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**FINAL SAFETY ANALYSIS REPORT ON THE
ZERO POWER PLUTONIUM REACTOR (ZPPR) FACILITY**

**H. Lawroski, R. G. Palmer,
F. W. Thalgott, R. N. Curran,
and R. G. Matlock**

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ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS

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Applied Physics Division

June 1972

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ABSTRACT

The Zero Power Plutonium Reactor (ZPPR) is a split-table-type critical facility operated by Argonne National Laboratory at the National Reactor Testing Station (Idaho) in support of the AEC's Liquid Metal Fast Breeder Reactor (LMFBR) Program. The ZPPR is designed for the study of the physics of power breeder systems and is capable of simulating fast-reactor compositions characteristic of the proposed 300-500-MWe demonstration plants and 1000-MWe commercial plants. This simulation is accomplished either by using fuel, coolant, and structural materials in the form of small plates stacked into drawers, or by employing fuel rods inserted into calandria assemblies.

This report describes the facility, the management philosophy used to ensure safety, and the reactor operating procedures, and provides a safety evaluation of the facility and its systems.

As part of this safety evaluation, a maximum credible accident and a hypothetical, incredible accident were calculated to demonstrate the capability of the containment structure. They both led to the conclusion that operation of the ZPPR will not cause undue risk to the health and safety of employees or to the general public.

I. SUMMARY AND INTRODUCTION

A. Summary

The Zero Power Plutonium Reactor (ZPPR) is a split-table-type critical facility operated by Argonne National Laboratory at the National Reactor Testing Station (NRTS) in Idaho. The ZPPR is designed for the study of the physics characteristics of power breeder reactors in support of the AEC's Fast Breeder Reactor Program and can accommodate

composition mockups characteristic of both the proposed 300- to 500-MWe demonstration plants and the 1000-MWe commercial plants. Studies will range from demonstration sizes to central-station sizes and will permit the simulation of reactor fuels proposed for these plants, including oxide, carbide, and metal fuels with various fuel, coolant, and structural-material compositions and volumes.

The physics of the systems will be investigated from the standpoint of the improvement of calculational techniques and input data to correctly predict (1) the performance characteristics of power-reactor systems, e.g., critical size or enrichment, power distribution, control-rod worth, and blanket and shielding optimization; (2) the safety characteristics of power reactors, e.g., Doppler coefficient, coolant density coefficient, and worth of fuel expansion or motion; and (3) the correctness of critical assembly simulation of detailed power-reactor geometries, e.g., heterogeneities of fuels, diluents, transition areas or plena, and fuel-blanket and support designs.

The ZPPR is similar in concept to the ZPR-3, -6, and -9, already in operation at Argonne. This design affords the flexibility, consistent with safety, necessary for the study of future central-station, fast-breeder power-reactor concepts. The facility consists of a large, cast-steel bed on which are mounted a stationary and a movable cast-steel table, each 168 in. wide and 108 in. long. The bed and tables are designed to carry 240 tons with a safety factor greater than 6 on the roller bearings, and much larger structural safety factors. The tables are separated by 7 ft during loading operations. Rectangular stainless steel tubes (nominal 2 in. x 2 in. x 4 ft) are stacked horizontally on the tables to form a honeycomb matrix, which may be varied, by adding or subtracting matrix tubes, from an 8- to a 14-ft maximum dimension. The matrix tubes are supported on each side by massive cast semisteel "knees" and are held in position from above by a "tiedown" system of I-beams bolted to the knees.

Reactor compositions to be studied are achieved by stacking reactor materials (for example, Pu-²³⁸U alloy, ²³⁸U oxide, sodium carbonate, sodium, graphite, steel, etc.), where necessary, jacketed in stainless steel, in drawers, and loading the drawers into the matrix tubes. The materials are color-coded to permit easy checking of the loaded drawer, and the drawers are number-coded for proper placement in the matrix.

A total plutonium inventory in excess of 3000 kg is envisioned for the facility. Approximately 2900 kg (the basic fuel) are in the form of stainless steel-jacketed plates (1/4 in. thick by 2 in. high by various lengths) containing a ternary alloy of uranium, plutonium and molybdenum (69.3 wt % uranium, 28.2 wt % plutonium, 2.5 wt % molybdenum). This composition ensures a negative Doppler effect on heating. Special fuel forms--such as mixed-uranium-plutonium oxide plates and rods, and

plutonium and plutonium-uranium-alloy metal plates of differing plutonium enrichments and isotopic compositions--are available in much smaller quantities for special experiments.

The reactor is controlled by fueled control rods, fueled safety rods, poison shim rods, and poison safety rods. At least one fueled control rod, one shim rod, and four safety rods are required in each half.

Each half of the assembly may contain up to 10 fuel-control safety rods. Three drives on each half are designed for precision positioning and readout to be used as control rods. At least one rod of the three in each half must be designated as a control rod to satisfy the interlock circuit and permit closure of the halves. The other drives are designated as fuel safety-rod drives and must be fully inserted (in their most reactive positions) to permit closure of the halves.

Additional rods are used when necessary to provide the required shutdown margins. These rods contain boron carbide and are designated as "poison" rods. An initial complement of 12 poison rods and drives is available, six for each half, but provisions are incorporated for adding four more to each half.

Six poison rods, three per half, are dual-purpose rods and can be used as poison safety rods or shim (poison-control) rods. At least one of the poison rods in each half must be designated as a shim rod to satisfy the interlock circuit and permit closure of the halves.

These rods and drives, both fuel and poison, are essentially identical in design to those used, and proven, in ZPR-6 and -9.

Rod speeds are limited to ensure that reactivity cannot be added at a rate greater than $0.06 \text{ } \$/\text{sec}$. A sufficient number of rods are used to ensure that, for each core studied, a minimum shutdown of $1.5\% \Delta k/k$ is available. The maximum excess reactivity permitted in any loading, with all control/shim rods in their most reactive position, is $0.5\% \Delta k/k$.

Neutron sources ($\sim 10^7 \text{ n/sec}$ each) are provided on ZPPR, although, for most cases involving plutonium fuel, they do not add significantly to the neutron background deriving from spontaneous fission in the fuel. The startup circuit is interlocked to prevent inward motion of the tables unless a minimum neutron flux is detected in the core.

The operating neutron-monitoring system comprises two neutron counter channels with low-level and period trips, two log-n channels with period trips, two linear channels with power-level trips, and two linear channels with bucking circuits. All channels cause a scram on loss of high voltage.

The reactor is housed in a 50-ft-dia, ~24-ft-high circular room with walls and floor of reinforced concrete. The cell roof is a filter composed of washed and graded layers of gravel and sand with a minimum thickness of 16 ft, supported on a catenary cable network. The reactor cell roof is enclosed by a backup containment structure of corrugated steel siding to house 285 AEC-type high-efficiency filters, each rated at 1000 cfm at 1 in. of water-pressure drop. The steady-state lifting pressure for the gravel-sand filter is 12.5 psig.

All areas where fuel is normally stored or handled (i.e., the reactor cell, fuel storage vault, and workroom) are situated under an earth mound.

Access to the cell is controlled through double blast and seal doors, and to the vault and workroom area through seal doors. When the reactor is shut down, 1000 cfm of conditioned air is fed to the reactor cell. During operation, the ventilation system is sealed by separate fire and seal valves.

The operation of the facility conforms with existing Laboratory safety directives. In all cases, the principle of concurrence by at least two competent and qualified scientists on a loading change or other significant change is followed. The ZPPR Reactor Manager has the primary responsibility for safe operation of the facility. The line of responsibility in ascending order for the management of the facility is: ZPPR Reactor Manager, Fast Critical Experiments Section Head, ZPR Program Manager (Idaho), Deputy Director Applied Physics Division, Director Applied Physics Division, and the Laboratory Director.

Reactor-safety evaluations include the calculation of reactivity coefficients for various materials and the central-core-drawer, expansion-coefficient, "fat-man," and personnel-shield worths. The effect of the gap between halves, and of gravel in the gap between halves, has been calculated. The Doppler temperature coefficient has been calculated for a number of typical cores and is negative in all cases.

No cores will be built where the calculated prompt power coefficient is positive.

A careful review of operational procedures and equipment (i.e., instrumentation, control- and safety-rod performance, reactor bed and tables, personnel shields, fuel-can rupture, and reactor cooling system) has revealed no plausible route to a severe accident. Power failure or earthquake does not initiate a severe accident.

In all cases involving credible reactivity addition rates, even involving gross loading errors and management failure, the reactor control and safety system shuts the reactor down with no damage.

The maximum credible accident is considered to be a fire. Hence, although the design and operation of the facility are such as to minimize the possibility of a fire in the proximity of fuel, the presence of electrical equipment in the cell does present opportunity for initiating a fire. Even if pessimistic burning rates are assumed, no significant pressure builds up in the cell.

To further illustrate the capability of the containment structure, the consequence of a hypothetical startup accident has been calculated. This accident is termed the design basis accident (DBA). It is in no way deemed credible.

The DBA is arrived at by assuming that:

1. The reactor becomes critical with the table halves moving together at that speed and separation that give the maximum reactivity addition rate. For a 1000-liter core, this occurs at a separation of approximately 4-8 in. (i.e., at the intermediate speed), and the maximum reactivity addition rate is 45 cent/sec. For a 6000-liter core, a maximum reactivity addition rate of 20 cent/sec is achieved at separations greater than 18 in. (i.e., on high speed). Both occurrences require a complete breakdown of administrative control so gross as to allow overloadings of 30 and 100%, respectively, with no veto from responsible supervisors.

2. All safety devices fail to operate (i.e., control and safety rods do not scram, and the halves do not scram, i.e., separate). In view of the multiplicity of required instrumentation, spring-loaded safety rods, and out-drive motors, this is incredible.

The accident continues with no shutdown mechanism allowed, except self-disassembly of the machine. Two conditions have been analyzed: one in which sodium is loaded adjacent to fuel, and one in which empty cans are loaded adjacent to fuel. Heat transfer to materials adjacent to fuel is permitted.

In the first case, fuel melts but does not boil; however, sodium does boil. Assuming instantaneous combustion of sodium vapor, and assuming further that all the energy of combustion goes to heat the air in the cell, cell pressures are always below that which will disturb the filter.

In the second case, plutonium is vaporized and is assumed to burn instantaneously without heat loss except to air. Again, cell pressures are below that which will disturb the roof filter. Quantities of plutonium released as vapor (39.95 kg) result in total body burdens below the acceptable burden, even at 55 m (distance to the control wing) under strong inversion and weak mixing conditions.

Gamma and beta whole-body doses have been calculated for the maximum DBA, assuming 100% release of fission gases from the fuel. Even at 55 m, under the assumption of complete exposure to the cloud, the maximum gamma dose under strong-inversion, weak-mixing conditions is 186 rad. Since the operators are in the Support Building and have a thick concrete corridor available as a shield, this dose is not considered excessive.

At other locations on the EBR-II site, the doses are much lower, and at the site boundary are essentially negligible.

Operating approval for the ZPPR was granted on the basis that the cell gravel-sand roof and the HEPA filters in the backup containment structure have a combined attenuation of 5×10^4 for plutonium aerosol. An attenuation factor of 100 was allowed for the backup containment structure. For the cell gravel-sand roof, an assumed attenuation factor of 500 was allowed on the basis of worst-point laboratory data. (Tests have indicated a minimum combined attenuation of 5×10^4 , based on worst-point data for both the plutonium and uranium filter tests. A combined attenuation of 5×10^5 is indicated from worst-point data for the plutonium tests only.) In experiments in which large amounts of ^{238}Pu or ^{241}Pu are used, a revised minimum acceptable attenuation will be determined using the calculation technique given in the DBA and will be such that the inhalation dose at the NRTS boundary will not exceed the nonoccupational maximum permissible body burden. The minimum acceptable attenuation will be lower than that demonstrated for the ZPPR containment structure.

The safety analyses for the ZPPR led to the conclusion that operation of the ZPPR at the NRTS will not cause undue risk to the health and safety of employees or to the general public, with the operation subject to the restriction that no assembly will be constructed for which:

1. The calculated prompt power coefficient is positive.
2. An analysis of the DBA, under the assumptions and with the calculational model used in this report, yields a cell pressure in excess of 12.5 psig.

B. Introduction

The programmatic aim of the AEC's Fast Breeder Reactor Program is to develop economic power reactors for central-station power. These plants will be plutonium-fueled with electrical outputs in the 1000-MWe range. There is no critical facility in the United States at this time capable of accommodating zero-power breeder-reactor experiments of this size fully fueled with plutonium. Argonne National Laboratory has designed such a facility for the AEC, and it has been constructed at the NRTS. It will be

used for the study of the physics of power breeder systems, initially, in the demonstration-reactor size (300-500 MWe), and subsequently, in the central-station size (1000 MWe).

Uncertainties in basic cross-section data, in inelastic scattering and absorption cross sections, and in calculational techniques require that critical assembly data for reactors of these sizes be available as checkpoints for theoretical predictions of fast-reactor processes. The ZPPR will be used to investigate the Doppler coefficient and the sodium-void coefficient of reactivity for these systems. It will permit evaluation of calculational techniques by mocking up reactors of specific composition and geometry for comparison with calculation, and in particular, will permit evaluation of the effectiveness of zoned reactor measurements by comparing the experimental results obtained in zones with those of full-size mockups.

The experimental determinations with each assembly may include: critical mass; control-rod calibration; neutron-flux and power distribution, by means of foils and counters; neutron-spectrum measurements and fission-ratio or threshold-detector techniques; reactivity coefficients of various materials; Doppler coefficient; temperature coefficient; effectiveness of control rods for large power systems; Rossi α or neutron lifetime; and transfer function.

The experiments will be restricted to assemblies fueled with ^{235}U or plutonium, primarily with an 11.5% ^{240}Pu content, although limited quantities of plutonium with other isotopic compositions will be used. The use of ^{233}U will be limited to quantities necessary for measurements of the reactivity coefficients or the Doppler coefficient. More extensive use of ^{233}U as a fuel will not be undertaken without additional analysis, review, and approval.

II. DESCRIPTION OF SITE

A. Location

The ZPPR Facility, approximately 300 m southeast of the Experimental Breeder Reactor II (EBR-II) Building, is in the East Area of the NRTS.¹ The East Area consists of the TREAT, EBR-II, and ZPPR sites and is located 3 miles north of U.S. Highway 20 and 11 miles west of the eastern NRTS boundary. The East Area is about 33 miles west of Idaho Falls, Idaho. A map of the NRTS and vicinity is shown in Fig. 1.

The ZPPR reactor cell, which is the prime concern of this safety analysis, is 150 m southeast of the EBR-II complex and 1000 m southeast of the TREAT Facility. A layout of the East Area is shown in Fig. 2.

The overall population within approximately 40 miles of the NRTS site boundary is about 70,000 and has not increased significantly within the last 15 years. The major increase has been in the Idaho Falls area. The areas in the vicinity of the NRTS are farming areas and are not expected to change appreciably in population. The present population distribution around the NRTS is given in Table I.

The facilities at the NRTS have a variable working population, depending upon the status of the program and the utilization of day or night shifts. A representative site attendance for December 1966 is given in Table II.

B. Hydrology

The NRTS is a sagebrush-covered desert area of volcanic origin.² The station is a relatively level plain with an average elevation of 5000 ft. The elevation at the ZPPR Facility area is 5126.5 ft, and the support-wing main-floor elevation is 5126 ft.

The water table is about 670 ft below grade, as evidenced by the drilling log of the EBR-II test well. Outcroppings of lava rock, which can be seen occasionally, extend below grade to the water table and are honey-combed with many multiple fractures. The natural underground reservoir under the NRTS originates primarily in the mountains to the north and travels from a northeast to southwest direction. The estimated flow of the underground reservoir is about 500 ft³/sec, resulting in an estimated flow of the main body of water through the lava of about 0.5 mile/year.

The exact flow path of surface water to the water table is unknown. In the ZPPR Facility area, the surface drainage is in a northwest direction. The rate of flow is dependent upon soil conditions. Drainage through areas containing clay would be around, rather than through, the surface soil. In porous lava rock, which is broken, and in gravel overburden, drainage would be rather rapid and diffuse.

Fig. 2. Argonne National Laboratory/East Area,
NRTS. ANL Neg. No. 103-11040.

TABLE I. Population Distribution around NRTS^a

Area	Population	Distance from ZPPR, miles	Direction from ZPPR
Atomic City	155	13	SW
Moore, Howe, and rural area	750	24	NW
Arco-Butte City	1,840	33	W
Mud Lake, Monte View, Terreton, and rural area	1,320	21	NNE
Roberts	456	27	NE
Basalt	300	29	ENE
Rigby	2,550	36	ENE
Idaho Falls area	50,000	33	E
Shelley	2,875	32	ESE
Blackfoot	8,120	34	SSE

^aReceived from Idaho Operations Office, January 1967.

TABLE II. NRTS Population Distribution^a

Area	Population	Distance from ZPPR, miles	Direction from ZPPR
Central Facilities	1068	15	WSW
SPERT	161	11	WSW
Army Reactor area	38	10	WSW
ANL-West	33	20	WSW
Test Reactor area	989	16	W
Chemical Processing Plant	289	14	W
Naval Reactor Facility	1872	14	WNW
Test Area North	563	18	N
ANL-East	383	<0.5	NW

^aReceived from Idaho Operations Office, January 1967.

The surface discharge of the aquifer beneath the NRTS occurs in springs along the Snake River Valley near Hagerman, Idaho. Considering locations and probable drainage, released contaminated water would reach the Snake River in approximately 240 years.³

C. Seismology

In the "Map of the United States Showing Zones of Approximately Equal Seismic Probability," the NRTS is in a region designated as Zone 2.⁴ It is located approximately 200 miles south of the Helena area (in Montana), which is a Zone 3 region.

The NRTS area is known to be traversed by a series of geologically recent lineaments and distinct linear surface features. These features are believed to be faults or fractures in the rock materials along which movement has taken place, i.e., one rock mass sliding past the other. There are several of these lineaments in the area of the ZPPR site. One of them is a proven fault located approximately 1 mile east of the ZPPR site and extending for several miles in a north-northeasterly direction from the east Twin Butte.⁵

Before the NRTS was established, J. Stewart Williams of Utah State University issued a report entitled Earthquake Risk at the Site Near Arco, Idaho. This report was written at the request of the firm of Smith, Hinchman and Gryth, Inc., 243 West Congress Street, Detroit, Michigan, and provided one of the bases for evaluating the present location of the NRTS. In this report, Williams states:

"Earthquake risk at this site is appreciable, but not great. Since isoseismal maps for principal earthquakes have been drawn, beginning in 1925, the isoseismals of only one earthquake reached Cerro Grande. Prior to this time several earthquakes recorded for surrounding areas may have been felt at Cerro Grande. There is no record of a major earthquake originating close to Cerro Grande.

"However, Cerro Grande is surrounded by areas of comparatively high seismic activity. Furthermore, it lies in a region of geologically young faults, any one of which must be considered potentially active. For these reasons earthquake risks at this site should not be dismissed from consideration in planning an industrial plant to be built at the site.

"Cerro Grande is situated within 150 miles of several areas of pronounced earthquake activity. Any one of these might produce a shock stronger than it has yet produced with a correspondingly greater intensity at Cerro Grande. The earthquake history of 100 years for this area is very short, from the geological point of view. An earthquake might occur any day that would alter substantially our ideas of the distribution of seismic activity in the area about the Snake River Plains."

In conclusion, Williams offered the following opinion:

"Earthquake risk in any area is relative to the types of structures to be built. Reinforced concrete buildings, well constructed in every way with high factors of safety and incorporating features recommended by engineers acquainted with earthquake-proof design, stand less risk of being damaged. Such buildings, set on the lava bedrock at Cerro Grande, certainly would be reasonably safe from earthquake damage."

Since the time the above report was written by Williams, an earthquake was felt at the NRTS on August 17, 1959. This earthquake was described as follows: August 17, 1959, 11:37 p.m., main shock of Hebgen Lake, Montana, series, epicenter 44°50' N, 111°05' W.⁶ This earthquake was perceptible over an area of approximately 600,000 square miles in the United States and Canada. Pasadena magnitude was 7.1. Maximum intensity in the epicentral region was X.⁷ In Idaho, intensity VII was reported from the Big Springs, Henry's Lake, and Island Park areas. Intensity VI was reported from Arco, Dubois, Howe, Rexburg, Rigby, Rupert, and St. Anthony. Intensity V was reported from Aberdeen, American Falls, Blackfoot, Burley, Carey, Idaho Falls, Richfield, Shelley, and Shoshone.

A survey of the reactor facilities at the NRTS indicated no visible damage to any reactors or buildings, although the earthquake was readily detected at the site.⁸

Fig. 3. Typical Wind Roses at EBR-II Area.
ANL Neg. No. 103-A11705.

D. Climatology

The NRTS is located in the Upper Snake River plain. The site is oriented in a southwest-northeast direction with the Continental Divide to the north and east. About 90% of the time, surface winds are either from the southwest or northeast. A comprehensive discussion of NRTS climatology is given by Yanskey *et al.*⁹

A monitoring tower at the ANL East Area of the NRTS has been used for the past two years to record wind conditions in this area. Within this short period of data accumulation, there is little, if any, difference in the wind patterns at ANL East Area compared to the rest of the NRTS. For information, wind roses of 20- and 240-ft elevation are shown in Fig. 3. These wind roses indicate that the most likely winds are from the southwest and the northeast. These considerations show that the ZPPR Facility is located in the most propitious direction with respect to EBR-II and TREAT.

III. DESCRIPTION OF THE ZPPR FACILITY

A. Introduction

The ZPPR Facility is divided into two areas: (a) the mound area and (b) the support wing. The mound area is where essentially all fissile materials are handled. The support wing contains the reactor control room, offices, and other necessary complements. A perspective of the facility is shown in Fig. 4.

Fig. 4. Perspective of ZPPR Facility. ANL Neg. No. 103-A11717.

B. Mound Area

The mound area consists of the reactor cell, fuel storage vault, workroom, and equipment rooms, as well as the access and escape tunnels. A floor plan of the ZPPR facility is shown in Fig. 5.

1. Reactor Cell

The reactor cell is a 50-ft-dia circular room with the floor and walls constructed of reinforced concrete. A blanket-material storage room is adjoined to the cell and is located directly behind the stationary half of the reactor. Normally, passage to the reactor cell from other portions of the mound area is through an access corridor. An escape tunnel is also provided for emergencies. Seal pits for reactor instrumentation, controls, and experiments are located in an alcove of the access corridor near the reactor cell.

A false ceiling to minimize dust and provide emergency-flow diversion of moisture in the event of a roof leak is located directly under

the reactor-cell roof. Two panels housing AEC-type high-efficiency filters are provided in the suspended ceiling to equalize the cell and atmosphere pressure during normal operation of the facility.

Fig. 5. Plot Plan of ZPPR Facility. ANL Neg. No. 103-A11723.

The reactor cell roof is composed of layers of washed-and-dried gravel and sand. A cross section of the reactor cell roof and backup containment structure is shown in Fig. 6. The gravel-sand roof is supported by a catenary cable network of 1.875-in. steel cables on 15-in. centers anchored in a 6-ft 11-in.-wide, 4-ft 6-in.-thick steel-reinforced concrete ring beam. The steel cables are held in place by cross-cable clamp assemblies. The bottom of the catenary is 23 ft 7 in. from the floor of the reactor cell. Four layers of 2 x 2-10/10 galvanized, welded-wire mesh are placed in a cross-hatch fashion on the 1.875-in. steel cables. A single layer of 1/2 x 1/2-10/10 galvanized, welded-wire mesh is placed on top of the coarser wire mesh. The first layer of roof fill is 2- to 1-in. gravel. A covering of 4 x 4 x 0.063-in. woven-wire cloth is then placed on the first layer of gravel. The next layer is a 6-in. depth of 2-in. to No. 10 gravel, which is then covered with a layer of 26 x 26 x 0.017-in. woven-wire cloth. The first layer of 20- to 50-mesh sand, 1 ft thick, is supported on this screen. A 4-ft layer of 2-in. to No. 10 gravel is located directly above the first layer of sand. This 4-ft layer of gravel is covered with another layer of 26 x 26 x 0.017-in. woven-wire cloth.

Fig. 6. ZPPR Cell-roof Cross Section. ANL Neg. No. 103-11064 Rev. 1.

Another layer of 20- to 50-mesh sand, 18 in. thick, is then placed on the 26 x 26 x 0.017-in. woven-wire cloth. The top layer of the gravel-sand roof is an 8-ft 6-in. bed of 2-in. to No. 10 gravel.

The reactor cell roof is enclosed by a backup containment structure, which is essentially a housing for 285 AEC-type high-efficiency filters, nominally rated 1000 cfm at 1-in. H₂O pressure. Corrugated steel siding confines the top portions of the gravel-sand roof and directs flow up to the AEC-type high-efficiency filters. The backup containment structure is designed to withstand 3-in. H₂O pressure. An enclosed peripheral catwalk with furnace filters, shielded by metal louvers, provides outside protection for the AEC-type high-efficiency filters. The roof of the backup structure prevents rain and snow from wetting the AEC-type high-efficiency filters and the gravel-sand roof. A gasketed door is provided for access to the interior of the backup containment structure.

The reactor cell, the access corridor, and the blanket storage room were coated with a layer of sealant compound before the earth fill was placed. This compound seals the concrete and prevents moisture from penetrating the walls. The earth fill encloses the cell, the access corridor, the blanket storage room, and the escape tunnel. The earth fill is covered with sealed 0.5-in. asphalt liner sheets to protect the mound from erosion and moisture.

The reactor cell floor has a 46-in.-deep pit to accommodate the reactor bed and tables. The bottom of the reactor pit is a 7-ft-thick, heavily steel-reinforced concrete slab. The sides of the pit are 3-ft-thick, heavily steel-reinforced concrete. The reactor bed is bolted to the pit floor, and the tops of the tables are flush with the cell floor.

The reactor cell, access corridor, and blanket storage room are supported on 54 steel pilings extending approximately 50 ft down into a support lava bed. The bottoms of the pilings are imbedded into a base of concrete at least 4 ft thick in each hole.

2. Fuel Storage Vault

The fuel storage vault is a 40 x 50-ft reinforced concrete room. The outside walls are 1-ft-thick, reinforced concrete, and the roof is composed of prestressed concrete beams covered by 7 in. of concrete. The wall between the fuel storage and the workroom is 9 in. of reinforced concrete. The floor of the vault is 18 in. of heavily reinforced concrete. An earth fill is compacted around the walls. The roof is covered by 4 ft of washed-and-dried sand fill, maximum size 0.375 in., with at least 45 wt % passing through a No. 20 U.S. standard, sieve-size screen. The sand and the earth fill are protected from the weather by an asphalt membrane.

The single entrance to the vault from the workroom is through a vault door and a 3-hr-rated fire-daygate door. Two ventilation penetrations are provided: one for the inlet air duct and one for the exhaust.

The heat load in the vault is removed by recirculating air through a direct-expansion-type air conditioner.

Fuel in the vault is stored in airtight canisters reposing in concrete blocks containing spaced holes compatible with established ANL criticality criteria. Materials in the vault are carefully controlled to minimize hazards.

3. Workroom

The workroom is used primarily to inspect fuel, to load and unload matrix drawers, and to check drawers. The walls of the workroom are 1-ft-thick, reinforced concrete, and the roof is prestressed concrete beams with an overlay of 7 in. of concrete. An earthfill is compacted around the walls, and the roof is covered with 4 ft of washed-and-dried sand, maximum size 0.375 in., with at least 45 wt % passing through a No. 20 U.S. standard, sieve-size screen. The floor of the workroom is 18 in. of reinforced concrete.

The workroom has access through seal doors to the outside of the mound area, the support wing, and the reactor cell. These seal doors are protected by fire doors. Normal fire doors are provided to the vault and to the inside equipment room. In addition, a vault door is provided to the fuel storage vault.

Two inspection hoods are positioned near the entrance to the vault. A four-station loading table is centrally located for convenience. These hoods and tables are treated as suspect areas and are integrated into the AEC-type high-efficiency filter air-handling system of the mound area.

A transfer tube from the workroom to the reactor cell provides a convenient method for transporting loaded drawers between these two areas. This tube has an electrically driven conveyor, which is interlocked to preclude improper usage. Seal covers for each end are provided for controlling airflow in the tube during reactor loading and unloading operations and during operation of the reactor.

4. Equipment Rooms

a. Inside Equipment Rooms. The inside equipment room contains the AEC-type high-efficiency exhaust filters and exhaust fans for the entire mound area. The main electrical-power control panel for the mound area is also located in the inside equipment room. The construction of the equipment-room floor, walls, and roof is the same as that of the workroom.

b. Outside Equipment Room. The outside equipment room houses the external air-conditioning equipment for the mound area. This includes air filters, supply fans, steam-air heaters, and air-conditioning compressors and condensers. The inlet air-handling system is connected to the mound area by a 24-in. steel duct. The outside equipment room is a concrete block building with a built-up metal decking roof.

5. Ventilation

All supply air for the mound area is filtered and conditioned before entering the facility. The exhaust air is filtered through AEC-type high-efficiency filters before leaving the mound area and is transmitted to the atmosphere through a 75-ft-high stack. An emergency sand filter, backed up by an AEC-type high-efficiency filter, can be used to control the release of contamination from the workroom area.

Total flow of air to the mound area is 4800 cfm. The airflow patterns are balanced to provide adequate air supplies to the various work areas. The mound area is also adjusted to operate at atmospheric pressure. The diagram of the airflow pattern in the reactor cell is shown in Fig. 7. The flow pattern in the workroom/vault/inside-equipment-room area is shown in Fig. 8.

a. Isolation. The main air ducts into and out of the mound area are equipped with separate fire and seal valves to effectively stop airflow when necessary. The emergency air-venting system from the workroom to the outside contains a normally closed vent valve. The three personnel and/or equipment access passages from the mound area are equipped with seal doors. Specifically, these seal doors are located in the connecting tunnel to the control room, in the escape tunnel from the workroom to the outside, and in the escape tunnel from the reactor cell to the outside.

All penetrations of electrical power, instrumentation, refrigerant lines, and air ducts and controls are through seal connections. These connections are at all bulkheads by the seal doors and in a seal pit in the access corridor to the reactor cell.

To isolate the reactor cell from the workroom, the air ducts between these two areas have separate fire and seal valves. Also, two sets of blast doors and seal doors are incorporated into the access tunnel to the reactor cell. The loading tube has seal covers to complete the isolation of the cell. No attempt is made to hinder airflow through the gravel-sand roof of the cell.

b. Flow Patterns. The 4800 cfm of filtered and conditioned air is separated into two main streams: (1) 1000 cfm goes to the reactor cell, and (2) 3800 cfm is distributed to the workroom/fuel-storage-vault/inside-equipment-room area. The reactor cell, the vault, and the inside equipment

Fig. 7. Airflow Patterns in Reactor Cell Area. ANL Neg. No. 103-A11826.

Fig. 8. Airflow Patterns in Workroom/Vault/Inside-equipment-room Area. ANL Neg. No. 103-A11827.

room have separate recirculating air-conditioning units to remove heat from these areas. The workroom airflow is a one-pass system, exhausting through the inspection hoods and the work table.

The reactor-cell ventilation system has two large, direct-expansion refrigeration coils, either of which can satisfactorily cool a normal reactor loading. The airflow for reactor cooling is adjustable to accommodate experimental core sizes and compositions. Air from the reactor core is filtered by AEC-type high-efficiency filters before being recirculated in the reactor cell.

While the reactor is shut down, 1000 cfm of conditioned air is fed to the reactor cell. Concurrently, 1000 cfm is exhausted through the filtering system. While the reactor is operating, the isolation valves in both air ducts are closed and both sets of seal doors are kept closed. The loading tube is also sealed during reactor operations. Therefore, during reactor operations any exchange of air in the reactor cell must pass through the gravel-sand roof and the backup containment filters.

c. Emergency Flow Patterns. Any time high-alpha activity or a fire is detected in the cell, the supply- and exhaust-air-duct valves to the cell are automatically closed. During normal functioning of the facility, i.e., when fuel is present in the reactor cell, one of the two seal doors is kept closed at all times. This same philosophy is applied to the transfer tube between the cell and the workroom. This would therefore direct any venting of the cell through the gravel-sand roof and the backup containment structure.

The exhaust air from the mound area is monitored for gaseous and particulate contamination. Detection of particulate contamination, on the downstream side of the exhaust filters, automatically closes both outlet and inlet air ducts to the mound area and automatically opens the valve in the emergency sand filter for the workroom. The main exhaust isolation valves for the mound area are interlocked with the main air-supply valves to prevent pressurizing the mound area by the main air-supply fan.

An indication of a fire alarm in the workroom/inside-equipment-room/fuel-storage-vault area activates the controls to automatically close the main exhaust isolation valves for the mound area and to open the emergency filter exhaust valve.

The emergency sand filter is a combination filter composed of a 5 x 5-ft, 26-in. layer of 20-50-mesh washed-and-dried sand, backed up with a nominally rated 1000 cfm at 1-in. H_2O ΔP AEC-type high-efficiency filter. The sand layer is supported on cross-hatched wire mesh. The filter structure is fabricated from welded 0.3675- and 0.25-in. steel plates backed up by angle irons for additional strength. The vent line to the emergency filter from the mound area and to the exhaust stack is 10-in. Schedule-40 welded steel pipe.

C. Support Wing

The support wing is a conventional one-story building with a fan loft and a fully-excavated basement and has a structural steel frame with non-load-bearing masonry walls. The support wing houses the main floor, service floor, fan loft, and the Argonne Fast Source Reactor (AFSR) room. The support wing is steam heated in the winter and is cooled in the summer with direct-expansion-type air-conditioning.

1. Main Floor

The main floor consists of the reactor control room, offices, an electronics shop, and a core-coating room. The main floor of the support wing is connected to the mound area by an access tunnel between the control room and the workroom.

The reactor control room contains the control console with necessary control switches and instrument readouts. A working office is located adjacent to the control room.

The core-coating room, located adjacent to the control room and at the entrance to the mound area, is used to clean reactor core simulants such as ^{235}U , depleted uranium, and stainless steel. The room contains two hoods for handling suspect materials and a fuel-coating machine, which is used primarily to coat and dry depleted uranium with a protective film. The coating is generally a fluorocarbon dissolved in a solvent. Any drainage from the core-coating room is collected in suspect retention tanks in the service floor.

The remainder of the main floor of the support wing is occupied by offices, an electronics shop, a change room, and an elevator, which also serves the service floor and fan loft.

The control room, the work office, and the core-coating room are treated as suspect areas, and the air from these rooms is exhausted through AEC-type high-efficiency filters located in the fan loft.

2. Service Floor

The service floor of the ZPPR facility is the basement of the support wing and consists of a computer room, a cable-routing room, a counting room, a laboratory-shop area, an electric switchgear area with an emergency diesel-powered electric generator, suspect-waste retention tanks, and air compressors.

The cable-routing room is directly under the reactor control room and has connecting openings for instrument and control leads. The

cable-routing room also houses a constant-power motor generator for reactor instrumentation. Since there are air passages between the cable-routing room and the control room, which is considered a suspect area, the exhaust air from the cable-routing room is also vented through an AEC-type high-efficiency filter in the fan loft.

The laboratory area has a hood for minor chemical analyses pertaining primarily to the counting room. The exit air from this hood and the counting room is exhausted through AEC-type high-efficiency filters.

3. Fan Loft

The fan loft is located above the center portion of the main floor of the support wing. The fan loft houses the air-handling equipment for the support wing. This includes the AEC-type high-efficiency filters for the suspect areas, the exhaust fans, the air-conditioning compressors, and the air heaters. Since all the air-conditioning condensers are air-cooled, the fan units with the heat exchangers of these condensers are located outside the fan loft on the roof of the support wing.

4. Argonne Fast Source Reactor Room

A room on the north side of the support wing houses the AFSR. The reactor core, blanket, cooling system, and instrumentation and control systems are from the original reactor and are covered in Ref. 10.

IV. DESCRIPTION OF THE REACTOR

A. General

The ZPPR is a split-half critical machine used to simulate fast-reactor cores. This type of critical machine has been shown to be a versatile experimental apparatus at Argonne National Laboratory by the performance of ZPR-3, which has been in operation since 1955, and ZPR-6 and -9, which started operation in 1963. Other organizations such as LASL, KAPL, ORNL, Pratt and Whitney, and Atomics International also have used the split-half concept for critical assemblies.

The basic element of the ZPPR is a bed and tables system, which holds the matrix assembly. The two tables, one movable and one stationary, are supported on a cast semisteel bed. One half of the matrix assembly, consisting of rectangular tubes firmly held in place by knees and a tiedown structure, rests on each table. The control- and safety-rod drives are mounted near the rear of each table and are connected to the control and safety drawers, or blades, by extension shafts through the plenum. Matrix drawers hold the simulation materials for the various reactor compositions and geometries.

The split-half machine was chosen primarily for safety and convenience in loading and unloading operations. Separating the reactor into two sections effects a large decrease in reactivity. Retractable shields are placed before the face of each half to protect personnel during reactor shutdown periods. A retractable loading platform is used as a support for loading and unloading operations.

A cooling system for the reactor draws cell air through the matrix tubes, filters the exhaust air, and cools the air before recirculation in the reactor cell.

An artist's rendition of the reactor is shown in Fig. 9. Comprehensive scale drawings of the stationary and movable halves are shown in Figs. 10 and 11.

B. Reactor Bed and Tables Assembly

1. Bed and Tables

The bed and tables assembly is shown in Fig. 12. This photograph was taken during testing and checkout at the manufacturer's plant. This assembly is designed to carry 240 tons with a static safety factor of 6.3 on the roller bearing. The structural safety factors for the bed and tables are those normally used in the machine-tool industry (700-1000).

Fig. 9

Artist's Rendition of ZPPR.
ANL Neg. No. 103-11248.

Fig. 10. Stationary Half of Reactor Assembly. ANL Neg. No. 103-A11715.

Fig. 11. Movable Half of Reactor Assembly. ANL Neg. No. 103-A11716.

Fig. 12. Assembled Bed and Tables during Manufacturer's Checkout. ANL Neg. No. 103-A11719.

The bed is a 130-in.-wide, 304-in.-long, 24-in.-thick, heavily ribbed Meehanite box-type structure. The bed is made in two pieces, matched, doweled, and bolted together. One piece, 108 in. long, is directly under the stationary table. The other piece, 196 in. long, supports the movable table. The bed contains five longitudinal raceways approximately equally spaced with the widest ways 120 in. apart. Each raceway contains removable, flame-hardened, accurately ground strips to provide a polished rolling surface for the movable table bearings. The section of the bed under the stationary table does not contain the hardened ground strips since no roller bearings are necessary.

The central raceway of the bed is accurately ground on each side to provide an alignment way for the movable table.

The bed is fastened to the reactor pit floor by 41 holddown bolts. There are 65 adjustable leveling jacks to accurately level the bed. Final adjustment of these leveling jacks is made when the bed and tables assembly is loaded with 180 tons before the reactor matrix is assembled.

Each table is a 108-in.-long, 168-in.-wide, 22-in.-thick, heavily ribbed Meehanite structure. Tee slots have been machined into each table, to be used in holding the matrix structures and rod-drive mounting assemblies. The stationary table is keyed and bolted to the bed to prevent any motion.

The movable table travels on 35 recirculating, antifriction, roller-bearing assemblies. Since the loads on the tables are primarily in the forward portions of each table, the bearings on the movable table are concentrated under the heavily loaded area. The locations of the bearings are shown in Fig. 13. A periodically pulsed oil pump supplies lubricating oil to the bearing surfaces. The movable table has two wiping assemblies for each bearing raceway to remove foreign materials and keep the bearing surfaces clean.

The movable table has tapered gibs on each end, mating with the alignment way of the bed, to ensure accurate alignment of the two tables as the reactor halves are brought together.

With the bed and table assembly in the reactor pit, the tops of the tables are at the same elevation as the reactor cell floor.

2. Table Drive System

The drive system for the movable table is a set of geared motors driving through clutches and gears to a precision-ground ball screw. A preloaded, nonrevolving, antifriction, ball-bearing nut is keyed and bolted

to the movable table at the rear end. Three motors (fast, medium, and slow speeds) are used to effect closure of the reactor halves. One unidirectional electric motor is used to separate the halves during a scram, and an emergency unidirectional scram air motor is also provided. The maximum separation of the tables with the drive system is nominally 84 in.

Fig. 13. Underside of Reactor Movable Table. ANL Neg. No. 103-A11825.

The ball screw is a 4-in.-dia, 0.75-in.-pitch, precision-ground, hardened steel shaft. A support bearing holds the ball screw in the transmission at the end of the bed. A thrust bearing anchors and supports the other end of the ball screw and is readily accessible when the movable table is fully retracted. The location of this thrust bearing can be seen in Fig. 11.

The table closure system drives through a specially designed unidirectional, jaw-type, adjustable slip clutch to prevent applying excessive forces during closure of the reactor halves. Slippage of the jaw clutch activates a signal switch as it rides up on the gear teeth. However, when this clutch is driven in the reverse direction (table separation), it operates as a positive nonslip gear.

The table closure system has three drive-in motors: fast, medium, and slow speeds. The fast- and medium-speed motors are powered through a hot-rail system. The slow-speed motor is wired directly to its power relay. The hot-rail system is composed of four track assemblies. The tracks are mounted on the bed, and the electrical pickoffs are mounted on the movable table. The hot-rail system mounted on the bed and tables is shown in Fig. 14.

Fig. 14. Hot-rail System for Movable-table Motor Drives. ANL Neg. No. 103-A11824.

Two of the track assemblies are wired to the power relays of the fast and medium motors in the control panel. The other two track assemblies are wired to the motors. The power pickoffs transmit the electrical power between the respective track assemblies.

Each track assembly contains three bus bars for the three-phase power and three corresponding sliding brush contacts. A cutaway of a track assembly is shown in Fig. 15.

Fig. 15. Electrical Pickup on Hot-rail System. ANL Neg. No. 103-A11718.

The fast-speed gear motor is coupled with an electric clutch to the drive transmission. The setting on the electromagnetic clutch is adjusted

above the torque required to cause slippage of the jaw slip clutch on the ball screw. This electric clutch is also set to slip when opposed by the electric-scam motor system. The fast-speed closure motor is only operable when the table separation is between 18 and 84 in.

The fast-speed closure motor power is interrupted at a table separation of 36 in. At this separation, all programmed safety rods must be cocked before proceeding with table closure. The fast-speed motor is stopped at a nominal separation of 18 in. by two limit switches in series before running off the hot-rail system.

The medium-speed closure motor is functional during table-separation distances between 18 and 3 in. Limit switches automatically deactivate the power to the medium-speed motor at a nominal separation of 3 in. before running off the hot-rail system. An electromagnetic clutch couples the motor to the transmission and is adjusted similarly to the fast-speed motor clutch.

The electrical contact, between the track wired to the fast-speed motor and its corresponding pickoff, ends at 17.4-in. separation of the reactor halves. Beyond this point, the pickoff runs off into air. For the medium-speed motor, the electrical continuity terminates at 2.5-in. separation of the reactor halves.

The slow-speed motor is used to bring the reactor halves together from the 3-in. separation. The functioning of the slow-speed motor electromagnetic clutch is the same as the other drive-in motors. Since the slow-speed motor has power for the entire distance of travel, it can be used to accurately position the reactor halves, regardless of the location.

The power to the motors in the table closure system is interlocked so that only one motor is operative at any given time. The power to the drive-in motors and the power to the drive-in electromagnetic clutches are automatically removed in the event of a scram. The scram motor automatically retracts the movable table upon receiving a scram signal. This scram motor is coupled to the ball screw through an electromagnetic clutch and gearing. Both the scram motor and clutch are capable of overriding any one of the drive-in motors' electromagnetic clutches. If the ball screw does not rotate in the scram direction within 1 sec, a time-delay switch automatically opens a solenoid valve in the emergency air-scam system. An air-activated clutch couples the unidirectional air motor through gears directly to the ball screw, and the air motor separates the two reactor halves. The emergency air-scam system will override any of the other drive-motor systems and will function automatically, regardless of the position of the movable table. An interlock on the supply air to the air-scam motor prevents startup or scrams the reactor if the air supply drops

below a preset level. This level is set so that the remaining supply air is sufficient to separate the halves with the air motor operating at its rated horsepower.

The separation action of the electric scram motor is terminated by actuation of an out-limit switch, which interrupts power to the scram motor. The emergency-air motor is stopped by a cam-operated air valve at the fully separated position of the movable table.

A continuous-readout system, consisting of a precision rack on the table and a pinion gear coupled to a selsyn transmitter on the bed along with an amplifier and a digital position indicator, determines the movable table position with an accuracy of 0.025 in. The control switches and read-out for the operation of the movable table are located in the control room.

A summary of the motors and their operations is as follows:

- a. Fast speed: Three-phase, 440-V, 0.75-hp motor operates at separation between 84 and 18 in.; table speed, 18.6 in./min.
- b. Medium speed: Three-phase, 440-V, 0.5-hp motor operates at separations between 18 and 3 in.; table speed, 6.0 in./min.
- c. Slow speed: Three-phase, 440-V, 0.25-hp motor operates at separations between 84 and 0 in.; table speed, 0.17 in./min.
- d. Scram: Unidirectional, three-phase, 440-V, 2-hp motor operates at separations between 84 and 0 in.; table speed, 30.0 in./min.
- e. Emergency scram: Unidirectional, 90-psi-nominal, 1.5-hp air motor operates at separations between 84 and 0 in.; table speed, 30.0 in./min.

C. Matrix Assembly

The matrix assembly consists of an array of horizontal, rectangular, Type 304 stainless steel tubes held in place by knees and a tiedown system. The matrix drawers are placed inside the tubes. More details of the assembly can be seen in Figs. 10 and 11.

Each tube has internal dimensions of 2.1 x 2.2 in. and is 48 in. long. The tube-wall thickness is 0.039 in. Rectangular notches are cut into the front vertical edges of each tube and are used to ensure accurate positioning of core drawers, which have matching tabs. The drawers are fabricated from perforated, 0.030-in.-thick Type 304 stainless steel. The nominal internal dimensions of drawers in the initial complement are 2 x 2 x 23 in. The front face of each drawer is Type 410 stainless steel, which is magnetic

so that it can be remotely removed by an electromagnet on an extension shaft. The initial complement of rear drawers has the same dimensions as the front drawer, but has no index tabs and has a curved extension of the backplate to allow removal from the rear of the tubes. Special narrow drawers are provided for matrix tubes, which also accommodate poison-safety-blade assemblies.

The simulation of reactor cores and blankets requires materials of many different shapes and sizes. Each material has a distinct color code or shape for each identification. Normally, reactor simulants are plate-type elements varying in thickness from 0.0625 to 2 in. and in length from 0.5 to 8 in. Special rod-type simulants and traverse materials have appropriate dimensions consistent with their intended functions.

The uranium plates are coated with a nominal 0.0005-in.-thick pigmented fluorocarbon to reduce oxidation, to decrease spread of contamination, and to identify fully enriched, natural, and depleted pieces. Other simulants are coated or glazed when necessary to protect personnel. Still others, such as sodium and sodium carbonate, are clad in stainless steel because of pyrophoric or hygroscopic characteristics.

The reactor-core simulants are usually placed longitudinally in the drawer, forming columns of similar plates. A special, segmented leaf spring is inserted in the rear of each drawer to hold the core materials firmly toward the front.

A photograph of a matrix tube, drawers, and core and blanket simulants is shown in Fig. 16.

Fig. 16. ZPPR Matrix Components. ANL Neg. No. 103-11524 Rev. 1.

The knees of the reactor are 121-in.-high, 46-in.-wide cast semi-steel. The insides are machined surfaces to prevent distortion of the tubes, and the outsides are heavily ribbed to provide lateral support for the matrix. The bottoms of the knees are held in place by bolting to the tee slots of the table; the tops of the knees are held together by the matrix tiedown beams. Experimental holes are provided in each knee for radial access to the simulated cores.

The matrix tie-down system for each half has six 2-in. I-beams bolted to the knees. An adjustable steel fixture for each beam is bolted to the knees and is used to apply the holddown forces. A 0.375-in. steel plate under the tiedown beams is used to distribute the holddown forces. The layout of the matrix tiedown system is shown on Figs. 10 and 11.

D. Fuel-rod Drives

Two types of fuel-rod drives are available on each half of the reactor assembly. The drives are designated as fuel control rods and fuel safety rods, and the drawers coupled to each drive are loaded similarly to regular reactor core drawers which contain fuel. The functional difference between a fuel control rod and a fuel safety rod is that, during reactor startup, the fuel control rods are fully withdrawn from the core until the reactor halves are brought together and the fuel safety rods are fully inserted to their most reactive positions before the reactor halves can be brought closer together than 36 in.

There are three fuel control-type drives on each reactor half. At least one of these three fuel-rod drives per half must be programmed as a control rod to be used in effecting the criticality of the assembly. The control-rod drives can also be programmed as fuel safety rods. The drives used as control rods have precision ball screws and selsyn-driven readouts, which permit accurate positioning.

The regular fuel-safety-rod drives have off-the-shelf ball screws with potentiometer readout systems having a normal accuracy of ± 0.125 in. However, to obtain a reproducible safety-rod position in the fully inserted condition, a precision linear transducer is used to measure the final 1 in. of travel of the rod.

A picture of a fuel-rod drive is shown in Fig. 17. The mechanical motions of these rods are the same as those used on ZPR-6 and -9,¹¹ i.e., a magnetic latch, a spring-driven scram action, and a ball-screw drive mechanism.

The full travel of the fuel drives is 24 in., and the maximum speed of a drive is a nominal 12 in./min. Since in larger cores the worth of the fuel rods may be less, several fuel safety rods may be moved simultaneously

Fig. 17. Fuel-rod Drive. ANL Neg. No. 103-A11822.

by ganging. However, at no time will the rate of reactivity addition be greater than 0.06 $\$/\text{sec}$. The fuel control-rod-type drives have an adjustable speed (maximum 12 in./min) to permit flexibility in making experimental measurements, but are wired to exclude ganging.

The holding latches of the drives are dc magnets operating at about 18 V. Tests on these magnets show conclusively that 130 V ac superimposed on the magnets does not impede the scram capabilities of the holding magnet. Loss of latch power will automatically scram the rods.

Each rod drive is mounted on a cast semisteel rod-drive mounting plate, firmly anchored to the reactor table and knees. The rear of each rod is also supported by a frame to permit alignment of the rod drives. This eliminates cantilevering of the rod-drive assemblies. A ball joint is interposed between the drive and the drawer to compensate for inadvertent misalignment of each drive and the reactor matrix.

Each rod drive has a continuous position indication as the rods are inserted into the reactor matrix. Since position indication is coupled to the drive mechanism, the intermediate position of the fuel drawer is not known during the scrambling action. Each rod drive has microswitches for indication of the rod in the "fully-out" and "fully-in" conditions.

The scram action of the as-built fuel-rod drives was timed using an unloaded drawer. A typical time-distance relationship of an empty rod drive is shown in Fig. 18. The delay time in rod movement is caused by the switching time, relay action, and residual holding of the magnet. This curve is typical for an empty control drawer (5.5 kg). With a 15-kg load, the total scram time is increased by about 0.045 sec.

Fig. 18. Typical Scram Performance Curve of Fuel-rod Drives. ANL Neg. No. 103-A11700.

The control and safety fuel drawers are fabricated from thicker materials (0.050 in.) to provide structural strength for the repeated movements of the rods. A fuel-movement eliminator covers each loaded fuel control or safety drawer and prevents vertical motion of the fuel in the drawer during scram action. The drawers are Scottsonized* to increase wear resistance.

The installed wiring for ZPPR will allow a total of 20 fuel-rod drives for experiments on large cores.

*A surface-hardening treatment.

E. Poison-rod Drives

Two types of poison-rod drives are available on each half of the reactor assembly. The drives are designated as shim (poison control) rods and poison safety rods. These drives have ^{10}B in the form of B_4C as the neutron absorber. The functional difference between a poison safety rod and a shim rod is that, during reactor startup, the poison safety rods are fully withdrawn from the core (in their most reactive position) before the reactor halves can be brought closer than 36 in. The shim rods are fully inserted into the reactor (their least reactive position) until the reactor halves are brought together.

There are three shim-rod drives on each half of the reactor. At least one of these three drives per half must be programmed as a shim rod to be used in establishing criticality of the assembly. The shim-rod drives can also be programmed as poison safety rods.

Position indication for the dual-purpose poison rods is by digital shaft encoder with a visual readout on the control console.

The poison-safety-rod drives insert an absorber blade into the reactor to help provide the mandatory 1.5% reactivity shutdown without requiring an inordinate number of fuel safety rods, since a poison safety rod has a larger reactivity worth than a normally loaded fuel safety rod.

A photograph of a poison-rod drive is shown in Fig. 19. The poison section, not shown, is a column of 14 plates set edge-to-edge, giving a nominal 28-in. active length. Each plate, nominally $2 \times 2 \times 0.25$ in., is formed by hot-pressing and sintering highly-enriched boron carbide. A single plate contains 25 g of ^{10}B , yielding 350 g of ^{10}B per blade. The boron carbide plates are fitted into a stainless steel container and held in place with a binder. The container, or blade, is attached to the drive and slides in a special guide tube, which penetrates the reactor core adjacent to specific narrow core and blanket drawers. The blades and sheaths are Scottsonized or chrome-plated to provide better wear and friction characteristics.

The travel distance of each poison-rod drive is 30 in. The poison-rod drive unit is similar to the fuel-rod drives in that ball screws are used to latch and position the rod and a dc magnet holds the rod against a spring force. However, the scram direction of the poison safety rods is toward the core. The drive is a dc motor, which moves the rod at 26 in./min. A geared, linear potentiometer indicates the position of each movement, and a limit switch indicates final inserted position after a scram. Linear transducers for the last inch of travel determine the position of the poison safety rods when the rods are fully withdrawn from the core.

POISON SAFETY ROD IS SHOWN IN COCKED POSITION

Fig. 19. Poison-rod Drive. ANL Neg. No. 103-A11823.

In the simulation of large cores in which a large number of safety rods are used, the simultaneous movement of rods during the cocking operation is provided by ganging. The capability of ganging rods is controlled in a locked control cabinet managed by reactor supervisory personnel and cannot be changed during reactor operation.

The installed wiring for ZPPR will allow a total of 20 poison-rod drives for experiments on large cores. If necessary, additional rod drives can be installed.

F. Source Assemblies

The use of external sources in reactor cores containing sizable quantities of plutonium fuel is superfluous due to the high spontaneous neutron background provided primarily by the ^{240}Pu . In the operation of ZPR-3, 390 kg of plutonium fuel with approximately 7 wt % ^{240}Pu provides a spontaneous source of 2×10^7 neutrons per sec. The quantities of plutonium to be used in ZPPR will be much greater and will mask out the normally provided external sources. The sources described below have utility only during initial stages of a critical experiment for instrument checkout.

The source assemblies provided in ZPPR are similar to those used on ZPR-6 and -9. Two americium-beryllium sources, each producing about 10^7 neutrons/sec, are provided, one for each reactor half. Each source is mounted on a flexible shaft and is driven from its coffin, mounted on the side of the assembly, through a guide tube into the outer portion of the core. Control switches and position indication are provided at the reactor control console.

When not in use, the sources are kept in their coffins to protect personnel from the radiation hazard.

G. Personnel Shields

The ZPPR is equipped with retractable personnel shields to cover the exposed reactor core during shutdown. Similar shields have been tested on the SEFOR (Southwest Experimental Fast Oxide Reactor) critical assembly in ZPR-3. The assembly of large quantities of plutonium results in a hazardous fast-neutron background between the reactor-core interfaces when the halves are separated. The experimental program also may require brief durations of increased-power (10-kW) reactor operations to make measurements above the inherent, high-neutron-source levels. The fission products then present short periods of high-radiation background, which may require gamma shielding.

The personnel shields move on an overhead track system and are shown in Fig. 9. During reactor operations, the shields are retracted and

are interlocked with the table movement. When the reactor halves are separated during shutdown, the shields can be driven into position in front of each reactor half by remote controls in the control room or at the entrance of the reactor cell.

Each shield is composed of four 96-in.-high, 36-in.-wide, 10-in.-thick, door-like segments, presenting a projected shielded area 12 ft long and 8 ft high. The segments are mechanized and can be separated by a nominal 4 in. for loading and unloading single drawers and by a nominal 15 in. for inspecting the loaded core. The shields can be positioned laterally to permit access to all parts of the reactor face.

The shield composition is shown in Fig. 20. The Benelex 70 is an impregnated pressed wood with a hydrogen content of 5.51×10^{22} atoms/cc and is used to degrade the neutron energy and increase absorption. The Boral sheet is placed on the core-interface side of the shields to minimize reentry of thermalized neutrons into the core. The boron-containing paint is used as a dispersed neutron absorber. The steel and the Benelex also provide gamma-ray attenuation.

Fig. 20. Cross Section of Personnel Shield. ANL Neg. No. 103-A11704.

H. Loading Platform

The loading platform spans the reactor pit in the area between the separated reactor tables and is used as a personnel support during reactor loading and unloading operations. The platform is retractable and is driven on its roller and track system by an electrically powered rack-and-pinion drive. The center section is 11 ft 7.5 in. long by 56 in. wide and elevates 8 ft. An electric motor drives ball screws, which raise the center section

to full height in 70 sec. Controls for the platform travel are located in the control room and in the reactor cell. Elevating controls are located only on the portion of the platform that elevates.

Both surfaces of the platform are supported by a network of reinforcing I-beams. The load-carrying capability of the elevating part of the platform is 4000 lb, with certain restrictions on distribution. The platform is supported and guided by raised roller supports on the bed as it is inserted between the reactor halves. A locking hook and wedge keys keep the platform securely fastened and aligned in the inserted position.

The platform system is interlocked with the reactor-table movement to prevent damage.

I. Reactor Cooling System

A reactor cooling system is incorporated in the ZPPR to remove the heat generated primarily by the natural radioactive decay of plutonium isotopes in the assembled experimental cores. Basically, cell air is pulled through the reactor core, filtered, and then cooled in a direct-expansion heat exchanger before being recirculated to the reactor cell (see Fig. 7).

As explained earlier, the reactor matrix tubes are rectangular in cross section, which results in a 0.10-in. gap above the loaded core drawers. Air passes through these gaps over the edges of the fuel plates into the plenums between the reactor matrices and the rod-drive mounting plates. Ducts from the plenums transmit the air to AEC-type high-efficiency filters, where particulates, if any, are removed. The manifolds of the filter are connected to the reactor fans, which push the air to the heat exchangers. Control vanes or valves are provided in the various duct work to direct and control the airflow.

During separation of the reactor halves, air from the cell is drawn through the clearances between the personnel shields and the matrix on each half and henceforth through each matrix to the filters. During reactor start-ups, as the reactor halves approach closure, a louver on the top of the movable-half plenum opens and the valve (SIV-105) between the movable-half plenum and the filters closes. At this time, the valve (MD-126) between the two manifolds of the filters opens and airflow is from the back end of the movable half through the reactor and to the filters.

The matrix tubes that do not contain fuel are blocked off by individual expandable Neoprene plugs (shown in Fig. 16). This system gives adequate flexibility for manifolding the irregular shapes of experimental cores. The rod-drive mounting plates and plenums are gasketed to minimize leakage.

The cooling system is essentially a reduced-pressure system to minimize personnel hazards due to plutonium particulates in the unlikely event of a ruptured fuel can. The rated flow of the system is about 6000 cfm with inlet to the reactor fans at a negative pressure of about 10 in. of H₂O.

The 6000 cfm at the 5000-ft elevation of the NRTS will remove 3.1 kW of heat per °C of temperature rise. The heat-exchange capability of the cell refrigeration system can accommodate about 20 kW of heat from the reactor.

Since there may be periods during long weekends and between shifts when operating personnel are not at the ZPPR Facility, the reactor cooling system is also monitored at the EBR-II Facility where personnel are on duty continuously.

J. Core and Blanket Materials

The major portion of the fuel^{12,13} for ZPPR is a 69.3 wt % uranium, 28.2 wt % plutonium, 2.5 wt % molybdenum ternary alloy, clad in Type 304 stainless steel. The clad fuel ranges from 1 to 8 in. in length and is 0.245 in. thick and 1.998 in. high. The cladding is 0.014-0.016 in. thick with 0.100-in. end plugs. The fabrication procedure for jacketing the fuel requires the end plugs to be in direct contact with the fuel core during the closure welding operation. This, with the end-to-end contact of the pieces, ensures prompt thermal expansion of the fuel columns loaded in the reactor-core drawers. The specification for the isotopic composition of the plutonium portion of the basic fuel is as follows:

²³⁹Pu: >85.8 wt %

²⁴¹Pu: <2.5 wt %

²⁴⁰Pu: 11.5 wt %

²³⁸Pu: <0.1 wt %

²⁴²Pu: 0.1 wt %

A number of materials are used in simulating reactor cores and blankets. Depleted uranium, aluminum, stainless steel, U₃O₈, sodium carbonate, sodium, beryllium oxide, and graphite are the prevalent simulants. Other special materials are also used. A photograph of some of the core simulants is shown in Fig. 16.

The experimental program requires specialized fuels and materials in minor amounts. These include high-²⁴⁰Pu fuels, thinner fuel plates, rod-type elements, and others.

Some of the core materials may contain thermocouples for sensing core temperatures.

K. Nuclear Instrumentation

The ZPPR is equipped with nuclear instrumentation as shown in Fig. 21. The eight channels of nuclear instrumentation are as follows:

- 1 and 2: neutron-counter-period safety channels;
- 3 and 4: log-n-period safety channels;
- 5 and 6: linear-power-level safety channels;
- 7 and 8: linear-power-level channels with bucking circuits.

The instrumentation is solid-state circuit design, except where voltage or input-impedance demands require the use of vacuum tubes. Circuit designs are fail-safe as far as possible and permit operating in an ambient temperature range of 0-50°C. All instrument outputs used in the scram and interlock system are through contacts of high-speed relays.

The instrumentation neutron detectors are located on or in the reactor matrix, and the readout instruments are mounted in control-room cabinets. The neutron detectors are divided so that each reactor half has one of each of the different types of instruments.

The usage and operating philosophy of the different channels are as follows:

a. Channels 1 and 2. Channels 1 and 2 are neutron-counter channels equipped with log count-rate and period meters to provide reactor information and protection during the startup range of neutron flux. Each channel provides a low-neutron-level table-drive interlock to provide protection against startup without adequate neutron levels in the reactor. The chamber voltage supply is interlocked to provide a scram signal on low voltage. Period trips are provided by each channel at +30 and +15 sec. The +30-sec trip is used to interrupt the table drive on all periods shorter than +30 sec and is automatically reset. The +15-sec trip scrams the reactor. These channels are used for multiplication measurements during approach to critical.

b. Channels 3 and 4. Channels 3 and 4 are log-n-period channels using ^{10}B -lined, compensated ionization chambers. The operating range for the log n circuits is 10^{-11} to 10^{-4} A, which provides neutron-flux information and period protection during startup and operating levels of the reactor. The outputs of the channels are recorded on a dual-pen recorder. Period trips at +30 and +15 sec are provided. The +30-sec trip interrupts the table drive if the observed period is shorter than +30 sec and is automatically reset. Activation of the +15-sec trip scrams the reactor.

Fig. 21. Schematic Diagram of ZPPR Nuclear Instrumentation. ANL Neg. No. 103-A11683.

c. Channels 5 and 6. Channels 5 and 6 are linear-power safety circuits using ^{10}B -lined, uncompensated ionization chambers. The operating range for the safety circuits is 10^{-10} to 10^{-4} A in decade steps. The range of these channels overlaps the upper ends of startup Channels 1 and 2. The power-level trips are manually set at a prescribed reactor power level and scram the reactor when this power level is exceeded.

d. Channels 7 and 8. Channels 7 and 8 are linear current amplifiers with bucking-current circuits. The detectors are ^{10}B -lined, compensated ionization chambers. The two circuits provide operational information of the operating reactor neutron-flux levels. The bucking-current circuits permit increased sensitivity of the instruments for accurate determination of power level. The outputs of Channels 7 and 8 are fed to the reactor panel meters and recorders.

The ZPPR nuclear instrumentation has been checked out for operability and performance in the Argonaut reactor. The trip circuits, a special ANL design, have been tested at $0-65^{\circ}\text{C}$ and have demonstrated trip operability at these temperatures. Additional tests have shown that opening or shorting out any one active component will result in a trip signal from the circuit. These capabilities provide, as far as practical, a fail-safe trip circuit for the ZPPR.

L. Reactor Controls and Interlocks

The reactor is operated from the ZPPR control room. The control console is an island-type with only the necessary switches and readout available to the reactor operator. Other secondary controls are mounted in the control cabinets.

Administrative control is exercised over the access to the instrumentation and control wiring. Three control keys are required to operate the reactor: (1) Control Power Key, (2) Reset Power Key, and (3) Table Drive Key. The backs of the cabinetry in the control room have locks.

One cabinet contains the selection equipment for ganging the fuel safety rods and the poison safety rods. The selection is performed on a visible panel, enclosed by a lockable, windowed door. The ganging of safety rods is administratively controlled and is only changed after analysis of reactivity worths while the reactor is shut down.

The movements of the shields, the loading platform, and the reactor cooling system can be controlled from the control room. The shields and the loading platform can also be controlled in the reactor cell. The isolation of the cell ventilation is controlled from the control room.

Extensive interlocks are provided in the ZPPR system. A simplified interlock diagram is shown in Fig. 22. An explanation of the interlock system is as follows:

1. Control Power Key. The Control Power Key supplies power to all control and interlock circuits, except those circuits requiring power during shutdown.

2. Reset Power Key. The Reset Power Key supplies power for reset and for the source-drive circuits.

3. Reset Circuit. The reset circuit performs two functions: (1) ensures that all safety devices and interlocks are in the proper condition for reactor startup, and (2) provides for a scram of the reactor for off-normal conditions of the safety devices or violation of the reactor interlocks during operation. The reset circuit is made up of the following safety devices and interlocks:

- a. Fifteen-sec period trips from Channels 1-4.
- b. Power-level trips from Channels 5 and 6.
- c. Loss-of-detector-voltage trips for Channels 1-8.

NOTE: If operation of startup Channels 1 and 2 is attempted beyond their upper limits, the detectors' lives are shortened considerably. To prolong the detectors' lives, level circuits in Channels 3 and 4 are used to permit deactivation of the high-voltage supplies of Channels 1 and 2 at the upper limits of the detectors. At this power level, Channels 5 and 6 are well within their operating range. As the reactor power is reduced, reactivation of Channels 1 and 2 must be initiated or the reactor will scram automatically.

- d. Two reactor-core/high-temperature trips.
- e. Low pressure in the emergency air-driven table scram-air-supply.
- f. Fire in cell area.
- g. Two high-alpha activity monitors in reactor-cooling-system trips.
- h. Earthquake trip.
- i. Cell-integrity interlocks, requiring all blast and seal doors, blast and seal valves, and fuel-transfer-tube doors be closed at startup and remain closed during operation.
- j. Safety-rod position interlock, requiring all rods to be in their least reactive position (scrammed) for reset. This is an indication that all rods will scram to their least reactive position.

Fig. 22. Wiring Diagram of ZPPR Interlocks and Controls. ANL Neg. No. 103-A11661.

k. Minimum control-rod and shim-rod interlocks, requiring at least one control rod and one shim rod to be programmed in each reactor half.

l. Two manual scrams in the system: one in the control room on the control console, and one in the cell near the entrance corridor.

If all safety devices and interlocks are in the correct condition for startup, the reset relays may be actuated and held in by the reset contacts around the reset pushbutton. The reset relays supply power to the table- and rod-drive circuits and the table-clutch and rod-latch power supplies.

4. Source Drives. Two sources are provided, one in each half of the matrix. The sources can be inserted or withdrawn when the control system is reset. If the control system is not reset, the sources may be inserted; however, they will automatically withdraw after 1 min. The sources automatically withdraw when a scram occurs or when the Control Power or Reset Power Keys are removed.

5. Table-drive Circuit. The table drive consists of three reversible drive motors, interlocked and controlled as follows:

a. The Table Drive Key provides power to the table-drive circuits.

b. The table scram interlock removes power from the drive-in circuit on reactor scram.

c. The platform interlock requires the loading platform to be fully retracted before operation of table.

d. The shield interlock requires personnel shields on each half of the matrix to be fully retracted before operation of the table.

e. The 30-sec period trip from Channel 1, 2, 3, or 4 interrupts table drive on any period shorter than 30 sec.

f. The neutron-level interlocks are two level trips provided by Channels 1 and 2, which require a minimum neutron level in the reactor (either from sources or spontaneous fission) before the tables can be driven together.

g. The rod-position interlock requires that all safety rods be in their most reactive position at any separation of the tables under 36 in.

h. The high-speed drive requires only initiation of the drive by pushbutton. High-speed "in" will continue automatically until the required speed-change point is reached if the 36-in. separation interlock of safety rods is satisfied. High-speed "out" will continue until the matrix is completely separated. High-speed separation requires only interlocks a, b, c, and d directly above for operation.

i. The intermediate-speed drive requires constant actuation by the operator. The "in" drive will terminate at the required speed-change point.

j. The slow-speed drive requires constant actuation by the operator. The "in" drive will terminate with complete closure of the matrix halves.

6. Table Scram Circuits. Loss of the table-scram-clutch power scrams the reactor and prevents startup. Upon any other scram signal from the reset circuit, an electrical scram motor will drive the reactor halves completely apart. If the normal scram system fails to operate during a scram, an emergency air-driven scram motor is activated and drives the halves apart. The air-scram motor becomes operative after the ball-screw drive stops rotating or if power to the reactor system is interrupted.

7. Table-clutch Power Supply. The table-clutch power supply supplies power through appropriate relays to the magnetic clutches, which connect the correct drive motors to the table-drive ball screw.

8. Control- and Shim-rod Drives. The six control- and six shim-rod drives are controlled by individual switches. Operation of these rods to insert reactivity requires the reactor controls to be reset. The rods can be operated for latching with only control power.

9. Poison-safety-rod Drives. The poison-safety-rod drives are selected by means of a selector switch and operated by a single rod-drive switch. Each rod has a gang-selector switch associated with it to allow administrative control of ganged rod-drive operations. Rods may be combined into one of four gangs or operated individually. Ganged operation functions for both addition and removal of reactivity. The "all" position functions only in the direction of negative reactivity and drives all rods, regardless of whether they are part of a gang or operated individually. The cabinet that houses the selection equipment is locked and is also interlocked with the table-drive circuit. Access to this cabinet is under administrative control.

10. Fuel-safety-rod Drives. The fuel-safety-rod drives function the same as the poison-safety-rod drives, except that the directions of reactivity addition and subtraction are reversed. Ganging of the fuel safety rod is accomplished similarly to the poison safeties, and the cabinet containing the selection equipment is locked. The cabinet is interlocked with the table-drive circuit and is under administrative control.

11. Rod-position Relays. The rod-position relays sense "in" and "out" positions of all rods and provide interlock information on the reset (three) and table-drive (five) circuits. Six control- and six shim-rod function switches are in the safety-rod circuits; this allows assigned control

and shim rods to function as safety rods. When a control or shim rod is assigned the function of a safety rod, it must fulfill the same requirements of position as a normal safety rod. Assignment of control or shim rods to fuel-safety or poison-safety-rod functions is limited by the minimum control- and shim-rod requirement as defined under item k on p. 62.

12. Minimum Control- and Shim-rod Requirement. This circuit senses the presence of at least one functional control and one shim rod in each half of the reactor and supplies this interlock information to the reset circuit.

13. Rod-latch Power Supply. The rod-latch power supplies provide power for all safety-rod latches, including all control and shim rods that have been assigned safety-rod functions. Power is supplied through scram contacts from the 15-sec-period scrams of Channels 1-4, the power-level scram contacts of Channels 5 and 6, and the reactor high-temperature trips. Each safety rod is also provided with an individual scram for testing purposes. The scram contacts are backed up by two reset-relay contacts, which remove primary power to the safety-rod-latch power supplies upon a scram condition.

14. Reactor Cooling System. The fire relay will operate when it senses any fire in the cell area. The reactor is scrammed and power to the reactor cooling fans is cut off, turning off the fans. All reactor-cooling-system dampers and louvers are closed. The core temperature is also monitored; if the core temperature rises over the maximum allowable value, the reactor is scrammed and the reactor cooling system is secured. The reactor cooling air is monitored for alpha activity before entering the AEC-type high-efficiency filters. A high-alpha-activity signal from the reactor cooling air scrams the reactor and shuts down the cooling system.

V. FIRE AND RADIATION MONITORING

A. Fire Monitoring

The fire-alarm system in ZPPR is an extension of the existing ADT McCulloh network in the East Area of ANL at the NRTS. The system includes automatic response units and manual alarms. The system is also connected to the NRTS network.

1. Types of Fire Detectors

Two types of detectors are used. The area detectors are ADT "Teletherm," which are of the radiant-heat-detecting type. The airflow systems have BRK smoke detectors, which sense combustion products.

2. Locations of Fire Detectors

The fire detectors are identified and located as follows:

<u>Location</u>	<u>ADT "Teletherm" Detector</u>	<u>BRK Smoke Detector</u>
Reactor cell	4	
Blanket storage room	1	
Access corridor	3	
Fuel storage vault	4	
Workroom	4	
Inside equipment room	1	
Outside equipment building	1	
Sodium storage building		1
Main floor of support wing	15	
Control room	2	1
Cable-routing room	1	1
Main area of service floor	12	
Fan loft	5	
AFSR room	4	

B. Radiation Monitoring

Radiation monitoring is provided in ZPPR to minimize hazards to operating personnel and to detect abnormal conditions that may affect other areas. A selected monitoring system is installed, but subsequent operations may indicate revisions, deletions, or additions to this system.

All radiation signals are monitored in the control room as well as at the instruments. Since there are periods between shifts and on long week-ends when no personnel are at the ZPPR Facility, a high-radiation alarm,

initiated by the two plutonium monitors in the reactor cooling system, is provided in the control room of the EBR-II Facility.

1. Types of Radiation Monitors

There are four types of radiation monitors in the ZPPR radiation-monitoring system. Each has a specific purpose and is located to provide the optimum protection of personnel.

a. Alpha Air Monitors. Alpha air monitors sample air to detect airborne-plutonium-particulate contamination. Basically, these are particulate collectors, detectors, discriminators, counters, and alarms.

b. Area Gamma Monitors. Area gamma monitors are used primarily to detect fixed contamination or radioactive reactor-core simulants, as opposed to airborne contamination. In specific cases, such as in the workroom, the area gamma monitor may also be used as a criticality monitor. The area gamma monitors are basically a detector, a counter, and an alarm system.

c. Beta-Gamma Air-particulate Monitors. Beta-gamma air-particulate monitors sample air to detect airborne radioactive particulates other than alpha emitters. These units contain a filter, a detector, a counter, and an alarm system.

d. Effluent Monitor. A monitor is installed with sensing units in the ventilation of the mound area before and after the AEC-type high-efficiency exhaust-air filters. The monitor samples alpha particulates and beta-gamma particulates on both sides of the filter and gaseous gamma activity of the filters in the mound area.

2. Locations of Radiation Monitors

a. Alpha Air Monitors. Five alpha air monitors provide personnel protection. Two of these are located in the reactor air-cooling system (one for each reactor half) to warn personnel of plutonium particulates from the fuel in the core that may become airborne. The other three alpha air monitors are portable and can be moved to optimum locations for personnel protection. One is located in-cell to monitor cell air activity. One alpha air monitor is also positioned in the workroom. A spare portable alpha air monitor is available for use as conditions warrant.

The high-level alarm signs for the reactor cell are located at the entrance to the reactor cell proper and at the access corridor entrance from the workroom. The alarm signs for the workroom area are at the entrance to the workroom from the cell and support wing and at the outside access corridor.

b. Area Gamma Monitors. Seven area gamma monitors are dispersed in the ZPPR Facility. Two gamma monitors are located in the reactor cell, two in the inside equipment room, one in the workroom, one in the core-coating room, and one at the suspect filters in the fan loft. As for the alpha air monitors, warning signs are located at the entrances of the listed areas.

Three gamma-criticality monitors are provided in the mound area. One monitor is located on top of the loading hood in the workroom, one in the storage vault, and one in the core-coating room. All three of these monitors have local and remote (ZPPR control room) alarms.

c. Beta-Gamma Air-particulate Monitors. Three beta-gamma air-particulate monitors are provided for detection of airborne contamination. One is in the reactor cell and one in the workroom. The third monitor is a spare and can be moved as needed. Again, warning signs are located at the entrances of the cell and the workroom.

d. Effluent Monitor. The effluent-monitor air-particulate sample points are in the ventilation duct to the AEC-type high-efficiency filters and in the downstream duct from the filters. The gaseous gamma detector is located in the exhaust stack of the mound area.

VI. MANAGEMENT OF ZPPR

A. General

The ultimate responsibility for safe reactor operations is vested in the Laboratory Director. The direct line of responsibility is shown graphically in Fig. 23.

Fig. 23. ZPPR Management. ANL Neg. No. 103-A11684 Rev. 1.

B. Philosophy

Safe operation of the facility is the overruling concern of all personnel associated with the use of ZPPR. The selection and training of qualified individuals for management and operations are the primary protection against accidents to ensure the safe operation of the facility. Responsible, safety-oriented action is expected of these individuals at all times. As fortification for safe operation, fail-safe equipment, as far as practical, is incorporated in the reactor and experimental equipment. This includes mechanical-shutdown equipment and electronic safety circuits associated with the nuclear instrumentation. Preventive maintenance of equipment on a scheduled basis and preoperational and routine checks for proper performance of reactor components and instrumentation are performed to ensure safe reactor operations. Critical reviews of experiments and agreement by at least two qualified individuals are required before the performance of the experiments to minimize the introduction of unsafe reactor conditions.

Prescribed logs of reactor operations and maintenance are maintained. Since the overall safety of the facility depends upon factors related to operating the reactor, loading the reactor, and preparing the experiments, each phase is evaluated periodically and changes are made if necessary to enhance the safe operation of the facility.

C. Management of the Facility

Primary responsibility for management of the ZPPR Facility is the function of the ZPPR Reactor Manager, whose relation to the other staff is shown in Fig. 23.

D. Organizational Positions

1. General

The Laboratory Director is responsible for the safe conduct of all the Laboratory's programs. Among the groups available to him for advice is the Reactor Safety Review Committee (RSRC) of the Laboratory. This group independently reviews each reactor system periodically. At the request of the Laboratory Director, it reviews Safety Analysis Reports and proposals for new experiments or procedures where a question of safety is involved. Other committees reporting to the Laboratory Director are appointed on an ad hoc basis to investigate special aspects of reactor safety.

The Director of the Applied Physics Division has primary responsibility for the programs carried out on the ZPPR. Individuals are formally authorized to various positions in the operating organization by the Division Director and/or the Laboratory Director, after suitable programs of training and examination.

The Deputy Director acts for the Division Director in the absence or unavailability of the Division Director and on matters specifically delegated to him by the Division Director. The responsibility and authority of the Division Director applicable to safe and reliable operation of the Idaho Applied Physics facilities are delegated to the Deputy Director, who also represents the Division Director at the Idaho Site.

2. Personnel

a. ZPR Program Manager. The ZPR Program Manager (Idaho) reports to the Division Director and has responsibility for all matters, including programmatic, operational, and safety matters for the three sections (fast critical experiments, theoretical support, and experimental support) in the Applied Physics Division-Idaho activities. He is appointed by the Division Director.

b. Section Head, Fast Critical Experiments. The Section Head, Fast Critical Experiments, is responsible for programmatic, operational, and safety matters relating to the operations of the ZPPR and the AFSR. He is appointed by the Division Director.

c. ZPPR Reactor Manager. The ZPPR Reactor Manager is appointed by the Laboratory Director, upon recommendation by the Division Director. His function is to guide and direct details of the experimental program and personnel. He ensures that operation is in compliance with approved procedures. He proposes such rules as are necessary for safety and sees that these rules are properly transmitted to the operating personnel. He provides long-term continuity of management at the staff level. He is responsible for a set of reactor keys, which he issues to the proper operating personnel. Installation and modification of equipment are subject to his safety review.

d. Reactor Supervisor. The person immediately responsible for the safe operation of the facility, in the control room, is the Reactor Supervisor. He must be authorized by the Laboratory Director as qualified. The Reactor Supervisor has the authority to veto any proposed reactor operation.

The designation of Reactor Supervisor is limited to staff members or salaried personnel who have been qualified as Reactor Operators and who, in addition, have sufficient training to have demonstrated their ability to estimate the reactivity effects of changes in the machine loading. Since the Reactor Supervisor is in sole charge during routine operations, his attitude toward safety must be dependable. He is therefore required to have a demonstrated appreciation of the factors connected with ZPPR safety. In particular, he is familiar with the safe operation of the normal complement of equipment and instruments, and with the safety background of the approved operations and procedures. All loading and experiment plans are subject to his review and approval.

Certain reactor operations require the presence of a second supervisor at the control console in addition to the Reactor Supervisor. This requirement is in force during any startups in which the reactor is in an unconfirmed state, as described in Sec. VII.G. The Reactor Supervisor still retains prime responsibility for safe operation of the facility, although, when the reactor is in an unconfirmed state, all experimental procedures require the concurrence of the second supervisor before they may be carried out. It is then the responsibility of both supervisors independently to evaluate and approve the safety of the operation of the reactor.

e. Chief Technician. The Chief Technician (not shown in Fig. 23) is a qualified Reactor Operator with several years experience and a complete knowledge of the operation and mechanical features of the machine. He

supervises all maintenance work on the mechanical features and safety interlocks of the machine as directed by the ZPPR Reactor Manager.

f. Reactor Operators. Reactor Operators work under the direction of, and are responsible to, the Reactor Supervisor in the performance of loading changes or reactor operation. They must pass a standard color-perception test, as must all personnel who load, unload, or check drawers. They repair, install, or check out reactor equipment under the direction of the Chief Technician.

Authorization of Reactor Operators is by the Applied Physics Division Director upon the recommendation of the ZPPR Reactor Manager, when the Division Director is satisfied that the candidate has met the requirements for this position.

As part of the qualification as Reactor Operator, an individual, whether Staff Scientist or Engineer, Scientific and Engineering Assistant, or Technician, must serve a period as Reactor Operator Trainee. Training is accomplished by individual instruction and practice under qualified Reactor Operators, classwork by staff personnel, and testing as considered necessary.

The requirements for Reactor Operators include a knowledge of operation of the system and its basic auxiliaries, and familiarity with the daily, weekly, and other checks on instruments and equipment. A Reactor Operator must be able to recognize when a piece of equipment is malfunctioning and must have a clear conception of the administrative lines for reporting such malfunctions. He must be familiar with routine safety precautions and emergency procedures.

g. Health Physicist. A Health Physicist is assigned to the ZPPR. He is responsible for the operation and maintenance of the personnel and area monitoring equipment. He is advisory in function and reports to the Reactor Supervisor about the current status of radiation fields, contamination areas, and possible personnel exposures. He also reports, through the Radiation Safety Section, to the ZPPR Reactor Manager the records of accumulated personnel doses. He is responsible for establishing barriers or notices of radiation hazards when necessary.

h. Special Materials Representative. The Special Materials Representative is the custodian of the vault containing fissionable material. He issues fuel to the loading crew at the direction of the Reactor Supervisor in accordance with loading plans previously approved by both supervisors assigned to the assembly.

VII. REACTOR OPERATIONS AND ASSOCIATED ACTIVITIES

A. Introduction

The primary consideration during operation of the reactor and associated activities is the safety of personnel. Subsequent consideration is given to the equipment and the facility. Since the ZPPR Facility uses large amounts of plutonium for experiments, studied, careful actions are required at all times. All operating personnel are instructed to be thoroughly familiar with the activities associated with reactor operations. A training program for both staff- and technician-level personnel is used to ensure systematic evaluations of these operations. Wherever possible, a combination of qualified experienced personnel and newly trained people is used to obtain optimum safety. The design of the reactor is such that scrambling is tolerable under any circumstance.

The associated activities of reactor operations include loading and unloading the reactor, checking out the reactor system, handling core simulants, and maintenance.

B. Loading the Reactor

The loading of the reactor includes planning, preparation of the drawer-loading diagrams and the loaded-matrix diagrams, checking and inspection of fuel before loading fuel and other core simulants into the drawers, checking the loaded drawers against the diagrams, loading the drawers into the reactor matrix, and finally checking the loaded matrix with the previously prepared diagrams.

The planning of critical experiments involves formulating core compositions, preparing the drawer-loading sheets, and preparing the reactor-matrix loading diagrams. The loading sheets are approved by two Reactor Supervisors. The general experimental loading plan must be approved by the ZPPR Reactor Manager.

In the initial handling of the reactor fuel, the integrity of the cladding is checked as it is brought from storage, before being loaded into the drawers. After the core simulants are placed in the drawers, the drawer loading sheet is checked against the loaded drawer by qualified personnel. The drawers are coded and fit into a specific location in the reactor matrix. The loaded drawers in the matrix are then checked against the matrix loading diagrams by qualified personnel.

C. Reactor Startup

The reactor startup is a studied procedure effected by trained, qualified operating personnel. A weekly operating checkout is made of the reactor

system. The details of the weekly checkout are given in the operating manual. The weekly checkout is made before operating the reactor during that calendar week. If the reactor is not operated during a given week, those items requiring startup are not checked. However, to provide continuity of records, it shall be noted in the reactor logbook that no reactor operation occurred. Responsible personnel signatures are required in completing the weekly checkout.

A daily operating checkout is made of the reactor control system before startup. This checkout includes all reactor interlocks, reactor controls, operability of the nuclear instrumentation, inspection of the reactor, verification of the minimum safety-rod shutdown capability, and a brief description of the experiment(s) to be performed. The list filled out during checkout is given in detail in the operating manual. Again, signatures of responsible personnel are required as part of the daily checkout.

Three keys are required to operate the reactor: a Control Power Key, a Reset Power Key, and a Table Drive Key.

To start up and operate the reactor, a qualified Reactor Operator or a supervised Reactor Operator Trainee must be at the control console. The Reactor Supervisor must be in attendance in the control room for immediate consultation or instructions. If the Reactor Supervisor must leave the control room for a short time, another Reactor Supervisor may be temporarily substituted.

The sequences required to start up are best explained from the point of view of an operator at the control console. These sequences are explained in more detail in the operating manual and are described here to illustrate a representative startup. These operations can be followed by referring to Fig. 22. Weekly and daily checkouts are completed. The Control Power Key is inserted and is used to activate the control system. The Reset Power Key is inserted to provide continuity to reset the scram string.

The interlock contacts are closed if the following conditions are satisfied:

1. Cell integrity is verified. (Access corridor: two blast doors and two seal doors are secured. Ventilation ducts: two blast valves and two seal valves are closed. Escape tunnel: two blast doors and two seal doors are secured. Loading tube: two doors are closed.)
2. Periods on count-rate startup Channels 1 and 2 and on log-n-period Channels 3 and 4 are longer than the 15-sec trip point.
3. Power levels on safety Channels 5 and 6 are lower than the trip power setting.

- 4... Reactor core temperature is below the preset trip level.
- 5... The detector voltage on the instrumentation is set at the prescribed value.
- 6.. Power to the table-separation clutch is operable.
- 7.. The emergency-scrum air pressure is higher than the minimum acceptable.
- 8.. The cell-fire signal is negative.
- 9.. The reactor-cooling-system alpha activity is below the alarm limit.
10. The earthquake detector is not tripped.
11. The safety rods (both fuel and poison) are in their least reactive positions; i.e., the fuel safety rods are out of the core, and the poison safety rods are in the core.
12. At least one control and one shim rod are programmed for each half of the reactor core.
13. The manual scrams are not actuated.

After these control interlocks are satisfied, the reset relays are activated.

The initial table-closure movement is interlocked by the reset relays, the positions of the reactor shields and loading platform, the neutron level, and the Table Drive Key. A period signal is taken from the log-count-rate channels (1 and 2) and the log-n channels (3 and 4) and is used to interrupt the table closure if a period of 30 sec or shorter is obtained. This circuit is in operation at any table position and is reset automatically when the reactor period exceeds 30 sec.

The high-speed table drive has a self-holding-type switch and will run as long as the interlocks are satisfied. At a 36-in. table separation a rod-position interlock must be satisfied. At this table position, a limit switch is actuated, and all safety rods must be cocked before the table can be moved any farther. Also, the programmed control and shim rods must be in their least reactive positions. At any table separation less than 36 in., all safety rods must be cocked to allow table movement.

Until the reactor halves are brought together, control and shim rods must remain in their least reactive positions. The programming panels for the control, shim, fuel safety, and poison safety rods are in a locked control cabinet, which is kept under strict management control by the use of an access key.

When the table separation reaches 18 in., two limit switches deactuate the power to the high-speed table-closure drive. The intermediate-speed motor drive is manually actuated by the Reactor Operator. The intermediate-drive control switch requires continuous actuation by the Reactor Operator, or it will automatically open and stop the table movement.

At a 3-in. separation, another set of limit switches deactuates the intermediate-speed drive. The slow-speed motor closes the reactor halves at 1/6 in./min. The reactor halves are brought together from the 3-in. separation by the slow-speed motor. The slow-speed motor-control switch also requires continuous actuation by the Reactor Operator.

When the reactor halves are approximately 2 in. apart, the reactor cooling-system valves and louver are changed to redirect the coolant flow.

The reactor halves are brought together by the slow-speed motor, and, upon making contact, a preset force is applied by the drive mechanism through the slip clutch on the ball screw. Slippage of this clutch is detected by a switch on the clutch and is indicated at the control console in the control room.

All normal loadings are planned to be subcritical with the two reactor halves together and at least two control rods out. After the halves are closed, the control rods are inserted and can be adjusted to reach criticality and desired experimental power. During the approach to power, the rods are normally adjusted to control the reactor period to not less than 30 sec.

D. Reactor Shutdown

The reactor is considered shut down when the safety rods are located at their least reactive positions and the halves are fully separated. The full-separation-of-the-halves criteria may be waived for carefully controlled conditions during maintenance of the bed-and-table system. These conditions are explained in detail in the operating manual.

The reactor shutdown can be effected through three sequences:

- (1) a scram, whether automatic or manual, in which the safety-rod magnets are deenergized, and the electric scram motor drives the halves apart;
- (2) the loss of power or failure of the electric scram system to separate the halves; this actuates the emergency air motor, which drives the halves apart; and
- (3) a controlled shutdown using the regular drive motors.

When a scram signal is received, the shutdown occurs automatically. Violation of any one of the reactor reset interlocks will yield a scram.

To perform a controlled shutdown, the sequence of operation is such that the interlock is not violated. This sequence is as follows:

1. One control rod in each half must be fully withdrawn.
2. The safety rods must be in the position of greatest reactivity worth.
3. The startup-counter, high-voltage interlocks must be reactivated at the proper neutron level, or the reactor will scram.
4. The slow-speed motor is used to separate the halves to 3 in.
5. The intermediate-speed motor separates the halves to 18 in.
6. The high-speed motor separates the halves to the fully open position.
7. The safety rods are then electrically driven to their normal shutdown position.

A fast shutdown is provided in the reactor controls for experiments requiring a rapid decrease in reactor power. This fast shutdown, without scrambling the reactor, is obtained by first reactivating Channels 1 and 2 and then removing the Table Drive Key. The reactor halves are driven apart without scrambling the safety rods.

When the reactor core is loaded, the reactor shields are moved in front of the reactor halves to protect personnel. The loading platform can also be driven into position between the halves.

If movement of the table is necessary for maintenance when the reactor is loaded, special permission must be obtained from the Division Director or Deputy Division Director. After maintenance has been completed, a comprehensive check of the reactor's response to interlocks and controls is made to verify proper functioning of the system.

E. Approach to Critical

1. General

The approach to critical may be generally divided into three categories: (1) new loading, (2) previous loading, and (3) minor changes in the reactor core. The weekly and daily checkouts, explained in Sec. VII.C., shall be in force in an approach to critical. The reactor safety interlock system is in full operation any time the reactor is started up.

2. New Loadings

A new critical experiment is thoroughly examined, as described in Sec. VII.B. As a general procedure, the blanket is installed first. When a new critical assembly is being loaded, the initial loading is between one-fifth and one-half the calculated critical mass. The unloaded fuel locations

are filled with diluent materials. The control rods and fuel safety rods are initially loaded with their proper complement of fuel.

At the end of each loading stage, the count rates on the flux-monitoring channels are recorded. Established graphical extrapolation methods are used to obtain a new estimate of the critical mass at each stage. Allowing for safety-rod and control-rod worths, a safe increment of fuel drawers is then substituted for the diluent materials to constitute the next stage.

This procedure of step-by-step loading and multiplication evaluation is followed until sufficient fuel is added to the core to obtain criticality with the control rods.

3. Previous Loadings

When a previous loading is repeated, the incremental steps for approach to critical may be somewhat larger since the uncertainties in the critical mass are not great. However, to check with the previous loading, multiplication measurements are made as the loading to critical is performed.

4. Minor Changes in Reactor Core

As in all loadings, calculations are made, checked, and approved prior to changes in the core. This mode of reactor experimentation is the major mode for most reactor cores and occurs after the initial approach to critical.

F. Limitations in Reactivity-insertion Rates

The normal reactivity-insertion rates are described here to clarify reactor startup and operating criteria.

The safety rods (both fuel and poison) are normally used for fast shutdown of the reactor. Scramming these rods removes reactivity in well under 0.5 sec. The total worth of these safety rods is at least 1.5% $\Delta k/k$ reactivity.

Reactivity can be added by movement of the halves, insertion of the fuel safety rods, withdrawal of the poison-safety or shim rods, and, finally, insertion of the control rods.

Movement of the reactor halves is limited to a maximum of 2 \$/sec reactivity insertion during closure.

Combined movements of the safety rods are limited to a 0.06 \$/sec reactivity-insertion rate.

The movement of control or shim rods is limited also to a 0.06 \$/sec reactivity-insertion rate.

The maximum excess reactivity permitted in each loading with all control/shim rods in their most reactive position is 0.2% $\Delta k/k$ for one pair of control or shim rods. A maximum of 0.5% $\Delta k/k$ is permitted for three or more pairs of control or shim rods.

When movable, remotely controlled experimental devices are in the reactor:

1. The maximum excess reactivity available with these devices in their most reactive positions shall be ≤ 0.50 \$.
2. The maximum reactivity-insertion rate for these devices, when the reactor is within 2% $\Delta k/k$ of critical, shall be 0.06 \$/sec for reactivity additions greater than 0.10 \$.

G. Confirmed and Unconfirmed States of Operation

The presence of two Reactor Supervisors shall be required during startups when the reactor is in an unconfirmed state of operation. An unconfirmed state exists after a loading change and until the halves are driven together and criticality verified. Examples of changes falling into the category of unconfirmed state of operation are:

1. Fuel or blanket loading changes in which the estimated change is greater than 30 lh.
2. Changes of location of control, shim, or safety rods.
3. Location changes of nuclear instrumentation used in reactor operation.
4. Whenever in the judgment of the Reactor Supervisor or Operators an unknown or large reactivity change has been made in the facility.

For the category of confirmed state of operation in which no loading change is involved and the geometry is understood, the facility may be operated by the Reactor Supervisor and one Reactor Operator. Examples of operation falling into this category are:

5. Foil activation or nuclear-emulsion irradiation.
6. Flux traverses or fission-ratio measurements by means of foils or small counters.
7. Reactivity-coefficient measurements when small reactivity changes are involved.
8. Pile-oscillator-type measurements after the reactor has been made critical with the oscillator sample in the reactor.

In general, this second category of operation may be undertaken only after the reactor has first been made critical with the necessary experimental equipment in the reactor.

During a confirmed state of operation, only one Reactor Supervisor is required in the control room.

H. Experimental Equipment

As far as practical, experimental equipment is checked out and operated before installation in the reactor. Whenever reactor management considers it advisable, scram signals from experimental equipment are incorporated into the reactor system.

I. Preventive Maintenance

The ZPPR Facility is operated with a preventive-maintenance schedule. Included in the preventive-maintenance schedules are all systems involving the reactor instrumentation and controls, the ventilation, and the isolation components. The schedules are based on the individual mechanical and electrical components in that there are daily, weekly, monthly, quarterly, yearly, and biennial checks. The specifics of the preventive maintenance are part of the reactor operating manual.

VIII. REACTOR SAFETY EVALUATION

A. Introduction

The evaluations presented in this section emphasize the use of plutonium as a fuel. Since the problems using ^{235}U are less serious than those of plutonium, the ZPPR Facility can be used for both plutonium and ^{235}U . It is anticipated that ^{233}U may at some time be used as a fuel in ZPPR experiments and that additional shielding may be needed for the protection of personnel.

ANL has had more than five years' experience with plutonium-fueled cores at the ZPR-3 Facility. During this period, valuable background and maturity have been obtained and will be applied to the experimental program of ZPPR. As in most safety reviews, a pessimistic attitude is assumed and the various operations involved in the performance of experiments in the reactor are evaluated. The items discussed in the evaluation are those believed to be pertinent, and, if new evaluations become necessary in the future, they will be treated in much the same way.

Fig. 24. Reactivity Worth of Gap in ZPPR for Various Cores. ANL Neg. No. 103-A11697.

reasonable agreement was found. Calculations for larger separations have been performed for the DBA studies using the 2-D transport-theory code SNARG.¹⁶ The results are given in Sec. X.

The primary purpose of this evaluation is to examine normal operations to ascertain whether the facility is designed, built, equipped, and operated to provide more than reasonable protection for the operating personnel, the facility, and the surrounding countryside.

B. Reactivity Effects

1. Worth of Gap between Halves

The reactivity worth of the gap between the core halves is plotted in Fig. 24 as a function of separation for carbide and metal cores of various sizes. Tamor's¹⁴ method was used in the calculations. Measured and calculated gap sizes on ZPR-6 Assemblies 1 and 2 have been compared by Rusch *et al.*,¹⁵ and

2. Expansion Coefficient of Reactivity

The rear ends of the drawers in ZPPR are fitted with leaf-spring metal plates to allow for longitudinal expansion of the drawer materials. The normal arrangement will be to have the spring separated from the core materials by a few inches of axial blanket to minimize its perturbation on the critical mass.

Although there will be several fuel plates per column of fuel in the drawer, experiments have shown that the fuel column, when heated, will behave as though it were one continuous plate.¹⁷ Heating will normally move fuel material from high importance (core average) to low importance (core edge), so that the reactivity effect is negative.

Expansion of other materials will in general give positive reactivity effects, but these will be small. The positive effect is due to a net movement of absorber from the core center and a net movement of scattering material into a flux gradient at the core edge. In transient situations, particularly the DBA for ZPPR, the temperatures of most of the nonfuel materials lag well behind the fuel-alloy temperature. Because of their fast-fission heating and their larger absorptive cross sections, depleted-uranium plates have the only other significant positive expansion effects, and this positive effect is less than its negative Doppler coefficient and much less than the expansion effects of the fuel.

The changes in reactivity due to a change in dimensions and temperatures of core plates and depleted plates for the two cores are given in Table III.

TABLE III. Reactivity Feedback Coefficients
for Expansion and Doppler Effects

	$\frac{\Delta k}{k} / \frac{\Delta l}{l}$ for Expansion of		Doppler Coefficient at 300°K	
	Fuel Plates	Depleted U	Fuel Plates	Depleted U
1000-liter Core	-0.315	+0.050	-1.03×10^{-6}	-3.10×10^{-6}
6000-liter Core	-0.393	+0.190	-2.30×10^{-6}	-8.60×10^{-6}

Expansion Coefficients of Alloy Fuel

300-500°K:	$15.4 \times 10^{-6}/^{\circ}\text{K}$
500-650°K:	$28.5 \times 10^{-6}/^{\circ}\text{K}$
650-818°K:	$28.6 \times 10^{-6}/^{\circ}\text{K}$
818-900°K:	$52.0 \times 10^{-6}/^{\circ}\text{K}$
900-1230°K:	$24.7 \times 10^{-6}/^{\circ}\text{K}$

Expansion Coefficient of Depleted Uranium

Average of $14.1 \times 10^{-6}/^{\circ}\text{K}$ from 300 to 900°K

3. Material Worths of Central Drawers

Table IV lists the effects on reactivity of removing the central drawers from both halves of the core in ZPPR and replacing them by drawers full of other materials. In all cases except fissile-material substitutions, the effects of the substitutions are negative. The greatest positive effect is gained by substituting a drawer full of enriched- ^{235}U plates. Eight fuel-alloy plates would be less than the ^{235}U substitution because of the ^{238}U diluent and the claddings.

TABLE IV. Worths of Various Materials in Central Drawer (both halves)

Removal of Central Drawer and Substitution of		Reactivity Change, %					
		Metal			Carbide		
Material	Density, g/cc	2000 liters	4000 liters	6000 liters	2000 liters	4000 liters	6000 liters
<u>L/D = 1.0</u>							
Paraffin	0.90	-1.07	-0.80	-0.65	-0.76	-0.64	-0.56
Beryllium	1.85	-0.38	-0.30	-0.25	-0.15	-0.11	-0.08
Graphite	1.85	-0.27	-0.20	-0.16	-0.17	-0.11	-0.07
Depleted U	19.7	-0.42	-0.29	-0.22	-0.55	-0.41	-0.34
Enriched U	18.9	+7.26	+6.08	+5.40	+7.14	+6.06	+5.4
<u>L/D = 0.5</u>							
Paraffin	0.90	-0.70	-0.50	-0.38	-0.48	-0.40	-0.35
Beryllium	1.85	-0.26	-0.20	-0.14	-0.06	-0.05	-0.04
Graphite	1.85	-0.18	-0.14	-0.10	-0.11	-0.05	-0.05
Depleted U	19.7	-0.26	-0.18	-0.12	-0.35	-0.25	-0.21
Enriched U	18.9	+5.40	+4.46	+3.93	+5.15	+4.32	+3.81

4. Personnel Shields and the "Fat-man" Effect

During manual operations between the core halves at shutdown, layered steel and hydrogenous shields protect the operating personnel from neutron and gamma leakage from the core. The neutrons are generated from spontaneous fission in the ^{240}Pu content of the fuel alloy. Some multiplication of these neutrons occurs, even under shutdown conditions, especially in the very large cores.

The details of the shields are given in Sec. IV.G. In essence, the shields are alternating layers of 2-in.-thick Benelex slabs and 0.5-in. stainless steel plates. To absorb the low-energy neutrons returning from the shield to the core, a 0.25-in. sheet of Boral is fixed to the core side of the shields. As designed, these shields also serve to isolate against inadvertent reactivity increases due to the "fat-man" effect.

Calculations have been made to determine the reactivity effects of placing these shields against the bare half-core faces. An effectively infinite thickness of water was assumed for the hydrogenous shield. (H_2O has about 14% more hydrogen atoms/cc than Benelex.) The calculations have been made for typical power-reactor mockups covering the size ranges of greatest interest to the ZPPR control program, namely 2000-, 4000- and 6000-liter metal and carbide cores having length-to-diameter ratios of 1.0 and 0.5. Multigroup diffusion theory using the Hansen and Roach cross-section library¹⁸ was the basis of the calculations.

The results of the calculations are shown in Table V. In all cases, the water-reflected half-cores have been calculated to be subcritical. The presence of the Boral shielding reduces the criticality to almost that of the bare core in every case. Thus there is no indication that the personnel shields or the presence of operators behind the shields will present an operational hazard.

TABLE V. Reactivity Effects of Personnel Shields

	Subcriticality Factor, $1 - k_{eff}$					
	2000 liters		4000 liters		6000 liters	
	Metal	Carbide	Metal	Carbide	Metal	Carbide
<u>$L/D = 0.5$</u>						
Half-cores	0.265	0.330	0.212	0.274	0.184	0.241
Half-cores reflected by 8 in. of H_2O	0.102	0.100	0.093	0.092	0.086	0.084
Half-cores reflected by 8 in. of H_2O , but separated by 1/64-in. Boral	0.211	0.254	0.175	0.221	0.155	0.197
<u>$L/D = 1.0$</u>						
Half-cores	0.137	0.177	0.105	0.140	0.088	0.120
Half-cores reflected by 8 in. of H_2O	0.099	0.102	0.084	0.090	0.074	0.080
Half-cores reflected by 8 in. of H_2O , but separated by 1/8-in. Boral	0.132	0.170	0.102	0.135	0.086	0.116

In an experiment¹⁹ attempting to measure the "fat-man" effect in ZPR-6 on a 950-liter carbide core with an L-to-D ratio of 0.33, the half-core was very subcritical with 8 in. of hydrogenous reflector material. As the ZPPR program progresses to larger cores, experimental checks of the calculational sizes will enable more reliable assessments to be made on the effects of the shields on the very large cores.

Interlocking of the startup controls with the shield-position indicator precludes any possibility of table motion with the shields in place. Even if the interlocks were not there, the two shields would effectively decouple the core halves.

5. Gravel and Sand between Core Halves

In the extremely improbable event of ceiling collapse as the reactor halves are moving together, calculations have been made to show that the presence of wet gravel between the halves at any separation could not cause large reactivity increases.

It is estimated that dripping wet gravel with 30% voidage would have a water content of several percent. A large 6000-liter carbide core with an L-to-D ratio of 0.5 was used as the model, and criticality was assumed with the halves together. The gravel was simulated by potassium, which has roughly the same nuclear cross section as silica. The Hansen and Roach cross-section library was again used. Based upon these assumptions, the reactor would be subcritical by the following amounts at 6-, 12-, and 24-in. separations:

	Separation		
	6 in.	12 in.	24 in.
Dry gravel	-6.3%	-9.4%	-13.5%
Wet gravel (3% water)	-8.6%	-11.2%	-16.1%

In smaller cores, the leakage effect would cause a greater degree of subcriticality. In low-density plutonium systems, the effect of hydrogen is negative and it is more so in metal-fueled cores than in carbide cores.

6. Doppler Coefficient of Fuel and Depleted Uranium

As described in Sec. IV.J, the bulk of the fuel for ZPPR will be in the form of a Pu-U-Mo alloy having the following composition: 28.2 wt % plutonium containing 11.5% ^{240}Pu , 69.3 wt % depleted uranium, 2.5 wt % molybdenum.

The Doppler temperature coefficient of reactivity in this fuel is always negative, and its magnitude depends upon the system and core size.

All the cores in the ZPPR will probably have aggregate U/Pu ratios greater than the alloy fuel value of 2.46. This means that a significant, if not larger, fraction of the ^{238}U will be in the form of depleted-uranium metal or depleted U_3O_8 . Consequently, there will be a large negative Doppler component from this ^{238}U , which, because of its small internal heating, will not play a major part in accident conditions unless large powers are reached as in the case of the hypothetical DBA. At this time the negative Doppler coefficient has a useful effect, since it manifests itself when the fuel-expansion effects have been lost due to melting and the fuel Doppler is beginning to saturate out.

Table VI shows the results of Doppler-coefficient calculations on a range of core sizes and compositions of interest to the ZPPR program. Diffusion theory and the ANL-224 set were used in the calculations.

TABLE VI. Doppler Effects in Typical ZPPR Cores at 300°K

	2000 liters		4000 liters		6000 liters	
	Metal	Carbide	Metal	Carbide	Metal	Carbide
Fuel Doppler Coefficient, ($\Delta k/k$)/°K	-1.2×10^{-6}	-6.8×10^{-6}	-1.9×10^{-6}	-8.2×10^{-6}	-2.3×10^{-6}	-9.0×10^{-6}
Depleted Doppler Coefficient, ($\Delta k/k$)/°K	-5.5×10^{-6}	-15.9×10^{-6}	-7.2×10^{-6}	-23.0×10^{-6}	-8.6×10^{-6}	-27.1×10^{-6}

C. Instrumentation

1. Nuclear Instrumentation

The reactor nuclear instrumentation has eight channels. The reactor will not be operated unless all eight channels are functioning properly. It is anticipated that failure of components may occur during operations. In such event, the reactor will be shut down immediately and repairs will be made. Operability of the instrumentation is verified during reactor checkout before startup.

During the startup of the reactor, four channels (1-4) are functioning as the reactor halves are brought together. Each channel has a +15-sec period trip to scram the reactor. Channels 1 and 3 are on the stationary half and Channels 2 and 4 on the movable half of the reactor. Any one-channel trip signal will scram the reactor. In addition, Channels 1-4 have +30-sec period trips, which stop the table motion if an apparent period of less than +30 sec is detected.

As the power level is raised, safety-power-level Channels 5 and 6 become operative before the upper operating limits of Channels 1 and 2 are reached. Therefore, at the higher power levels, Channels 3-6 protect the reactor system with period and power-level scrams. At these higher power levels, Channels 1 and 2 are deactuated to prolong detector life.

On loss of high voltage, a trip circuit on each channel supplies scram signals to the reactor.

The nuclear instrumentation has been tested between 0 and 65°C and has been found to be fail-safe in providing reactor trip signals. Normally, this instrumentation operates in areas maintained at about 22°C. Circulation fans in the cabinetry prevent stagnation of air in the cabinets and provide cool air for the instruments. Since the major portion of the nuclear instrumentation is solid-state electronics, the heat loads are minimal.

One linear power-level instrument, Channel 7, contains an internal power supply (battery-operated), which provides power for normal operation of the channel for at least 2 hr after loss of external power.

Wherever possible and practical, the channels are separated to ensure independent functioning of each channel.

Daily and weekly checkout of the instrumentation and periodic preventive maintenance provide assurance that the nuclear instrumentation will function, if necessary, in an emergency.

2. Fire

The reactor core contains temperature sensors to monitor both halves during operations and shutdown. A core-temperature trip circuit for scrambling the reactor is included in the reset power of the reactor controls. The normal temperature trip point in the core is set below the melting point of sodium, 98°C. Additional temperature monitoring may be included in special experiments to provide additional protection.

Fire detectors are dispersed in the reactor cell. A signal from any of these detectors automatically closes the valves in the air ducts to and from the cell. Because of the containment provided by the closed seal valves and seal doors, the sealed piping and instrument bulkheads, and the sealed covers on the loading tube, the venting of the combustion products from the cell is directed through the cell roof. If a fire signal is obtained during reactor operations, this signal also scrams the reactor.

3. Cell Isolation

The reactor cell has seal doors in the access and escape tunnels, seal valves in the supply and exhaust ventilation ducts, gas seals on all bulkhead penetrations, and seal covers on the loading tube. During reactor operations, all seal doors, seal valves, and seal covers are maintained in a closed position and are interlocked with the reactor controls. While the loaded reactor is shut down and personnel are traversing in or out of the cell, at least one seal door is automatically kept closed by interlocking between the control mechanism of the doors. Indicator lights are provided in the reactor control room to provide monitoring for the doors, valves, and covers of the cell penetrations.

During major modifications in the reactor cell, when no plutonium or uranium fuel is present in the cell, it may be practical to open both sets of access doors to the cell simultaneously to permit unencumbered transportation of materials and personnel. Since the lack of fuel in the cell precludes any release of contamination, this deviation under special conditions is acceptable.

The pressure capability of the seal doors is 12.5 psig, and the seals are tested periodically. The seal valves and bulkheads are also tested periodically.

4. Radiation Monitoring

a. Alpha Air Monitoring. Two alpha air monitors in the reactor cooling system are used to detect alpha-emitting particulates that may originate from the core due to ruptured fuel jackets, test samples, etc. An above-normal background trip from the alpha air monitors is provided in the scram relay string and automatically shuts down the reactor and the cooling system.

A portable, high-sensitivity alpha air monitor is also located in the cell to monitor background to warn personnel of respirable alpha contamination. Since the cell is secured while the reactor is operating, a slight buildup of radon from the concrete may occur if the reactor operations are extended. Based on experience with the EBR-II containment building, the buildup should not be extensive and will not mask the detection capabilities of the alpha air monitors.

The readouts of the alpha air monitors are in the control room and at the instrument locations. Warning signs are located at the cell entrance. The proper use of the alpha-air-monitoring equipment precludes personnel from inhaling alpha-producing particulates.

b. Gamma Monitoring. Two area gamma monitors in the cell warn personnel of radiation background from the reactor or from other materials. These instruments are kept in operation continuously and are used to evaluate the propriety of entering the cell after reactor operations, in particular. This minimizes unnecessary direct-radiation hazards.

c. Beta-Gamma Air-particulate Monitoring. The portable beta-gamma air-particulate monitor in the cell is used primarily for detection of airborne fission products from the fuel or fertile materials and provides personnel protection.

D. Safety-, Control- and Shim-rod Performance

The fuel safety rods are withdrawn from the core during shutdown to remove reactivity. The rods are held cocked against a compressed spring by a dc magnet. Removal of dc power to the magnet releases the rod, and the springs push the rod out of the reactor in less than 0.3 sec. A typical distance-time relationship for an empty fuel safety rod is shown in Fig. 18. The empty drawer weighs 5.5 kg, and the time to move 90% of the travel distance is 0.155 sec. These tests were performed on a rod drive in the laboratory under somewhat idealized conditions. To determine the rods' scram characteristic in the core, the tests will be repeated when the rods are mounted in the matrix.

Scheduled periodic checking of the performance of the fuel safety rods is made. These data are compared to original and previous data to

determine continuous satisfactory performance of the rods. Any rod performing unsatisfactorily will be repaired or replaced before the next startup of the reactor.

Since each rod has a dc holding magnet, the release characteristics of the magnets were determined with 130 V ac superimposed. This voltage does not hold the rod and does not measurably affect the release time of the magnets.

The control rods are similar to the fuel safety rods in their scram characteristics.

The drawers for the rod drives are Type 304 stainless steel and are Scottsonized or chrome-plated, giving a hard surface coating to minimize galling of the matrix tubes. The rod drives are supported at the back end and are aligned with the matrix tubes to give a straight, unrestricted scram movement. A covering over the loaded fuel-rod drawers prevents vertical motion of the pieces during movement of the rods.

Loss of power to the reactor system will deenergize the holding magnets, and all rods will scram.

Each fueled safety rod has two springs, and conceivably a spring could break. When a scram signal is received, the remaining spring would be able to at least partially drive out the rod.

The substantial frame around the rod drives protects them and precludes significant damage to the rods from falling objects. The reactor cooling plenum encloses the connecting portions between the drawers and the drive mechanisms and protects them from outside interference.

The poison safety rods are filled with highly enriched ^{10}B carbide plates. The rods have dc holding magnets and use a compressed spring to drive the delatched rod into the core to remove reactivity. The same alignment and protection are provided by the reactor frame and the reactor cooling plenum. The poison blades and the sheaths are Scottsonized or chrome-plated to reduce friction and wear.

Since the poison safety rods are physically lighter than the fuel safety rods, the travel time during movement is considerably shorter. The travel time for 90% of the 30-in. stroke is about 100 msec. The scram times are measured with the rods mounted on the reactor assembly and are periodically checked.

The shim rods are similar to the poison safety rods in their scram characteristics.

The worths of the individual rods will be estimated for each new critical experiment. Before the reactor is loaded, the worths of the control and shim rods will also be estimated and checked by reactivity measurements.

Since the reactor may have a complement of 40 safety, control, and shim rods, failure of one rod, or even several rods simultaneously, will not appreciably affect the fast shutdown of the reactor by scrambling the rods.

If the selector switch or the rod-gang-movement controls were to malfunction and all ganged rods were to be moved to insert reactivity, this additional reactivity would be less than that considered in the DBA (see Sec. X).

E. Reactor Bed and Tables

The reactor bed and tables rest on a 7-ft-thick pad of heavily reinforced concrete with a ridge beam on each side, 3 ft wide and at least 11 ft 10 in. thick. Six piling posts (three on each side) support the ridge beams. This substantial support minimizes deflections of the bed as the movable table supporting one reactor half is driven 7 ft during reactor startup or shutdown. The leveling jacks of the bed provide leveling adjustments, and the holddown bolts securely anchor the bed to the reinforced-concrete pad.

Measurements for levelness and flatness are made periodically to ensure proper attitudes for the support bearings of the movable table.

The fixed table is bolted to the bed, which precludes any movement of the table during operations.

The movable table rolls on 35 recirculating bearing assemblies, which are oiled to reduce friction. The drive ball screw is coupled to the movable table by a preloaded antibacklash ball nut. The ball nut is keyed to the table with a 4-in.-long, 1-in.-thick steel key and is bolted with four 1-in.-dia steel bolts. There is a minimum clearance between the ball nut and the table so that the key can only be removed by dismantling the ball nut or by lifting the table sufficiently to clear the key.

Since the holding bolts are installed from the top, they cannot loosen by themselves and fall out. The horizontal separation of the ball nut and the table would require shearing the key and the four steel bolts simultaneously.

The ball bearings of the ball nut would have to be removed before the nut would be able to slip along the ball screw. The retaining rings of the ball bearings are mechanically locked in place.

The hardened surfaces of the bedways, which support the roller bearings of the movable table, are wiped and kept clean by Neoprene wipers mounted on each end of the movable table.

The table-closure motors are interlocked so that only one drive motor at a time can be used. When the table separation reaches 18 in., limit switches in series stop the power to the fast-speed motor. If the two limit switches fail to operate and the motor continues to move the table, the hot-rail electrical pickup runs off the hot rail into air, removing power to the motor. The coasting of the movable table after the fast-speed motor was shut off was measured at the plant of the bed and tables manufacturer. The coast of the movable table traveling at 18.6 in./min was 0.046 in.

The hot-rail system is also used on the medium-speed drive motor.

In the event of a scram, the table-closure motors and clutches are automatically deenergized and the electric scram motor and clutch are energized and separate the reactor halves at a speed of 30 in./min. If the particular table-closure motor and clutch fail to deenergize or disengage, the electric-scram-motor system is strong enough to override the closure motor system and separate the halves as required.

The electric-scram-motor system is backed up by an emergency air motor. If the ball screw does not rotate in the "drive out" direction within 1 sec after the scram signal is given, the air-motor system is activated and drives the halves apart. With the air-motor system engaged, the output of the air motor is geared directly to the ball screw and will override any of the other drive systems.

Both the electric scram motor and the emergency-air motor are unidirectional to separate the halves.

The loading platform and the positions of the personnel shields are interlocked so the table cannot be moved unless the platform and the shields are in the fully retracted position.

The structure and drive system of the bed and tables are well constructed, and with the interlocks and safety features, present an easily operable, and above all, a safe support machine for the reactor matrix and controls.

F. Personnel Shields

The personnel shields are used primarily to reduce the fast-neutron dose to personnel working between the loaded halves of the reactor. The shield section can be separated for inspection of the reactor matrix. Shields of similar thickness and composition have been tested on the SEFOR critical assembly in ZPR-3.

The SEFOR critical assembly contained 19.7 kg of ^{240}Pu (6%) out of a total loading of 329 kg of plutonium. The RAPSODIE critical assembly, which was studied previously, had about 1.23 kg of ^{240}Pu . Measurements indicated that neutron dose between the halves in the SEFOR critical assembly was only nine times as great as the dose between the halves for the RAPSODIE, although the amount of ^{240}Pu was 16 times as great. This can probably be attributed to self-shielding due to positioning of fuel plates in a critical assembly.

The health-physics instruments indicated a field of about 130 mR/hr between the bare halves of the SEFOR critical assembly, which were 60 in. apart. The insertion of the shields in front of the halves reduced this field to about 4 mR/hr.

The separation of the halves in the ZPPR is 84 in. It is estimated that the dose rate would be not greater than 50 mR/hr for these large cores. Although this dose rate is not desirable, controlled personnel access to the high-flux area can be used to prevent excessive doses.

If experience on the larger core indicates the doses may become excessive, additional shielding can be added.

G. Fuel-can Rupture

The fuel for the ZPPR is encased in a 0.015-in., Type 304 stainless steel cladding. The fuel cans are checked as they are removed from storage and also rechecked if abnormal handling occurs during loading of the drawers. Special pieces of fuel are processed in the same manner.

The fuel is an alloy of U-Mo-Pu, which is nonpyrophoric and requires a temperature of about 600°C before ignition occurs in air. The oxidation rate at cell temperature (approximately 20°C) is nil.¹³

The cladding barrier and the nonpyrophoricity of the fuel preclude a major problem if a fuel can ruptures in the reactor. However, an illustrative analysis was made.

After the fuel drawers are loaded into the reactor core, the fuel continues to be monitored for alpha by detectors located in the reactor cooling-system ducts. AEC-type high-efficiency filters are interposed in the cooling system downstream from the alpha monitors.

Data from the filtration tests of AEC-type high-efficiency filters located downstream from the filter sands indicate a minimum attenuation of 200.²⁰ In these tests, the AEC-type high-efficiency filters were subjected only to the high-penetration aerosol passing through the sand. Therefore, the use of an attenuation of 200 for calculations is a conservative estimate of the capabilities of the AEC-type high-efficiency filters.

For calculational purposes, it was assumed that 1 kg of fuel ruptured and oxidized. Measurements made during the oxidation of plutonium indicate that 0.05 wt % becomes airborne.²¹ The amount of plutonium that becomes airborne is therefore 0.14 g (282.8×0.0005).

With an attenuation of 200, the amount of plutonium exhausted to the cell is 0.0007 g. The cell contains $58,000 \text{ ft}^3$ of air (1640 m^3), so that the concentration in the air is $4.3 \times 10^{-7} \text{ g/m}^3$. This yields the following isotopic concentrations in the air:

$$^{239}\text{Pu}: 3.7 \times 10^{-1} \mu\text{g/m}^3 \text{ or } 2.3 \times 10^{-8} \mu\text{Ci/cc};$$

$$^{240}\text{Pu}: 5 \times 10^{-2} \mu\text{g/m}^3 \text{ or } 1.1 \times 10^{-8} \mu\text{Ci/cc};$$

$$^{241}\text{Pu}: 7 \times 10^{-3} \mu\text{g/m}^3 \text{ or } 8 \times 10^{-7} \mu\text{Ci/cc}.$$

With a breathing rate of $3.47 \times 10^{-4} \text{ m}^3/\text{sec}$ and a bone retention factor of 0.2, the bone accumulation in 10 min by breathing the above concentrations would be:

	MPBB (Bone) (nonoccupational), μCi
$^{239}\text{Pu}: 1.5 \times 10^{-2} \mu\text{g} \text{ or } 9.6 \times 10^{-4} \mu\text{Ci}$	4×10^{-3}
$^{240}\text{Pu}: 2.1 \times 10^{-3} \mu\text{g} \text{ or } 4.7 \times 10^{-4} \mu\text{Ci}$	9×10^{-2}
$^{241}\text{Pu}: 2.9 \times 10^{-4} \mu\text{g} \text{ or } 3.3 \times 10^{-2} \mu\text{Ci}$	4×10^{-3}

If the release occurred in a short period of time, the response of the alpha monitors of the reactor cooling system would indicate a high-level alarm within 1 min. The low-level alpha monitor in the cell would alarm in about 5 min after the initial release. An alarm from any one of these three alpha monitors, i.e., the two alpha monitors in the reactor cooling system and the cell alpha monitor, would automatically isolate the cell ventilation system and allow personnel to evacuate before appreciable inhalation of air contamination.

In summary, there is no significant danger to personnel in the event of a fuel-can rupture in the reactor.

H. Reactor Cooling System

The heat generation in the reactor core comes primarily from spontaneous nuclear emissions in the fuel. A compilation of the isotopic heat generation rate per kilogram of basic plutonium fuel is as follows:

^{239}Pu : 1.59

^{241}Pu : <0.35

^{240}Pu : 0.79

^{238}Pu : <0.57

^{242}Pu : nil

<3.30 W/kg Pu

After two years, an additional 0.3 W/kg will be generated by americium from the decay of ^{241}Pu . This will increase as the fuel ages.

A representative 6000-liter core fueled with 3000 kg of plutonium will have 15,000 kg of depleted uranium, 7000 kg of steel, and 1450 kg of sodium. The heat capacity of this core is 19.5×10^5 cal/°C. Using a heat generation rate of 3.3 W/kg Pu and assuming no heat losses, 17.6 hr would be required to heat the core from 20 to 98°C (the melting point of sodium), and an additional 5.5 hr to melt the sodium. Another 90 hr would be required to bring the temperature of the core to 500°C, for a total of 113.1 hr.

The heat generation in the core is uniform so that hot spots would not be prevalent. Previous experience on a ZPR-3 plutonium loading²² indicates reasonable heat loss, especially with the halves separated.

These calculations indicate there would be ample time to unload the reactor before reaching significant core temperatures due to fuel heat-up, and therefore there is no particular operating hazard if the reactor cooling system fails.

Special test pieces of different isotopic composition will not add appreciably more heat to the systems, so that the problem of core heat-up due to these pieces is insignificant.

I. Power Failure

The loss of power would immediately scram the reactor. The safety rods would delatch, since they are held by electromagnets, and the scram springs would drive them to their least reactive position. The emergency air-scram-drive system of the reactor halves would be actuated by loss of power to the solenoid valve connecting the scram drive and the ballast tank. This would separate the halves at a speed of 30 in./min.

Channel 7 of the nuclear instrumentation contains an emergency battery power supply and remains actuated automatically to monitor reactor power. Portable, battery-operated, health-physics radiation detectors are also available for personnel use.

An emergency diesel-motor-driven generator is automatically started and engaged to provide power in event of an outside power

interruption. This is primarily a convenience to maintain airflows, cooling, and lights in the facility until normal electric power is again available. The facility is designed so that failure of the emergency power system will not cause unacceptable health hazards to the personnel.

J. Earthquake

The design of the reactor cell is based on two criteria: (1) Zone "2" earthquake susceptibility (0.2 g), and (2) ability to withstand a 75-psi internal cell pressure. The strength of the support structure needed to hold the earth mound and gravel-sand roof exceeds the strength requirement to withstand the earthquake effects in a Zone "2" area.

An earthquake detector is included in the reactor scram system and is actuated at an acceleration greater than 50 cm/sec^2 (0.05 g). As in all scrams, the safety rods remove reactivity within 0.3 sec and the reactor halves are separated. The wipers on the bedways remove any materials that might impede the movement of the reactor half.

The safety rods are protected by the reactor frame and the cooling plenum from falling objects so that the rods can scram satisfactorily.

The safety factor of the roof catenary cable is at least 2.5, so that failure of the roof is highly unlikely due to an earthquake. The matrix hold-down structure protects the core from falling objects.

The strength and design of the reactor and the cell structure preclude significant hazards due to an earthquake in this area.

K. Maximum Credible Accident

1. Description

The maximum credible accident in the ZPPR Facility is considered to be a fire in the core, since the core of the reactor will contain the highest concentrated accumulation of plutonium. Although unlikely, one might postulate, for example, the use of electrically heated samples in the core to set up the conditions for initiating a fire by either failure of the heating element or arcing due to a short circuit.

The fuel is encased in 0.015-in. stainless steel cladding, and penetration of the cladding would be necessary before the fuel could oxidize. Normally, electrical currents required for melting the cladding would trip the safety overloads on the electric switchgear, removing the primary source of energy to raise the temperature.

Other materials, such as sodium, could burn, but since they present no radioactive contamination hazards by themselves, primary attention will be concentrated on an accident involving fuel only.

2. Events

In the event of a plutonium fire, the alpha monitors would alarm, scram the reactor, and shut off the reactor cooling system. The securing of the cooling system precludes a ready supply of oxygen to the accident area. If the accident occurred when the reactor was shut down, the ventilation valves would automatically close, thus securing the isolation of the cell area. Venting of particulate formed would occur only through the gravel-sand roof and the backup filters of the containment structure.

The postulated case in which any appreciable amount of fuel might be involved is that in which Doppler measurements are being made in a heated section of the core. This might be as many as the central 25 drawers of the core, which would contain approximately 25 kg of plutonium, 125 kg of depleted uranium, and 12 kg of sodium.

The oxidation of the plutonium, uranium, and sodium is assumed to be the heat-energy input of the accident calculation. It is further assumed that no heat is lost to the surroundings and that all the heat is used to raise the temperature of the air in the cell.

Since there initially is insufficient energy to vaporize the materials, the oxidation rates are quite low. If one assumes that all the sodium is liquid, the oxidation rate is on the order of $0.1 \text{ (g/cm}^2\text{)/min}$.²³ Then the 12 kg of liquid sodium, if spread on the reactor pit floor, would burn within 1 min, raising the temperature of the air in the cell about 66°C . The rising temperature increases the pressure in the cell, causing the air to vent through the cell roof. The maximum pressure rise in the cell would be only 2.75 psi. At this pressure, the venting rate through the cell roof is about 1100 cfm or 0.56 ft/min superficial velocity.

Burning tests on uranium-plutonium-molybdenum samples by Baker²⁴ indicate that the exposed plates must be heated to $600\text{--}660^\circ\text{C}$ before oxidation will be self-sustaining. The plates, upon reaching this temperature, oxidize completely at a steady rate in about 50 min. If the plates are heated above 660°C and the external heat source is removed, the oxidation is self-extinguishing and the piece cools down. Since, the external heat source is removed in the proposed accident, it is assumed that fuel and depleted uranium burn at a steady rate and are completely oxidized in 50 min.

The oxidation of 150 kg of fuel and uranium in 50 min releases $2.9 \times 10^6 \text{ cal/min}$ of heat to the cell air. This heats the air in the cell at a rate of 7.2°C/min . The corresponding venting rate out of the cell is 1400 cfm, which requires a velocity of about 0.7 ft/min through the gravel-sand roof at the first layer of sand and about 0.45 ft/min at the upper layer of sand.

The airborne plutonium in this accident is only 12.5 g out of the 25 kg in the drawers. Taking the most conservative situation, in which all the airborne plutonium is subjected to the filtration of the gravel-sand roof and the high-efficiency filters (total minimum attenuation = 5×10^4),²⁰ only 250 μg of plutonium is released to the atmosphere.

The plutonium concentration level near the ZPPR control room during the release of 250 μg in 50 min was calculated to be $\chi = 152 \times 10^{-6} \mu\text{g}/\text{m}^3$ ($\chi/Q = 1.82 \times 10^{-3}$, $Q = 83.3 \times 10^{-3} \mu\text{g}/\text{sec}$). If a man was in the cloud during the entire release, he would retain by inhalation only $32 \times 10^{-6} \mu\text{g}$ of plutonium. (Breathing rate = $3.47 \times 10^{-4} \text{ m}^3/\text{sec}$, breathing time = 50 min, retention factor = 0.20.) This calculation is based on strong inversion and weak mixing conditions and would indicate lower doses if other atmospheric conditions were considered or if greater distances downwind were taken into account (see Sec. X).

Even if 3000 kg of plutonium burned, the inhalation dose at the control room, using the minimum acceptable attenuation of 5×10^4 for the combination of the gravel-sand roof and the backup containment structure, would be only $3.8 \times 10^{-3} \mu\text{g}$ of plutonium. The inhalation dose at the NRTS boundary under these conditions would not exceed the maximum permissible total body burden (nonoccupational).

The maximum credible accident presents no unacceptable risks to the personnel or surrounding facilities. Furthermore, no special evacuation is required. Such an accident would necessitate an extensive cleanup of the reactor cell and its equipment.

IX. SAFETY EVALUATION OF OTHER FACILITY AREAS

A. Fuel-storage Vault

The safety aspects of storing 3000 kg of plutonium require reasonable isolation of the fuel and adequate shielding. Criticality considerations are checked and approved by the ANL Criticality Hazards Control Committee before the fuel is placed into the ZPPR fuel-storage vault.

The fuel pieces (clad in stainless steel), which will eventually include more than 16,000 pieces, are kept in gastight canisters. Each canister containing approximately 3 kg of ^{239}Pu + ^{241}Pu weighs about 10 kg. The neutron source from each canister, due primarily to the ^{240}Pu content, is about 2×10^5 neutrons/sec. At present, similar quantities are processed during the operation of ZPR-3, and handling of the individual canisters by personnel results in an acceptable body dose.

Should the experimental program require appreciable power operation of the reactor, shielding from fission-product decay may be necessary to limit personnel exposures during handling of the fuel canisters.

The unshielded storage of large quantities of plutonium in the fuel-storage vault could produce an untenable neutron background in the vault. Therefore the canisters of fuel are stored in boron-impregnated concrete bins. Each bin has 12 steel storage tubes, which penetrate through the bin and are separated by a minimum of 8 in. of concrete. Each tube may contain two fuel canisters, end to end. The ends of the storage tubes have hinged, bolt-locked doors.

In essence, the fuel-storage vault is a closely aligned array of concrete columns with storage holes and access aisles, which effectively reduce the neutron background of the fuel and allow normal personnel access without excessive exposure.

The plutonium fuel generates heat by spontaneous nuclear disintegration so that each fuel-containing canister will produce 6-9 W of heat. The heat from the fuel is transmitted to the air in the room by conduction and convection. A direct-expansion air conditioner removes the heat from the storage vault. If the heat-removal system is inoperative, the earth mound surrounding the storage vault and the massive concrete storage structure provide a heat sink until the system is repaired. The well-spaced storage precludes any generation of hot spots; therefore no significant rise in stored fuel temperature is expected.

Administrative controls are used to minimize the presence of flammable materials in the fuel-storage vault.

The use of airtight canisters for storage of fuel and the isolation of these canisters by the concrete holders provide a reasonable minimal safety hazard to operating personnel in the fuel-storage vault.

B. Workroom

1. Fuel Handling

The fuel, as it is brought from the vault in airtight canisters, is placed in an inspection hood or table near the entrance of the vault. The inspection hood is a suspect area, and the air is drawn from the room into the hood and is exhausted from the hood to the outside of the mound area through AEC-type high-efficiency filters. The fuel is inspected and checked for alpha contamination before being removed from the inspection hood.

After inspection, the fuel is transported to the loading hood, where it is loaded into matrix drawers. The loading hood is also treated as a suspect area, and room air is drawn into the loading hood and exhausted through AEC-type high-efficiency filters to the atmosphere outside the mound area.

The loaded drawers are transported into the reactor cell to be placed into the reactor matrix.

The handling of fuel in suspect inspection hoods and in the loading hood precludes any appreciable inhalation hazard to personnel. Should direct exposure to the fuel result in unacceptable radiation doses to parts of the body, shielding will be provided to minimize this hazard.

2. Control of Fissile Materials

The accountability of fuel is maintained by Special Materials Representative assigned to the facility. This Special Materials Representative removes the fuel from the vault and inspects the fuel before releasing the plutonium-alloy pieces to the operating personnel who load the fuel into the reactor core drawers. Both the Special Materials Representative and reactor operations personnel check to ensure that the allowable inventory of fuel at the loading table is not exceeded.

Other core and blanket simulants are color-coded and can be easily recognized. The operations personnel are periodically checked for color perception to ensure color-discrimination capabilities.

The inventory of sodium in the workroom is kept at a minimum, even though the sodium is clad in stainless steel. When not in use, the sodium is stored in an external building close by, but outside, the mound area.

Other materials that might be combustible, such as paraffin and polyethylene, are under administrative control and also kept at a minimum.

Boxes or crates used to transport nonfissile core and blanket simulants are removed as soon as practical. These also are under administrative control.

3. Accidental Hazards

The primary accidental hazard involved in the workroom is a fire. The probability of the ignition of materials is very low since combustibles are carefully controlled and reactor simulants that could ignite spontaneously are clad in stainless steel. Nevertheless, an electrical short circuit in a motor is possible, and therefore a minor incident is possible. The ventilation system for the mound area is secured automatically when a fire alarm is sensed in the workroom. The emergency exhaust valve for the emergency sand filter is opened. The three exits from the workroom allow personnel to rapidly evacuate the area.

The involvement of fuel in a fire has even less probability. However, for illustrative purposes only, a fire involving 10 drawers is considered. The 10-drawer unit was selected because this is the largest anticipated transportation cask for movement of drawers between the workroom and the cell.

The 10 drawers contain approximately 10 kg of plutonium in the form of alloy, 50 kg of depleted uranium, partially in the form of the alloy and partially as unclad plates, and 6 kg of sodium canned in stainless steel. The workroom, vault, and inside equipment room are connected through the ventilation system so that the volume of air used for calculation is 54,000 ft³ at about 300°K and 12.5 psia. The heat capacity (C_p) of the air is 3.97×10^5 cal/°C, since the air flows out of the area during the process.

The calculated heat-generation rate* of the burning fuel and depleted uranium is 1.32×10^6 cal/min (1.2 kg/min), and of the sodium is 6.7×10^5 cal/min (307 g/min), for a total of 1.99×10^6 cal/min. Assuming no heat losses and a pressure-rise rate of 0.208 psi/min, the rate of temperature rise in the area is therefore 5° C/min. The venting rate required to release pressure in the workroom vault area is calculated to be 765 cfm at 300°K and 14.7 psia or a superficial velocity of 0.51 ft/sec through the sand filter. This would require a pressure drop of 0.93 psi, so that the maximum pressure in the vault-workroom area would be less than 1 psig.

The emergency sand filter can easily withstand 1.9 psig without disrupting the sand bed. The concrete structure backed by the sand-and-earth

fill also is sufficient to withstand the calculated pressure. The seal doors are rated at 12.5 psig and tested at 5 psig, so that venting through the emergency filter is ensured.

Only 0.05 wt % of the plutonium, or 5 g out of the total of 10 kg, would become airborne. Plutonium release calculations from the stack show that an attenuation factor of 40 for the emergency filter will result in an acceptable plutonium dose (0.1 mpbb) at 150 m. Since the HEPA filter alone provides an attenuation of 200 for plutonium aerosols,²⁰ the mound-area emergency ventilation system is more than adequate to accommodate this postulated workroom accident. The exhaust stack is 75 ft high and the duration of a fire would be about 50 min, so that the vented plutonium would be distributed.

Although a fire would contaminate the workroom area, there would be no unacceptable hazard to the public.

Assuming that one drawer containing 1 kg of plutonium burns in the workroom in 50 min (20 g/min of plutonium) only 0.01 g/min of plutonium would become airborne. Assuming uniform plutonium dispersion in the workroom air, and no ventilation, the plutonium concentration would be sufficient to set off the plutonium monitor alarm within 2 sec. More realistically, allowing up to 30 sec for the plutonium monitor to alarm and an additional 30 sec to evacuate, the following doses by inhalation would be received by personnel in the workroom:

	MPBB (Bone) (nonoccupational), μCi
^{239}Pu : 0.034 μg or $2.1 \times 10^{-3} \mu\text{Ci}$	4×10^{-3}
^{240}Pu : 0.0045 μg or $10^{-3} \mu\text{Ci}$	4×10^{-3}
^{241}Pu : 0.00062 μg or $7.1 \times 10^{-2} \mu\text{Ci}$	9×10^{-2}

These doses assume a breathing rate of $3.47 \times 10^{-4} \text{ m}^3/\text{sec}$ and a bone retention factor of 0.2. With the ventilation system operating and recirculating filtered air, the inhalation doses would be significantly smaller.

C. Other Areas

Since only minor amounts of material are handled under administrative control in the support wing and all suspect areas are exhausted through AEC-type high-efficiency filters, no unacceptable hazard is involved in these areas.

X. THE DESIGN BASIS ACCIDENT (DBA)

A. Introduction

Even though an excursion leading to core damage is considered incredible in the ZPPR reactor design, a study has been made to estimate the consequences of a hypothetical startup accident.

The ZPPR design allows a fast speed of 18 in./min until the core separation is 18 in., an intermediate speed of 6 in./min to a separation of 3 in., and a slow speed of 1/6 in./min to the final closure. The use of a hot-rail system on the ZPPR bed and tables drive system prevents over-run of these speeds past the above control points.

Operation of the reactor is administratively controlled so that the reactivity insertion rate will never exceed 2 β /sec at any speed. For the ZPPR, this 2 β rate would be possible in small cores at high and intermediate table speeds. The function of ZPPR is to study the physics of cores in the range of 1000-6000 liters, in which the reactivity insertion rates vary from 50 to 28 cent/ β /sec at high and intermediate speeds.

A fuel overloading of about 100% would be required for criticality to be possible when the halves are 18 in. apart. At increased separations, the overloading necessary for criticality increases correspondingly. More than a 30% fuel overloading would be required for criticality to be possible in the intermediate-table-speed range. Mandatory procedures on approach to criticality with new core loadings or with significant changes to existing core loadings render this management failure extremely unlikely. Even then, core damage is still conditional upon complete failure of all safety circuits or reactor-scrum equipment.

The cases considered in the DBA are cores that have been overloaded to the extent that delayed, and then prompt, criticality would be achieved in either the fast- or intermediate-speed ranges in the absence of any safety circuits.

B. Nuclear Calculations

In Ref. 25 two metal-fueled cores, 1000 and 6000 liters, with length-to-diameter ratios of 0.5, were used for the DBA calculations. Metal cores rather than simulated ceramic cores were considered because the lower Doppler feedback coefficients make them slightly more conservative. The higher leakage associated with the ceramic cores would make the reactivity insertion rates up to 30% greater, but this will not significantly affect the results when the increased Doppler coefficient is taken into account. The L-to-D ratio affects the insertion rates, but only slightly. For very small gaps ($d \ll D$), the flatter core would have smaller insertion rates. As the

gap size increases, however, the effect of increased leakage area begins to gain on the opposing effect of reduced escape angles, and eventually a cross-over point is reached. In the range 3-18 in., corresponding to the intermediate table speed, the flatter core would have about a 10-20% lower insertion rate in the 6000-liter size and about 5% lower in the 1000-liter size. Beyond 18 in., the differences decrease rapidly and eventually change size. It will be shown later in these discussions that more than a 50% increase in reactivity insertion rate can be tolerated in the large core.

The gap worths for the 1000- and 6000-liter metal cores were estimated using the two-dimensional transport-theory code SNARG¹⁶ in three energy groups. The results are shown in Fig. 25. At the designed table speeds, the maximum reactivity insertion rates are about 45 cent/sec for the 1000-liter core in the intermediate-speed range, and 20 cent/sec for the 6000-liter core in the fast-speed range.

Fig. 25
Reactivity Worth of Gap in ZPPR
for Two Metal-fueled Cores. ANL
Neg. No. 103-A11694.

Figures 26-29 show the fuel worth and power along the axial axis for different gap sizes for the 1000- and 6000-liter metal cores as calculated from three-group, two-dimensional transport theory. Molten fuel flowing into the gap would not maintain its same horizontal position, but would fall to the reactor floor, which is a point of much lower worth. All the gaps correspond to the intermediate speed range of the table drive. Curves for the fast speed range would be close to those for infinite separation of the halves. The point of maximum fuel worth corresponds, at all times, to the point of maximum power.

In Ref. 25 the model used in the DBA studies was an adiabatic one that did not allow for heat conduction out of the fuel into the diluents.

Fig. 26. Fuel Worth and Power Distribution:
1000-liter Core, 10-cm Gap. ANL
Neg. No. 103-A11687 Rev. 1.

Fig. 27. Fuel Worth and Power Distribution:
1000-liter Core, 20-cm Gap. ANL
Neg. No. 103-A11691.

Fig. 28. Fuel Worth and Power Distribution:
1000-liter Core, 40-cm Gap. ANL
Neg. No. 103-A11685 Rev. 1.

Fig. 29. Fuel Worth and Power Distribution:
6000-liter Core, 10-cm Gap. ANL
Neg. No. 103-A11686 Rev. 1.

A computer code has been developed for DBA studies in Argonne's zero-power fast reactors. In essence, it is a transient heat-transfer code in semi-infinite slab geometry, coupled to a modified version of the space-independent neutron-kinetics code, R101. Quasi-space dependence is introduced by the use of spatial zones over which the relevant physical quantities can be represented adequately by average values.

The current CDC-1604 version of the ZPR accident code assumes that each core zone has the same characteristic heat-transfer cell, i.e., the same plate arrangement in each core drawer. Use of a larger machine than the CDC-1604 would permit greater generalization. Since the local temperature gradients along the axis parallel to the plane of the plates are much less than across the axis perpendicular to the plates, the use of one-dimensional slab geometry is considered adequate.

Each material in the cell is allowed up to 10 temperature ranges over which the heat-transfer properties are represented by average values. The two latent heat ranges (fusion and vaporization) can be included as ranges. As a coding artifice, the specific heats in the latent heat ranges are designated very high values ($>10^6$ cal/g) in order to maintain a constant temperature during the change-of-state process. The temperature boundaries of the ranges do not have to be the same for each material.

The temperature of node N in plate J at the end of a time step is related to the temperature at the start of the time step by

$$T'(J, N) = \frac{1}{M(J)} [T(J, N-1) + T(J, N+1)] + \left[1 - \frac{2}{M(J)}\right] T(J, N) + G(J),$$

where

$G(J)$ = temperature rise due to heat production in the plate (during time step) and $M(J)$ is a coefficient given by

$$M(J) = \frac{c(J)\rho(J)\Delta x(J)}{k(J)\Delta t},$$

where

$c(J)$ = specific heat of material in plate J,

$\rho(J)$ = density of material in plate J,

$k(J)$ = thermal conductivity of material in plate J,

$\Delta x(J)$ = node interval in plate J,

and

Δt = time step.

The temperature equation is modified when the node is a boundary node. For example, the temperature of the last node in the J plate is given by

$$T'(J, N) = \frac{2N(J)H(J+1)}{M(J)} T(J+1, 1) + \frac{2T(J, N-1)}{M(J)} + \left\{ 1 - \frac{2}{M(J)} [1 + N(J)H(J+1)] \right\} T(J, N) + G(J),$$

where

$$N(J) = \frac{\Delta x(J)}{k(J)}$$

and

$H(J+1)$ = boundary conductance between plates J and J + 1.

For mathematical stability,

$$\Delta t \leq \frac{c(J)\rho(J)[\Delta x(J)]^2}{2k(J)[1 + \Delta x(J)H(J)/k(J)]}$$

The accident code proceeds as follows:

Until sensible heating has taken place, the neutron-kinetics part operates alone, the rising power being a function of the reactivity ramp rate, the delayed-neutron characteristics, and the lifetime of the reactor. When the energy has reached a point where sensible heating starts (specified as input), the code goes into the heat-transfer routine at specified energy increments after that. The time taken to realize the energy increment is divided into a number of time steps, so that each step is less than the minimum time step required for stability as given by the properties of the first temperature range. Initially, a specified air gap is assumed between every plate, and the equivalent boundary conductance, k , is derived for this gap. As the core materials heat up, the expansion of the plates is assumed to decrease the air gap until it is zero. At this point, a fixed boundary conductance is introduced.

At the end of each time step, the code tests each average plate temperature to see if the next range of constants should be used and, if so, whether it is one of the latent heat ranges. Once a plate is in its latent heat-of-fusion range, the code tests after each time step to determine how much heat has been added since the change of state started, and when this added heat equals the latent heat of fusion, the next range is used. Before the calculation goes back into the neutronic routine at the end of the specified integrated energy interval, the change in reactivity Δk due to the temperature-induced expansion and Doppler coefficients in each spatial zone is computed

and summed to be applied as a reactivity step. Once a material has entered the liquid stage; no further expansion coefficient is attributed to it. A T^{-1} or a $T^{-3/2}$ temperature dependence is assumed for the Doppler feedback.

The shutdown mechanisms for the accident can be flow of molten fuel out of the core and/or vaporization of the fuel. In the former case, a simple hydraulic-flow model is assumed, taking into account the hydraulic height of the molten fuel in the drawer and the loss in worth of the molten fuel as it leaves the core. This shutdown mechanism is allowed to take place when a certain threshold temperature is achieved in the fuel. This could be, for example, when the fuel temperature reached the melting point of the cladding. If the fuel vaporizes, the reactivity effect of movement of fuel vapor out of the core is taken into account.

At a specified frequency, the code will print out the average temperature of each plate in the cell for each core zone. When required, the code will also print out the temperatures of each nodal point in the cell for each zone. Frequent printout of the neutronic parameters of power, energy, period, k_{eff} , and reactivity-feedback components is also given. Each time a material changes its thermal-property range, identification of the plate, zone, and range number is printed out. When boiling occurs, printout of the amount boiled off is printed at the same rate as the neutronic parameters.

The excursion model is, of course, far from realistic in the later stages where rapid attack of the stainless steel cladding by high-temperature fuel is inevitable. The purpose of the somewhat idealistic model used in the DBA calculations is to illustrate how far the excursion could be allowed to continue beyond the point where rapid shutdown would have occurred (1700°K or less), before stored heat in the core could boil off unacceptable quantities of sodium vapor.

The arrangement of materials in the drawers for the heat-transfer calculations is shown in Fig. 30. A sodium plate is positioned on each side of the fuel plate to present the most conservative conditions. In a simulated

Fig. 30

Heat-transfer Models for
DBA Calculations. ANL
Neg. No. 103-A11703.

ceramic-fueled core, the most likely neighboring materials to the fuel plates would be graphite, sodium carbonate, or iron oxide. The situation in which the sodium plates have been taken out of the core, as in the sodium-void experiment, will be considered later in this discussion.

Kato et al.²⁶ have reviewed the measured data on boundary conductances between plane interfaces as a function of pressure. They conclude that a heat-transfer coefficient, H , of 1500 Btu/hr-ft²-°F is a reasonable value between plates in ZPR drawers. In the transient-heat transfer code, the minimum time step for mathematical stability of the finite-difference temperature equations is dependent on the maximum value of H in any part of the cell. When the fuel or sodium has become molten, wetting of the boundary increases the boundary conductance in those plates to a value of about 100,000 Btu/hr-ft²-°F. Unfortunately, however, a value of $H = 100,000$ would make the computer running time impractical, and so an approximate resistance-sharing method has been used. For example, when the adjacent fuel and sodium are molten, the boundary conductances across the fuel- and sodium-cladding interfaces would be 100,000 Btu/hr-ft²-°F. If X is the contact conductance between the two clad surfaces, then the total resistance due to boundaries between the fuel and the sodium is

$$R_1 = \frac{1}{100,000} + \frac{1}{100,000} + \frac{1}{X}.$$

If Y is the value of H we assume at each boundary, then

$$R_2 = \frac{1}{Y} + \frac{1}{Y} + \frac{1}{Y}.$$

Equating R_1 with R_2 , the equivalent value of X is $1/(3Y^{-1} - 0.00002)$. Two values of 3700 and 7400 Btu/hr-ft²-°F were used in the calculations giving equivalent X 's of 1264 and 2595. These values bracket the Kato et al. selected value of 1500.

The above resistance-sharing method would give the correct temperatures if the heat capacities of the claddings were negligible. The effect of the finite cladding capacities causes an error of less than 50°K in the sodium at fuel temperatures in the later stages of the accident, with somewhat higher errors in the cladding temperatures.

The other thermal data used in the heat-transfer calculations are given in Table VII. Because of the nonexistence of data at high temperatures, many of the values are guessed extrapolations. In the change-of-state regions, the specific heats have been assigned values of 10⁷ cal/g, but again this is purely a coding artifice to keep the temperature constant until the amount of heat absorbed equals the latent heat.

TABLE VII. Selected Thermal Data for Reactor Materials

Range	Upper Temp, °K	Heat Capacity, cal/g-°C	Thermal Conductivity, cal/sec-cm ² -°C	Latent Heat, cal/g	Range	Upper Temp, °K	Heat Capacity, cal/g-°C	Thermal Conductivity, cal/sec-cm ² -°C	Latent Heat, cal/g
Fuel Alloy					Stainless Steel				
1	500	0.03	0.064	-	1	500	0.121	0.040	-
2	650	0.035	0.074	-	2	700	0.134	0.045	-
3	818	0.037	0.082	-	3	900	0.141	0.051	-
4	900	0.039	0.090	-	4	1100	0.147	0.057	-
5	1230	0.040	0.110	-	5	1300	0.160	0.063	-
6	1230	10 ⁷	0.120	20	6	1500	0.175	0.069	-
7	1900	0.044	0.160	-	7	1670	0.190	0.075	-
8	2800	0.046	0.240	-	8	1670	10 ⁷	0.079	50
9	3700	0.048	0.300	-	9	2500	0.250	0.080	-
10	-	0.050	0.300	450	10	-	-	-	-
Sodium					Depleted Uranium				
1	370	0.338	0.212	-	1	500	0.03	0.064	-
2	370	10 ⁷	0.207	27.05	2	650	0.035	0.074	-
3	500	0.325	0.200	-	3	818	0.037	0.082	-
4	650	0.312	0.184	-	4	900	0.039	0.090	-
5	750	0.309	0.169	-	5	1400	0.040	0.110	-
6	850	0.301	0.157	-	6	1400	10 ⁷	0.120	20
7	950	0.300	0.146	-	7	1900	0.044	0.160	-
8	1050	0.301	0.134	-	8	2800	0.046	0.240	-
9	1155	0.304	0.120	-	9	4200	0.048	0.300	-
10	-	10 ⁷	0.115	1005	10	4200	10 ⁷	0.300	450

The reactivity-feedback coefficients for expansion and Doppler used in the DBA are given in Table III.

The assumption is made that the excursion is stopped by movement of fuel liquid or fuel vapor to regions of lower importance, either to the rear of the core halves or out through the front face, or both. Since the axial power peak is much closer to the front face of the half core, then the fuel will quickly flow into the gap between the halves once the fuel cladding has been destroyed.

If the fluid fuel has not already destroyed the cladding before its temperature reaches the melting point of the cladding (1675°K), then it will destroy the cladding very rapidly at and above that temperature. Measurements by Savage²⁷ indicate just how rapidly molten fuel alloys attack stainless steel at temperatures around 1673°K. Rates of 1 in. or more per second are evident. In the time scale of the accident, significant penetration of the cladding will not take place at the eutectic formation temperature.

More or less simultaneous destruction of the front end of the fuel cladding and drawer will make an easy path for fuel flow into the gap between the halves. If the molten eutectic has also destroyed the sodium cladding before the fuel starts to flow, then the resulting sodium vapor pressure produced by the fuel-sodium contact would rapidly accelerate the displacement of the liquid fuel. Further sodium vaporization would be slowed down by vapor barriers and displacement of the heat source.

In the excursion model, the fuel is mathematically allowed to flow out when a certain trigger temperature is reached by the fuel. The flow rate

assumed is that through a 1/4-in.-dia hole in the drawer front due to a hydrostatic head of fuel in the core of about 4 cm. Neglecting friction, this amounts to 0.52 kg of fuel per second per drawer. Zones 1 of the 1000- and 6000-liter cores have fuel reactivity worths (relative to the flow) of 6.2 and 2.2 cent/kg, respectively. The corresponding number of drawers in the zones are 88 and 456, respectively. Thus, the reactivity shutoff rates due to fuel flowing from Zones 1 of the two cores are 2.8 and 5.1 \$/sec, respectively.

Figure 31 shows the average core neutron density and the excess reactivity in the 6000-liter core with 20-cent/sec ramp rate corresponding to the fast table speed. It is hypothesized that a gross management error coincidental with complete safety-circuit failures caused the condition of supercriticality to be achieved when the gap between the halves was greater than 18 in.

Fig. 31. 20-cent/sec Excursion in 6000-liter Metal-fueled Core. ANL Neg. No. 103-A11695.

As usual, the initial prompt critical excursion is turned around by the first sensible heating in the fuel-alloy plates from Doppler and expansion feedback. Continuation of the 20-cent/sec ramp rate raises the temperature of the fuel until the melting point is reached. At this point, the expansion coefficient of reactivity for the fuel is reduced to zero and further Doppler feedback is postponed until the melting is complete. Since the calculations are done on a zonal basis and the depleted-uranium Doppler is still active, there is no time during the accident when some negative feedback is not active. The dashed curve in Fig. 31 refers to the case in which the reactivity feedback coefficients have been arbitrarily reduced by one-half in the 6000-liter core. When the fuel reaches an average temperature in the zone of 2000°K, the fuel flowing out gives a negative ramp rate of 5.1 \$/sec, which reduces the power level by a factor of about 25 in the first second.

The fuel and sodium temperatures for the feedback coefficients as calculated are shown in Fig. 32. Two curves are shown here, one with a boundary conductance of 3700 Btu/hr-ft²-°F and the other with twice that

Fig. 32. DBA in 6000-liter Metal-fueled Core for 20-cent/sec Feedback Coefficients as Calculated. ANL Neg. No. 103-A11698 Rev. 1.

value. In the first case, the sodium in Zone 1 begins to boil about $3/4$ sec after fuel flow has started to shut down the excursion. Assuming the heat transfer model is still valid, then the total amount of sodium boiled off would be about 6 kg. For higher values of H , the sodium starts to boil about $1/10$ sec after the shutdown mechanism operates and the fuel temperature is at its peak of 2120°K . The amount of sodium boiled off under these conditions approaches 30 kg.

A trigger temperature of 2000°K is very pessimistic, considering Savage's experiments. If the mechanism operates at the melting point of the fuel cladding (1673°K), then the peak fuel temperature would be only about 1700°K and the stored heat would not, under the assumptions of the model, boil off any sodium. Fuel eutectic contact with sodium would boil off some sodium, but probably not very much.

Figure 33 shows the fuel and sodium temperatures and the sodium boiling rate for the 6000-liter core with the feedback coefficients reduced by one-half. A 2000°K trigger temperature causes no boiling, whereas a 2500°K trigger causes the boiling-off of just over 30 kg of sodium from Zone 1. A further calculation was made on the 6000-liter core with a 50% increase in the ramp rate, 30 cent/sec. The temperatures and boiloff rates are shown in Fig. 34. In this case, only a few kilograms of sodium are boiled off, mainly because the sodium is well below the boiling point when the shutoff mechanism starts.

Figures 35 and 36 describe the calculated results of the 45-cent/sec accident in the 1000-liter metal core for various values of trigger temperatures and feedback coefficients. The worst case of sodium boiling is about 30 kg or more for a 2500°K trigger temperature at half the calculated coefficients, both conditions being pessimistic.

It will be shown in Sec. C below that the above sodium boiling rates do not jeopardize the ZPPR containment. Since the fuel is unlikely to stay in its cladding at temperatures above 1673°K , the amounts of sodium boiled off will be much less than calculated above.

Calculations have also been made to see if operation of the high-level trips could prevent damage to the core in the startup accident. It was assumed that 80 msec after prompt critical, a shutdown reactivity of 1.5% was applied in 205 msec. This was followed by a high rate loss of reactivity due to the halves moving apart. In the 6000-liter, 20-cent/sec accident with feedback coefficient halved, the maximum fuel temperature is only 450°K ; in the 1000-liter core, with 45 cent/sec and again only half the coefficient, the maximum fuel temperature is only 480°K . In neither case will core damage result.

One of the standard measurements in large, dilute ZPPR cores will be that of the sodium-void coefficient. This measurement is performed by

Fig. 33. DBA in 6000-liter Metal-fueled Core for 20-cent/sec Feedback Coefficients Halved. ANL Neg. No. 103-A11692 Rev. 1.

Fig. 34. DBA in 6000-liter Metal-fueled Core for 30-cent/sec Feedback Coefficients as Calculated. ANL Neg. No. 103-A11693 Rev. 1.

removing all or some sodium plates from all or some drawers in the core, and replacing them by empty cans having the same mass of stainless steel as the sodium cans. If both sodium plates adjacent to the fuel plates in Fig. 30 were removed, then the hydraulic height of the fuel melt after destroying its cladding would only be 1.67 cm. This is probably still adequate to force fuel out of the core at a rate that will shut off the excursion. If it is not adequate, then the ultimate shutdown will be vaporization of fuel as in the adiabatic case in Ref. 25. The fuel-vapor release rates for the two cores are given in Tables VIII and IX. As will be shown in Sec. C below, these rates do not jeopardize the containment.

TABLE VIII. Calculated Cell Overpressures Caused by Heat Released from Oxidation of Vaporized Fuel during DBA While Performing a Sodium-voided Core Experiment; 6000-liter Metal Core; 20-cent/sec Reactivity Insertion Rate; 11.5% Release of Cell Air to Atmosphere

Time after Vaporization Begins, sec	Total Fuel Vaporized, kg	Cell Overpressure, psig	Time after Vaporization Begins, sec	Total Fuel Vaporized, kg	Cell Overpressure, psig
0	0	0	10	128.0	4.68
1	52.2	6.91	11	132.0	4.04
2	65.8	7.63	12	136.0	3.52
3	75.8	7.71	13	139.6	2.98
3.2	77.8	7.713	14	142.2	2.56
4	85.5	7.78	15	145.0	2.23
5	94.5	7.52	16	147.4	1.90
6	103.2	7.14	17	149.8	1.61
7	111.5	6.63	18	152.0	1.39
8	118.0	6.00	19	154.2	1.22
9	123.5	5.30	20	155.6	1.04

NOTE: 28.8% of air was left in the cell at end of 20 sec.

TABLE IX. Calculated Cell Overpressures Caused by Heat Released from Oxidation of Vaporized Fuel during DBA While Performing a Sodium-voided Core Experiment; 1000-liter Metal Core; 45-cent/sec Reactivity Insertion Rate

Time after Vaporization Begins, sec	Total Fuel Vaporized, kg	Cell Overpressure, psig	Time after Vaporization Begins, sec	Total Fuel Vaporized, kg	Cell Overpressure, psig
0	0	0	6	38.2	3.1
1	21.8	3.0	7	39.8	2.9
2	27.5	3.4	8	41.1	2.6
3	31.3	3.53	9	42.2	2.4
4	34.1	3.5	10	43.2	2.2
5	36.2	3.3			

When only one sodium plate is removed from an adjacent position to the fuel plate, then on cladding failure the hydraulic height of the fuel melt would be 2.5 cm. This would cause a flow rate sufficient to shut down the excursion, and the sodium boiling rates in the idealistic model would be about halved, since one plate has been removed.

C. Venting Characteristics of the Reactor Cell during DBA Conditions

The pressure-time relationship of the ZPPR cell has been calculated for selected cases of the DBA's. A computer program for the IBM 1620 has been developed and used in describing the venting characteristics of the cell's gravel-sand roof.

The basic input to the program is as follows:

1. Cell volume: 58,000 ft³
2. Initial cell pressure: 12.5 psia
3. Initial cell air temperature: 300°K
4. Initial heat capacity of the air: 4×10^5 cal/°K
5. Initial inventory of cell air: 6.24×10^4 g-moles
6. Venting rate of lower layer of sand: 947 g-moles/sec-psi (across the layer)
7. Venting rate of upper layer of sand: 1010 g-moles/sec-psi (across the layer)
8. Heat input from oxidation and condensation of fuel or sodium
9. The gas relationship: $PV = NRT$.

The incremental time for each calculation is 0.1 sec. The temperature rise of the air is computed from the heat input due to oxidation and condensation and the heat capacity of the air in the cell. This temperature rise is then converted to a pressure rise assuming no venting during the time increment. The average cell pressure is computed, and the venting of gases during the time period is calculated. A material balance of the cell is made to obtain the revised quantity and heat capacity of the remaining air. The separation of the two layers of sand is considered in the calculation by an iteration process on the cell pressure, the pressure between the two layers of sand, the respective venting rates, and the inventory of gas between the two layers of sand.

The heat release to the air from the burning of uranium or plutonium vapors and their subsequent condensation is obtained by the summation of the heat of vaporization of uranium or plutonium, 450 cal/g, and the heat of formation of the oxide (UO₂ or PuO₂), 960 cal/g (total = 1410 cal/g). The total heat released to the air per kilogram of fuel vaporized is therefore 1.41×10^6 cal. The heat release used for calculation from the oxidation of sodium vapor to Na₂O is that given in the Chemistry and Physics Handbook (47th edition, 1966-1967, p. D-26), which is essentially the heat of oxidation of the vapor and the condensation of the oxide. The heat release to the air per kilogram of sodium vaporized is 3.4×10^6 cal. The calculations use

the pessimistic assumption of no heat loss to the cell surroundings. Further, it is assumed that all the vaporized fuel or sodium is instantaneously mixed with sufficient air for rapid, complete combustion.

For simplicity, the venting rate is assumed linear with pressure drop. The flow characteristics of a 12-in.-deep sand bed are shown in Fig. 37. An examination of flow characteristics and the maximum pressures generated in the DBA's indicates a spread of about 10% between the calculated pressures, assuming linearity and the calculated pressures using the detailed curve. This is within the spread of data usually obtained from pressure drop-flow measurements through random packed beds.

Fig. 37. Pressure Drop vs Flow for a 12-in.-deep ZPPR Sand Bed. ANL Neg. No. 103-A11701.

The calculated results of the DBA using the normal feedback and heat-transfer coefficients for a 6000-liter metal core with a 20-cent/sec reactivity insertion rate as described in Sec. B above are given in Table X. The maximum cell overpressure is 2.06 psig and occurs about 1.4 sec after vaporization of sodium starts. At the end of 10 sec, the pressure in the cell is 0.77 psig. The total amount of sodium vaporized is only 6.8 kg and occurs in 1.7 sec. The gross rupture of sodium cans is not expected since there is not enough energy to create appreciable sodium pressure. However, even if sufficient sodium cans were ruptured and liquid sodium covered the entire reactor pit area, the burning rate of sodium liquid would only be 0.8 kg/sec. Coupling this pool burning of sodium with the DBA does not increase the peak pressure of 2.06 psig. At the end of 20 sec, the cell contains 75% of the original inventory of air.

TABLE X. Calculated Cell Overpressures Caused by Heat Released from Oxidation of Vaporized Sodium during DBA Using Normal Feedback and Heat-transfer Coefficients (6000-liter metal core, 20-cent/sec reactivity insertion rate)

Time after Vaporization Begins, sec	Total Sodium Vaporized, kg	Cell Overpressure, psig	Time after Vaporization Begins, sec	Total Sodium Vaporized, kg	Cell Overpressure, psig
0	0	0	2.0	6.8	1.95
0.2	1.8	0.63	3	6.8	1.74
0.4	3.3	1.14	4	6.8	1.55
0.6	4.5	1.54	5	6.8	1.38
0.8	5.3	1.78	6	6.8	1.23
1.0	6.0	1.98	7	6.8	1.10
1.2	6.3	2.04	8	6.8	0.98
1.4	6.5	2.06	9	6.8	0.87
1.6	6.7	2.04	10	6.8	0.77
1.8	6.8	2.00			

NOTE: 89% of the air was left in the cell after 10 sec.

Parametric studies of the DBA and the effects of varying the coefficients were determined. An example of the calculated cell overpressure is shown in Table XI, where the heat-transfer coefficient to the sodium cans is doubled. Other coefficients and the reactivity insertion rate of 20 cent/sec are kept the same for this variance of the DBA. The calculated maximum cell overpressure is 8.55 psig and occurs at 1.4 sec after sodium vaporization starts. At the end of 20 sec, the cell overpressure is 1.08 psig and 43% of the original air remains in the cell.

TABLE XI. Calculated Cell Overpressures Caused by Heat Released from Oxidation of Vaporized Sodium during DBA Using Normal Feedback Coefficients and Twice the Normal Heat-transfer Coefficient (6000-liter metal core, 20-cent/sec reactivity insertion rate)

Time after Vaporization Begins, sec	Total Sodium Vaporized, kg	Cell Overpressure, psig	Time after Vaporization Begins, sec	Total Sodium Vaporized, kg	Cell Overpressure, psig
0	0	0	1.0	24.6	7.95
0.1	3.5	1.24	1.1	25.7	8.20
0.2	6.5	2.29	1.2	26.6	8.37
0.3	9.5	3.32	1.3	27.5	8.54
0.4	12.5	4.33	1.4	28.0	8.55
0.5	15.5	5.33	1.5	28.3	8.50
0.6	18.0	6.12	1.6	28.6	8.46
0.7	20.0	6.72	1.7	28.7	8.34
0.8	22.0	7.31	20	28.7	1.08
0.9	23.4	7.68			

The estimated cell overpressures are shown in Table XII for the DBA variation in which the feedback coefficients are halved and a trigger temperature of 2500°K is used. At a time lapse of 2.1 sec after vaporization

begins, the peak pressure is 8.38 psig. At the end of 20 sec, the pressure is 0.33 psig and 53.7% of the air remains in the cell.

TABLE XII. Calculated Cell Overpressures Caused by Heat Released from Oxidation of Vaporized Sodium during DBA Using Feedback Coefficients Halved, Normal Heat-transfer Coefficient, and a Trigger Temperature of 2500°K (6000-liter metal core, 20-cent/sec reactivity insertion rate)

Time after Vaporization Begins, sec	Total Sodium Vaporized, kg	Cell Overpressure, psig	Time after Vaporization Begins, sec	Total Sodium Vaporized, kg	Cell Overpressure, psig
0	0	0	8.0	31.1	3.08
1.0	20.2	6.61	9.0	31.1	2.56
2.0	29.0	8.36	10.0	31.1	2.13
2.1	29.5	8.38	12.0	31.1	1.48
3.0	31.1	7.60	14.0	31.1	1.02
4.0	31.1	6.32	16.0	31.1	0.71
5.0	31.1	5.26	18.0	31.1	0.49
6.0	31.1	4.37	20.0	31.1	0.33
7.0	31.1	3.64			

The calculated cell overpressure effects of the DBA for a 1000-liter metal core are less than those for the 6000-liter core, as can be seen by comparing the results in Fig. 36 with those in Fig. 32.

The parametric case, in which the heating of the fuel is adiabatic, such as for a sodium-voided core for studying sodium-void coefficients, is shown in Table VIII for a 6000-liter core and in Table IX for a 1000-liter core. The plutonium associated with the 155.6 kg of uranium-plutonium alloy vaporized in the 6000-liter core study is 39.95 kg. (This alloy differs slightly in the uranium-to-plutonium ratio from the normal ZPPR fuel.) Both of these cores are identical to those of the DBA studies considered in Ref. 25.

The ZPPR gravel-sand roof can accommodate steady cell overpressures up to 12.5 psig without disturbing the roof or altering its containment characteristics. The pressure capability is due to the mass of sand and gravel of the roof. As shown in the previous discussions, none of the DBA's studied with their parametric variations exceeds 8.55 psig, which is well within the capabilities of the ZPPR roof.

An additional series of calculations has been made with work being performed in venting air from the cell. The calculations were based on the 6000-liter-core DBA in which fuel (approximately 28.2 wt % plutonium, 69.3 wt % uranium, 2.5 wt % molybdenum) was vaporized and burned as the reactor halves were brought together. To vaporize at this rate, the fuel temperature reaches its boiling point (3800°K).

The previous calculations, summarized in Table VIII, were based on a 25 wt % plutonium fuel. Since the actual fuel composition is 28.2 wt % plutonium, a corrected vaporization rate was used for the present calculations. The corrected vaporization rate indicated that 144 kg of fuel were vaporized in 20 sec, including the 2.5 wt % molybdenum not contained in Table VIII.

The results of the calculations are summarized in Table XIII, and a curve of pressure versus time is shown in Fig. 38. The calculated peak pressure is 7.4 psig at 1.2 sec when the work performed in venting by the expanding gas is included. After 20 sec, the cell pressure drops to 0.6 psig and 48.4% of the air is left in the cell. A calculation of this type will be used in the future in determining the feasibility of proposed reactor loadings. The equations and calculational technique used are described in Appendix A.

TABLE XIII. Calculated Cell Overpressures Caused by Heat Released from Oxidation of Vaporized Fuel during DBA While Performing a Sodium-voided Core Experiment; 6000-liter Metal Core; 20-cent/sec Reactivity Insertion Rate; 3.3% Release of Cell Air to Atmosphere

Time after Vaporization Begins, sec	Total Fuel Vaporized, kg	Cell Overpressure, psig	Air in Cell, lb moles
0	0	0	125.13
1.0	47.0	7.37	118.79
1.2	50.4	7.40	117.15
1.4	53.1	7.28	115.51
2.0	59.7	6.69	110.83
3.0	69.7	5.83	103.89
4.0	79.7	5.24	97.74
5.0	89.2	4.71	92.17
6.0	96.8	4.05	87.29
7.0	103.5	3.49	83.09
8.0	109.0	2.93	79.51
9.0	114.0	2.52	76.48
10.0	119.0	2.27	73.83
11.0	122.2	1.81	71.56
12.0	125.2	1.52	69.72
13.0	128.2	1.34	68.13
14.0	131.2	1.23	66.71
15.0	134.2	1.16	65.38
16.0	136.6	1.01	64.15
17.0	138.6	0.86	63.12
18.0	140.6	0.78	62.20
19.0	142.6	0.73	61.36
20.0	144.0	0.594	60.60

NOTE: 48.4% of the air was left in cell at end of 20 sec.

Fig. 38. Cell Overpressure vs Time: 6000-liter Core, 20-cent/sec
Reactivity Insertion Rate. ANL Neg. No. 103-A11702.

D. Environmental Effects from DBA's

The unique shape of the ZPPR cell structure and its proximity to other buildings, such as the support wing and the EBR-II Laboratory and Service Building, indicate the need to make modified diffusion calculations for cloud-release behavior. These calculations²⁸ were made using the meteorological data for the NRTS.⁹ A brief description of the calculations is given below.

1. Close-in Concentrations

Dispersion studies near a building shaped like ZPPR indicate that eddies occur to the lee side for a distance about three times its height.^{29,30} It is assumed that in this zone (50 m downwind), the air is completely mixed and the concentration may be calculated from $X = Q/cAu$, where X is the concentration in units/ m^3 , Q is the release rate in units/sec, u is the horizontal wind speed (m/sec), A is the vertical cross-section area of the building (550 m^2 for ZPPR), and c is the building wake factor (1/2 for poor mixing, 2 for strong mixing).

Values for X/Q to 50 m downwind are listed in Table XIV. The parameters used in the calculations are given in Table XV.

TABLE XIV. Values of x/Q

d, meters	Lapse	Neutral	Weak Inversion	Strong Inversion
<u>Building Wake Factor, $c = 1/2$</u>				
50	5.19×10^{-4}	7.27×10^{-4}	1.21×10^{-3}	1.82×10^{-3}
150	1.56×10^{-4}	5.17×10^{-4}	9.72×10^{-4}	1.72×10^{-3}
300	3.97×10^{-5}	2.40×10^{-4}	5.69×10^{-4}	1.49×10^{-3}
4700	2.27×10^{-7}	2.18×10^{-6}	8.20×10^{-6}	9.84×10^{-5}
<u>Building Wake Factor, $c = 2$</u>				
50	1.30×10^{-4}	1.82×10^{-4}	3.03×10^{-4}	4.55×10^{-4}
150	8.21×10^{-5}	1.65×10^{-4}	2.85×10^{-4}	4.48×10^{-4}
300	3.23×10^{-5}	1.21×10^{-4}	2.36×10^{-4}	4.31×10^{-4}
4700	2.27×10^{-7}	2.16×10^{-6}	8.04×10^{-6}	8.46×10^{-5}

TABLE XV. Parameters Used in Table XIV

	u, mps	n	C_y	C_z	$\frac{\pi}{2} C_y C_z$
Lapse	7	0.20	0.32	0.32	0.16077
Neutral	5	0.25	0.15	0.15	0.03556
Weak Inversion	3	0.33	0.28	0.07	0.03077
Strong Inversion	2	0.50	0.28	0.035	0.01538
d - 50	$(d - 50)^{1.80}$	$(d - 50)^{1.75}$	$(d - 50)^{1.67}$	$(d - 50)^{1.50}$	
100	3,981	3,162	2,208	1,000	
250	20,700	15,700	10,120	3,954	
4650	3,917,000	2,570,000	1,312,000	312,600	

C_y and C_z are in (meters)^{n/2}.

2. Extended-distance Concentrations

Dispersion for distances beyond this mixing zone, i.e., for $d \geq 50$ m, can be computed from a modified Sutton formula:

$$\frac{x}{Q} = \frac{1}{u \left[\frac{\pi}{2} C_y C_z (d - 50)^{2-n} + cA \right]}$$

where $\pi = 3.14$; n, C_y , and C_z are Sutton's parameters; and d is the horizontal distance from the reactor building (in meters).

The results of these calculations for up to 4700 m, which is the closest site boundary, are also given in Table XIV. These data are plotted on Fig. 39.

3. Gamma Dosages

According to Refs. 30 and 31, the gamma dose rate (in rads/sec) for total immersion in a steady-state cloud with infinite dimensions may be given as

$$D_Y = 0.246 \bar{E}_Y \chi (\rho_0/\rho),$$

where

D_Y = dose rate, in rads/sec;

$\bar{E}_Y$ = average energy, in MeV per disintegration;

χ = concentration of gamma-emitting radioactive materials, in curies/m³;

ρ_0/ρ = ratio of air densities, sea level to NRTS $\approx 1.0/0.85$.

This holds for a ground source, and since no high stack is involved, the assumption of a ground source for ZPPR is valid.

The total integrated dose, over a period ΔT , may be computed from

$$\int_{t_1}^{t_1+\Delta t} D_Y dt = 0.27 \bar{E}_Y \int_{t_1}^{t_1+\Delta t} \chi_Y dt.$$

If we let $B = \chi_Y/Q_Y$ and integrate, the total dose becomes $D_Y \Delta t = 0.27 \bar{E}_Y B Q_Y \Delta t$, where Q_Y is the actual emission rate of gamma energy in curies/second. This relationship is shown in Fig. 40.

4. Beta Dosages

A similar calculation can be made for beta dosages, although, in general, beta dosages are not as important. This relationship, $D_\beta = 0.17 \bar{E}_\beta Q_\beta \Delta t$, is given in Fig. 41. A factor of 0.64 is included because of the differences in beta radiation received from head to toe by a person standing on the ground.

5. Calculated Dosages from the DBA

The DBA studies included six variations of the 6000-liter core and four variations of the 1000-liter core. The maximum number of fissions in the DBA cases was 1.18×10^{20} fissions, of which more than 90% occurred in less than 3 sec. The calculations performed for the venting characteristics indicated that the minimum amount of air remaining in the cell after

Fig. 39. Calculated Downwind Concentrations during Short-time Releases from the ZPPR Backup Containment Structure. ANL Neg. No. 103-A11690.

Fig. 40. Gamma Dosages for a 20-sec Release per Unit Release and Unit Energy. ANL Neg. No. 103-A11688 Rev. 1.

Fig. 41. Beta Dosages for a 20-sec Release per Unit Release and Unit Energy. ANL Neg. No. 103-A11689.

20 sec was 28.8% and the cell pressure was 1.08 psig or less. Also, the largest amount of plutonium that became airborne in the cell was calculated to be 39.95 kg (same as the Ref. 25 DBA). Using this basis, the dosages due to inhalation of fission products and plutonium and direct body exposure to the gamma and beta radiation of the cloud were calculated.

The fission-product inventory calculations were performed using the "Curie" code and standard "Curie" input deck. Fission products that were assumed to be released were all of the prompt noble-gas fission products (7.3% yield) and their daughters. The released fission products of any significance were:

^{88}Kr , ^{89}Kr , ^{90}Kr , ^{91}Kr , ^{92}Kr , ^{93}Kr

^{89}Rb , ^{90}Rb , ^{91}Rb , ^{92}Rb , ^{93}Rb

^{92}Sr , ^{93}Sr , ^{94}Sr

^{137}Xe , ^{138}Xe , ^{139}Xe , ^{140}Xe

^{139}Cs , ^{140}Cs , ^{141}Cs , ^{142}Cs

^{141}Ba , ^{142}Ba .

Although iodine isotopes, formed as prompt fission products, will probably stay in the cell in an actual accident, they are treated as vented fission products here. The iodine isotopes considered are: ^{131}I , ^{134}I , ^{135}I , ^{136}I , ^{137}I , ^{138}I , and ^{139}I .

The total air volume of the cell is 58,000 ft³. The void volume in the gravel-sand roof is 16,700 ft³. In the DBA, 28.8% of the air, or 16,700 ft³, remains in the cell, 16,700 ft³ stays in the gravel roof, and 24,600 ft³ enters the backup containment structure. Assuming perfect mixing as the vented fission products enter the backup containment structure, the concentration in the backup containment increases as the venting occurs. The equivalent vented volume from the backup containment structure is only 6670 ft³, and 18,000 ft³ remains in the structure. Therefore, only 11.5% of the fission products released, i.e., the primary noble gases produced (7.3% yield) and their daughters, leave the backup containment structure and are dispersed to the atmosphere. This also applies to the airborne plutonium filtered by the gravel-sand roof.

A particulate attenuation of 5×10^4 has been allowed for the cell roof and backup containment structure (5×10^2 for the roof and 10^2 for the high-efficiency filter in the backup structure).²⁰ As shown above, only 11.5% of the original cell air is vented out of the overall ZPPR containment. By applying these attenuations to the 39.95 kg of plutonium vaporized in the DBA, the calculated release to the atmosphere is therefore 0.092 g ($3.995 \times 10^4 \times 0.115 \times 2 \times 10^{-5} = 0.092$ g).

In the calculation of inhalation dose, it is assumed the person on the ground level is unprotected by any structure and is exposed to the entire cloud passage. Since the venting of the DBA is over a 20-sec period, the exposure, for calculational purposes, was 20 sec, both for inhalation and direct-radiation doses. A breathing rate of $3.47 \times 10^{-4} \text{ m}^3/\text{sec}$ and a bone-retention factor of 0.2 is used in all calculations.

The calculated inhalation doses from plutonium due to the DBA are given in Table XVI and were obtained for a wind velocity of 1 m/sec by applying twice the cloud concentrations for strong inversions and weak mixing from Table XIV. Dosages during other meteorological conditions would be even lower.

TABLE XVI. Plutonium Isotope Inhalation Doses from DBA during Strong-inversion and Weak-mixing Meteorological Conditions
(breathing rate = $3.47 \times 10^{-4} \text{ m}^3/\text{sec}$; retention factor = 0.2; $u = 1 \text{ m/sec}$)

Location	Total Pu		239Pu		240Pu		241Pu	
	μg	μCi	μg	μCi	μg	μCi	μg	μCi
ZPPR Support Wing, 55 m	2.4×10^{-2}	6.8×10^{-2}	2.0×10^{-2}	1.2×10^{-3}	2.8×10^{-3}	6.0×10^{-4}	6.0×10^{-4}	6.8×10^{-2}
Laboratory and Service Building, ^a 150 m	2.2×10^{-2}	6.4×10^{-2}	1.8×10^{-2}	1.1×10^{-3}	2.4×10^{-3}	5.6×10^{-4}	5.6×10^{-4}	6.4×10^{-2}
EBR-II Reactor, ^a 300 m	1.9×10^{-2}	5.6×10^{-2}	1.6×10^{-2}	1.0×10^{-3}	2.2×10^{-3}	4.8×10^{-4}	4.8×10^{-4}	5.6×10^{-2}
Nearest site boundary, ^a 4700 m	1.3×10^{-3}	3.8×10^{-3}	1.1×10^{-3}	6.8×10^{-5}	1.5×10^{-4}	3.2×10^{-5}	3.2×10^{-5}	3.6×10^{-3}
Maximum permissible total body burdens, occupational (bone as critical organ)	1.94×10^{-1}	5.7×10^{-1}	6.5×10^{-1}	4.0×10^{-2}	1.8×10^{-1}	4.0×10^{-2}	8.0×10^{-3}	0.9
Emergency maximum permissible body burden (bone as critical organ)	1.94×10^{-2}	5.7×10^{-2}	6.5×10^{-2}	4.0×10^{-3}	1.8×10^{-2}	4.0×10^{-3}	8.0×10^{-4}	9.0×10^{-2}

^aUsing a containment attenuation of 5×10^4 for the gravel-sand roof and the backup containment structure.

The calculated gamma and beta dosages are shown in Table XVII. The doses assume complete release of the primary fission-product noble gases from all of the fuel in the core. For strong inversion, a wind velocity of 1 rather than 2 m/sec was used.

TABLE XVII. Gamma and Beta Body Doses due to Radiation from Fission-product Cloud in DBA for Voided Core

Location	Gamma, rads				Beta, rads			
	Strong Inversion ^a		Lapse ^b		Strong Inversion ^a		Lapse ^b	
	Weak Mixing	Strong Mixing	Weak Mixing	Strong Mixing	Weak Mixing	Strong Mixing	Weak Mixing	Strong Mixing
Outside ZPPR Control Room, 55 m	186	46	51	13	68	17	19	5
Laboratory and Service Building, 150 m	76	20	12	6	28	7	4.4	2
EBR-II Reactor, 300 m	29	8	2	2	11	3	0.7	0.7
Nearest site boundary, 4700 m	0.13	0.12	0.001	0.001	0.05	0.04	0.0004	0.0004

^a $u = 1 \text{ m/sec}$.

^b $u = 7 \text{ m/sec}$.

The calculated iodine doses are given in Table XVIII. These doses assume that 25% of the prompt iodine fission products formed become airborne in the cell. The cloud concentrations used in these calculations are twice those obtained from Table XIV (i.e., a wind velocity of 1 m/sec was used for the iodine-dose calculations).

TABLE XVIII. Iodine-isotope Inhalation Doses to Thyroid from DBA during Strong-inversion and Weak-mixing Meteorological Conditions^a
(breathing rate = 3.47×10^{-4} m³/sec; retention factor = 0.23; $u = 1$ m/sec)

Location	¹³¹ I, Ci	¹³⁴ I, Ci	¹³⁵ I, Ci	¹³⁶ I, Ci	¹³⁷ I, Ci	¹³⁸ I, Ci	¹³⁹ I, Ci
ZPPR Support Wing, 55 m	0.96×10^{-9}	0.45×10^{-6}	0.29×10^{-6}	0.43×10^{-4}	0.78×10^{-4}	0.24×10^{-5}	0.93×10^{-11}
Laboratory and Service Building, 150 m	0.91×10^{-9}	0.41×10^{-6}	0.27×10^{-6}	0.19×10^{-4}	0.49×10^{-5}	0.64×10^{-10}	-
EBR-II Reactor, 300 m	0.78×10^{-9}	0.38×10^{-6}	0.22×10^{-6}	0.49×10^{-5}	0.62×10^{-7}	-	-
Nearest site boundary, 4700 m	0.53×10^{-10}	0.88×10^{-8}	0.13×10^{-7}	-	-	-	-

^aUsing an attenuation factor of 10^2 for the HEPA filters in the backup containment structure and no attenuation in the gravel-sand roof.

The calculated dosages due to the cloud release of a DBA include the assumption that personnel are completely exposed, and therefore no credit is given to the effects of buildings for shielding purposes. Furthermore, the normal wind patterns are primarily from the southwest, which would then cause the hypothetical cloud to go out over an area of the NRTS that does not normally have occupancy. It is therefore concluded that even with the occurrence of a hypothetical DBA, the danger to the public is not significant.

The preceding calculated doses were obtained using an 11.5% release of cell air to the atmosphere. The 11.5% release results when using the data of Table VIII in which 28.8% of the cell air remains in the cell.

An additional series of release calculations has been performed using the data of Table XIII, in which 48.4% of the cell air remains in the cell. The results of these calculations show that only 3.3% of the air originally in the cell escapes to the atmosphere (to be compared with 11.5% from the previous calculation). These results also show that (using an overall particulate attenuation of 5×10^4) only 0.0106 g of the plutonium is released to the atmosphere (to be compared with 0.092 g from the previous calculation). The equations and techniques used in release calculations are described in Appendix B.

Note that when the results of these supplementary calculations are used:

- The plutonium inhalation doses in Table XVI are reduced by a factor of 8.7.
- The gamma and beta doses in Table XVII and the iodine doses of Table XVIII are reduced by a factor of 3.5.

The results of these supplemental calculations testify to the conservativeness of the data used in estimating the consequences of the DBA on the ZPPR. These calculations are considered most appropriate to the DBA analysis and dose calculations and will be used in the future in determining the feasibility of proposed reactor loadings.

XI. TESTING THE CONTAINMENT SYSTEM

A. Isolation and Leak Testing

1. Seal Doors

The seal doors in the access corridor and escape corridor will be tested yearly for air leakage. During the checkout of the facility, the doors will also be tested by the construction contractor. A gasketed test cover is bolted to the frame around the seal door on the cell side. The volume between the test cover and the closed door is pressurized to 5 psig, and the leak rate is determined by the rate of loss of air pressure. Leakage rates exceeding the maximum allowable value of $1 \text{ m}^3/\text{min}$ at 5 psig will be corrected before further operation of the reactor will be permitted.

If any damage to a door and its seals occurs during facility operations, the seal door involved will be repaired and retested for leak integrity.

2. Ventilation Valves

The seal valves in the air supply and exhaust ducts to the reactor cell and mound area are tested yearly for air leakage. These valves are also tested by the construction contractor during checkout of the facility. The flanged section of the duct near the cell is removed, and a blind flange with an air connection and gauge is bolted to the pipe. The pipe and valve are pressurized with air to 5 psig, and the leak rate is determined by the rate of loss of air pressure. Failure of the system to satisfactorily comply with the maximum allowable air leak rate of $0.1 \text{ m}^3/\text{min}$ at 5 psig will require repairs and retesting before reactor startup.

3. Piping Bulkheads at the Seal-door Frames

The bulkheads at the seal doors are double-walled, with a void between the two walls. The bulkheads are tested yearly by applying 5 psig air pressure to the volume between the two walls and determining air leakage by loss of air pressure. Work performed on the bulkhead seals between normal testing periods will require retesting and repairs if necessary. The allowable leakage rate of the piping bulkheads is $0.1 \text{ m}^3/\text{min}$ at 5 psig.

4. Instrument Pit Seals

The instrument wiring is brought through a bulkhead seal located in a pit in the access corridor just off the cell entrance. The bulkhead contains hermetically sealed connectors, which are leak-tested yearly. The reactor side of the connector is disconnected and placed aside. A test plate covering the panel of connectors is bolted to the frame, and 5 psig air

pressure is applied to the void between the test plate and the bulkhead. Leakage is determined by rate of loss of pressure. Unacceptable leakage rates higher than the acceptable value of $0.01 \text{ m}^3/\text{min}$ at 5 psig will require correction before reactor startup can commence.

5. Electrical and Instrument Bulkheads at Seal-door Frames

The electrical and instrument bulkheads at the seal-door frames are checked yearly for leakage rates. The tests are performed identically to those for the instrument pit seals in that the connections are disconnected and a test plate is bolted to the frame. Air is pressurized to 5 psig, and leakage rates are determined and corrected if the leakage is greater than $0.05 \text{ m}^3/\text{min}$ at 5 psig.

6. Loading Tube

Both ends of the loading-tube fixtures house gasketed covers, which are checked yearly for leakage. A test cover is bolted to the frame set in concrete, and the void is pressurized to 5 psig with air. Leakage rates are determined by the rate of loss of pressure and corrected if the leakage is greater than $0.05 \text{ m}^3/\text{min}$ at 5 psig.

7. Backup Containment Structure

The backup containment structure above the gravel-sand roof is normally leak tested every 3 years. Initially, it will be tested at the conclusion of construction and also 1 year later. If the test after 1 year indicates a serious deterioration of the structure, the scheduled testing will be more frequent than 3 years. A 0.006-in.-thick polyethylene sheet is placed over the gravel and is taped to the wall structure. Seal-cover plates are placed on the external surfaces of the AEC-type high-efficiency filters, which are left in place for the test. A blower and an airflow meter are connected in series to the test port located just below the access door of the backup containment structure. A water manometer is used to determine building air pressure. The leakage rate is determined by measuring the makeup air through the blower when the building is pressurized to 2 in. H_2O gauge. The acceptable leakage is 3000 cfm at 2 in. H_2O gauge, and failure of the test to obtain this leakage rate or lower will require a determination of locations and repair of the excess leakage areas.

8. Criteria for Leakage Rates

The criterion for establishing the maximum allowable leakage rates for Secs. XI.A.1-XI-A.6 is that operating personnel would receive less than a maximum permissible body burden (occupational) from breathing mixed air in adjacent rooms for 30 min after the DBA. The 30 min are considered adequate to take any action that might mitigate the consequences of the DBA.

The criterion for the leakage from the backup containment structure is that the amount of material escaping through leakage would equal the amount that escaped through the filters, assuming an attenuation of 200 for the filters. The effects of leakage are included in the composite minimum acceptable attenuation of 5×10^4 for the gravel-sand roof and the backup containment structure, i.e., 500 for the gravel-sand roof, and 100 for the backup containment structure.

B. Testing of AEC-type High-efficiency Filters

1. Backup Containment Structure

The AEC-type high-efficiency filters in the backup containment structure will be tested using the accepted di-octylphthalate (DOP) test aerosol every 3 years. Initially, the filters will be tested before startup of the reactor, and the second test will be made 1 year after. If the 1-year lapse time between the initial test and the second test indicates significant deterioration of the system, testing will be done on a more frequent schedule.

The equipment for testing these filters consists of a portable manifold (which fits over the outlet of an individual filter), an exhaust fan, a DOP generator, and a smoke detector.

The DOP aerosol is applied to the upstream side of an inplace filter in an area application, by means of a 4-in.-dia nozzle and duct attached to the smoke generator. The exhaust manifold is placed over the downstream side of the filter and attached to the exhaust fan. The exhaust air is sampled with the smoke detector to determine the outlet DOP concentration while the nozzle is scanned over the inlet side of the filter.

The sensitivity of this technique is more than adequate to determine the minimum acceptable filter efficiency of 99.97% for DOP.

During the tests, air is drawn through an inplace filter by means of a fan, adjusted to 500 cfm, and a portable exhaust manifold attached to the downstream side of the filter. A probe from the smoke detector is placed in the exhaust manifold to sample the exhaust air. A 4-in.-dia nozzle, attached to the DOP generator through a duct, is moved across the face of the filter on the upstream side. The entire filter face is scanned several times during one filter test. The DOP generator is operated at an airflow rate of 20 cfm (564 liters/min), with a measured DOP concentration of 100 $\mu\text{g/liter}$ ($5.64 \times 10^4 \mu\text{g/min}$).

The minimum DOP concentration detection capability of the smoke detector is $7 \times 10^{-4} \mu\text{g/liter}$.

Using the above numbers, the maximum HEPA filter efficiency that could be detected in this test is 99.982%; i.e.,

$$100 \times \frac{\text{Inlet DOP supply rate} - \text{Outlet DOP supply rate}}{\text{Inlet DOP supply rate}} =$$

$$100 \times \frac{5.64 \times 10^4 \mu\text{g}/\text{min} - 7 \times 10^{-4} \mu\text{g}/\text{liter} \times 1.466 \times 10^4 \text{ liters}/\text{min}}{5.64 \times 10^4 \mu\text{g}/\text{min}} =$$

$$100 \times \frac{56400 - 10.3}{56400} = 99.982\%.$$

The smoke-detector sensitivity is such that reading the meter scale to within one-tenth of a division corresponds to a filter-efficiency uncertainty of 0.002%.

Scanning the upstream side of the filter with the DOP generator nozzle, as is done in these tests, increases the probability of detecting small, local filter damage.

Unacceptable filters found during the tests are replaced and the new filters tested after mounting.

A visual inspection of these filters will be performed on a monthly basis, as part of the monthly maintenance check, to detect filter deterioration from aging, etc.

2. Testing of the Suspect AEC-type High-efficiency Filters

AEC-type high-efficiency filters are also used in the reactor cooling system, the exhausts from the mound area, the core-cleaning-room hoods, the laboratory hood in the basement, and the air exhausts from the control room, the cable routing room, and the counting room. Since these filters are used in suspect areas and are operating continuously, the efficiency of each filter is tested every 6 months using the standard DOP aerosol. The pressure drop across each filter at full flow is also measured to allow timely replacement of filters.

C. Efficiency Testing of the Gravel-Sand Roof

The efficiency of the gravel-sand roof is normally tested every 3 years. Initially, two tests will be performed 1 year apart. The purpose of these initial tests is to verify the adequacy of a surveillance interval of 3 years by demonstrating that no significant deterioration occurs within a 1-year interval.

The test will be conducted using DOP aerosol as described in the ZPPR Gravel-Sand Roof Testing Procedure.³¹ Briefly, the test occurs as follows: A test blower, located in the workroom (see Fig. 8), directs conditioned air from the ZPPR support wing and workroom to the reactor cell. The blower is connected to the 12-in.-dia reactor-cell air exhaust and supply ducts. At the entrance of the cell, flexible tubes are used to route the test air from the exhaust and supply ducts to the frangible ceiling so that test aerosol is not dispersed throughout the cell but is kept above the frangible ceiling. The two HEPA filters in the frangible ceiling provide for pressure equalization between the cell and the volume above the frangible ceiling. A DOP aerosol is ejected into the inlet air and its concentration measured before entering the gravel-sand roof. The test air that discharges out of the gravel-sand roof is diverted by a sealed plastic membrane through a hole in which the exhaust DOP concentration is measured. The inlet and exhaust DOP concentrations are compared to obtain the DOP penetration of the gravel-sand roof.

The minimum acceptable DOP penetration as determined by the above described test is $\leq 0.003\%$.

D. Testing of the Workroom Emergency Sand Filter

The efficiency testing of the emergency sand filter is normally performed every 3 years. Initially, two tests will be performed 1 year apart. The purpose of these initial tests is to verify the adequacy of a 3-year surveillance interval by demonstrating that no significant deterioration occurs within a 1-year interval.

The test will be conducted using DOP aerosol as described in the ZPPR Emergency Sand Filter Test Procedure.³² Briefly, the test occurs as follows: A test blower is located in the workroom and directs conditioned air from the workroom through the vent line to the emergency sand filter. DOP aerosol is injected into the test air, and its concentration is measured before entry into the bottom of the sand filter and again after leaving the filter. The upstream and downstream DOP concentrations are compared to determine the filtration efficiency.

The minimum acceptable DOP penetration as determined by the above-described test is $\leq 0.02\%$.

APPENDIX A

Calculation of Cell Pressure versus Time1. Assumptions

a. The heat input to the cell is determined by the vaporization rate of fuel alloy. (This rate is supplied by DBA calculations.)

b. Temperature and pressure changes are uniform throughout the cell.

c. The venting rate through the gravel-sand filter is 13.78 lb-moles/sec at 12.5 psig and is linear with pressure.

d. Initial conditions are: atmospheric pressure = 12.5 psia, ambient temperature = 540°R.

2. Derivation of Equations

a. The total internal energy of the cell, as a function of time is

$$U(t) = N(t) \int_0^T C_v(T) dT,$$

where

$N(t)$ = number of lb-moles of air in cell at time t ,

$C_v(T)$ = molar heat capacity of air at constant volume,

and

T = temperature.

Since $C_v(T)$ is not a strong function of temperature, the following approximation is used:

$$U(t) = N(t)C_vT(t). \quad (1)$$

b. The first law of thermodynamics is

$$dQ = dU + dW,$$

where

dQ = heat added to system,

dW = work performed by system,

and

dU = internal energy of system.

Substituting Eq. 1 into the above equation for dQ and rearranging gives

$$dN(t)C_vT(t) + dT(t)C_vN(t) = dQ(t) - dW(t). \quad (2)$$

c. The work term, $dW(t)$, is the work performed by the air in the cell in forcing $dN(t)$ moles of air from the cell through the gravel-sand filter and may be expressed as

$$dW(t) = p(t) dV'(t),$$

where

$p(t)$ = pressure in cell

and

$dV'(t)$ = volume occupied by $dN(t)$ moles of air.

Use the perfect gas law,

$$p(t)v(t) = RT(t),$$

where

$v(t)$ = molar specific volume,

R = gas constant = 1545 ft-lb/mole-°R,

and the fact that

$$dV'(t) = v(t) dN(t);$$

it follows that

$$dW(t) = dN(t)RT(t);$$

or, in Btu's,

$$dW(t) = dN(t)\frac{R}{J}T(t), \quad (3)$$

where

J = 778 ft-lb/Btu.

d. Substituting Eq. 3 into Eq. 2 and rearranging gives

$$dT(t)C_vN(t) = dQ(t) - dN(t)T(t)\frac{7}{5}C_v, \quad (4)$$

since for a diatomic gas $C_v = \frac{5}{2} \frac{R}{J}$.

e. The number of moles, vented, $dN(t)$, is determined as follows: The venting rate is 13.78 lb-moles/sec at 12.5 psig (1800 lb/ft²) across the filter and is assumed to vary linearly with pressure. The average venting rate during a period in which the cell pressure changes by an amount $dp(t)$ at pressure $p(t)$ is

$$\frac{dN(t)}{dt} = \frac{\left[p(t) + \frac{dp(t)}{2}\right] 13.78 \text{ moles/sec}}{1800 \text{ lb/ft}^2} - \frac{p_0 \times 13.78 \text{ moles/sec}}{1800 \text{ lb/ft}^2},$$

where

$$p_0 = \text{atmospheric pressure} = 1800 \text{ lb/ft}^2.$$

Since the calculations will be carried out on 0.1-sec time intervals, $dt = 0.1 \text{ sec}$. Substituting for dt and p_0 and rearranging gives

$$dN(t) = \frac{\left[p(t) + \frac{dp(t)}{2} - 1800 \text{ lb/ft}^2\right] 1.378 \text{ moles}}{1800 \text{ lb/ft}^2}. \quad (5)$$

Using the approximation

$$p(t) + \frac{dp(t)}{2} = \frac{N(t)R \left[T(t) + \frac{dT(t)}{2}\right]}{V}, \quad (6)$$

where

$$V = \text{cell volume} = 5.8 \times 10^4 \text{ ft}^3,$$

and substituting Eq. 6 into Eq. 5 results in

$$dN(t) = \frac{\left\{ \frac{N(t)R}{V} \left[T(t) + \frac{dT(t)}{2}\right] - 1800 \text{ lb/ft}^2 \right\} 1.378 \text{ moles}}{1800 \text{ lb/ft}^2}. \quad (7)$$

f. Substituting Eq. 7 into Eq. 4 and collecting terms in $dT(t)$, we obtain

$$dT(t) \left[C_v N(t) + \frac{N(t) R \times 1.378 \text{ moles}}{2V \times 1800 \text{ lb/ft}^2} \times \frac{7}{5} C_v T(t) \right] =$$

$$dQ(t) - \left[\frac{N(t) R T^2(t)}{V} \times \frac{1.378 \text{ moles}}{1800 \text{ lb/ft}^2} \times \frac{7}{5} C_v \right]$$

$$+ 1.378 \text{ moles} \times \frac{7}{5} C_v T(t).$$

Solving for $dT(t)$ gives

$$dT(t) = \frac{\frac{dQ(t)}{N(t)} - \frac{R T^2(t) \times 1.378 \text{ moles}}{V \times 1800 \text{ lb/ft}^2} \times \frac{7}{5} C_v + 1.378 \text{ moles} \times \frac{7}{5} C_v \frac{T(t)}{N(t)}}{C_v + \frac{R \times 1.378 \text{ moles}}{2V \times 1800 \text{ lb/ft}^2} \times \frac{7}{5} C_v T(t)} \quad (8)$$

Using the constants

$$R = 1545 \frac{\text{ft-lb}}{\text{mole-}^\circ\text{R}},$$

$$C_v = 4.96 \frac{\text{Btu}}{\text{mole-}^\circ\text{R}},$$

and

$$V = 5.8 \times 10^4 \text{ ft}^3;$$

Eqs. 7 and 8, in incremental form, become

$$\Delta T_i = \frac{\frac{\Delta Q_i}{N_{i-1}} - 1.427 \times 10^{-4} T_{i-1}^2 + 9.569 \frac{T_{i-1}}{N_{i-1}}}{4.96 + 1.427 \times 10^{-4} T_{i-1}},$$

and

$$\Delta N_i = 2.093 \times 10^{-5} N_{i-1} (T_{i-1} + \frac{1}{2} \Delta T_i) - 1.378 \text{ (moles)},$$

where

$$N_i = N_{i-1} - \Delta N_i \text{ (moles)},$$

$$T_i = T_{i-1} + \Delta T_i (^{\circ}\text{R}),$$

$$p_i = 1.85 \times 10^{-4} N_i T_i (\text{psia}),$$

$$\Delta p_i = p_i - 12.5 (\text{psig}),$$

and

$$\Delta Q_i = 5.6 \times 10^3 F_i (\text{Btu}),$$

where F_i = kilograms of fuel burned in i th interval and where the initial conditions are

$$T_0 = 540^{\circ}\text{R},$$

$$N_0 = 125.13 \text{ lb-moles},$$

and

$$p_0 = 12.5 \text{ psia}.$$

APPENDIX B

Release Calculations

Supplementary analyses have been performed using the data from Table XIII to determine the plutonium release as a function of time and the cell air release as a function of time from the backup containment structure. This type of calculation will be used in the future in determining the feasibility of proposed reactor loadings. These calculational techniques and results are for information only and have not been used in arriving at the tabulated dose rates presented in Chapter X above.

1. Plutonium Vented from Backup Containment Structure (BCS) as a Function of Time during a DBA

a. Assumptions

- (1) Perfect mixing of oxidized fuel and air occurs in the reactor cell.
- (2) The cell effluent displaces the air in the gravel-sand roof without mixing.
- (3) Perfect mixing of effluent from reactor cell occurs with the air in the BCS upon entering the BCS.
- (4) Effluent from cell cools rapidly to ambient temperature when traversing the gravel-sand roof.
- (5) Average pressure in the gravel-sand roof is atmospheric during a DBA. (This is a conservative simplification.)
- (6) The void volume in the gravel-sand roof is $1.67 \times 10^4 \text{ ft}^3$. The volume of the BCS is $2.8 \times 10^4 \text{ ft}^3$.
- (7) The void in the gravel-sand filter contains 36.0 lb-moles of air. The BCS contains 60.4 lb-moles of air.

b. Development of Equations

Let

f_i = the fraction of total fuel burned during the i th interval,

N_i = the number of moles of air in the cell during the i th interval,

and

ΔN_i = the number of moles vented from the cell during the i th interval.

At the end of the first interval, the fractional concentration of fuel in the cell air is

$$\frac{f_1}{N_1} \text{ (moles}^{-1}\text{)}.$$

The portion that leaves the cell by venting during this interval is

$$F_1 = \Delta N_1 \frac{f_1}{N_1}.$$

At the end of the second interval, the fractional concentration is

$$\frac{f_1 - F_1 + f_2}{N_2} = \frac{f_1 \left(1 - \frac{\Delta N_1}{N_1}\right) + f_2}{N_2}.$$

Rearranging (note that $N_1 - \Delta N_1 = N_2$) gives

$$\frac{f_1 \frac{N_2}{N_1} + f_2}{N_2}.$$

The portion that leaves the cell by venting during this interval is

$$F_2 = \frac{\Delta N_2}{N_2} \left(f_1 \frac{N_2}{N_1} + f_2 \right).$$

Similarly, for the third interval,

$$F_3 = \frac{\Delta N_3}{N_3} \left(f_1 \frac{N_2}{N_1} \frac{N_3}{N_2} + f_2 \frac{N_3}{N_2} + f_3 \right).$$

And, in general,

$$\begin{aligned} F_i &= \frac{\Delta N_i}{N_i} \left(f_1 \frac{N_2}{N_1} \frac{N_3}{N_2} \frac{N_4}{N_3} \frac{N_5}{N_4} \dots \frac{N_i}{N_{i-1}} + f_2 \frac{N_3}{N_2} \frac{N_4}{N_3} \dots \frac{N_i}{N_{i-1}} + \dots + f_{i-1} \frac{N_i}{N_{i-1}} + f_i \right) \\ &= \frac{\Delta N_i}{N_i} \left(f_1 \frac{N_i}{N_1} + f_2 \frac{N_i}{N_2} + f_3 \frac{N_i}{N_3} + \dots + f_i \right), \end{aligned}$$

or

$$F_i = \Delta N_i \left(\frac{f_1}{N_1} + \frac{f_2}{N_2} + \frac{f_3}{N_3} + \dots + \frac{f_i}{N_i} \right).$$

During the first interval in which effluent from the cell enters the backup containment structure, a fraction of fuel, F_1 , also enters. The fractional concentration, assuming perfect mixing, is

$$\frac{F_1}{M},$$

where

$$\begin{aligned} M &= \text{number of moles of air in BCS} \\ &= \text{constant.} \end{aligned}$$

The fractional amount of fuel leaving the BCS during this interval is

$$\phi_1 = \frac{\Delta N_1}{M} F_1.$$

The fractional concentration during the second interval is

$$\frac{F_1 - \frac{\Delta N_1}{M} F_1 + F_2}{M} = \frac{F_1 \left(1 - \frac{\Delta N_1}{M}\right) + F_2}{M}.$$

The fractional amount of fuel leaving the BCS is

$$\phi_2 = \frac{\Delta N_2}{M} \left[F_1 \left(1 - \frac{\Delta N_1}{M}\right) + F_2 \right].$$

Similarly, for the third interval,

$$\phi_3 = \frac{\Delta N_3}{M} \left[F_1 \left(1 - \frac{\Delta N_1}{M}\right) \left(1 - \frac{\Delta N_2}{M}\right) + F_2 \left(1 - \frac{\Delta N_2}{M}\right) + F_3 \right],$$

and, in general,

$$\begin{aligned} \phi_i &= \frac{\Delta N_i}{M} \left[F_1 \left(1 - \frac{\Delta N_1}{M}\right) \left(1 - \frac{\Delta N_2}{M}\right) \dots \left(1 - \frac{\Delta N_{i-1}}{M}\right) \right. \\ &\quad \left. + F_2 \left(1 - \frac{\Delta N_2}{M}\right) \left(1 - \frac{\Delta N_3}{M}\right) \dots \left(1 - \frac{\Delta N_{i-1}}{M}\right) + F_{i-1} \left(1 - \frac{\Delta N_{i-1}}{M}\right) + F_i \right]. \end{aligned}$$

The total amount escaping up to the i th interval is

$$T = \sum_i \phi_i.$$

2. Cell Air Vented from BCS

The fraction of fission-product gases formed that are released from the cell and the BCS is calculated assuming all fission-product gases are released into the cell immediately upon melting the fuel.

The molar concentration of fission products in the cell is

$$\frac{G}{N_0},$$

where

N_0 = initial number of moles of air in cell,

and

G = quantity of fission-product gases.

The amount of fission-product gas leaving the cell during the i th interval is

$$g_i = \frac{\Delta N_i}{N_0} G.$$

In the BCS, the concentration of gas at the end of the first interval is

$$\frac{g_1}{M}.$$

The amount vented during this interval is

$$\frac{\Delta N_1}{M} g_1.$$

At the end of the second interval, the concentration is

$$\frac{g_1 - \frac{\Delta N_1}{M} g_1 + g_2}{M} = \frac{g_1 \left(1 - \frac{\Delta N_1}{M}\right) + g_2}{M}.$$

The amount vented during this interval is

$$\frac{\Delta N_2}{M} \left[g_1 \left(1 - \frac{\Delta N_1}{M}\right) + g_2 \right].$$

Similarly, for the i th interval, the amount vented is

$$G_i = \frac{\Delta N_i}{M} \left[g_1 \left(1 - \frac{\Delta N_1}{M} \right) \left(1 - \frac{\Delta N_2}{M} \right) \dots \left(1 - \frac{\Delta N_{i-1}}{M} \right) \right. \\ \left. + g_2 \left(1 - \frac{\Delta N_2}{M} \right) \left(1 - \frac{\Delta N_3}{M} \right) \dots \left(1 - \frac{\Delta N_{i-1}}{M} \right) + \dots \right. \\ \left. + g_{i-1} \left(1 - \frac{\Delta N_{i-1}}{M} \right) + g_i \right].$$

Using $g_i = \frac{\Delta N_i}{N_0} G$ gives

$$G_i = \frac{\Delta N_i}{M} \frac{G}{N_0} \left[\Delta N_1 \left(1 - \frac{\Delta N_1}{M} \right) \left(1 - \frac{\Delta N_2}{M} \right) \dots \left(1 - \frac{\Delta N_{i-1}}{M} \right) \right. \\ \left. + \Delta N_2 \left(1 - \frac{\Delta N_2}{M} \right) \dots \left(1 - \frac{\Delta N_{i-1}}{M} \right) + \dots \right. \\ \left. + \Delta N_{i-1} \left(1 - \frac{\Delta N_{i-1}}{M} \right) + \Delta N_i \right].$$

The total amount vented up to and including the i th interval is

$$T = \sum_i G_i.$$

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