	15	2	-	1170	14.1	1227	14.8
SW	-	35	570	388	26	10	-	995	12.0	1030	12.5
W	-	32	481	502	66	18	-	1066	12.9	1098	13.3
NW	-	24	461	489	34	7	1	993	12.0	1017	12.3
TOTAL	59	317	4532	3111	193	44	1	7881	-	8198	-
PER CENT	.7	3.8	54.9	37.7	2.3	.5	-	95.2	-	-	-

JANUARY

N	-	5	45	34	4	1	-	83	6.6	87	7.0
NE	-	10	84	32	4	3	-	124	9.9	134	10.7
E	-	20	132	43	-	-	-	175	13.9	196	15.6
SE	-	7	93	35	-	-	-	128	10.1	134	10.7
S	-	5	130	53	1	1	-	184	14.7	190	15.1
SW	-	-	92	66	2	3	-	162	12.9	162	12.9
W	-	-	85	100	19	5	-	211	16.8	211	16.8
NW	-	1	78	49	7	3	-	137	10.9	138	10.9
TOTAL	-	48	739	412	37	16	-	1204	-	1252	-
PER CENT	-	3.8	32.8	32.8	2.9	1.3	-	95.8	-	-	-

FEBRUARY

N	-	1	55	77	3	-	-	134	11.5	135	11.6
NE	-	7	91	90	-	-	-	182	15.5	189	16.1
E	-	8	95	64	-	-	-	159	13.5	167	14.2
SE	-	3	83	28	-	-	-	106	6.3	113	9.7
S	-	9	78	45	2	-	-	119	10.6	134	11.3
SW	-	5	37	43	4	-	-	85	7.1	89	7.8
W	-	3	40	63	14	8	-	123	10.5	127	10.8
NW	-	2	94	113	5	1	-	215	18.3	217	18.3
TOTAL	3	38	573	523	28	8	-	1133	-	1171	-
PER CENT	.3	3.2	48.8	44.5	2.4	.8	-	96.5	-	-	-

MARCH

N	-	1	48	75	4	-	-	126	15.8	128	16.0
NE	-	-	68	66	6	-	-	145	17.5	140	17.5
E	-	5	70	57	2	-	-	130	16.1	135	16.8
SE	-	2	43	20	-	-	-	44	5.2	46	5.7
S	-	1	24	34	-	-	-	78	9.6	79	9.9
SW	-	3	51	37	-	-	-	89	11.0	90	11.3
W	-	-	34	47	9	-	-	80	11.2	92	11.5
NW	-	1	36	46	4	-	-	86	10.4	86	10.7
TOTAL	5	13	374	382	27	-	-	778	-	796	-
PER CENT	.6	1.6	46.7	47.7	3.4	-	-	97.1	-	-	-

Table XL (Cont'd.)

DIR/VEL	CALM	1-3	4-12	13-24	25-31	32-46	47 & TOTAL OVER OBS.	4 MPH & OVER %	TOTAL OBS.	ALL OBS. %
<u>APRIL</u>										
N	-	3	27	61	7	1	96	13.5	99	14.0
NE	-	2	61	80	1	-	141	19.9	143	20.1
E	-	2	67	63	-	-	131	18.5	133	18.8
SE	-	2	43	20	-	-	63	8.8	65	9.1
S	-	4	48	24	3	-	74	10.5	78	11.1
SW	-	4	39	29	1	-	70	9.8	74	10.3
W	-	1	23	24	3	2	52	7.4	53	7.5
NW	-	2	10	42	4	1	57	8.1	59	8.3
TOTAL	5	20	318	343	19	4	684	-	704	-
PER CENT	.7	2.8	44.9	48.4	2.7	.6	96.5	-	-	-
<u>MAY</u>										
N	-	2	35	33	2	-	70	12.7	72	13.1
NE	-	3	58	24	-	-	75	13.6	72	14.1
E	-	2	41	11	-	-	63	11.3	56	12.9
SE	-	2	42	24	-	-	66	9.3	72	10.2
S	-	4	29	23	1	-	53	11.9	72	13.0
SW	-	4	30	47	1	-	78	9.7	82	10.4
W	-	2	24	32	-	-	56	14.0	58	14.8
NW	-	-	-	-	-	-	-	10.1	58	10.5
TOTAL	6	35	297	212	4	-	513	-	548	-
PER CENT	1.1	6.3	53.6	38.3	.7	-	92.6	-	-	100.0
<u>JUNE</u>										
N	-	1	35	17	-	1	52	10.9	53	11.1
NE	-	3	64	15	-	-	80	16.8	83	17.3
E	-	3	68	8	-	-	76	15.8	79	16.4
SE	-	5	24	3	-	-	27	5.7	32	6.7
S	-	5	46	16	-	-	62	12.9	67	14.0
SW	-	2	50	24	1	1	73	15.2	75	15.6
W	-	3	23	29	-	-	54	11.3	57	11.8
NW	-	2	18	8	1	-	28	5.7	30	6.2
TOTAL	5	24	328	120	2	2	452	-	476	-
PER CENT	1.0	5.0	68.2	24.9	.4	.4	94.1	-	-	-
<u>JULY</u>										
N	-	2	19	7	-	-	26	12.4	28	13.4
NE	-	3	19	2	-	-	21	10.4	23	11.5
E	-	-	15	-	-	-	15	7.6	16	7.8
SE	-	2	16	-	-	-	16	7.9	18	8.7
S	-	1	19	3	-	-	22	10.6	23	11.3
SW	-	2	30	6	2	-	37	18.1	39	19.1
W	-	4	21	6	-	-	28	14.0	32	16.0
NW	-	-	13	9	-	1	23	11.3	23	11.3
TOTAL	2	14	152	33	2	1	188	-	190	-
PER CENT	1.0	6.9	74.5	16.2	1.0	.5	92.3	-	-	-

Table XL (Cont'd.)

DIR/VEL	CALM	1-3	4-12	13-24	25-31	32-46	47 & OVER	TOTAL OBS.	4 MPH & OVER %	TOTAL OBS.	ALL OBS. %
<u>AUGUST</u>											
N	-	3	27	9	2	-	-	39	11.4	41	12.3
NE	-	4	26	7	-	-	-	33	9.8	37	10.9
E	-	4	32	5	-	-	-	37	11.0	42	12.4
SE	-	2	48	3	-	-	-	51	15.2	53	15.6
S	-	6	38	15	-	-	-	43	12.7	49	14.5
SW	-	-	29	15	-	-	-	44	13.2	44	13.2
W	-	2	18	6	1	-	-	24	7.0	26	7.5
NW	-	5	21	10	1	-	-	33	9.8	38	11.2
TOTAL	8	26	239	60	4	1	-	304	-	330	-
PER CENT	2.4	7.7	70.7	17.8	1.2	.3	-	90.1	-	-	-
<u>SEPTEMBER</u>											
N	-	5	22	40	2	-	-	66	14.8	71	15.9
NE	-	3	37	11	-	-	-	47	10.6	50	11.1
E	-	5	59	1	-	-	-	60	13.5	65	14.6
SE	-	4	40	3	-	-	-	44	9.8	48	11.0
S	-	3	53	23	-	-	-	75	16.9	78	17.5
SW	-	1	38	19	-	-	-	57	12.7	58	13.0
W	-	4	17	18	1	-	-	37	8.2	41	9.1
NW	-	2	15	9	2	1	-	25	5.8	27	6.2
TOTAL	7	27	281	124	5	1	-	411	-	438	-
PER CENT	1.6	6.1	63.1	27.9	1.1	.2	-	92.3	-	-	-
<u>OCTOBER</u>											
N	-	1	24	22	5	-	-	46	9.9	48	10.1
NE	-	2	11	7	-	-	-	19	4.1	21	4.6
E	-	4	17	14	-	-	-	31	6.6	35	7.4
SE	-	1	42	9	-	-	-	50	10.7	51	11.0
S	-	1	61	59	-	-	-	121	25.8	121	25.9
SW	-	2	51	27	1	1	-	79	17.1	82	17.5
W	-	2	34	16	-	-	-	50	10.7	52	11.1
NW	-	1	35	16	-	-	-	52	11.1	52	11.3
TOTAL	4	14	271	170	6	1	-	448	-	462	-
PER CENT	.9	3.0	58.2	36.5	1.3	.2	-	96.0	-	-	-
<u>NOVEMBER</u>											
N	-	3	33	28	3	-	-	65	8.2	68	8.6
NE	-	4	41	42	-	-	-	82	10.4	86	11.0
E	-	3	34	17	-	-	-	51	6.4	54	6.8
SE	-	1	40	27	2	-	-	69	8.7	69	8.8
S	-	1	55	73	7	1	-	136	17.2	138	17.4
SW	-	6	45	40	11	6	-	102	13.0	108	13.7
W	-	4	49	77	1	1	-	133	16.8	137	17.4
NW	-	4	51	69	4	-	-	123	15.6	127	16.2
TOTAL	2	26	348	373	32	8	-	761	-	787	-
PER CENT	.3	3.3	44.1	47.3	4.1	1.0	-	96.3	-	-	-

Table XL (Cont'd.)

DIR/VEL	CALM	1-3	4-12	13-24	25-31	32-46	47 & OVER	TOTAL OBS.	4 MPH & OVER %	TOTAL OBS.	ALL OBS. %
<u>DECEMBER</u>											
N	-	1	34	43	3	-	-	81	7.7	82	7.8
NE	-	-	29	5	-	-	-	33	3.2	33	3.2
E	-	3	61	13	-	-	-	75	7.2	78	7.5
SE	-	3	114	23	1	-	-	137	13.2	140	13.1
S	-	12	121	62	1	1	-	186	17.8	197	17.0
SW	-	7	81	58	5	-	-	144	13.9	152	14.6
W	-	4	105	69	10	1	-	185	17.8	189	18.1
NW	-	2	67	86	7	-	-	159	15.3	161	15.5
TOTAL	9	32	612	359	27	2	-	1000	-	1032	-
PER CENT	.9	3.1	58.8	34.5	2.6	.2	-	96.1	-	-	-

Joint Frequency Distribution of Wind Direction
and Wind Speed During Hours of Precipitation
July 1949 - June 1954

Table XLI

LEVEL	DIRECTION (16 POINTS)	150 - FOOT																TOTAL	
		CALM	1-3	4-7	8-12	13-18	19-24	25-31	32-38	39-46	> 46	MISSING					TOTAL		
WIND SPEED (MPH)																		MISSING	TOTAL
VARIABLE	CALM	14																14	
	NNE	7	15	64	93	62	14	24	1									255	
	NE	2	21	45	67	42												202	
	ENE	1	24	53	82	54	21	4										248	
	E	9	31	129	101	36	11	1										321	
	ESE	6	27	77	50	19	3											186	
	SE	6	23	50	43	21	7	1										151	
	SSE	3	15	61	89	40	11	4										191	
	S	6	48	105		54	27	8										342	
	SSW	5	24	68	81	36	15	6										236	
	SW	9	12	40	68	38	14	1										188	
	WSW	2	19	54	62	24	5											173	
	W	4	19	59	47	34	21	11										199	
	WNW	6	17	41	51	22	9	4										152	
	NW	2	10	33	52	20	6											124	
	NNW	6	15	60	45	28	16											170	
N	6	27	75	127	77	31	3										347		
TOTAL		2	18	33	21	15											126		
		90	378	1049	1137	622	235	44	11	1							3648		

For the three-year period from July 1951 - June 1954 a study was made of the relationship between wind, precipitation and stability. Tables XLII and XLIII summarize this investigation. Worthy of note is that in periods of no precipitation the atmosphere was stable in only 24% of the measurements. The undesirable combination of atmospheric stability and precipitation occurred on only 2.1% of the observations but the wind speed exceeded 7 miles per hour on over 3/4 of these cases.

The data indicate that when precipitation is occurring there are appreciable differences in the distribution of wind speeds and wind direction at 150 feet during periods of stable and unstable conditions. For example, with precipitation there is a substantially higher proportion of winds from the northeast sector during unstable conditions. Also, the wind speeds average higher during unstable conditions.

IV. Atmospheric Dispersion of Noxious Gases (or Particulate Matter of Less than 10 μ) from a Point Ground Source.

Table XLIV presents ground concentrations in units per cubic meter for a continuous point source located at the ground emitting 1 unit per second. These calculations are based on the equation developed by Sir Oliver Graham Sutton of England which follows:

$$\psi(x,y) = \frac{2Q}{\pi C_y C_z u x^{2-n}} e^{-\left(\frac{y^2}{x^{2-n} C_y^2}\right)}$$

ψ = concentration in units per cubic meter

Q = source strength in units per second

C_y = horizontal diffusion coefficient

C_z = vertical diffusion coefficient

u = wind speed in meters per second

x = distance in meters from source along plume of axis

n = turbulence index ($0 < n < 1$).

Calculations of concentrations were made for values of the diffusion coefficients and the turbulence index prevailing under stable and unstable conditions. The parameters used were as follows:

Table XLII
 Joint Frequency Distribution of Wind Direction and Wind Speed
 During Hours of Precipitation and Stability as Indicated
 150-Foot Level July 1951 - June 1954

DIRECTION (16 POINTS)	WIND SPEED (MPH)										MISSING	TOTAL
	CALM	1-3	4-7	8-12	13-18	19-24	25-31	32-38	39-46	> 46		
STABLE - TEMPERATURE CONSTANT OR INCREASING WITH HEIGHT												
CALM	5											5
NNE		1	5	12	7	7	4					36
NE		2	2	7	3	1	2					17
ENE			4	6	4	1	1					16
E		3	6	22	6							37
ESE		2	7	15	7	3						34
SE		1	5	16	9	4	2					37
SSE		2	2	15	9	9	1	1				39
S			16	25	13	6	4		1			65
SSW		2	6	15	27	5	4	2				61
SW		1	4	7	11	10	1					34
WSW			6	7	8	3						24
W			2	12	1	1	2					18
WNW			5	2	7	2						16
NW			2	6	1	3	1					13
NNW		1	2	9	6	2	2					22
N		1	2	8	20	5	2					38
VARIABLE		2	6	1	1							10
MISSING											1	1
TOTAL	5	18	82	185	140	62	26	3	1		1	523
UNSTABLE - TEMPERATURE DECREASING WITH HEIGHT												
CALM	4											4
NNE		3	6	40	57	39	6					151
NE			10	24	40	29	13					116
ENE												
E		1	10	25	41	33	10					120
ESE		2	14	50	51	16	6	1				140
SE		4	10	38	26	6	1				3	88
SSE		3	9	16	19	11	4	1				63
S		1	6	27	24	21	7	3			1	90
SSW		2	13	31	42	16	9	5	1		1	120
SW		2	10	22	22	13	4	2				76
WSW		2	2	12	30	18	4		1		4	74
W			5	21	33	16	4		4		2	85
WNW		3	7	25	25	21	7	4		1		94
NW		3	4	20	23	11	3	3				67
NNW		1	4	17	34	9	2					67
N												
VARIABLE		3	8	25	22	21	9	3				88
MISSING												184
TOTAL	4	33	137	432	558	321	109	22	9	1	2	1639

Table XLIII

Joint Frequency Distribution of Wind Direction and Wind Speed
During Hours of No-Precipitation and Stability as Indicated
150-Foot Level July 1951 - June 1954

[illegible]

	C_y	C_z	n
Stable	$\frac{0.10}{0.37}$	$\frac{0.06}{0.21}$	$\frac{0.5}{0.2}$
Unstable			

These values have not been definitely determined for the atmosphere under the various meteorological conditions and must be estimated. Also, there is some question as to how valid the equation is over long distances. Measurements have been made at distances up to 14 miles and were within the same order of magnitude as calculated. The equation assumes that the wind direction remains constant throughout the time it takes for the cloud to reach the distance from the source in question. The calculated values will, therefore, be too high except in the case of a persistent stable regime. Also, the concentrations and cloud widths are given for a 3-minute mean. Instantaneous concentrations will be higher and cloud widths smaller. Because of the various possibilities of error, it is felt that the calculations of the above values from Sutton's equation can only be correct to a factor of ten.

Estimates for width and height of the cloud at various distances downwind are also presented, calculated from the same equation. The width is defined as the distance between points on the skirts of the crosswind concentration curve at which the concentration is $1/10$ of the peak value. Similarly, the height is the distance from the ground at which the concentration is $1/10$ of the value on the ground.

This equation must be used with some caution. It has been carefully checked with observations under average weather conditions at distances close to the source. However, the concentration and width of cloud depend upon the values chosen for the diffusion coefficients and turbulent index.

APPENDIX G

REACTOR SUPPORT AND CONTAINMENT STRUCTURE

The problem of preventing damage to the gas-tight building in case of an explosive metal-water reaction in the core of the reactor is complicated and requires analysis by experts. The Armour Research Institute has evaluated the problem.

From a mechanical standpoint it was convenient to consider the reactor support and the reactor explosion containment design jointly. The reactor was to be supported near the top in such a manner that it could absorb the impact of heavy objects, such as accidental dropping of the pressure vessel cover, without extensive damage. It must also be free to expand thermally. Prevention of rupture of the pressure vessel during a violent chemical reaction is not practical. Protection of the gas-tight integrity of the surrounding building was the objective. The most promising approach appeared to be to provide strong structural members above the reactor pressure vessel to prevent fragments from being blown upward. Explosion pressures would be allowed to relieve themselves in a downward direction, where thick concrete protects the gas-tight shell. Radial destructive effects must not destroy the anchorage of the top restraining structures. A description of the design adopted to satisfy these requirements follows.

The elevation of the top of the pressure vessel was determined by the requirement that the water level in the pressure vessel be the same as the normal water level in the condenser hotwell. This fixed the top of the pressure vessel closure flange at 24 in. below the main floor level. This depression makes consideration of the possible consequences of dropping heavy objects such as the pressure vessel cover from considerable heights necessary. The following is a list of heavy objects and the maximum distances they might fall before striking the pressure vessel as used in design calculations on the reactor support structure.

<u>Component</u>	<u>Weight (tons)</u>	<u>Maximum Fall (in.)</u>
Fuel coffin	15.0	18
Pressure vessel cover	8.5	30
Top shield indexing plugs	6.0	42
Carrier for indexing plugs	10.0	30

A network of structural steel framing (Figure 53) was considered for static support of the pressure vessel and absorption of impact loadings. Calculations for impact loading conditions followed rules outlined by Roark.*

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*R. J. Roark, "Formulas for Stress and Strain" (2nd ed., McGraw-Hill Book Co., New York) Articles 84 and 85, pp. 318-321.

UPPER VESSEL STRUCTURAL AND SHIELDING ARRANGEMENT

FIG. 53

Results of these calculations clearly indicated it would be difficult to use ordinary structural steel (A-7) under impact conditions. The total stresses would exceed the yield strength for the beams considered and in some cases would exceed the ultimate strength. This is because the short sections of A-7 are very stiff when sized to handle the maximum static load. The use of a high-strength structural steel such as Carilloy T-1 (yield strength 90,000 psi, ultimate strength 105,000 psi) seemed advisable. It was found that the Carilloy T-1 beams would absorb the impact loading but with a low (1.25) factor of safety, based on the ultimate tensile strength. To increase the factor of safety and prevent permanent deformation of the reactor support structure under impact loading, C-shaped springs, rolled from 3/4-in. T-1 plate, were placed between the pressure vessel support lugs and the structural steel framing beams (See Fig. 7B). The C-springs would deform under heavy impact, but the design is such that they can be replaced.

The C-springs are mounted in such a way that they can roll radially to accommodate thermal expansion without allowing a lateral shift in the centerline of the pressure vessel.

In addition to better fulfillment of functional requirements, use of T-1 steel instead of A-7 reduced the cost of the structure. The greater strength allowed use of thinner plates, thereby reducing the welding and erection costs.

The reactor support structure could have been anchored in the upper part of the concrete shield if it were not for the requirement of explosion containment. This problem necessitated the use of heavy beams across the top of the reactor, whose anchorages are not destroyed by the explosion. Vertical columns extending downward through the shielding to the bottom of the building were adopted for anchors. It was convenient to attach the horizontal support beam structure to these vertical steel columns.

Reaction conditions between molten fuel and water are still not well defined. If they are finely dispersed, zirconium and uranium-zirconium-niobium alloys oxidize completely with considerable violence in water. Without dispersion, only surface oxidation occurs. Whether the state of subdivision of fuel following melting of EBWR fuel plates would promote an explosive reaction is not known. Since the heat generation rate in the center of the core is about 2.5 times as high as the average throughout the reactor, it seems highly unlikely that the entire core could melt and react with water. The center of the core would melt and react first, blowing the remainder of the core apart before it could melt. Assumption of a violent reaction between 25% of the core metal and water seems adequately pessimistic.

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*R. J. Roark, "Formulas for Stress and Strain" (2nd ed., McGraw-Hill Book Co., New York) Articles 84 and 85, pp. 318-321.

Reaction of 25% of the core metal with water would release 8×10^8 cal. The reaction was assumed (pessimistically) to be a high-order detonation, similar to that of TNT. The energy was assumed to be released in the 4 ft dia x 4 ft high core area. It is interesting to note that the volume of TNT required to release 8×10^8 cal is about the same as the reactor core volume. A preliminary evaluation report* listed the following conclusions:

1. The lower section of the pressure vessel (in contact with the water phase) will fail under reflected shock wave pressure of 40,700 atm at the sides and 13,800 atm at the bottom.
2. The top portion of the vessel (in contact with the vapor phase) will not fail, since the maximum shock wave pressure is 260 atm (3800 psi). (Although the static strength of the vessel is 3200 psi, the actual strength under impact loads of short duration is much greater because of a large increase in strength at high strain rates.)
3. The transmission of the shock wave horizontally through the lower portion of the concrete biological shield, and axially through the lower shield plug, will reduce these masses to rubble.
4. The durations of the applied forces are very low, i.e., 10 msec or less.

Free-field pressure calculations by Armour indicated that unprotected vertical steel columns could not withstand the outward shock wave pressures. Compressible materials to absorb some of the shock energy and reduce the column loading were considered. High-pressure crushing tests on various kinds of wood led to selection of redwood. Redwood can undergo a large (factor of ~2) reduction in volume while offering large and essentially constant (2000 psi) resistance.

A blast shield (Figure 3C) consisting of alternate layers of wood and steel was adopted to protect the steel columns. Each layer of the blast shield consists of three inches of wood and one inch of steel. The first three layers utilize redwood. Celotex was used in the outer two layers because the pressure at this point should be below the crushing pressure for redwood. The total thickness of the blast shield is 21 in. Its composition is such that its gamma-ray and neutron attenuation properties are comparable with the surrounding concrete shield.

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* "Design Evaluation of the BER in Regard to Internal Explosions, Phase Report No. 1" ARF-No. D090 (M073) (May 10, 1955), Preliminary Report by F. B. Porzel.

In the blast shield, shock energy is absorbed in the compressible material. The next steel sheet then converts the material velocity component of energy back into a shock wave which is absorbed in the following layer of wood. A reduction in pressure by a factor of about 2 occurs in each layer. At the back of the shock shields (beginning of steel columns) the pressure is expected to be about 500 psi.

Consideration of dynamic loading factors on the vertical steel columns indicated that conventional structural steel columns were inappropriate. The calculated section sizes for A-7 steel were extremely large and heavy. The moment of inertia should be as small as possible consistent with strength requirements. The high strength alloy Carilloy T-1 was therefore adopted. Even with T-1 steel, the required columns were so large that they had to be built up (welded) from plate material.

To develop the required anchorage force for the six vertical columns, the columns are welded to 16-ft long anchor beams buried deep in the concrete floor of the gas-tight building. Reinforcing rods are welded to the anchor beams so that failure of the concrete would be along planes 45 degrees from the vertical, beginning at the edges of the anchor beams. Failure strength of the concrete along these planes exceeds the strength of the six steel columns.

Because of the curvature of the bottom of the gas-tight shell, the outboard anchor beam could not be buried as deeply in concrete as the inboard anchors. Two cross beams, used as compression members, prevent movement of the outboard anchor beam from under the mass of concrete below the fuel storage well.

A final report by the Armour Research Foundation has been issued.*

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*F. B. Porzel, "Design Evaluation of BER (Boiling Experimental Reactor) in Regard to Internal Explosions," ANL-5651 (January 1957).