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HAZARD SUMMARY REPORT ON THE
ARGONNE LOW POWER REACTOR (ALPR)

Reactor Engineering Division

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HAZARD SUMMARY REPORT ON THE ARGONNE LOW POWER REACTOR (ALPR)

I. INTRODUCTION

Evaluation studies indicate that nuclear power plants are well suited for applications in isolated arctic areas. The Army Reactors Branch has therefore undertaken a study of reactor power plants designed for such a purpose.

Work done by Argonne National Laboratory has resulted in the recommendation that a heterogeneous boiling reactor plant be built at the National Reactor Testing Station in Idaho. The Argonne Low Power Reactor (ALPR) is to be a prototype for the purposes of testing the operation and serving as a training center.

Transportation requirements and climatic conditions in the arctic are considered in the prototype design. The building design is based on an ultimate installation located in the permafrost tundra region.

The net electric power demand is to be normally 200 kw, but it can go as high as 260 kw intermittently. An equivalent of 400 kw of space heating will be supplied by the reactor for the installation.

The ALPR is closely related to the Borax boiling water reactors. The operating and control experience obtained by Argonne from the Borax experiments in Idaho was incorporated in the ALPR design.

It is the intention to develop the ALPR controls a step further so that the power generation rate of the reactor can be automatically adjusted to satisfy the demand of the station.

The reactor fuel charge and the control system are to be such that refueling will be required only at long intervals, several years, if possible.

At present, the Argonne Staff is working in collaboration with the architect-engineering firm of Pioneer Service & Engineering Company, Chicago, Illinois, to complete the design and detail drawings.

Critical experiments and operation of the prototype plant are scheduled for early 1958.

II. CONCLUSIONS

It is considered that ALPR is an inherently safe nuclear power plant. Its ability to operate with extremely remote probability of a dangerous nuclear incident is based on the following facts:

1. The characteristics of ALPR are similar to those of BORAX-II, which was stable at a higher power density than required in ALPR.
2. The boiling water reactor will tend to shut itself down with an increase of steam voids. In the event of increasing power densities, the moderator will be expelled, thus causing reactor shutdown.
3. It is conceived that the most hazardous operation will be during the critical assembly work or initial startup. However, at this stage of operation, the fission products have not started to accumulate and thus an accident at this time will produce minimum radiation hazards of long half-life. Also adequate staff supervision is provided throughout the pre-operational program.

III. DESCRIPTION OF ALPR FACILITY

A. Design Data

General

Reactor Heat Output	3 Mw (nominal)
Turbine-Generator Rating	300 kw
Frequency	$60 \pm 1/2$ cps
Voltage	$120/208 \pm 5\%$
Phase (4-wire)	3
Power Factor	0.8
Standby Equipment Capacity (diesel electric)	60 kw
Design Domestic Heat	400 kw
Steam Production	9020 lb/hr
Operating Pressure	300 psig
Operating Temperature	420F
Feedwater Temperature (hotwell)	134F
*Ambient Temperature	-60 to 60F
Wind Velocity (maximum)	125 mph
Building Height (maximum above ground)	50 ft
Transportability	Airlift
Building site (simulated)	Permafrost
Site Materials for Construction	Local gravel

Fuel

Length of Active Core	25.8 in.
Maximum Horizontal Cross Section of Core	34.5 in. x 34.5 in. (corners removed)
Number of Fuel Assemblies	
Minimum Loading	40
Maximum Loading	59
Total Thickness of Fuel Plates (0.050-in. meat, 0.035-in. clad)	0.120 in.
Number of Plates per Assembly	9
Average Water Channel Gap	0.310 in.
Cladding Alloy for Plates (Al-Ni) Alcoa	X-8001
Core Heat Transfer Area (40 assemblies)	475 sq ft
Fuel in 40 Assemblies, kg U ²³⁵	14.0
"Meat" Dimensions	25.8 x 3.5 x 0.05 in.
Meat Volume per Plate	74 cm ³
Approx. Weight Aluminum in Meat per Plate	200 gm
Weight U ²³⁵ per Plate	38.9 g
Weight Uranium (91% U ²³⁵ wt) per Plate	42.7 g

*Not incorporated in the building design.

Fuel (Cont'd.)

Approx. Wt % Uranium in Meat	17.7
Approx. Atom % Uranium in Meat	2.21
Weight U ²³⁵ per Assembly (approx.)	350 gm
Weight B ¹⁰ per strip	0.4 gm (.021" strip)
	0.5 gm (.026" strip)
Weight % B ¹⁰ in Boron Strips	0.42

Nuclear Data

Average Thermal Flux in Fuel at Full Power (fresh reactor)	$7.5 \times 10^{12} \text{ n(cm) / (cm}^3\text{)(sec)}$
Max/Avg Flux Ratio in Hot Fresh Reactor Radial	1.6
Control Cell	1.3
Axial (control rods out)	1.3
Axial (control rods halfway in)	1.7
Reactor, Control Rods Out	2.7
Reactor, Control Rods Halfway In	3.5
Reactivity Changes, % k	
a. Temperature	1.5 to 2
b. Xe + Sm	3
c. Steam Voids	1.3 to 2
d. Xenon Override	1 to 1.5
Neutron Lifetime (sec)	$4 \text{ to } 8 \times 10^{-5}$ (over range of core conditions)

Heat Transfer and Fluid Flow for 40-Assembly Reference Core

Average Power Density in Core Coolant	17.5 kw/l
Steam Flow at 3000 kw	9020 lb/hr
Average Steam Voids in Heated Channel	9%
Average Steam Voids in Moderator	7%
Exit Steam Quality	0.85%
Water Recirculation Ratio (lb water per lb of steam)	130
Feedwater Inlet Temperature	175F
Subcooling at Channel Inlet	1.8F
Average Boiling Length of Core	20 in.
Total Heat Transfer Area per 40-Element Core	475 ft ²
Average Heat Flux at Minimum Loading	21,500 Btu/(hr) (sq ft)
Average Fuel Temperature at Center Line	450F
Average Surface Temperature of Plates	440F

Control Rods

Number of Crosses	5
Additional Spaces for Tee Rods	4
Spacing	8-13/16 in.
Length of Cadmium Section	32 in.
Thickness	
Cadmium (meat)	0.060 in.
Aluminum-Nickel Alloy (clad)	0.080 in.
Total Travel	31 in.
Scram Time	< 2 sec
Withdrawal Rate	~0.01%k/sec in region of maxi- mum differential worth of central rod
Weight of Control Rod	
Four-blade	49 lb
Three-blade	37 lb

Pressure Vessel

Diameter, OD	4.5 ft
Wall Thickness	
SA-212 Base Material	3/4 in.
Type 304 Stainless Steel Clad	3/16 in.
Height (less head)	14-1/2 ft
Design Pressure	400 psig
Thermal Shield	3/4 in. Stainless Steel
Pressure Relief Size and Settings	
First stage to condenser	2 in. - 350 psi
Second stage to atmosphere	2 in. - 385 psi
Level of Water above Core	4 ft 3 in.
Total Weight (empty)	26000 lb
Average Volume of Steam Dome	80 cu ft
Total Weight of Contained Water	~ 6500 lbs.
Maximum Thermal Stresses Due to Gamma Heat in Vessel	750 psi
Maximum Thermal Stresses Due to Gamma Heat in Cover	600 psi
Design Stress	14000 psi
Type of Closure	Double gasket- ing with leakoff
Design Life	20 yr
Water Cleanup Circulation Rate	3 gpm

B. Site

1. General

The ALPR is to be built at the National Reactor Testing Station. Figure 1 shows the location of the plant on the NRTS Site in relation to other reactor test facilities. The plant's coordinates are N 675 398; E 325 990 and its location is Butte County, State of Idaho, on the eastern edge of Bingham County. It is $3/4$ mile north of the new Idaho Falls Highway, US-20, and 3 miles east of the junction of US-20 and US-26. ALPR will be located about 9 miles east of the BORAX and EBR-I facilities and about 6 miles east of the NRTS Central Facilities. The nearest off-site population centers are Howe 21 miles N.N.W. and Arco 24 miles W.N.W. from the reactor site.

The ALPR plant area will be a fenced-in square, about 350 ft on a side. As shown in Fig. 2, access to the enclosure will be by hard-top road, past a guard station. The reactor building with the power equipment lies beyond an auxiliary building, about 200 ft northeast of the guard shack. The auxiliary building will provide office space, eating facilities for the crew, shower and rest rooms, and auxiliary power and heating facilities. The end of this building which is nearest to the reactor contains the ALPR control room, and simulates the control module of the ultimate installation.

Just south of the plant, an air-cooled heat exchanger is installed to simulate the 400-kw domestic heat load. Adjacent to this heat exchanger is the air-cooled resistor bank to absorb the ALPR-power output.

A domestic water well is located on the site, together with a pump and a 50,000-gallon ground level storage tank. A public utility power line is brought in from the north to the site substation.

2. Hydrology

The NRTS lies on a level plain, the surface of which is covered by a thin layer of soil. The subsurface of much of the reservation is a layer of gravel varying in depth up to about 50 feet or more.¹ The ALPR site, however, lies on lava rock. Test borings indicate that from 1.5 feet below the surface down to about 14.5 feet the material is solid lava rock occasionally honeycombed with small voids and sometimes marred by larger cracks and voids. The concrete slab will bear directly on the bed rock and is designed so that the load to be carried by the rock is well within its load-carrying capacity. Drainage through this rock is not nearly as rapid as that through gravel overburden at other test sites, but is appreciably greater than that observed through sands or clays.

¹ "Hazard Summary Report, Experimental Breeder Reactor II (EBR-II)," ANL-5719, May 1957, L. J. Koch, et al.

3. Seismology

The available seismic history of this region is summarized in ANL-SM-236, Appendix K.² There exists some risk of seismic activity at the NRTS. It has been designated as Zone 2 Area by "The Pacific Coast Uniform Building Code." Therefore, it is considered that structures designed for Zone 2 are acceptable.

4. Meteorology

The most recent survey of meteorological data for the National Reactor Testing Station is contained in ANL-5719.¹ This data was taken at NRTS Central Facilities and is equally applicable to the ALPR site.

C. Building

The reactor building is a circular, continuous welded, steel tank measuring 38 ft 7 in. diameter by 48 ft high (Fig. 3). The material for the tank is ASTM Spec. A-7 steel, suitable for service to -40F, having a yield strength of 30,000 psi. The tank is supported, two feet above grade, on short concrete piers which bear on a concrete slab which in turn bears on lava. This construction simulates the eventual arctic installation where timber piles, extending into the permafrost, are used to support the tank structure. Inside the tank, structural steel is independent of the tank walls and supports two floors - an operating floor at elevation 19 ft 4 in. and a fan floor at elevation 32 ft 0 in. above the tank bottom.

The tank building is filled with gravel to a point two feet below the operating floor. The gravel serves as a biological shield for the centrally located reactor. The gravel shield also contains three fuel storage wells, a tank for contaminated water, a purification vault, two storage sumps, reactor instrument wells, ducting, and piping.

The exposed gravel area, which is roughly one-half of the operating floor area, is covered with a coating of approximately 1/4-in. thick coal tar pitch water emulsion. The pitch material is suitable for spraying, does not support combustion, bonds, and conforms readily to the gravel and the surrounding structural members. It forms an impervious membrane to restrict outleakage of radioactive gas or dust. Such material has been tested at Argonne for application technique and subjected to temperatures ranging from -20F to 400F with satisfactory results.

²"Hazards of the Materials Testing Reactor," ANL-SM-236, June 15, 1950, S. McLain, R. K. Winkleblack

The operating floor (Fig. 4) contains the top shielding of the reactor, the turbo-generator (supported on an independent slab), a 1000-gallon storage tank for water, the motor control panel, two feedwater pumps, the hotwell, the steam separator, the boron injection system, the bridge crane, the purification system, and miscellaneous minor equipment. The floor material is 1/4-in. checkered plate except for the area around the reactor, where concrete shielding serves as the floor. Personnel access to the building is by means of one exterior enclosed stairway connected with the control room and an additional open emergency exit on the opposite side of the building. A cargo door in the tank wall is serviced by a monorail crane.

The fan floor (Fig. 5) contains the main steam condensing system consisting of condenser, fan, air mixing chamber, building ventilation equipment, and ductwork. The fan will maintain a suction of about two inches of water. Personnel enter the fan floor through a simple air lock which is reached by ladder from the operating floor. A large hatch for the passage of equipment is provided in the floor.

The instruments and controls necessary to operate the reactor are grouped in the support facilities building. This building is connected with the reactor building by a short passageway. Control of the reactor will be from the control room, but access to the reactor operating floor is possible at all times.

D. Reactor

1. Mechanical Design

a. Core

(1) Structure

The entire core is fabricated of an aluminum-nickel alloy, with the exception of certain minor items, such as fuel element lifting knobs and spacer springs which are of stainless steel. General views of the ALPR core are shown in Figs. 6 and 7. Weights and surface areas of the fuel elements and related components are given in the Design Data Section.

As is indicated in Fig. 8, the core structure is divided into sixteen boxes. Twelve boxes (each to contain four fuel elements) at the center and sides are of square cross section; the four corner boxes are contoured to contain three fuel elements each. The maximum core capacity is 59 fuel assemblies plus one source assembly. The sides of the boxes serve as shrouds to define the control rod channels. There are five full-cross channels and four tee channels to be used, if necessary.

Dummy elements fabricated of aluminum-nickel alloy will be placed in the unused fuel positions to retain the active fuel elements. Figure 8 shows the forty fuel element core loading for the reference design and the extra fuel element positions.

(2) Fuel Assemblies

The fuel assemblies, shown in Fig. 9, consist of nine 0.120-in. thick flanged fuel plates assembled to side plates by spot welding. The flanged edges secure correct spacing of the plates. The fuel plate consists of a 0.050-in. thick center portion (meat) of aluminum-nickel-uranium alloy, clad with 0.035-inch thick aluminum-nickel. The "meat" is $3\frac{1}{2}$ in. wide by 25.8 in. long. The finished clad plate width is 3.710 in. wide (exclusive of the flanges) and 27.8 in. long. One aluminum-nickel strip containing approximately 0.42 wt. % B^{10} is spot welded to a side plate of each fuel assembly. Two thicknesses of poison strips have been fabricated for use in zero power experiments, each strip containing nominally 0.4g B^{10} , or 0.5g B^{10} . The 0.4g strip is 25.8 in. long, 3.875 in. wide, and 0.021 in. thick. The 0.5g strip is 0.026 in. thick. On the basis of the zero power reactivity studies, poison strips will be assigned to specific core locations, one strip per fuel assembly. If additional poison strips should be required, it is expected that half-length strips will be welded to the lower half of the opposite side plate of some assemblies. In the case of the 59-assembly core, it may prove useful to add poison strips containing a greater concentration and a greater mass of boron, nominally 0.8g B^{10} per strip. Such decisions will be reserved until the zero power studies have yielded sufficient information.

To maintain fuel assembly spacing, stainless steel springs are fastened on the four sides of the fuel elements at the top. A hold-down device keeps the assemblies from creeping upward during operation.

The aluminum-nickel alloy, Alcoa X-8001, has been selected as the cladding material for the ALPR fuel element.

(3) Control Rods

In the initial core configuration there are five cross control rods. These rods are $1\frac{1}{4}$ in. wide across the blade edges with approximately 14-in. cadmium "meat."

Additional spaces for four more control rods, T-shaped, are provided in the event that a larger number of fuel assemblies than the reference loading of 40 elements is used.

Figure 10 shows a full-cross control rod. The active portion of each blade is 0.060-in. thick cadmium sheet, 31 in. long by 7 in. wide. The cadmium is confined between two 0.080-in. thick aluminum-nickel sheets, edge-welded. The cadmium is perforated at intervals by $1\frac{1}{2}$ -in. diameter holes, through which the aluminum-nickel cladding is dimpled and spot-welded to provide support. The centrally located rod is furnished with a 19-in. follower section made of solid aluminum-nickel plate. A 3-in. solid aluminum-nickel, cross-shaped rod extension, 32 in. long, is attached to the upper end of the control blade. The over-all length of a complete control rod assembly is approximately 90 in.

The length of cadmium provides overlap beyond the active length of the fuel element. When fully inserted into the core, all of the follower section extends beyond the bottom of the fuel elements (Fig. 6). The follower section remains in the control rod channel when the cadmium section is withdrawn from the active zone and serves to reduce the thermal neutron flux peaking.

A shock absorber and stop arrangement, located within the control rod drive mechanism, functions during rod drops. However, a final positive mechanical stop is provided near the top of the rod extension which comes to rest against the top of the shroud in the dropped position.

b. Control Drive Mechanism

The mechanical drive incorporates the basic concept of transmitting rotary motion into linear motion by means of a rack and pinion (Fig. 11). A rotary shaft pressure seal is used. The pressure seal is of the positive clearance, break-down seal ring type, which will have controlled leakage and requires a minimum of maintenance. The leakage will be bled to the precoolers and the condensate return tank. Approximately 0.1 gpm of water will be bled continuously to each control drive housing or thimble. This will result in;

- (1) purging of thimbles to prevent accumulation of decomposition gases produced in the reactor;
- (2) reducing wear and corrosion of internal working parts such as rack, pinion, and bearings; and
- (3) leakage of fluid through the seal is water and not steam.

The external drive is positively engaged with the pinion shaft by means of a magnetic clutch. Failure of clutch current automatically inserts the rods rapidly into the core by gravity. Rapid insertion (scram) is accomplished within 2 sec over the full travel or an acceleration of $1/6$ g. The mechanism is so designed that a scram signal will not only release the magnetic clutch but the downward drive will further try to drive the rod down by positive action through a "free wheeling" clutch. In the event of a power failure, the control rod motor current is supplied by the emergency power system. The speed of normal rod travel is restricted by mechanical gearing. At the beginning of the zero power work, the speed will be set at 3 in./min. Prior to power operation, this rate will be adjusted (by replacing gears) so as to limit the reactivity insertion rate to approximately 0.01%/sec., as determined from period measurements of differential rod worth. A shock absorber and stop arrangement, located within the drive mechanism, functions during rod drops. Position indicating devices are directly installed on the pinion shaft to indicate position of the rods at all times within 0.05 in. over the full travel of 31 in. Since the drive is external, all motors and critical parts are readily accessible for maintenance or replacement.

The drive and the pressure seals were developed by Argonne National Laboratory and Alco Products, Inc. Tests on these mechanisms, conducted by Alco, have demonstrated successful performance. Additional tests conducted at Argonne have been conducted in a facility shown in Fig. 12, in which an actual ALPR prototype mechanism was operated for 1550 hr at ALPR operating conditions of 300 psi and 420F.

Over 8000 cycles and 250 scrams were successfully made after which time visual inspection of the parts indicated satisfactory performance.

c. Reactor Pressure Vessel

The reactor pressure vessel will be fabricated of Type SA-212, Grade B Firebox steel clad with Type 304 stainless steel, 4 ft 6 in. OD by $14\frac{1}{2}$ ft long, in accordance with ASME Code for Unfired Pressure Vessels UW-2 Lethal Vapor (Radioactive steam) and Code for power boilers, Section I, 1956 Revision. The vessel (Fig. 13) is designed for 400 psig at 450F for water and steam and 500F metal temperature. The lower end of the vessel is equipped with an ellipsoidal dished head. The upper end of the vessel is equipped with a flange and cover plate bolting. The stainless steel clad cover plate is equipped with nine flanged 6-in.

nozzles of Type 304 stainless steel to accommodate control rods and drives, one 4-in. opening for liquid level control, and one $2\frac{1}{2}$ -in. opening for source handling. The joints are sealed with gaskets. The main closure flange is equipped with continuous leak-off groove for collecting leakage in the event leakage occurs past the inner gasket.

The vessel is equipped with internal base pads to support the thermal shield and core structure, and two feedwater spray rings with nozzle connections outside the vessel. The various types of connections to the vessel are summarized below:

1 Feedwater	$1\frac{1}{2}$ in. diameter
1 Shutdown Cooling Water	$1\frac{1}{2}$ in. diameter
1 Steam	4 in. diameter
1 Separator Return	1 in. diameter
1 Outlet to Purification System	1 in. diameter

The vessel is insulated thermally with a 3-in. layer of magnesia which is banded to the vessel and protected by a $1/4$ -in. steel cover. This cover serves as a protection during shipment and installation. The insulation will be placed on the vessel prior to shipment from the fabricator.

The inner thermal shield (see Fig. 13) serves two purposes: to absorb gamma radiation (see Section III-D-1-e), and to support the reactor core structure which rests on eight bearing pads welded to the thermal shield.

The lower frame of the core structure is shown in vertical section (Fig. 6) and is shown dotted in the horizontal section (Fig. 7). It is made of $1/2$ -in. by 3-in. bars, notched and welded at cross points.

d. Water-Cooled Support Cylinder

The pressure vessel is installed inside a steel cylinder which consists of two half shells bolted together along vertical seams. The vessel is suspended with its upper flange resting on the top edge of this support cylinder. The cylinder itself rests on the bottom steel structure of the reactor building.

The support cylinder has one important purpose other than that of carrying the weight of the reactor. It also forms a substantial part of the thermal shielding. For this reason the wall thickness is increased by $1\frac{1}{4}$ in. of lead and the heat generated is removed by a system of water cooled copper coils built into the lead wall. Outside of the lead is a steel jacket $1\frac{1}{4}$ in. thick at the level of the reactor core and $1/4$ in. thick above and below this level.

At the location between the reactor and three vertical instrument tubes the lead thickness is increased by several inches to improve the gamma shielding.

The cooling water for the shield is circulated through a heat exchanger cooled by water which is pumped from the condenser hot-well through this heat exchanger and back to the condenser.

e. Shielding

The main bulk of the shield is gravel, since this material is expected to be a natural deposit at the ultimate plant site. The use of local gravel greatly minimizes the actual weight of the material which must be airborne to the final site. Local gravel is also specified as aggregate for additional concrete shielding.

The gravel for the biological shield has a density approaching 120 lb/cu ft. Since the building structure is about 38 ft in diameter, approximately 16 ft of shielding is provided for the reactor, although 12 ft has been estimated to be sufficient to attenuate the reactor radiation to AEC tolerance level. Conveniently, several components which might contain radioactivity are embedded in the gravel shield, such as the ion exchangers, contaminated water storage tank and spent fuel storage wells. The outer cylindrical thermal shield is composed of a $1\frac{1}{4}$ -in. layer of lead sandwiched between the $\frac{7}{8}$ -in. thick pressure vessel support tank and a $1\frac{1}{4}$ -in. thick cylindrical steel plate at the reactor core level. The bottom thermal shield is composed of a $1\frac{1}{4}$ -in. layer of lead sandwiched between the $1\frac{1}{2}$ -in. thick pressure vessel support tank bottom and a 1-in. thick steel plate. The shield cooling tubes lie in this $1\frac{1}{4}$ -in. lead layer.

A boron sheet is used below the reactor vessel to capture thermal neutrons and thus reduce the possibility of thermal neutron activation of air and possible airborne dust below the reactor building.

Directly below the reactor there is a $6\frac{1}{2}$ ft by $6\frac{1}{2}$ ft square, 17-in. deep layer of steel punchings or shot, in addition to the thermal boron shield, and outside of this is a 4-in. layer of shot extending to a 6-ft, 6-in. radius, all supported by 4-in. thick wooden planks. Fine gravel will be mixed with the shot to fill the voids. A steel retaining plate underlies all the bottom shielding. Below this array there is a 2-ft air space around the pilings (a requirement in permafrost-tundra regions), and a gravel pad is placed on the tundra and permafrost. At the NRTS, a concrete slab will replace the permafrost, but all construction above the ground level will be similar to the requirements for a remote arctic site.

During reactor operation the air space below the reactor is inaccessible because of radiation level; however, hazardous activation of argon or dust by neutrons is not anticipated. The thermal neutron

flux in this region is expected to be about 8×10^5 neutrons/cm²/sec, and the fast flux about 4×10^4 neutrons/cm²/sec. Gamma doses are of the order of 100 r/hr directly below the reactor. One hazard to personnel might be due to scattered radiation emerging from the air gap at the perimeter of the gravel tank, although estimates indicate a safe level of radiation. A radiation monitoring survey of this region is planned on the prototype to evaluate the design. Space is provided for additional earth or gravel, if required.

The top biological shielding consists of steel plates and shielding blocks made by pouring concrete into prefabricated steel frames. The blocks are shaped with steps, as illustrated in Fig. 14, to minimize radiation leakage through gaps. Access to the instrument holes is possible after removing only one of the blocks. The top shielding covers an area with a 6-ft radius from the reactor centerline.

The pressure vessel lid is covered with a layer of thermal insulation. Above this layer and around the control rod nozzles is a mixture of shielding material, each cubic foot containing 100 lb of steel punchings, 30 lb boric oxide, and the balance being local gravel. The mix is dry, but it is a satisfactory material for attenuating gamma rays while producing a negligible capture gamma ray dose.

Additional shielding directly above the control rod drives is provided by a stack of steel and masonite plates; this reduces the neutron and gamma doses to less than 100 mr/hr. It has been estimated that less than 10% of the gamma ray energy is reflected from the ceiling. Thus, the reactor operating room is expected to be within tolerable radiation dose levels but access to the top region is to be prohibited during operation of the reactor. Some cooling air is circulated in between the blocks during operation although the heat generation rates are negligible.

The outer thermal shield is designed to limit the amount of radiant energy which enters the gravel to low values. The large mass of gravel with its low thermal conductivity, 0.35 Btu/(hr)(ft)(F), acts like a large cylindrical insulator packed around the reactor vessel. Although some heat will pass through the outer tank wall, the floor, and the bottom of the structure, it is assumed that practically all the heat generated within the shield gravel is carried away by the thermal shield coolant water. Thus the gravel temperature will rise to a maximum, at a certain distance from the shield cooling coils, ultimately reaching an equilibrium temperature as the heat generation rate approaches zero. Heat flow to the outer wall accounts for the decrease in temperature beyond the peak.

Treating the gravel as an infinite insulating medium, with heat generation decreasing exponentially with radius the predicted temperature rise in the gravel is shown in Fig. 36. Cooling water in the thermal shield may reach 180F. The hottest point in the lead and steel thermal shield may

reach 270F. The gravel temperature will be higher, but it is expected to be less than 485F. This is considered to be within the safe limit for the structural steel members within the gravel. Gravel samples have been furnace heated to 750F and subjected to temperature gradients of the order of 1400F/ft without any apparent ill effect.

The inner thermal shield serves to support the core, but its primary function is to protect the outer lead layer from melting immediately if the shield cooling flow should fail. It removes 45% of the gamma energy reaching the pressure vessel and delivers this heat directly to the reactor water instead of the outer shield cooling system. However, secondary gammas born in the thermal shield add about 12% to the energy flux at the vessel. These hard gammas do penetrate relatively far into the shield. The net reduction of gamma ray energy flux at the outer thermal shield (about 30%) effectively reduces the rate of temperature rise which would take place in the lead if shield coolant flow were suddenly stopped. This gives the operator time to investigate and possibly correct the difficulty.

Table 1 shows a summary of neutron and gamma levels in the reactor shield. Calculations of thermal stresses at the most critical points were made and the maximum thermal stresses indicated are:

Pressure Vessel Wall	750 psi
Pressure Vessel Lid	600 psi

f. Fuel Handling and Storage

It is anticipated that the entire reactor core will be replaced at one shutdown and at a time when the expected three-year core life has elapsed. Unloading of the spent fuel elements is readily accomplished with the cover on, using the control rod nozzle openings as passage ways. To attenuate core radiation, the pressure vessel is filled with demineralized cold water from the contaminated water storage tank. Water will be taken from the discharge side of the contaminated water pump through a hose to an ion exchange bed in the waste storage sump. Then the decontaminated water is pumped through another hose to the hotwell where the feedwater system pumps it to the reactor. Temperature in the reactor vessel can be adjusted by circulating the water through the purification system. Additional cold water is obtained from the 1000-gallon fresh water storage tank. With a water height of 9 ft above the reactor core, the dose rate at the cover surface two hours after shutdown is less than 2 mr/hr except directly in the thimble openings where a dose rate of 140 mr/hr is anticipated.

After "flooding" of the core, the control rod drives are removed and the extension shafts detached from the control rods. This leaves an unobstructed region above the core, while the control blades themselves remain fully inserted.

Table 1

RADIATION FLUX LEVELS AT FULL POWER

Location	Nominal Distance from Centerline (cm)	Neutron Flux (n/cm ²)(sec)		Gamma Ray Flux	
		Fast	Thermal	mev/(cm ²)(sec)	r/hr
Radial (at level of midplane of core)					
Core outside surface	41.5	8.28×10^{12}	1×10^{13}		
Inside surface					
Thermal shield	62.86	1.1×10^{10}	3.5×10^{11}	5×10^{13}	
Pressure vessel	66.35	6.5×10^9	1.8×10^{11}	2.3×10^{13}	
Outer thermal shield (support cylinder)	79.4	2.4×10^9	9×10^{10}	6.2×10^{12}	
Gravel	85.4	1×10^9	3×10^9	2.7×10^{11}	1.6×10^6
One foot into gravel	115.9	1×10^8	2×10^9	4×10^{10}	
Twelve feet into gravel	14.8 ft	10	1×10^3		0.001
Outside of instrument shield	93 cm	10^8		1.4×10^9	4×10^3
Axial (along core vertical centerline)					
Underside of vessel lid	11 ft	9×10^6	5×10^7	2×10^{12}	
Floor level	13 ft	2×10^3	3×10^5		750
Above top shield plates	20 ft	80	1×10^3		0.10

The fuel element transport coffin is placed on top of one control rod thimble, while the other thimbles are used for the insertion of the fuel manipulator, viewer, and lighting equipment (Fig. 15).

The viewer is used to observe the handling of the fuel elements through one of the thimble openings. This viewer is essentially a tube with a lens on each end, completely filled with high purity water and closely fitting within the thimble opening. Thus, the radiation at the point of viewing is attenuated by the column of water within the viewer so that the dose rate will be less than 7.5 mr/hr even with a fuel element 2 ft above the core.

The operator extracts a fuel element from the core, first by attaching the coffin cable gripper to the selected fuel element by means of a long-handled tool (manipulator) and, secondly, by hoisting the element into the coffin. The coffin (Fig. 16) provides ten inches of lead shielding, reducing the dose rate of a spent element to 100 mr/hr at the coffin surface. When the coffin gate is closed, the coffin will be transported to the fuel storage containers by means of the building crane.

The loading of new fuel elements is accomplished by the above procedure in reverse. However the coffin is not employed, and one or two control rods are held part way out. The new fuel is lowered into the reactor through one of the nozzles until it is immediately above the core shrouds. The manipulator then takes hold of the fuel element gripper and guides the element to the desired position.

Fuel elements are stored in three storage wells located within the shielding gravel at a point approximately 11 ft from the center of the reactor. The wells are stainless steel cylinders, $19\frac{1}{2}$ in. ID and 15 ft long. Each well has a cluster of seven 6-in. ID tubes which serve as containers for the fuel elements. Three fuel elements may be stacked on top of each other in each tube for a total of 21 elements per well.

In the region (≈ 9 ft long) containing the fuel assemblies (Fig. 17) the central storage tube and three alternate tubes around it are rolled from 1/8-in. boral. The other three tubes are rolled from 1/8-in. 2S aluminum. Above the fuel storage zone aluminum tubing (6-in. ID, 1/8-in. wall thickness) is welded to the boral tubes to extend them to the full height of 14 ft 9 in. The boral and aluminum storage tubes are held in place by aluminum insulator separators which provide paths for the natural circulation of the fluid in the tank and also prevent galvanic corrosion between the aluminum and the water stainless steel tank. Water purity will be checked by sampling periodically from the bottom of the well by way of the water inlet pipe. If it should prove to be advisable to purify the water in the wells, a pump and ion exchange column can be installed to serve all three wells on a routine rotational basis.

Water is admitted manually to the storage wells for cooling the spent fuel. The evaporated water will be condensed in a finned tube radiator on the fan floor and returned by gravity. The rate of water evaporation is shown in Fig. 18 as a function of the shutdown time.

The thermocouples will be located in each storage well, one near the bottom of the tank and one near the top of the well to measure temperature of the cooling water.

The covers on each well are steel cans filled with 9 in. of concrete and 3 in. of steel. The water above the elements plus this lid give adequate shielding. The lid may be rotated with an eccentrically located plug serving as an access for loading.

g. Poison Injection System

A back-up shutdown system has been incorporated in the design of the ALPR plant, which provides for the addition of boric acid to the reactor water. At the discretion of the operating personnel, a concentrated boric acid solution may be pumped into the reactor through the lower or upper feedwater spray rings. It is expected that the boric acid concentration in a 120 gallon storage tank will be maintained at 100 grams H_3BO_3 per gallon. The manually operated pump has a capacity of at least 25 gal/hr when the reactor is at operating pressure, corresponding to a reactivity removal rate of at least $2\frac{1}{2}\%$ per hour. With the vessel at atmospheric pressure, the solution can be introduced through the upper feedwater spray ring by gravity feed through a bypass hose.

A secondary use of the poison injection system is that of special shutdown cooling in the event of a major loss of water from the vessel by an accident which would not exclude personnel from the immediate area of the hand pump.

This system is not considered to be an emergency safety measure since it requires the presence of operating personnel in an area which might be extremely hazardous in the event of radioactive contamination.

The boric acid can be removed from the reactor water by circulating it at room temperature through an auxiliary anion resin bed.

h. Shield Cooling

The shield cooling circuit, shown in Fig. 20, is a closed forced circulation loop capable of removing 65 kw of heat. Normally, only 40 kw of heat is removed at full power operation of which 35 kw is estimated to be due to heat generated in the shield by neutron and gamma attenuation and 5 kw due to thermal heat leakage from the pressure vessel.

In addition a natural circulation radiator and bypass section are incorporated for emergency shield cooling. Six kw of heat can be removed in the event of shield coolant pump failure or when condensate water flow is interrupted.

The shield coolant is circulated through two parallel sets of copper cooling coils, embedded in the lead portion of the thermal shield. One of the parallel sets is adequate to carry the heat load, thus the reactor can still operate should one of the parallel circuits fail.

2. Instrumentation and Control

a. Rod Positioning and Position Measurement

Two methods for operating the control rods will be incorporated in the reactor design, namely, (1) manual control, and (2) automatic steam pressure control. It is anticipated that automatic pressure control will be feasible; however, the transient behavior of the reactor under this type of operation needs to be more fully established. Consequently, plans call for running the reactor initially by manual control.

The control rod drive motors are actuated manually from the control panel by two control switches, one of which controls the center rod and one of which controls the eight peripheral rods in conjunction with a selection switch. Interlocking circuits are provided to prevent the simultaneous withdrawal of two or more rods.

Most rapid insertion of the control rods is effected by deenergizing magnetic clutches on the control rod drives, thus permitting all control rods to fall by gravity into the position of maximum poison. Additional means are provided to permit declutching of each individual drive for the purpose of experimentally determining the rapid insertion time for each rod. "Rod position" indication is accomplished by a synchro transmitter and receiver.

Limit switches are provided at both extremes of travel to (1) stop the rod drive motor at its end of travel, and (2) light the "rod in" and "rod out" pilot lamps. The limit switch circuit is connected with the shut-down circuit in such a way that, following a shutdown, the rods must be fully inserted before they can be withdrawn.

b. Safety and Alarm Circuits

In Table 2 are listed the abnormal system conditions resulting in release of the control rod drive clutches and subsequent rapid insertion of all rods together with the abnormal conditions that result in annunciation. In some cases (condenser pressure, reactor water level, reactor pressure) an abnormal condition first actuates the annunciator, and, if allowed to progress further, causes reactor shutdown. Key-operated switches are provided for bypassing the reactor period and low pressure shut-down circuits.

In addition to the condition that all scram circuits be satisfied, interlocking circuits prevent initial withdrawal of control rods during reactor startup unless (1) control power is switched on and (2) all rods are completely inserted.

c. Nuclear Instrumentation (See Figs. 21 and 22)

Measurement of instantaneous reactor power level will be accomplished through the use of six neutron flux detection channels as follows:

Channel I. An uncompensated, boron-lined ion chamber will be used to drive a panel-type microammeter and a sensitive moving coil relay in series. This channel will furnish rough reactor power information in the power range above a minimum of approximately one per cent of full power, and in addition the relay can be adjusted to provide a high-flux shutdown at a preset power level.

Channel II. A second uncompensated, boron-lined ion chamber will be connected to drive a multi-range amplifier-type microammeter and a sensitive moving coil relay in series. This channel will furnish reactor power information in the operating power range and below over a range of approximately five decades. In addition, a duplicate of the Channel I shutdown circuit actuation is provided.

Channel III. A gamma-compensated, boron-lined ion chamber will be employed in conjunction with logarithmic and period amplifiers to furnish log flux level and period information. The period circuit will be arranged to shut down the reactor in the event of a period shorter than a preset minimum (10 sec) when reactor power level is within 3 decades of full power. It is expected that this channel will be used only in the prototype.

Channel IV. A second gamma-compensated, boron-lined ion chamber will be coupled to a power level indicator and recorder through a d-c amplifier. This will furnish reactor power information over approximately seven decades. It is contemplated in the Idaho prototype installation only.

Table 2
OPERATING CONDITIONS RESULTING IN REACTOR
SHUTDOWN OR ANNUNCIATION

Item	Operating Conditions	Conditions Causing Annunciator Operation	Conditions Causing Automatic Reactor Shutdown
1.	Reactor neutron flux, high - Channel I	x	x
2.	Reactor neutron flux, high - Channel II	x	x
3.	Reactor neutron flux ion chamber high voltage supply, Channel I deenergized	x	x
4.	Reactor neutron flux ion chamber high voltage supply, Channel II deenergized	x	x
5.	Reactor water level, high or low	x	x
6.	Reactor steam pressure, high	x	x
7.	Main steam pressure relief valve, steam flow	x	x
8.	Main steam safety valve, steam flow	x	x
9.	Condenser pressure, high	x	x
10.	Reactor period, short	x	x
11.	Actuation of manual "Scram" pushbutton	-	x
12.	Control power, deenergized	-	x
13.	Rod drive clutch rectifier, failure	-	x
14.	Bypass steam flow, high	x	-
15.	Bypass valve discharge pressure, high	x	-
16.	Condenser outlet air temperature, high	x	-
17.	Condenser inlet air temperature, low	x	-
18.	Feed pump discharge pressure, low	x	-
19.	Purification water temperature, high	x	-
20.	Air ejector after condenser temperature, high	x	-
21.	Shield cooler outlet water temperature, high	x	-
22.	Hotwell water level, high or low	x	-
23.	Feed pump, automatic switchover	x	-
24.	Reactor pressure, low	x	x
25.	High pressure condensate tank water level, high	x	-
26.	Low pressure condensate tank water level, high	x	-
27.	Turbine oil inlet temperature, high	x	-
28.	Thermal shield temperature, high	x	x
29.	Shield cooler pump discharge pressure, low	x	-

x Yes
- No

Channels V and VI. Two identical channels consisting of BF_3 proportional counters, pulse amplifiers, and scaler-counters will be provided. These channels will be employed during core loading and initial reactor startup for the detection of very low neutron levels.

In Fig. 21 are shown the effective ranges of the various channels of nuclear instrumentation. The scale is logarithmic and is in terms of reactor heat generation in watts. Figure 22 shows the instrument block diagram. In addition to the above channels, Channel VII, which is employed to monitor air ejector exhaust gamma ray activity, is shown.

Four vertical instrument tubes are provided for the neutron flux detectors associated with the above instruments. These tubes are disposed in a circular arc about the reactor pressure vessel support shell. Air cooling and drains are provided. The flux detectors are positioned in the tubes by means of adjustable locating devices.

d. Reactor Pressure Interlocks

(1) Maximum Pressures

Under normal operating conditions, reactor steam pressure will be regulated to 300 ± 5 psig by the automatic steam by-pass valve (with a maximum change of electrical load of ± 60 kw). The bypass valve is a motor-operated valve controlled by the pressure on the inlet side. It is a proportional valve, adjusted so that it will be closed for pressures up to 280 psig and fully open at 310 psig. At the maximum opening the valve capacity is 10,000 lb of steam per hour.

Under average operating conditions with a pressure of 285 psig, the by-pass steam flow will amount to 1400 lb/hr. This corresponds to a reactor pressure of 300 psig and a turbine inlet pressure of 275 psig, the difference being due to pressure drop in the piping system.

At approximately 330 psig alarm contacts in the reactor steam pressure indicator will open and the corresponding annunciator signal will be actuated. If reactor pressure should continue to increase to approximately 340 psig, another set of contacts in an independent bourdon tube-type gauge will open, causing the reactor to shut down. A further increase in pressure will actuate a pressure relief valve, set to open at 370 psig, and a safety valve, set to open at 385 psig. Steam flow through either of these valves is detected by means of temperature-sensing instruments which incorporate contacts connected to the shut-down circuit.

(2) Reactor Minimum Pressure

The reactor steam pressure indicator can also actuate a low pressure warning and scram. This circuit is not put into operation until the reactor is operating at power. The pressure for this scram will be set once sufficient operating experience is gained, but is expected to be about 250 psig. If reactor pressure drops, the steam bypass and turbine trip valves will close to keep from blowing down the vessel. A hand operated valve in the main steam line can be closed, if necessary, to isolate the reactor from the rest of the plant.

e. Reactor Water-Level Measurement and Control

The water-level-sensing device is inserted into the vessel from a mounting flange on the top cover. This device consists of a "float" or buoyant member which extends vertically downward to a point level with the bottom of the reactor core and is shielded from the turbulence incident to boiling by a still well. The float is suspended from a crank arm by means of an extension rod attached to its upper end. Rotary motion of the crank shaft is proportional to the buoyant force on the float, and this movement is transmitted to an external electrical transducer through a torque tube assembly. The torque tube acts as a motion-restricting spring and also eliminates the need for a pressure seal between fixed and rotating parts.

The output from the liquid level transducer is a low-energy electrical signal which is fed into the reactor level indicating controller. The output from this controller constitutes one input to the three-element feedwater valve control. In addition, two sets of independent contacts actuate the alarm and shut-down circuits in the event of abnormally high or low water level. Failure of either the liquid level transducer or an instrument amplifier component will indicate a maximum water level and result in immediate reactor shutdown.

Recognizing the importance of determining reactor water level and of maintaining it within certain limits, and further that failure of the primary level instrument is possible, independent means are provided for making approximate water level measurements. For this purpose four 1/2-in., open-ended tubes extend vertically into the reactor vessel to different depths. Three of these tubes, which extend to normal water level, two feet above it, and two feet below it, respectively, lead to manually-operated valves which in turn feed into a manifold and orifice assembly. The orifice discharges into a line leading to the condenser. A pressure gauge, calibrated "water," "mixture," and "steam," is connected to the manifold. Upon opening one of the three valves, a gauge indication corresponding to the state of the fluid being vented indicates qualitatively the water level relative to the open end of the respective tube. This device

will be employed for periodic checks of the primary level instruments. The fourth tube extends to a point at the level of the top of the core and may be used with a bubbler device to determine the head of water above the top of the core.

In addition to reactor water level, control of the feedwater valve is made a function of both total steam flow and feedwater flow. These latter two quantities are measured by means of flow nozzles, differential pressure transducers, and instruments whose outputs are proportional to the magnitude of the steam and water flow rates, respectively. The three instrument output signals are mixed and, together with a signal derived from the feedwater valve drive unit, are fed into the feedwater valve drive servo amplifier. In this connection it should be noted that errors in the instrument measuring steam and water flow rates result in incorrect feedwater valve position only until the reactor water level changes slightly and the level instrument generates a balancing signal. Thus under steady-state conditions feedwater flow is exactly equal to total steam flow. Switches are provided by means of which the steam and water flow signals may be disconnected from the circuit and feedwater valve position controlled as a function of reactor water level alone. In the event of a reactor scram, the feedwater valve controls revert automatically to "manual." Further the control system is arranged so that the valve is closed automatically by the emergency power supply, following a reactor scram.

f. Temperature and Pressure Measurements

Feedwater temperature, condenser inlet air temperature, and condenser outlet air temperature are indicated and recorded in the control room. Another group of temperatures, not requiring continuous surveillance, may be measured individually (also in the control room) by means of a single-point indicating instrument equipped with pushbutton selectors. Locally-mounted indicators are employed for a third group of temperatures where only infrequent supervision and/or temperature alarm contacts are required.

Similarly, the more important process pressures are displayed in the control room, while those requiring less supervision are indicated locally.

3. Physics

a. Neutron Diffusion Models for Reactivity Analyses

Diffusion theory was applied to the calculation of overall reactivity effects in the reactor. The thermal utilization of fuel in the assembly lattice was calculated by a one-group (P_3) method, as was the effectiveness of control rods in a "control cell." Two-group diffusion theory

was used to calculate the macroscopic or reactor problems once the parameters of a homogenized core were obtained. In this theory the effects of epithermal absorptions and fissions are ignored. It has been found that these effects are small in a reactor of this type operating at a relatively low temperature ($kT_{\text{coolant}} = 0.042$ ev), for the macroscopic thermal neutron absorption cross section is small and there is little epithermal neutron absorption.

A plot was made of the experimentally determined age* versus the volume ratio of aluminum to light water of unit density. The age $\bar{\tau}$ for an equivalent homogenized core was obtained for the volume averaged material compositions:

$$\bar{\tau} \approx \frac{\tau(\text{Al}/\text{H}_2\text{O of unit density})}{(1 - \text{core void fraction})^2}$$

(see Table 3).

Table 3

TWO-GROUP AGE: $\bar{\tau}(\text{cm}^2)$

	Core	Reflector (H ₂ O)
Cold	51.0	31.4
Hot (417F): $\rho(\text{H}_2\text{O})$ 0.85 g/cm ³	68.4	43.5
Operating (417F) (10% mean steam void in coolant channels)	78.9	-

In computing thermal neutron absorption effects, it was assumed that in fuel lattices the thermal neutron flux spectrum is essentially Maxwellian but with a mode, kT_{eff} , shifted from the moderator kT to a value set by the ratio of absorption to scattering.** The microscopic absorption and fission cross sections of the various materials were read from the tables and graphs in BNL-325; $\eta = 2.08$; $\alpha = 0.184$ for U²³⁵.

*Reactor Handbook, Vol. 1, Physics, AECD-3645.

**ORNL-1958, "Effect of Moderator Temperature Upon Neutron Flux in Infinite, Capturing Medium" R. R. Coveyou, et al., October 20, 1955.

b. Control Cells

The term "control cell" denotes the region comprised of four fuel assemblies and half of the surrounding control channels. (See Fig. 23.)

The control cell is made up of two regions called the "effective control channel" and the "effective fuel zone." By the "effective control channel" is meant the region of the cell extending from the outer boundary to the nearest surface, or edge, of fuel meat. The cold metal-to-water ratio in this zone is 0.75. The "effective fuel zone" is the remaining region; it occupies 72% of the area of the control cell, and the cold metal-to-water ratio there is 0.40. The reactor averaged cold metal-to-water ratio is ≈ 0.49 .

One-group (cell) calculations were made to obtain thermal neutron flux distributions within these cells. The control cell model for the computations was one of concentric circular cylinders, the corresponding cell regions having areas equal to those in the actual cell. It was found that one-group calculations of infinite slab lattices would yield nearly the same channel disadvantage factor. In conjunction with one-group P_3 calculations of slab lattices (with and without the presence of rods), this method was used to verify some of the cylindrical model calculations of rod worth.

Flux distributions in the operating fresh control cell are presented in Fig. 24. Significant flux peaking occurs when rods are withdrawn. In the presence of rods, the effective control channel tends to flatten the flux in the effective fuel zone.

If the reactivity effect of rod insertion in a control cell is defined in terms of these thermal utilizations, then, in the fresh reactor:

$$(\text{cold control cell}) \frac{\Delta k}{k} (\text{rods out} \rightarrow \text{rods in}) \approx \frac{0.77 - 0.91}{0.77} = -0.18$$

and

$$(\text{operating control cell}) \frac{\Delta k}{k} (\text{rods out} \rightarrow \text{rods in}) \approx \frac{0.74 - 0.92}{0.74} = -0.24$$

In a control cell of the operating depleted reactor,

$$\frac{\Delta k}{k} (\text{rods out} \rightarrow \text{rods in}) \approx \frac{0.65 - 0.90}{0.65} = -0.38$$

The concept of $\frac{\Delta k}{k}$ is of doubtful value when applied to large perturbations. The results listed are to be interpreted only as an indication of the change in rod effectiveness with temperature and fuel depletion.

In the following sections it is shown that the five cross rods adequately control the cold fresh reactor. The rod worth increases with fuel burnup, as indicated in Table 5, and the control of the cold reactor is demonstrated for the entire core cycle.

Table 4
FLUX RATIOS IN CONTROL CELLS
(Not Referenced)

Moderator Conditions	Fuel Burnup	Thermal Disadvantage Factor of Effective Fuel Zone		Max/Avg Flux in Effective Fuel Zone
		Rods In	Rods Out	
68F	Fresh reactor	---	1.7	---
417F; operating voids	Fresh reactor	0.7	1.5	1.3
417F; operating voids	Depleted reactor	---	1.35	---

Table 5
THERMAL UTILIZATION OF EFFECTIVE FUEL ZONE
IN CONTROL CELL
(Based on Data in Table 4)

Moderator Conditions	Fuel Burnup	Absorptions in Effective Fuel Zone	
		Absorptions in Control Cell	
		Rods In	Rods Out
68F	Fresh reactor	0.77	0.91
417F; operating voids	Fresh reactor	0.74	0.92
417F; operating voids	Depleted reactor	0.65	0.90

c. Reactivity Effects in Fuel-Water Lattice

In calculating the effective fuel zone parameters, the boron side strips were considered as uniformly distributed in that zone. It is found that the local thermal flux peaking in the water gaps between cell assemblies is reduced by the boron strips and the average flux in the boron raised to such levels that the approximation of uniform distribution is adequate.

The thermal disadvantage factor of the fuel plates in a lattice of fuel plates and water channels was calculated on the AVIDAC by a one-group P_3 routine. The results are presented in Table 6.

Table 6

THERMAL DISADVANTAGE FACTOR IN PLATE LATTICE:

$$\frac{\bar{\phi}_s(\text{Al} - \text{H}_2\text{O})}{\bar{\phi}_s(\text{meat})}$$

Moderator, Temp. °F	Local Fuel Depletion	$\frac{\bar{\phi}_s(\text{Al} - \text{H}_2\text{O})}{\bar{\phi}_s(\text{meat})}$
68	Fresh fuel	1.071
420	Fresh fuel	1.044
420	20% depleted	1.037
420	50% depleted	1.023

These results indicate that a positive temperature effect arises from the change in flux peaking in the fuel lattice. Fuel depletion also reduces this peaking, a positive reactivity contribution.

d. Partial Insertion of Control Rods: Flux, Rod Worth

The "window-shade" technique was used to evaluate the worth of the five cross rods as a function of their depth of core penetration as a bank. In the hot, fresh, forty-assembly core, the rods are worth approximately 20% k; this was simulated in terms of uniformly distributed thermal neutron absorption in the regions penetrated by the rods.

Multi-region, reflected reactor problems were solved on the UNIVAC to obtain the two-group fast and thermal neutron flux distributions axially (shown in Fig. 25) as well as the rod worth (shown in Fig. 26). Axially non-uniform steam void distributions were computed and shown to have small effect on the flux distributions. However, the partial insertion

of control rods reduces the heat production locally and increases the mean steam void in the coolant channels, a negative reactivity effect. This negative $\frac{\Delta k}{k}$ (steam) is smaller than the steam void reactivities attained in stable operation of the BORAX-II and BORAX-III reactors, operating at 150 psig to 300 psig, respectively.

e. Radial Flux Distribution

The radial components of the flux distributions were computed for the hot fresh reference core (Fig. 27). The core radius, 39.9 cm, is appropriate to a circular cylinder of the same horizontal area as ten control cells. The production of steam tends to flatten the thermal flux distribution radially since fewer neutrons slow down to thermal in a less dense moderator.

f. Reactivity in Steam Void

The interaction effects of steam void distribution may be summarized as follows:

(1) Within the range of steam void fractions estimated for ALPR at full power, there is little distortion of the radial and axial neutron flux distributions.

(2) The mean steam void in the fuel assembly coolant channels is approximately 9 to 10%. This number applies to the reactor with control rods fully withdrawn. The resulting reactivity is:

$$\frac{\Delta k}{k} (\text{steam}) \approx -1.3 \text{ to } -1.5\% k$$

When rods are inserted as a bank, the steam void (negative) reactivity effect may be as much as $-2\% k$.

A correction factor was applied to the coolant void because of the presence of water in the control channels. The net mean steam void in the core fluid is then 7 to 8%.

Feedwater return is through nozzles in the lower spray ring, located vertically just below the top of the active core. The feedwater inlet temperature is 175F. When mixed with the reactor fluid circulating in the vessel, the incoming feedwater subcools the reactor fluid entering the coolant channels 1.8F. Approximately one-fourth of the core heat is required to raise the water temperature to saturation. The result is that both the mean steam void and the reactivity in steam are approximately 25% smaller in the subcooled case than for saturated inlet fluid.

g. Temperature "Coefficient" of Reactivity

An increase in moderator temperature to 417F will reduce the reactivity of the reactor. The size of this reactivity effect is strongly dependent on the location of the control rods, as shown in Table 5. Heating the fresh reactor to operating temperature results in a net reactivity effect of approximately -2% k, when control rods are withdrawn fully. The reduction in thermal flux peaking in the control channels contributes $\approx +1\%$ k, and the reduction in the lattice disadvantage factor is worth 1% k. However, the latter effects are small in comparison with a change of as much as -4 to -5% k arising from an increase in the worth of fully inserted rods accompanying the rise in temperature.

h. Perturbation Theory Calculations

Two-group perturbation theory was used to calculate the reactivity effects of fuel depletion and of boron-10 burnup during the core life. The shape of the real reactor flux was approximated by the product of the radial and axial flux components in the hot fresh reactor. The adjoint flux shapes were assumed to be given by the product of the equivalent bare core radial and axial flux components. The reactor was partitioned axially into five (symmetric) regions of equal thickness, and radially into eight regions of equal thickness.

For each of these forty regions an average thermal neutron flux and average weighting factors were computed. The boron-10 was assumed to burn up six times as fast as the U^{235} (since $\frac{\bar{\sigma}_a B^{10}}{\bar{\sigma}_a U^{235}} \approx 6$):

$$\frac{B^{10}(r,t)}{B^{10}(r,o)} \approx \left[\frac{U^{235}(r,t)}{U^{235}(r,o)} \right]^6$$

The variation of fission product absorption with burn-up was estimated on the basis of a plot by R. W. Deutsch.* The reactivity components of burnup and the net reactivity are plotted in Fig. 28, based on the use of enough boron in the side strips to control 11% in the fresh reactor. The plot of net reactivity variation does not include the time variations of the following: thermal utilization of fuel plates in the assembly, relative absorptions of neutrons by the control channel material, thermal neutron leakage from the reactor, and capture of neutrons by "equilibrium" fission products.

*"Fission-Product Buildup in Enriched Thermal Reactors,"
R. W. Deutsch, Nucleonics, Vol. 14, No. 9, September, 1956.

i. Xenon and Samarium

Assuming a core flux at the center equal to 2.7 times the average thermal neutron flux (1×10^{13} n cm/cm³ sec) in the fuel of the depleted reactor, the perturbation method used was adapted to the evaluation of the statistical worth of full power equilibrium xenon. The result was 20% higher than the number obtained by using a volume average flux of 1×10^{13} n cm/cm³ sec.

Similarly, the variation of xenon after complete shutdown from equilibrium full power operation in the depleted reactor was evaluated by statistical weighting methods. This variation is shown in Fig. 29.

It is estimated that override of maximum xenon is possible at all times if 1 to 1.5% k is available. Otherwise the reactor can be operated at a lower power level until the xenon concentration approaches equilibrium again.

j. Control Requirements in Fresh ALPR

(1) Reference Core (40 Fuel Assemblies)

Loss of operating steam voids	1.3 to 2.0% k
Loss of equilibrium xenon and samarium	$\approx 3\%$ k
Temperature drop from 417F to 68F (rods out)	$\approx 2\%$
Operating margin	<div style="border-top: 1px solid black; display: inline-block; width: 100%;"> 2 to 3% k 8 to 10% k </div>

NOTE: The control capacity of the five cross control rods in the hot fresh forty assembly reactor is approximately 20%, and in the cold fresh reactor approximately 15%.

(2) Fifty-nine - Assembly Core

NOTE: The control capacity of nine control rods is approximately 22% in the cold fresh reactor. This is sufficient to control the cold fully loaded reactor at all times during the core cycle.

k. Additional Remarks on Reactivity Effects in the Various Core Loadings

It has been calculated that the five control rods can not quite control the reference 59-assembly reactor, but the capacity of the nine rods is more than adequate. If additional control should be decided upon in the course of the zero power investigations, for example in the form of poison strips of greater boron content, it should be possible to control the fully loaded core with only the five cross rods.

In the 59-assembly core operating at 3 mw, the reactivity control in equilibrium xenon and samarium is roughly 2.5% in comparison with approximately 3% in the 40-assembly core at 3 mw. In the latter core the reactivity in steam voids is $\approx 1.5\%$; for the same fractional mean steam void in the coolant water in the core, the 59-assembly reactor produces 4.5 mw with $\Delta k/k$ (steam voids) $\approx 1.2\%$.

The worth of the isolated fully inserted central cross rod is reduced from $\approx 10\%$ in the operating 40-assembly system to $\approx 7\%$ in the fully loaded 59-assembly core.

4. Heat Transfer

Heat generated in the reactor results in a lighter steam and water mixture within the fuel channels than in the downcomer area around the core. The difference in density between the mixture in the fuel channels and the downcomer causes upward natural circulation through the core.

If the reactor is operating at saturated conditions, boiling would start at the entrance of the fuel channel. However, incoming feedwater lowers the temperature of the water entering the core, hence the nonboiling length of the fuel is a function of the incoming feedwater temperature. For the ALPR, the nonboiling length is estimated to be approximately 6 in. when the incoming feedwater is at a temperature of 175F.

The method for calculating the heat transfer characteristics of the ALPR core essentially consists of summing the pressure drops within and over the core and around the downcomer. This summation must be equal to zero to satisfy steady-state flow condition. However, due to the variation of two-phase friction factors and relative steam and water velocities, the critical parameters used in evaluating boiling performance are of an empirical nature and must be obtained from experimental data.

The empirical factors used in the performance evaluation are those obtained from boiling experiments on electrically heated channels, 1/4 in. and 1/2 in. wide, and 5 ft long.*

*"A Method of Analysis of Natural Circulation Boiling Systems," P. A. Lottes and W. S. Flinn, Paper Presented June, 1956 before the American Nuclear Society.

In the experiments, the electrical heat input is known and the steam volume fraction, flow rate and pressure drop are measured. From these measurements the two-phase friction drop (liquid and steam) and slip ratios (velocity steam/velocity water) are obtained. The laboratory tests have been made over a range of pressures from 150-600 psi and up to 80% exit steam voids.

The predicted boiling performance is shown in Fig. 30.

E. Power Plant

1. Steam System

a. Equipment

The power plant performs essentially a dual function, that is (1) it provides space heating, and (2) it generates electrical power.

The equipment necessary in connection with space heating is a heat exchanger through which 40 psig superheated steam is passed to maintain hot water in the heating system. A conventional water trap on the heating side serves to pass only condensate, thus utilizing the latent heat of vaporization of the steam. A temperature-sensitive throttle valve controls the amount of steam to the domestic heat exchanger.

The major components comprising the power plant outside of the reactor vessel are listed as follows:

(1) Steam turbine and generator	(6) Shield cooling water system
(2) Steam by-pass system	(see Section III-D-1-h)
(3) Air cooled condenser and fans	(7) Boron injection system
(4) Reactor feedwater system	(see Section III-D-1-g)
(5) Primary water purification system	

b. Steam By-pass System

Referring to the flow diagram in Fig. 31, steam from the reactor flows through the steam separator and then through the reactor pressure control valve to either the turbine or the turbine pressure regulator. The turbine pressure regulator by-passes a set amount of steam so that a decrease in turbine inlet steam pressure causes the valve to close; similarly an increase in turbine inlet steam pressure causes the by-pass valve to open. Thus, the reactor pressure is not greatly affected by changes in turbine load. This is important since turbine load changes could result in inverse effects on reactor pressure and moderator voids, thus changing reactivity. The by-pass valve is capable of passing all of the turbine steam load in the event of a turbine trip.

The steam by-pass valve controls revert to "manual" control condition in the event of a reactor scram. Also the valve control system is arranged so that the valve is automatically driven closed by the emergency power supply, following a reactor scram.

A second back pressure regulating valve serves to maintain a pressure of 40 psig in the space heat exchanger and downstream from the above-named by-pass valve. A relief valve allows discharge to the condenser when the steam line pressure exceeds 50 psig.

c. Air Cooled Condenser and Fans

Exhaust steam from the turbine is condensed within an air cooled, finned tube-type condenser operating at 5 in. of mercury absolute pressure. The air steam condenser was selected as the most practical and feasible solution to the problem of potential arctic operation where cooling water is not available.

Condenser cooling air is supplied at the proper temperature and flow rate to provide constant turbine back pressure while avoiding freezing of condensate in the tubes.

This is accomplished by controlled mixing of the circulating and incoming air streams.

Condenser outlet air is exhausted from the building by a centrifugal blower, inducing an equal flow of colder outside air to enter the mixing chamber. The recirculation rate is automatically adjusted to maintain the condenser inlet air temperature at 40F. Since this cooling method is rather unique, its performance will be thoroughly evaluated in the prototype installation. Aluminum is the principal material used in the condenser system.

Air and gases are removed by a conventional air ejector system and discharged into the stack chimney. Activity measurement of the condenser gases will be made to determine the extent of possible radiological hazard at this point.

The condenser is provided with a safety valve discharging to the air outlet duct in the event that a maximum permissible pressure for this system of 5 psig is exceeded.

d. Reactor Feedwater System

Condensed water from the condenser, air ejectors, and space heating system is collected in the hotwell tank. This water is returned to the reactor by one of two boiler feed pumps. Automatic changeover is provided in case the running pump stops. Condensate is also used as cooling water in the shield cooling heat exchangers and the air ejector after-condensers. A separate condensate circulating pump is used to supply these systems with water. The plant can run without this pump for some short time; that is, the thermal shield will not melt at once nor does the lack of condensate to the

air ejector after-cooler need an immediate shutdown of the plant. However, from an operating standpoint, the plant load should be reduced to zero as soon as possible.

Prolonged stoppage of this pump will result in a gradual temperature rise of the thermal shield, a condition which will cause the reactor to shut down automatically.

Water level in the hotwell is maintained to provide adequate submergence of the feedwater pumps. Water can be added manually for make-up reasons from the demineralized water storage system.

The returning feedwater serves as the coolant for the purification water cooler. In this cooler the feedwater of 135F is heated to 175F by the heat supplied from the purged reactor water. The feedwater is passed through a filter and then enters the reactor through a spray ring located at the level of the top of the reactor core.

e. Primary Water Purification System

Reactor water is continuously recirculated through a purification system at the rate of 3 to 5 gpm. This system removes suspended and dissolved impurities in order to prevent buildup of coatings in pipes and on the fuel elements and to minimize the radioactive carryover into the turbine. It is expected that reactor water purity can thus be held to below 1 ppm of suspended matter.

Water from the reactor is taken out near the top of the core and returned through the feedwater line. The water, coming from the reactor, first passes through a 5-gallon purge water holdup tank to reduce the N^{16} activity. Then the water is cooled by regenerative heat exchange with the feedwater.

A self-operated temperature control valve is used to control the flow of purge water in order to keep its temperature below the maximum allowable temperature of 200F, imposed by the ion exchange resins.

After cooling, the water is pumped through a filter, a cation exchanger, and a mixed-bed exchanger and returned to the feedwater line. Part of the flow by-passes the mixed bed to maintain pH between 6.5 and 7.

BORAX-III experiments have shown that the resin activity corresponds to the activity of sodium-24. Extrapolating to ALPR gives an activity at the ion exchange column surface of 5 to 20 r/hr during operation and 0.1 r/hr four days after shutdown.

Due to the small amount of uranium in the fuel meat (17.7 wt %), the uranium-aluminum alloy has a fair degree of corrosion resistance. Rapid propagation of a localized clad failure is unlikely and thus there is small likelihood that large amounts of fission products would escape from the reactor to be absorbed by the ion exchange system.

For the purpose of replacement of ion exchange resins, appropriate valving is provided to isolate the purification system from the high pressure portion of the plant. After isolation, resin is removed by opening a connection to the demineralizer resin chamber and pumping water from the "contaminated water" storage tank through the chamber. Resin is thus forced out of the vessel through a pipeline into a resin disposal container, surrounded by shielding gravel within a 55-gallon storage drum. It has been estimated that for a dose rate of 100 mr/hr at the drum wall, the dose rate at the center of the waste can would be 1.1 r/hr. This would correspond to a total source strength of 1.23×10^9 mev/sec/cu ft of resin or about 1.6×10^{-7} times the total accumulated fission product activity of the reactor core.

The purification water cooler, cation exchanger column, and mixed-bed demineralizer are located in a common vault below the surface of the gravel fill, but the instrumentation and valves are accessible from a panel on the operating floor.

2. Electrical System

a. General

The essential components and connections of the electrical system associated with the ALPR plant are shown in Fig. 32. The components of the system may be grouped functionally under five general headings:

- (1) Main Generator
- (2) Plant Auxiliaries
- (3) Simulated Plant Load
- (4) Emergency Power Supply
- (5) Auxiliary Diesel Engine Generator
- (6) External Utility Line Connections.

Each of these main divisions is treated individually below.

b. Main Generator

The main generator is a 300-kw, 0.8 power factor, 120-208-volt, 60-cycle, 3-phase, 4-wire, 1200-rpm self-ventilated unit driven by a steam turbine through a reduction gear. It is equipped with the

standard features and accessories. However, the rather stringent frequency regulation requirements dictated by the proposed ultimate use of a plant of this design necessitate a speed governor having exceptional speed regulation. A directly connected exciter, controlled by an automatic voltage regulator, supplies excitation.

Associated with the generator, but located in the Support Facilities Building, is the generator control cubicle which contains the main generator and load bus switching equipment, generator controls, and miscellaneous instruments and generator protective equipment. The generator may be connected to either or both of the two load buses, which are designated "Utility" and "Equipment." The compensating transformer and induction voltage regulator provide extremely close voltage regulation on the equipment bus.

The main generator circuit breaker is rated at 1200 amperes and has an interrupting capacity of 25,000 amperes. It is electrically operated with closing power obtained from the 208-volt a-c system and tripping power from the 24-volt station battery. The generator is protected from external faults by overcurrent trip units in the main circuit breaker, and from internal faults by differential relays. Means are provided for synchronizing the generator with the load bus, with the auxiliary diesel engine generator, and the external utility lines.

c. Plant Auxiliaries

The plant auxiliaries include various motor-driven equipment, plant lighting, and the instrument load. The feedwater pump motors and condenser-cooling fan motor are equipped with reduced-voltage starters. All other motors are started across-the-line. Motor starters are of the combination circuit breaker type and are centralized in a motor control load center in the reactor building. This load center may be fed from either of the two 120-208 volt load buses.

Reactor building lighting circuits are fed from a lighting panel incorporated in the auxiliary motor load center. Lighting circuits in the Support Facilities Building are fed from a lighting panel situated in the load control switchgear in this building. Emergency lighting is provided by small individual storage battery units which include integral chargers. These units normally operate on a floating charge with the associated lamps extinguished. A power failure causes the lamps to be automatically switched on.

Non-critical plant controls and instrumentation (both nuclear and process) are supplied through an arrangement of two interlocked circuit breakers from either the utility bus or the equipment bus. Critical nuclear and process instrumentation and controls are supplied from the

emergency power supply (see below). The deep well pump and water supply pump which serve the site water supply system are connected to the external electric utility lines only.

d. Simulated Generator Load

In order to study reactor, generator, and voltage regulator system operation under various steady-state and transient loading conditions, a dummy electrical load is provided. It consists of fan-cooled load elements and is arranged such that various combinations of load may be connected to the equipment and utility buses, respectively, up to a total equal to the full rated output of the plant.

e. Emergency Power Supply

ALPR is provided with a source of emergency electric power. This consists of a small alternator which normally is driven by a motor connected to the plant electrical system but which upon failure of normal electric power is driven by a motor connected to the station battery. Thus uninterrupted power is available to accomplish the following objectives:

1. To close the electrically-operated steam by-pass and feedwater valves with a view toward preventing pressure buildup in the condenser and consequent loss of reactor water through the condenser safety valve.
2. To drive the control rods inward in the event of rod sticking or failure to scram.
3. To operate essential nuclear and process instrumentation.
4. To inject high pressure boric acid into reactor vessel for shutdown in the event this is desired.

The equipment connected to the emergency power system is listed in Table 7.

f. Diesel Engine Generator Standby

An auxiliary diesel engine generator, rated 60 kw, 0.8 pf, is also installed and may be connected to either the equipment or the utility bus. This generator is not adequate to carry the total plant auxiliary load, and is provided primarily to train operating personnel in synchronizing and load transfer procedures.

Table 7

EQUIPMENT CONNECTED TO THE EMERGENCY POWER SYSTEMProcess Instrumentation:

1. Main steam pressure indicator
2. Main steam flow recorder
3. Reactor steam pressure recorder
4. By-pass steam flow recorder
5. By-pass valve drive circuits
6. Condenser vacuum recorder
7. Feedwater flow recorder
8. Feedwater valve drive circuits
9. Reactor water level recorder
10. Multipoint temperature indicator

Nuclear Instrumentation and Control:

1. Channel I high voltage supply
2. Channel II high voltage supply
3. Channel II amplifier
4. Instrument well exhaust fan drive motor
5. Control rod drive motors

System Valves (drive close)

1. Main steam by-pass valve
2. Feedwater control valve

g. External Utility Line Connections

Public utilities electrical power is delivered to the ALPR site by a single 13.8 kv transmission line having a nominal capacity of 2500 kva. This same line will be used to supply electric power to other experimental facilities which are expected to be constructed in the vicinity at a later date. An outdoor transformer substation is employed to reduce the transmission voltage to 2400 volts for the deep well pump and 120-208 volts for utilization in the ALPR plant. Connection to the ALPR plant proper is made through current-limiting disconnecting fuses and an electrically-operated air circuit breaker having an interrupting capacity of 25,000 amperes. Energy from the line is used to supply general facilities during periods of reactor shutdown, and, in addition, the plant auxiliaries during startup.

IV. OPERATING PROCEDURE

A. Preliminary

After the core support structure has been bolted into position and the control rod drives installed, the rods will be engaged and checked for smoothness of motion in their guided channels. The scram time and the accuracy of rod positioning will be checked, as well as instrumentation, alarms, interlocks, and limit devices.

An antimony (Sb^{124})-beryllium source will be located in a specially designed dummy assembly. During the cold reactor work this source will be inserted in various core locations as the loading proceeds, so as to provide a multiplied signal for the compensated ion chambers and BF_3 proportional counters which are placed in temporary positions in the core, enclosed in aluminum tubes. The permanent location of the source assembly will be in the quadrant of the core essentially diametrically opposite the mean location of the chambers and counters. This ensures that the neutron flux and current readings reflect the multiplication effect of the core.

A typical safe loading procedure will be followed for the early core loadings. Having determined that a particular array is subcritical, with the control rods partially inserted, the rods will be withdrawn, one at a time to new banked positions until criticality is attained, if possible. A relative multiplication curve can be drawn and extrapolated to an estimate of criticality. A log will be kept of similar plots from previous loadings, and a prediction of the next safe loading increment will be inferred. Before adding or replacing fuel assemblies, all but one or two control rods will be fully inserted. The remaining rods will be moved to a pre-determined position to provide an emergency, or back-up control such that a loading revision cannot bring the reactor to criticality.

Fuel assemblies will be inserted through one of the free nozzles in the pressure vessel lid. The operation will be observed through a viewer inserted through another nozzle. Concurrently, a measurement of relative multiplication will be made by means of a scaler connected to a BF_3 proportional counter. Unless the safety of such a procedure has been demonstrated directly from earlier loadings, no more than one assembly will be added or replaced during any loading revision. Subsequent to each loading change, another reactivity test will be conducted by withdrawing rods in preparation for the next increment of fuel.

The major pre-operational program in the prototype facility consists of the determination of the (differential and integral) rod worth in, and the reactivity of various fuel loadings in reactors at various

moderator temperatures up to 420F and 300 psig. Period measurements will be made to estimate the effect of temperature on rod worth, and on critical rod positions.

Prior to the nuclear heating of the water, the control rods will be calibrated in the hot reactor in terms of period measurements. This calibration will be used to determine the maximum rate of rod withdrawal to match a desired maximum rate of reactivity insertion. In this pre-power phase of the program, heat will be supplied to the water in the pressure vessel by passing steam from a boiler through coils inserted in the downcomer.

B. Normal Startup

It will be necessary to make a complete check of the equipment prior to the actual operation of the reactor to clear the parameters for which abnormal conditions will be indicated on the annunciator panel. Subsequent steps follow:

1. Set flux instruments to operate on the most sensitive scales.
2. Verify that water levels in the reactor and power system are in their normal ranges.
3. Start condenser cooling fans.
4. Start one feedwater pump to recirculate water back to the hotwell.
5. Check supply of control drive seal water.
6. Start shield cooling pump and check system.
7. Start purification system pump.
8. Start instrument well exhaust fan.
9. Start condensate circulating pump.
10. Put reactor on a positive period by withdrawing rods and bring up the power level to permit a gradual increase in reactor pressure. Change flux instrument scales as necessary to observe increasing flux.
11. Open main steam line stop valve to supply steam to air ejector and gland seal system. Build up vacuum before rolling turbine.

12. Open and adjust feedwater valve gradually to maintain water levels in their normal ranges.
13. Open by-pass valve to by-pass sufficient steam for warm-up.
14. Open turbine throttle valve and proceed with warm-up and turning of turbine rotor.
15. Build up vacuum to the highest possible level.
16. Bring reactor power up gradually to full pressure.
17. Increase reactor power to by-pass approximately 10% of full power.
18. Place feedwater valve, condenser fan, and by-pass valve on automatic controls.
19. Bring turbine up to rated speed and increase by-pass flow to design capacity.
20. Synchronize turbine generator and close generator breaker.
21. Gradually load generator and at the same time raise reactor output to maintain pressure.

C. Normal Operation

Operation of the steam plant is automatic where changes of load are small. Thus, small changes in turbine load merely result in compensating adjustments of the by-pass valve, and the reactor continues to operate at the same power level and only slight change in reactor pressure.

Larger changes in load, up to approximately 60-kw electrical, can still be accommodated by the by-pass control system. However, reactor power must be adjusted after substantial load changes in order to maintain a relatively constant average by-pass flow (1400 lb per hour design by-pass).

D. Normal Shutdown

After the plant has been in normal operation as a unit without external power supply the shutdown procedure may be as follows:

1. Open circuit breakers feeding artificial load.
2. Synchronize turbine generator with the public utility power supply and close tie circuit breaker (Fig. 32).
3. Gradually reduce generator load to zero, thus transferring the auxiliary load to public utility power supply. Decrease reactor power output accordingly.
4. Open generator circuit breaker.
5. Trip turbine throttle valve and let turbine coast down.
6. Scram reactor.
7. Lower reactor temperature and pressure to the desired conditions by manually by-passing steam to the condenser. Maintain reactor water level by manual feedwater control.
8. Close steam valve to air ejectors.
9. When atmospheric pressure is reached in reactor, condenser fans may be stopped, feedwater to control rods shut off, and feedwater pumps stopped.
10. Further shutdown cooling can be accomplished as described in "Reactor Shutdown Cooling" section on page 58.

E. Scram Actuated Shutdown

Upon the actuation of the control rod shutdown push button in the control room, the following sequence of events automatically takes place:

1. De-energization of control rod drive clutches (resulting in rapid insertion of control rods causing reactor shutdown).
2. Turbine trip throttle valve is tripped - emergency power supply feed changed to battery.
3. Close steam by-pass valve and revert to manual operation.
4. Close feedwater valve and revert to manual operation - condenser fan damper control reverts to manual control - condenser fan speed control reverts to manual.

V. REACTOR SAFETY EVALUATION

A. Turbine Load Changes

Normally, changes in turbine load will cause a change in turbine speed which in turn will be adjusted automatically by the steam turbine throttle valve until normal speed is recovered. The governor characteristics are such that speed variations are held within $\pm 0.16\%$ corresponding to a frequency variation of ± 0.1 cps. Transient load changes of as much as 60 kw change the frequency by less than ± 0.5 cps and steady state speed is re-established within 1.5 seconds. If the overspeed should exceed 10% (corresponding to 6 cps), the turbine trip valve will automatically close, thus protecting the equipment from overspeeding.

B. Loss of Power

A total loss of power may be visualized in the case in which the main generator circuit breaker trips during an outage of the public utility power supply. Upon loss of electrical power, the magnetic clutches on the control rod drives would be de-energized, causing the control rods to fall into the reactor core, and thus causing the reactor to shut down.

All instrumentation and equipment drive motors will be without current, except for essential equipment and instruments which are on the "emergency power supply" (see Table 7, page 47). Consequently, feed-water flow to the reactor is stopped and the feed-water control valve and steam by-pass valve will be driven closed automatically and then switched to "manual" control.

The steam pressure will rise, due to the decay heat in the reactor, until the pressure relief valve to the condenser opens at 370 psig differential pressure. Steam discharging into the condenser will increase the pressure there to 5 psig at which time the condenser relief valve will discharge to the atmosphere. It has been calculated that it will require 3 hr to boil off one foot of reactor water due to decay heat.

If the control rods do not drop in under gravity and if the emergency power system should fail to drive them in, a further increase in pressure to 385 psig will cause the reactor pressure vessel safety valve to open to atmosphere. It is estimated that it would take approximately one-half hour to evaporate the $4\frac{1}{2}$ ft of water above the core, if the nominal reactor steam rate of 9,000 lb/hr is maintained. Loss of water above the core will ultimately shut the reactor down due to the loss of top reflector. Failure of the rods to scram will be confirmed by the position indicators and limit lights.

C. Rupture of System Piping

1. Main Steam Line

A major rupture of the steam line upstream of the control valves would result in expulsion of steam and water from the vessel. The sudden drop in pressure would result in flashing of approximately 20% of the water to steam and a corresponding drop in water temperature to 212 F. Provided sufficient water remains in the vessel, the reactor will still be critical. The temperature drop supplies a positive reactivity contribution of $\sim 1\%$. Unless the reactor is shut down by insertion of the control rods, the net reactivity to be compensated by the steam would be about 2.5%.

It is probable that "chugging" would occur. In BORAX-II (a reactor similar to ALPR), the largest $\Delta k/k$ (steam) consistent with stable operation was 1.6% at 75 psig. At atmospheric pressure this value would be smaller. If the chugging continued, the amplitude of the power oscillations would increase rapidly until sufficient fluid was expelled to shut the reactor down.

Another possibility is that during this divergent oscillation, some of the fuel plates might burn out, since they would be inadequately cooled by convection heat transfer. If fuel is melted and drops to the bottom, the system will be subcritical.

A metal-water chemical reaction is unlikely since the metal must be very hot and finely divided to initiate a reaction.* Even if such a reaction should occur, the resulting incident will not exceed the Maximum Credible Incident cited in Appendix C.

It is possible that when the break occurs the escaping steam would carry all the water out with it. The dry core would be subcritical, but the decay heat must be removed or the core will melt. Radiation and convection of superheated steam can transfer heat to the vessel wall; however, the insulated vessel would not dissipate the heat with sufficient rapidity to prevent ultimate melting of the core. A pool of molten core material at the bottom of the vessel would be unmoderated, unreflected, and subcritical. The quantity of volatile fission products that escape would be less than that released in the Maximum Credible Incident.

The consequences of a major steam line rupture pose a serious potential hazard. The water and steam released will contain a high concentration of short-lived isotopes of oxygen and nitrogen, along with radioactive sodium and other corrosion products. In the event of a

*W. C. Ruebsamen, F. J. Shon, and J. B. Chrisney, "Chemical Reactions between Water and Rapidly Heated Metal," NAA-SR-197, October, 1952.

coincident fuel cladding rupture, some fission products will be released and since the building is not gas-tight, volatile products will escape. The area will be monitored and, if necessary, the site will be evacuated.

2. Feedwater Line Ruptures

In the event of a rupture of the lower feedwater line outside the vessel, water will be forced out until its level drops at least to the level of the lower spray ring. Then steam will be flashed through the ruptured line until the temperature of the remaining water is reduced below the point of atmospheric boiling. This process will drop the water level below the top of the active core causing reactor shutdown. The consequences of such an event are discussed in Part K.

3. Failure of Piping Below Operating Floor Level

Figure 36 shows the location of piping and components below the operating floor level, the respective leakage containment and recovery systems, and method of venting the gravel bed. The scale superimposed indicates the distance from the core centerline and the corresponding limiting temperature estimated for maximum summer and minimum winter conditions.

Leakage from reactor vessel piping (1) will be confined to the piping trough and drain to the waste storage sump (2). In the event of a major rupture with the reactor at pressure, the fluid will be expelled through the trough opening at the floor level, and some of it will flash to steam within the building at atmospheric pressure. The remaining water will return, via the floor drains (3), to the waste storage sump (2), both of which are beyond the critical temperature region.

Leakage from any of the pressure vessel support cylinder thermal shield cooling coils (4) will be confined to the same trough and will drain to the waste storage sump.

The pipe trough drains (5) pass through regions below atmospheric boiling temperature. In the event a pipe line failure in the trough is accompanied by a drain line failure, any resulting vapor will exhaust through the gravel vent (6) to the atmosphere.

The purification system piping (below floor level) is contained within the purification system vault (7) which drains to the waste storage sump. The piping is located beyond the critical temperature region.

In order to ensure a moisture-free atmosphere, the instrument wells (8) feature a ventilating system (exhaust fan) and a

gravity drain (9), to the outside. The experimental tube (10) drains to the instrument well. In the event of a break in these lines any vapors formed will exhaust to the atmosphere through the gravel vent annulus.

The fuel storage vapor vent lines (11), the space-heating water lines (12) and the building unit heaters water lines (13), are located beyond the critical temperature region.

D. Feedwater Pump Failure

Two feedwater pumps are installed in parallel: one pump is used under normal operating conditions; the other is available as a standby. If the discharge pressure of the operating pump should for any reason (excepting Stop switch actuation) fall below a set pressure, the standby pump will start automatically. The Process Annunciator will be actuated, giving an alarm and indication that the transfer has occurred. After the standby pump has come up to pressure, a subsequent drop in its discharge pressure will initiate a similar sequence of events to transfer back to the original pump.

Following an electrical power failure the standby pump will start automatically only if power is restored within a preset time. Otherwise a pump must be started manually.

The standby pump will also start if the discharge pressure of the operating pump falls below the set pressure without stopping the pump. Both pumps can be run simultaneously by manually starting the standby pump while the first pump is operating.

Should one feedwater pump fail when the other is not operable, an alarm is indicated at the control panel. The reactor power level tends to decrease, and the reactor water level will fall until the Low-water-level scram ultimately shuts the reactor down.

E. Chugging in the Reactor Core and Sudden Increase of Cold Feedwater Flow

Under certain extreme operating conditions, natural circulation boiling reactors have experienced oscillatory instabilities. These phenomena, generally described as "chugging," have been studied in the series of BORAX experiments and, more recently, in the EBWR. The similarity in hydrodynamic characteristics between BORAX-II and ALPR lends strong support to stable operation of ALPR at design conditions. The metal-to-water volume ratio in the BORAX-II core has been calculated to be 0.42. The metal-to-water ratio in the ALPR core is 0.49. The volume fraction occupied by the coolant is essentially the same in both reactors.

The similarity includes flow channeling except for the downcomer volumes (larger in BORAX-II). The fuel assembly volume in BORAX-II is ~60% of the volume of an ALPR assembly. Consequently, 67 BORAX-II assemblies contain essentially the same cooling volume as 40 ALPR assemblies. Unpublished BORAX-II data indicate that the 67-assembly core produced at least 6 Mw stably at a power density of 36 kw/liter of coolant (corresponding to ~3% k in steam void).

Therefore, it is anticipated that ALPR will operate stably at power densities considerably higher than the value of 17.5 kw/liter specified for 3 Mw with 40 fuel assemblies. It is intended that eventually the reactor will be operated at power levels higher than 3 Mw. Extrapolation to the 59-assembly core indicates that at least 6 Mw could be produced without "chugging" at a power density of ~24 kw/liter.

It is believed that power excursions promoted by a rapid addition of reactivity will not induce "chugging" provided the new equilibrium power level is not much above 3 Mw. If reactivity is added at reasonable power levels (e.g., 1 Mw), the reactor power will rise to compensate for this excess reactivity by the formation of additional steam void in the core.

An interruption of feedwater flow will effect changes in the temperature of the moderator and degree of subcooling, and, consequently, the reactivity and power level. A stoppage of feedwater flow would eliminate the flow of sub-cooled water into the core. If the flow is restored to its original rate, an initial transient will occur after which the reactor power will return to its previous level. The rise in reactivity is gradual since fresh feedwater represents a nominal percentage of the circulating water, and mixing occurs in the downcomer.

A sudden increase in feedwater flow may be occasioned by the simultaneous operation of both feedwater pumps. Calculations indicate that the resulting increase in feedwater flow rate to 30 gpm at a discharge pressure of 356 psig will not increase the equilibrium power level to beyond 4 Mw; hence it is unlikely that "chugging" will occur. If the transient causes a significant rapid overshoot to a high flux level, the high flux signal will scram the reactor. If the duration of the power excursion is sufficient to produce an excess steam pressure, the high-pressure scram will shut the reactor down.

F. Cladding Failure

A ruptured fuel element can be expected to contaminate both the reactor water and the steam by the release of fission products. This situation will be detected automatically by monitoring the gases being discharged at the air ejector. The detector is a scintillation crystal, the signal of which actuates an annunciator if it exceeds a preset level. The

signal may be fed to a calibrated single channel pulse height analyzer so that identification of the observed activity may be attempted. Five area monitor instruments on the operating floor drive meters with alarms on the control panel. The turbine, hotwell, and purification system are monitored several times daily.

Since the fuel meat is an alloy of uranium, aluminum, and 2 wt % nickel, containing only 17.7 wt % uranium, the corrosion resistance of the meat precludes the probability of a rapid corrosion rate which would release a dangerous concentration of fission products before the cladding failure would be detected.

When an unusual rise in activity indicates a ruptured element, the reactor must be shut down and cooled off. The water may have to be decontaminated, or replaced, and the pressure vessel refilled before the core can be visually inspected (through the cover plate nozzle) and the fuel replaced.

G. Cooling Fan Stops

The flow of cooling air through the condenser is essentially the only heat sink in the system. In case the condenser fan should fail, the exhaust steam pressure, which is normally 5 in. Hg abs, would rise to 15 in. Hg abs at which time an alarm is sounded. A further increase in pressure to 3 psig will initiate the automatic reactor shut-down system and drop the load on the generator. A further increase of pressure to 5 psig will open the exhaust safety valve.

H. Water Level Device Out of Order

1. Faulty Operation Causing Water Level to Rise

If the level float should be stuck in a low position the feedwater valve will receive a signal to open even after the normal water level has been reached in the reactor. If the trouble is not detected, all the water from the hotwell tank will ultimately be transferred to the reactor.

The insurance against having the reactor overfilled in this case is that the hotwell would reach a low level and actuate an alarm before the reactor water level reaches a dangerously high point.

2. Faulty Operation Causing Water Level to Fall

This could mean that the level float is stuck in its high position. The level signal would cause the feedwater regulating valve to close too far and the reactor water level would continue going down while the hotwell level would increase.

A high hotwell level would actuate an alarm before the reactor level would become dangerously low. However, a second safety presents itself in the form of approximately $2\frac{1}{2}\%$ drop in reactivity at the time when the top reflector is lost.

I. Shield Cooling Pump Fails

Under normal conditions the shield cooling water is circulated through a water cooler by means of a centrifugal pump, as shown on the flow diagram in Fig. 20. The secondary cooling medium in this heat exchanger is condensate circulated from the hotwell.

The shield cooling system must be kept working as long as the reactor is operating at normal power. Even after the reactor is shut down a certain degree of cooling must be maintained. This is accomplished by means of natural circulation of the shield cooling water through the air-cooled radiator, which is located under the ceiling of the operating room. This system, which goes into action immediately when the pump has stopped, can take care of about 6% of the normal operating requirement without boiling taking place in the cooling coils. For special emergencies water from the water storage tank can be drained through the shield cooling coils from which it will overflow by gravity into the retention tank. No electric power is required for either of the two emergency cooling systems, but their capacity will be adequate only after the reactor is shut down. The safety of the shield cooling is therefore dependent on the reactor being shut down in case this cooling should fail. There are three points in the system which could set off the alarm: (1) a failure of the pump pressure; (2) a rise in water temperature above a certain safe limit; and (3) a rise in the temperature of the shielding plate above normal.

It has been estimated that the reactor can continue to operate without shield cooling for at least $1\frac{1}{2}$ hr without melting the lead which surrounds the shield cooling coils. The reactor should be shut down after the alarm, unless the error can be corrected quickly and adequate coolant flow restored. In any event, a continued increase in shielding plate temperature will automatically scram the reactor.

J. Reactor Shutdown Cooling

At the time the reactor is shut down, the reactor core produces heat equivalent to 6% of operating power (equivalent to a steam rate of approximately 550 lb/hr). This decay heat production decreases 1% approximately six hours after shutdown (~ 90 lb/hr).

After shutdown the steam is passed through the pressure regulated bypass valves and is condensed in the space heating heat exchanger. This cooling off can be accomplished without operation of the condenser and air-removal ejectors after the turbine is shut down.

The condensate from the heat exchanger is returned via the flash tank and hotwell to the reactor. By manually adjusting the steam bypass valve, the reactor pressure and temperature can be reduced below atmospheric boiling conditions within an estimated time of five hours. Then further cooling can be accomplished by bypassing 3 gpm from the reactor water purification system to the heat exchanger. This water is returned to the reactor reduced in temperature by approximately 50 F. At this time the reactor feed pumps may be stopped since steam is no longer being condensed outside the reactor. Figure 33 shows the reactor shutdown temperature as a function of time after shutdown based upon the shutdown procedure described above.

K. Danger of Core Meltdown

Rupture of the main steam line or feedwater line (Part V.C) will result in the loss of large quantities of reactor water. The Low Reactor Pressure Scram, and Low Reactor Water Level Scram will be actuated in addition to the Low Hotwell Level Alarm. Should the control rods fail to go in, the reactor will ultimately be shut down when the reactor water level reaches the top of the fuel plates. Meanwhile the operating power would be decreased due to decrease in moderator density, loss of subcooling and partial loss of top reflector.

Whether shutdown is caused by rod scram or loss of top reflector water, the decay heat generated in the core will have to be removed. Assuming feedwater flow cannot be re-established, the normal shutdown cooling procedure is not possible. Water cannot be added by gravity feed if the system is still under pressure or if occupancy of the operating floor is hazardous.

The evaporation of water will ultimately expose the tops of the fuel plates. As the water level decreases, the exposed fuel plates are cooled by the steam rising between them, and by axial conduction to the immersed section. If no heat is transferred by convection to steam, or by conduction to side plates, the axial temperature distribution in the plate is given by:

$$\Delta T(x) \cong \frac{Qx^2}{2k}$$

where

Q = average heat generation rate per unit core volume

x = axial distance above the water level

k = thermal conductivity of aluminum at 420F = 124 Btu/(ft)(hr)(F).

The results are tabulated below. Decay heat is computed from data by Untermeyer and Weills (ANL-4790). The time of shutdown is chosen to be when the water level is at the tops of the plates.

FUEL PLATE AXIAL TEMPERATURE DIFFERENCE RELATIVE
TO TIME AFTER SHUTDOWN AND WATER LEVEL

x(ft) Distance Below Fuel Plate Tips	t(hr) Time After Shutdown (x assumed 0 at t = 0)	Decay Heat % of Full Operating Power	ΔT (F)
0.5	1.5	1.4	61
0.75	2.8	1.2	116
1.0	4.0	1.1	190
1.5	7.2	0.95	368
2.0	11.0	0.88	606
2.3	12.7	0.86	774

Ample time is available for the crew to take remedial action, but if this is not done, the water level will ultimately drop below the core. Eleven hours after shutdown the decay heat is about 25 kw or about 180 Btu/(hr)(sq ft) of core heat transfer surface. Thus if any water remains in the vessel a heat transfer coefficient even lower than 1 Btu per hour per square foot is sufficient to keep plate temperatures below melting. Once the vessel is at atmospheric pressure and access to the building is possible, cooling water may be added by gravity feed.

In the event that additional flashing occurs after fuel plates have been exposed the resulting time for the crew to take action is reduced. Even if a foot of water is flashed the time available can only be shortened by two to three hours.

APPENDIX A

REACTIVITY OF STORED FUEL

The reactivity of a storage well filled with fuel assemblies was estimated as follows. The arrangement of boral is such that only a small area of each of three alternate outer tubes is not covered by a thermally black absorber. These areas face a thin region of water and steel and are separated from the next well by at least $1\frac{1}{2}$ ft of a poorly reflecting, poorly moderating medium of gravel. Interaction effects between storage wells, and with the reactor are small. A Cell calculation was made to determine the fraction of thermal neutrons available which are captured by the boral and the water. Assuming that the fuel assemblies are undepleted (350 grams U^{235} per assembly), unpoisoned, and that the boron-aluminum side strips are missing from all assemblies, the calculation of the cold cell showed that $k_{\infty} < 0.9$. That is, even the most reactive infinite array of such tubes would be subcritical. Estimating the aluminum-to-water ratio to be at least 0.2 in the "core," using a radius of 23 cm (≈ 9 in.) as an overestimate of the effective core size, and adding in a radial reflector savings of 8 cm, the neutron non-leakage factor is found to be smaller than 0.8. Thus $k_{eff} \lesssim 0.7$ is obtained, which allows a large margin of error for uncertainties of calculation and of future fuel loadings.

The temperature and void coefficients of reactivity are negative. Addition of boron side strips to the assemblies would subtract another 10-11% k. Since boral partially surrounds the fuel, the reflector savings is overestimated. Therefore, with the exception of the assumptions that a cell calculation is appropriate and that each fuel assembly is completely bounded by a thermally black absorber, it may be seen that all assumptions lead to an overestimate of reactivity.

APPENDIX B

EVALUATION OF HAZARDS ARISING FROM: I. MALFUNCTIONING OF RODS DURING COLD STARTUP. II. IMPROPER INSERTION OF FUEL ASSEMBLIES

I. Malfunctioning of Rods During Cold Startup

An antimony (Sb^{124})-beryllium source will be located at the periphery of the core, diametrically opposite the BF_3 counter in the shield. The size of the source was chosen so that the subcritical neutron flux is detectable by the counter. The intensity of this source, at least 1.5×10^9 neutrons/sec, combined with a multiplication of ten (i.e., $k_{\text{ex}} = -0.10$), is expected to be adequate for this purpose.

Analysis

Place a fission neutron point source of effective intensity AS n/sec at the center of a subcritical spherical bare reactor. Close to criticality, the fundamental mode predominates, and the solutions to the two-group diffusion theory equations are approximately:

$$\phi_s(r) = -\frac{AS}{2R^2 k_{\infty} \Sigma_{a_s} k_{\text{ex}}} \frac{\sin \frac{\pi r}{R}}{r}$$

$$\phi_f(r) = \frac{\Sigma_{a_s}}{\Sigma_{a_f}} \left(1 + \frac{L^2 \pi^2}{R^2} \right) \phi_s(r)$$

In these formulas, ϕ_f and ϕ_s are the two-group fast and slow neutron fluxes, respectively; R is the radius of the equivalent bare core; Σ_{a_s} and Σ_{a_f} are, respectively, the slow macroscopic absorption cross section, and the fast slowing-down cross section; S is the source intensity (neutrons/sec); A represents the relative statistical worth of the actual source in the reactor.

The cold ALPR reactor may be approximated as a spherical core of equal volume (radius $r_c \approx 43$ cm for the forty-assembly core) augmented to a bare-equivalent core of radius $R \approx 51$ cm. The actual source emits 0.03 Mev neutrons and is considered as a point source in the calculations. Using the two-group perturbation theory result that the worth of a fission neutron source varies, spatially, as the adjoint fast flux, $A \approx 1/6$ accounts for the peripheral location of the source.

Then $\phi_s(0) \approx 2 \times 10^5$ and $\phi_f(r_c) \geq 1 \times 10^5$, if $k_{\text{ex}} = -0.10$.

The (space-)asymptotic thermal neutron flux in the water reflector is proportional to the fast flux, since the direct thermal leakage from the reactor is attenuated more rapidly. After six inches of water, or 58 cm from the core center:

$$\phi_f(58) \approx \exp[-0.18 \times 15] \times \frac{43}{58} \times \phi_f(r_c) \geq 5 \times 10^3$$

and

$$\phi_s(58) \geq 1.2 \times 10^4$$

An attenuation factor for the remaining thicknesses of water, steel thermal shield, water, pressure vessel, and support cylinder may be inferred from slab experiments reported by R. Rickert and A. D. Rossin.* The composition of this factor ($\approx 10^{-3}$), geometry corrections, and the level of thermal neutron flux deduced above yield the result $\phi_s \geq 10$ at the position of the BF_3 counter. Even with a further flux depression in the counter this flux is easily detected.

During a startup from a cold subcritical condition, reactivity is added at the maximum rate of 0.0001 k/sec. Consider a case of a ramp insertion of reactivity: $k_{\text{ex}}(t) = -0.05 + 0.0001 t$. It will be seen that the maximum reactivity possible during such a startup is less than β ($\equiv$ prompt criticality). Therefore, for the range of prompt mean neutron lifetimes anticipated during the core cycle, the neutron kinetics are essentially independent of the lifetime, ℓ^* . The value $\ell^* = 4 \times 10^{-5}$ sec was selected as representing the shortest lifetime expected. In Fig. 34, the solution of the kinetics problem is depicted. The AVIDAC was used to solve for the thermal neutron density, $n(t)$, using six delayed groups (Hughes' data), $k_{\text{ex}} = -0.05 + 0.0001 t$, and a source of such intensity that $n(0) = 1$. Also plotted in Fig. 34 is the ratio

$$n(t) / \frac{dn(t)}{dt}$$

If the flux were rising exponentially, this ratio would be the reactor period; in this case it is simply an indication of the rate of flux increase.

It may be seen that the flux at criticality ($t = 500$ sec) is thirty times the level at time zero, hence sixty times the equilibrium level at $k_{\text{ex}} = -0.10$. The flux at the center of the core is then

$$\phi_s(t = 500) \approx 10^7 n \text{ cm}^3/\text{sec},$$

and the power level is of the order of two watts. If this ramp insertion is continued, full power is reached approximately 57 sec later, at which time the reactor is on a one-second "period," and the reactor is 0.57% supercritical.

*ANL-5601, Reactor Engineering Division Quarterly Report Section II, April, May, June, 1956.

It should be noted that "two watts" is only an order of magnitude. If this number should actually be 0.2 watts or 20 watts at criticality, to a good approximation the only change involved is a shift of several seconds in the time required to reach full power. Such an uncertainty does not alter the inferences to be drawn from a plot of heat output after startup, shown in Fig. 35, namely: approximately 4.5 megawatt seconds of heat will have been produced by the time the flux reaches a high-flux level corresponding to 150% of full power, and less than 35 mws would be released in the following two seconds. Since the rods will be some distance into the core at this time, the 0.6% k to be controlled by rod scram will be achieved within two seconds after the high-flux level is reached.

If the rods are scrambled within the next two seconds, the reactor flux will not rise to the flux level otherwise reached at 560 sec, corresponding to a power level of 90 mw. Assuming that the ratio of maximum to average thermal flux is less than five, the maximum heat transfer rate is less than 150 times the average rate, or less than 4×10^6 Btu/(hr)(ft²). As may be seen* even under these circumstances, the surface temperature stays well below the melting point of aluminum. The conductivity of the thin aluminum plates and the relatively long period of the transient, ≈ 1 sec, ensure that the plates do not melt.

An extrapolation from experimental data is possible, although these data were obtained for reactors (BORAX, SPERT I) in which the fuel plates were 0.060 in. thick whereas the ALPR plate thickness is 0.120 in. From the available information** several trends may be deduced.

First, for a given reactor configuration, the maximum plate surface temperature attained during short-period transients starting from low power in a cold reactor increased as the period decreased. For periods longer than 0.1 sec, this maximum temperature did not exceed 260F in the BORAX I and BORAX II experiments reported. It is believed that bubble formation controlled the excursions for periods shorter than 0.1 sec, though it is

*ANL-5532, "Results of Recent Analyses of BORAX II Transient Experiments," A. J. Ulrich, Fig. 5.

**ANL-5211, "Transient and Steady State Characteristics of a Boiling Reactor," The BORAX Experiments, 1953.

ANL-5323, "Experimental Investigation of the Self-Limitation of Power During Reactivity Transients in a Subcooled, Water-Moderated Reactor," BORAX I Experiments, 1954.

ANL-5532, "Results of Recent Analyses of BORAX II Transient Experiments," A. J. Ulrich.

IDO-16285, "Experimental Investigation of Reactor Transients," W. E. Nyer, et al, April 20, 1956.

debatable whether these bubbles arose from radiolytic decomposition of the water or from steam formation. It has been found, both in BORAX and in SPERT, that for such excursions there arise negative reactivity effects not easily explainable in terms of the heating of water. In the very short period excursions, e.g. 30 msec, it is obvious that the power rise was terminated rapidly by the large negative reactivity effects of bubble formation, almost certainly steam, considering the large heat transfer rates attained at the peak of the excursion.* Even in these cases, however, the maximum plate surface temperature did not exceed 400F.

Secondly, as mentioned in the preceding paragraph, the total energy transferred to the water before the first excursion is terminated does not appear to raise the bulk fluid temperature enough to introduce a significant negative reactivity. Until the mechanism of heat transfer is shifted to boiling heat transfer, the temperature distribution in the plate is essentially flat unless the period of the excursion is substantially smaller than one second. (The solution of the heat conduction problem: clad fuel plate in a stagnant coolant is available.** The "flatness" of the temperature distribution was demonstrated by the substitution of the ALPR constants in those formulas.) Thus, unless the surface temperature exceeds saturation temperature, there is no possibility of fuel plate melting. It is concluded that the magnitude of the negative temperature coefficient is of little importance to these calculations of melt-down.

In view of the very small energy requirement to shut down the reactor by the rapid formation of steam at the surface of a hot plate, it is reasonable to conclude that the heating of only a few fuel plates to as much as 300F (surface temperature) will shut down the reactor. Within the range of negative void coefficients of BORAX II, and ALPR, this shutdown will occur rapidly, and the exact magnitude of the coefficient is of little importance. For the same reason, it is not necessary to worry about the exact ratio of heat transfer area to coolant volume. It remains to be shown that the maximum temperature of the meat is less than the melting point of that alloy when the heat input is exponential, with a one-second period, and the surface temperature reaches 300F.

If the plate attains a surface temperature as high as 300F, the heat transfer rate, Q/A , is approximately 3×10^6 Btu/(ft²)(hr). This transient heat transfer was shown by A. J. Ulrich*** to satisfy a relationship similar to those obtained for equilibrium boiling heat transfer. The rate is proportional to $(\Delta T)^n$, where n is somewhat larger than unity, and ΔT is the difference between the surface temperature of the plate and the saturation

*ANL-5532, Ibid.

**Reactor Handbook, Vol. 2, Engineering, AECD-3646.

***ANL-5532, Ibid.

temperature of the coolant. Therefore, if the boiling heat transfer boundary condition is replaced by the condition

$$-k \left. \frac{\partial(\Delta T)}{\partial x} \right]_{\text{surface}} = h \Delta T \Big|_{\text{surface}}$$

and a value of h is selected for which $h \max \Delta T \Big|_{\text{surface}}$ is at least $3 \times 10^6 \text{ Btu}/(\text{ft}^2)(\text{hr})$, the temperature gradient in the plate is overestimated, and so is the temperature at the center of the fuel meat for this range of surface temperatures.

The purpose of these arguments is to show that a relatively trivial calculation can prove the obvious: the fuel plates will not be melted even if the rods should fail to scram within several seconds after the high-flux level is reached. What is desired is the asymptotic solution of the problem of a clad fuel plate in a flowing coolant for an exponentially rising ($e^{\alpha t}$) heat source in the meat. This solution was provided by H. Greenspan.* The ratio of the center meat temperature to the temperature of the cooled clad surface is found to be

$$\frac{T_1(0,t)}{T_2(b,t)} = \frac{\cosh \left[\sqrt{\frac{\alpha}{\kappa}} \left(\frac{k}{h} + b \right) \right] - \cosh \left[\sqrt{\frac{\alpha}{\kappa}} \left(\frac{k}{h} + b - a \right) \right]}{\sinh \left[\sqrt{\frac{\alpha}{\kappa}} a \right] \sinh \left[\sqrt{\frac{\alpha}{\kappa}} \frac{k}{h} \right]}$$

where α is the reciprocal of the period of rise in the level of the heat source, k is the conductivity and κ the diffusivity of the heat and clad (assumed to be identical with respect to these properties), and a is the half-thickness of the meat and b the clad thickness (in feet, since h is measured in that system of units). The functions T_i ($i = 1, 2$) represent the rise in temperature above saturation.

For $\alpha \approx 1/\text{sec}$ and $h = 3 \times 10^4 \text{ Btu}/(\text{ft}^2)(\text{hr})(\text{F})$, this ratio is approximately 2, and the maximum meat temperature is less than 450F when the surface temperature reaches 300F. Thus it is concluded that shutdown occurs before the plates can melt. The maximum reactivity inserted by rods to this point cannot make the reactor prompt critical.

II. Improper Insertion of Fuel Assemblies

A standard procedure will be followed during loading of fuel assemblies. Before loading an additional assembly in the cold reactor, or before replacing one, all but one or two control rods will be fully inserted. This is a scheme usually adopted to provide a margin of safety during loading changes. The neutron flux multiplication is checked by means of the BF_3

*Reactor Handbook, Vol. 2, Engineering, AECD-3646, p. 149,
H. Greenspan

counter to ensure reactor subcriticality. Then the loading change is effected slowly while maintaining a check on multiplication. If this procedure is followed, the reactor is always subcritical during a loading alteration.

According to results of detailed theoretical analyses, the five cross control rods can handle the most reactive core comprised of forty fuel assemblies containing boron strips. The addition of one fuel assembly at the periphery of the reference core is a reactivity effect considerably smaller than 0.5% k. The analysis of the previous section indicates that even a step-insertion of that much reactivity will not melt the fuel plates. The accidental replacement of the (statistically) most important assembly in the forty-assembly core by one not containing a boron strip has been calculated to produce a reactivity increase of less than 0.5% k.

Thus, even if these reactivity insertions should arise from a loading procedure in direct violation of the rules, there is no likelihood of a meltdown.

APPENDIX C

MAXIMUM CREDIBLE INCIDENT

The problem chosen for an evaluation of a "maximum credible incident" is one in which the complete destruction and dispersal of the entire core is visualized. It is also assumed that this event takes place after the reactor has been operating for an infinite length of time. Although these highly pessimistic conditions were selected, the results of the calculations fell within previously established limits.

The reaction could be best imagined to be initiated by an explosive charge producing sufficient heat to melt the aluminum and disperse the hot material in water and steam. The actual effect of an aluminum-water reaction would be to produce more heat, and more pressure on the vessel, and in turn on the building if the vessel fails. A nuclear excursion could be predicated; however, it is shown in this report that it is not likely that a transient could be produced in ALPR which would heat the plates to melting temperature before the reactor shuts itself down by expulsion of moderator. A faster addition of more reactivity than was added in BORAX-I would be required. It might be mentioned that the longer-lived radiation hazard resulting from a BORAX-I nuclear transient is at least two orders of magnitude less than the case discussed below.

The maximum hypothetical incident can then be described as follows:

Heat required

to boil all H ₂ O in core	85.7 x 10 ⁶ calories
to heat and melt all aluminum in core	<u>57.4 x 10⁶ calories</u>
Total to initiate reaction	143.1 x 10 ⁶ calories
	or 600 megawatt-seconds
	or 1.9 x 10 ¹⁹ fissions
	(Approx. 4- $\frac{1}{2}$ x BORAX)

The following reactions may take place:

There are 3×10^5 gm aluminum in the core so 5700 moles Al₂O₃ are formed yielding 2.2×10^9 cal. if the metal-water reaction takes place and goes to completion. Oxygen is limited so assume little H₂O recombination. It takes 1.4×10^9 calories to vaporize the rest of the water in the vessel so despite heat losses, the resultant pressure should burst the vessel and probably the building wall.

If one-fourth of the core material is now released from the building in a cloud, the fission product activity may be treated as follows:

$$\text{Total activity} = 0.1 (0.25 P_0) (t + 10)^{-0.2*}$$

P_0 = megawatts of operating power

t = time after end of operation in seconds

If the volume of the cloud (in cubic meters) is $L \times W \times H$, the specific activity is:

$$A_{sp} = \frac{0.1 (0.25 P_0) (t + 10)^{-0.2}}{LWH} \frac{Mw}{m^3}$$

Then if the distance from the reactor is d (in meters), W can be taken as $d/7$ as a rough approximation. If an observer receives 60% of the dose in the cloud (assuming dose equals source if cloud is large) over the time T during which cloud is passing him at wind velocity V (meters/sec):

$$\text{Total Integrated Dose} = T A_{sp} = \frac{0.105 P_0 (t + 10)^{-0.2}}{dVH}$$

Since $d = VT$, and if the 10-second correction is ignored (which is all right for times of the order of minutes or more),

$$\text{Total Integrated Dose} = \frac{(0.105) (0.923 \times 10^8) P_0}{(d)^{1.2} (V)^{0.8} (H)} \text{ roentgens.}$$

H is taken to be that distance where the diffusion gives a concentration of $1/10$ that at the cloud center, by Sutton's equation.**

$$H = \left[\ln 10 \right]^{1/2} C_y \left(\frac{2 - n}{2} \right)$$

The exclusion radius is that value of d in meters for which R equals 300 roentgens.

$$d^{(2.2 - \frac{n}{2})} = \frac{(0.97 \times 10^7) P_0}{(\ln 10)^{1/2} C_Z (300) (V^{0.8})}$$

*S. Untermyer and J. T. Weills, "Heat Generation in Irradiated Uranium," ANL-4790, February, 1952.

**"Meteorology and Atomic Energy," AECU-3066.

The most dangerous situation involves low wind velocities and stable meteorological conditions, allowing the cloud to diffuse slowly along the ground, in the direction of the wind. For this case the Sutton parameters are taken as $C_Z = 0.05$, $n = 0.5$, and $V = 1$ meter/sec (2.24 mph).

Thus:

$$d^{1.95} = \frac{(2.13 \times 10^4) (3)}{0.05}$$

$$d = 1352 \text{ meters} = 0.84 \text{ miles.}$$

This distance, d , is the standard exclusion radius for an integrated radiation dose of 300 roentgens.

For these atmospheric conditions the exclusion radius is seen to be roughly proportional to the square root of the power level (or proportion of the core which is dispersed in the cloud) and inversely proportional to the square root of the limiting dose level. Thus if the entire core is dispersed the exclusion radius is only doubled (to 1.7 miles). An exclusion radius for an integrated dose rate of 30 roentgens, which is within the permissible 30-year cumulative dose, would be 2.74 miles. Obviously these relatively small exclusion radii would even be smaller if wind velocities (and the diffusion parameter, C_Z) are greater.

Since doubling the reactor power results in an increase of about 40% in the exclusion radius, the maximum accident considerations should not limit possible increases in operating power if no other safety provisions are violated. Considering the isolated location of the ALPR, it is clear that no appreciable hazard to off-site population is envisioned.

The total beta dose calculated at a point 1000 meters downwind from the site (about the distance to Highway U.S. 20) under the above conditions, is 35 rep to an unshielded body. Presence of clothing effectively cuts this value down roughly by a factor of ten.

It takes fifteen minutes for the cloud to reach an observer at the above mentioned point 1000 meters downwind. With this simple model of the cloud, it is calculated that the iodine activity accumulated at the thyroid due to inhalation is about $70 \mu\text{c}$, assuming 15% of the iodine inhaled goes to the thyroid. The maximum permissible concentration at this organ is given as $0.06 \mu\text{c}$ for the concentration of I^{131} which results in an average dose of 0.3 rem/week for the rest of a lifetime.* Twenty-five rem is given as the maximum permissible one-shot dose, without causing appreciable

*W. C. Redman, J. Thie, L. Dates, "Hazard Summary Report on the Oxide Critical Experiments," ANL-5715 (April, 1957).

detrimental effect on the human body. An initial concentration of about $2.4 \mu\text{C}$ of I^{131} will yield a dose of 25 rem to the thyroid. Under the conditions of this model the result is a dose of about 750 rem, a lethal dose. It is clear that the conditions chosen are highly pessimistic even for the unlikely event described. Choosing average daytime wind condition at NRTS results in a diminution of this dose by two orders of magnitude at 1000 meters from the reactor (and not in the direction of the road).

FIG. 2
ACCESS TO ENCLOSURE

FIG. 3
REACTOR BUILDING

1. TURBINE-GENERATOR	11. WASTE STORAGE TANKS
2. HEAT EXCHANGER	12. FEED WATER LINE FILTER
3. WATER STORAGE TANK (OVERHEAD)	13. PURIFICATION SYSTEM AREA
4. MOTOR CONTROL BOARD	14. CONTAMINATED WATER STORAGE TANK
5. COVERED STAIRWAY	15. BORON STORAGE TANK
6. CONDENSATE CIRCULATING PUMP	16. COVERED EMERGENCY STAIRWAY
7. HOTWELL (OVERHEAD)	17. CONTROL ROD DRIVE MOTORS
8. FEED WATER PUMPS	18. CONCRETE SHIELD
9. FUEL STORAGE WELLS	19. SUPPORT FACILITIES BUILDING ✓
(CONTROL ROOM LOCATION)
10. EQUIPMENT DOORS	

FIG. 4
OPERATING FLOOR PLAN

FIG. 5
FAN FLOOR PLAN

0 2 4 6
SCALE - inches

FIG. 6
CORE VERTICAL SECTION

FIG. 7
CORE PLAN SECTION

FIG. 8
CORE LOADING PATTERN

BORON STRIP FULL LENGTH
OF SIDE PLATE

FIG. 9
FUEL ELEMENT

FIG. 10
CROSS TYPE CONTROL ROD

FIG. 11
CONTROL ROD DRIVE

FIG. 12
CONTROL ROD DRIVE TEST FACILITY

- LEGEND
- 1. LAMINATED TOP SHIELD
 - 2. CONCRETE SHIELD
 - 3. DRY SHIELD MIXTURE
 - 4. INSTRUMENT WELL
 - 5. CONTROL ROD
 - 6. CORE STRUCTURE
 - 7. GRAVEL
 - 8. DRY SHIELD MIXTURE
 - 9. THERMAL SHIELD
 - 10. PRESSURE VESSEL - 4'-6 O.D.
 - 11. INSULATION
 - 12. AIR SPACE
 - 13. SUPPORT CYLINDER
 - 14. LEAD THERMAL SHIELD AND COOLING COILS
 - 15. FEED WATER SPRAY RING
 - 16. SEPARATOR RETURN LINE
 - 17. PURIFICATION PURGE LINE
 - 18. FEED WATER INLET
 - 19. BORON SPRAY RING
 - 20. STEAM LINE
 - 21. CONTROL ROD DRIVE MOTOR
 - 22. CONTROL ROD DRIVE
 - 23. BORAL SHIELD

FIG.13
REACTOR INSTALLATION
VERTICAL SECTION

FIG. 14
CONTROL ROD DRIVES AND TOP SHIELDING

FIG. 15
FUEL ELEMENT TRANSFER SYSTEM

FIG. 17
FUEL ELEMENT STORAGE WELL

FIG. 18
EVAPORATION FROM FUEL STORAGE WELL
(1/3 REACTOR CORE)

FIG. 19
BORIC ACID INJECTION SYSTEM

FIG. 20
SHIELD COOLING SYSTEM

POWER LEVEL IN WATTS
BY DECADES

INSTRUMENT CHANNELS

- I MICROAMMETER WITH SAFETY TRIP
- II MICRO-MICROAMMETER WITH SAFETY TRIP
- III LOG LEVEL AND PERIOD METERS
- IV ELECTROMETER
- V AND VI COUNTERS, NO'S. 1 AND 2

FIG. 21
FLUX INSTRUMENTATION

FIG. 22
NUCLEAR INSTRUMENT BLOCK DIAGRAM

FIG.23
CONTROL CELL

FIG. 24
THERMAL NEUTRON FLUX IN CYLINDRICAL MODEL OF CONTROL CELL
(FRESH REACTOR OPERATING AT 3 mw)

FIG. 25
 AXIAL VARIATION OF FAST
 AND THERMAL NEUTRON FLUX

FIG. 26
FRACTIONAL CONTROL ROD WORTH
VS FRACTIONAL CORE PENETRATION
BY ROD BANK

FIG. 27
RADIAL VARIATION OF FAST AND THERMAL
NEUTRON FLUX

FIG. 28
REACTIVITY VARIATION DURING CORE LIFETIME

FIG. 29
 STATISTICALLY-WEIGHTED XENON EFFECTIVENESS AFTER
 COMPLETE SHUTDOWN OF DEPLETED REACTOR FROM
 3 mw EQUILIBRIUM POWER

FIG. 30
BOILING PARAMETERS 300 psi 130-175 F FEED WATER
(40 ELEMENTS)

FIG. 31
FLOW DIAGRAM

FIG. 33
REACTOR SHUTDOWN TEMPERATURES

FIG. 34a
 NEUTRON KINETICS DURING ROD
 WITHDRAWAL IN COLD REACTOR

FIG. 34b
 NEUTRON KINETICS DURING ROD
 WITHDRAWAL IN COLD REACTOR

FIG. 35
REACTOR ENERGY OUTPUT DURING COLD STARTUP
(NORMALIZED TO 2 WATTS REACTOR POWER AT 500 sec.)

FIG 36
REACTOR SUB-OPERATING FLOOR
COMPONENTS AND CALCULATED
GRAVEL SHIELD
TEMPERATURE DISTRIBUTION