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FINAL SAFEGUARDS ANALYSIS HIGH TEMPERATURE LATTICE TEST REACTOR

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FINAL SAFEGUARDS ANALYSIS

HIGH TEMPERATURE LATTICE TEST REACTOR

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FINAL SAFEGUARDS ANALYSIS

HIGH TEMPERATURE LATTICE TEST REACTOR

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FINAL SAFEGUARDS ANALYSIS

HIGH TEMPERATURE LATTICE TEST REACTOR

I. INTRODUCTION

The High Temperature Lattice Test Reactor (HTLTR), located at the Pacific Northwest Laboratory, in the 300 Area of the Hanford Works, is a versatile, low-power research reactor designed to obtain nuclear data for application to large, high thermal efficiency central power stations; the data derived have indirect application to small special purpose reactors operating at very high temperatures. The HTLTR provides a unique combination of advantages for the experimental determination of a) the nuclear characteristics of high temperature thermal reactor cores and b) basic reactor physics parameters at high temperatures:

Precision and Accuracy

The HTLTR operates on the established principle of the Physical Constants Testing Reactor (PCTR), in which a void is substituted for a small central cell (test cell) of an experimental lattice (test core) driven by the reactor (driver section). Suitable buffer zones surround the test core so that the test cell is in the same flux environment as it would be in a critical assembly of its own kind. This method has been shown[1] to give values of the neutron multiplication factor of an accuracy and precision equal to those obtained from a complete critical assembly.

1 See References, page 12.1

Economy

Typically, a lattice test of a low-reactivity system in the HTLTR can be made with 40 to 80 cubic feet of test volume. In comparison, an exponential assembly would require 400 to 600 cubic feet, and a critical assembly 8000 cubic feet. The ratios are not so large in high-reactivity systems, but a considerable advantage is still maintained. When the effects of changes in the multiplication factor caused by small changes in geometry, composition, or operating parameters are to be explored, the economy of working with the small test volume is apparent.

Because of the small quantities of special materials to be procured and fabricated, experimental data may be obtained more quickly and with much less lead time in HTLTR than in other ways. Experiment times of two to six months will suffice for nearly all experimental lattice programs.

Safety

Many nuclear safety features are inherent in the HTLTR concept. Loading and startup hazards, usually considered as leading to the most severe consequences in low power test reactor accident analyses, are minimized, since the overall nuclear characteristics of the reactor are almost entirely established by the enriched uranium driver region. Only a small, subcritical assembly is of unfamiliar arrangement and characteristics for each test.

Fission product inventory will be low because power levels no greater than two kilowatts for short time intervals will be required. The control of fission products is therefore only of minor concern.

Total excess reactivity will be minimum, as excess reactivity for xenon transients or fuel lifetime is not necessary.

A negative prompt temperature coefficient of reactivity is inherent in the driver fuel. On the other hand, reactivity changes upon heating or cooling the reactor are very slow because of the large heat capacity of the graphite moderator stack.

These inherent safety factors make the HTLTR easy to control by simple means.

II. SUMMARY

II. SUMMARY

The High Temperature Lattice Test Reactor (HTLTR), located in the 300 Area of the Hanford Works, is a low power, high temperature research reactor. It is to be operated by Battelle-Northwest to obtain basic reactor physics data and design data applicable to large, high power nuclear reactors. The experimental programs will parallel those of the Physical Constants Test Reactor (PCTR), except that measurements in HTLTR can be made at high temperatures.

The building site is shown to be suitable for the reactor as designed.

The reactor building consists of two major sections, the reactor enclosure and the services structure. The reactor enclosure is of ordinary reinforced concrete. The reactor is on the ground floor, and the various process sytems are in the basement. The walls and roof of the reactor enclosure serve as the radiation shield. Shield doors provide access to the reactor room and basement. The reactor enclosure is neither sealed nor designed to withstand internal pressures; it does, however, have a separate ventilation system exhausting through high-efficiency filters which maintains the enclosure at a slight negative pressure for contamination control. The services structure is an abutting two-story steel building, which contains the control room, experimental assembly room, offices, and necessary utilities and services.

The reactor consists of a ten-foot cube of graphite moderator, pierced by holes which are either loaded with slightly enriched uranium driver fuel, experimental test fuel, poison shim rods, control rods, and graphite heater bars, or are plugged with graphite. The central 5 x 5 x 10 foot section can be completely removed and altered to any composition or configuration desired for an experiment.

The driver fuel is 5% enriched UO_2 ceramic pellets clad in graphite. This fuel surrounds the test core, on the ends as well as radially. The critical mass will vary, typically, from 30 to 60 kg of U-235. Shim rods containing gadolinium oxide will be loaded as needed to compensate the slow negative temperature coefficient of the reactor. The test core will contain from zero to six kg of fissionable isotopes such as Pu-239, U-233, and U-235, and varying quantities of other materials such as uranium, thorium, copper, nickel, iridium, rhodium, samarium, etc. These materials will be in forms suitable for the experiment temperature.

The reactor is electrically heated to the design temperature, 1000 C, with graphite heater bars powered with 475 kW of low voltage ac. The moderator block is thermally insulated and is contained in a steel shell which serves as the envelope for the nitrogen blanket and the primary barrier to the escape of contamination. A Boral liner in the steel shell reduces neutron leakage to the room and capture gamma-ray production in the steel shell.

The reactor flux level is controlled by 8 shutter-type horizontal control rods and 4 vertical blade-type safety rods. The control rods also act as a safety system since they may be closed quickly, driven by springs external to the reactor shell.

The startup neutron source is a positive ion accelerator which focuses a beam of deuterons onto a beryllium target.

The reactor neutron kinetics are largely established by the driver fuel. The U-238 in the driver provides a prompt negative power coefficient of reactivity. The neutron lifetime of about 1 millisec and the delayed neutron fraction of about 0.6% are also determined largely by the uranium driver fuel.

The instrumentation for the entire facility is centered in the programmed measurement and control system (PMACS). This system includes a high-speed digital computer, and performs the following functions:

- controls the gas system and the electrical heaters to programmed set points
- controls the operation of the various pieces of experimental equipment
- collects and analyzes experimental and process data inputs
- generates output signals for the control of process systems and for the display and recording of the values of process variables in real time
- generates output signals to the safety circuit.

Peripheral equipment of the PMACS includes two low-speed magnetic tape memory units, input and output typewriters, and a large three-color cathode ray tube for display of data.

Initially the PMACS will not have a feedback loop for automatic control of the power level and period of the reactor. The reactor operator will control all motion of the control rods, except for automatic scram.

Besides the output signals from the PMACS, there are two independent conventional neutron flux channels providing inputs to the safety circuit. These channels and the safety circuit are not a part of PMACS. The safety circuit is a separate direct current loop, consisting of a power supply, a transformer-rectifier combination driven by a transistorized amplifier, and the holding magnet coils of the control and safety rods. Input signals to the safety circuit are

- reactor flux level and period signals from each of the two conventional channels

- a "reactor-normal" signal from PMACS
- three manual trip buttons.

The PMACS "reactor-normal" signal signifies that important process variables do not exceed their set points, that various interlocks are properly set, that functional diagnostic tests of the computer operation are satisfactory, and that the reactor flux level and period derived from two additional, independent, and dissimilar channels are within set limits. This safety circuit combines the features of redundancy, dissimilar components, and frequent testing which are required for best reliability.

The experimental equipment auxiliary to the reactor includes two oscillator mechanisms, one to move the test cell or the adjoining cell into and out of position, the other to move small specimens in the test cell or adjoining cells. They have cooling chambers for the removal of specimens from the test cell without the necessity of cooling the reactor. A neutron chopper and time-of-flight spectrometer are provided; the neutron detectors, at the end of a 25-meter flight tube, are in an adjoining small building. Test cores may be assembled on a core dolly having a load capacity of 14,000 lb. Two wire traverse mechanisms are provided for measurements of flux distribution.

Procedures for maintenance, operation, and safeguards control of experiments are described, and the pre-startup and preliminary nuclear testing program is outlined.

The experimental program for HTLTR does not require operation of the reactor at significant nuclear power levels. The maximum energy release now planned is 2 kW for 8 hr., 16 kW-hr. Routine operation generates from 0.01 kW-hr. in period measurements or low-level irradiations to 1.0 kW-hr in neutron-beam experiments.

The experimental program for HTLTR includes, but is not limited to, measurement of

- k_∞ of a lattice or medium, and its variation with temperature
- the temperature coefficient of k_∞ when only a fuel element is heated.
- the worth of a lattice heterogeneity in a supercell.
- various lattice parameters by foil activation,
- neutron energy spectra in multiplying or non-multiplying media, using the chopper system.
- effective cross-sections of materials as a function of temperature

Procedural restrictions on the amount and the rate of addition of reactivity provide safe operation of the reactor. These controls are backed up by a safety circuit and mechanical safety system which have a low probability for failure. The inventory of fission products in the fuel is always so small that it does not constitute a potential hazard either on- or off-site.

Accidents considered in the analysis are nuclear excursions, mechanical and utility failures, instrument failures, escape of fuel materials from their containers, critical assembly outside the reactor, spread of fuel material in disposal of wastes, and non-nuclear incidents. Most of these are shown to be improbable because of design features of the reactor.

The maximum credible accident is postulated to occur because of an error in loading. Two sequences of events leading to an inadvertent criticality are described. These are shown to be of very low probability, and their consequences are shown to be negligible. It is highly improbable that the HTLTR will ever cause radiological hazards to the general public.

III. SITE

III. SITE

A. Location and Environment

The High Temperature Lattice Test Reactor is located south of the 300 Area of the Hanford plant, about seven miles north of the center of Richland, Washington. The location of the Hanford Works in the State of Washington is illustrated in Figure III-1, and the location of the 300 Area in the Hanford plant complex is shown in Figure III-2. The reactor building is 700 feet south of the 300 Area exclusion fence, and about 2000 feet west of the west bank of the Columbia River.

Since the reactor has no potential for large releases of fission products, only the immediate surroundings are discussed herein. The nearest buildings to HTLTR are those of other Pacific Northwest Laboratory facilities: the Low Level Laboratory about 500 feet west, the Technical Library about 1000 feet northwest, three research buildings at about 800 feet north, and the PRTR about 650 feet northeast. The nearest dwellings are farmhouses along the east bank of the Columbia River and one mile east; the newly extended north boundary of the City of Richland is about one mile south. The area between the new city boundary and the presently closest dwellings within Richland, about 2-1/2 miles south, is zoned industrial. (Figure III-3)

B. Geology

The terrain along the west bank of the Columbia for several miles above and below the 300 Area is fairly level. The nearest significant topographic features are the bluffs across the river to the northeast, which rise 400 to 600 feet. The nearest other obstructions to air flow are the Horse Heaven Hills and Red Mountains at a distance of about ten miles to the south and southwest.

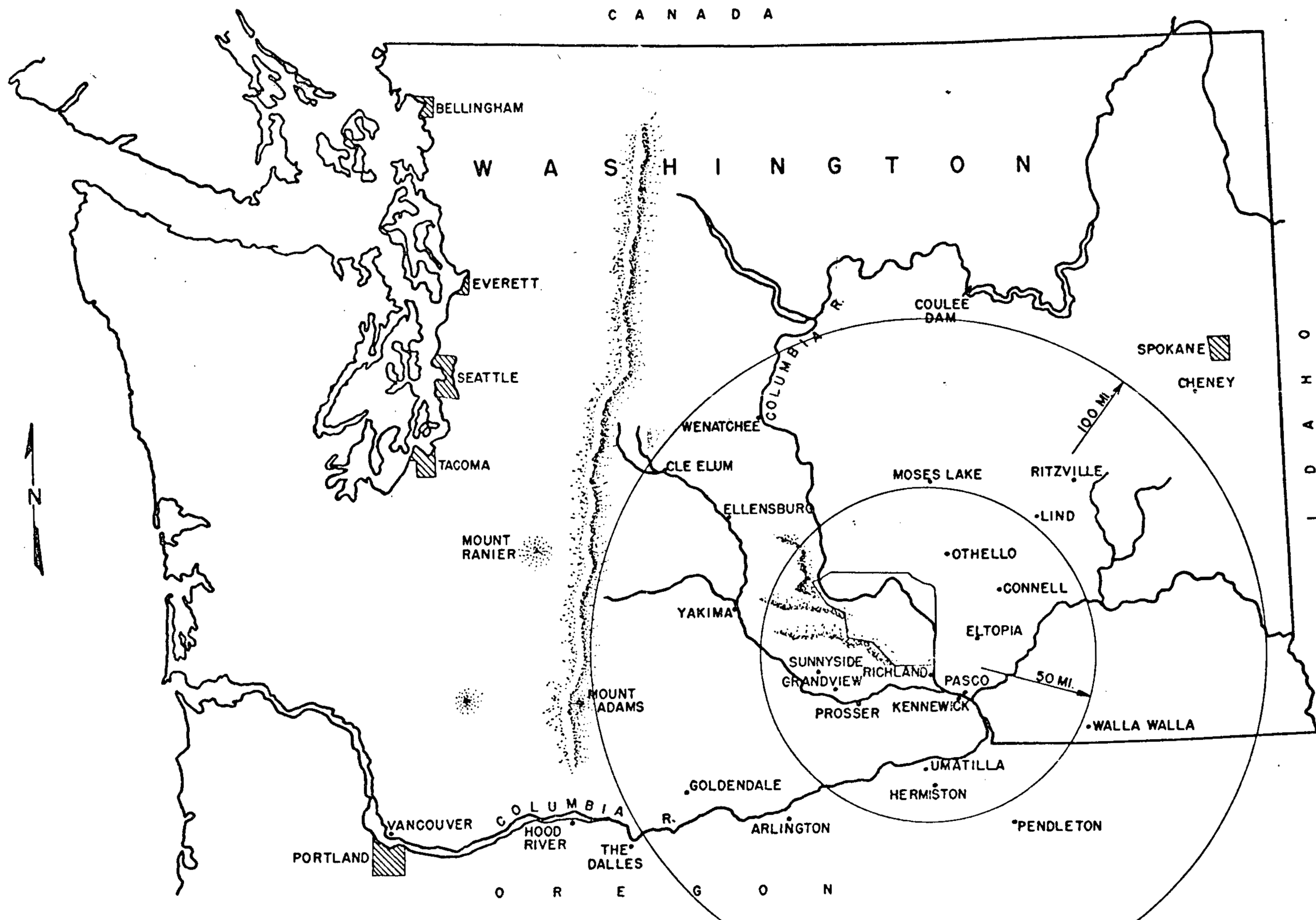


FIGURE III-1

Hanford Site Location

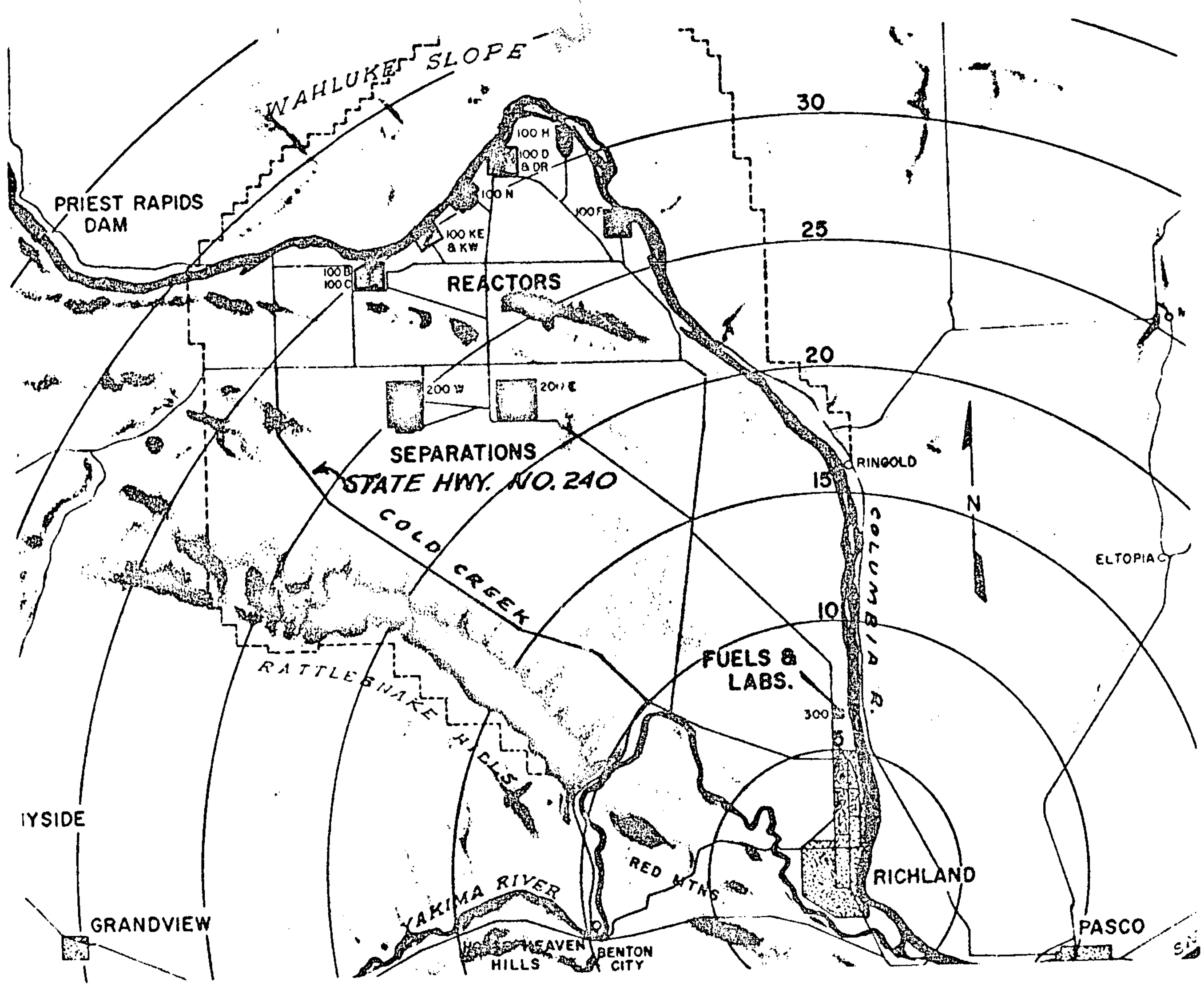
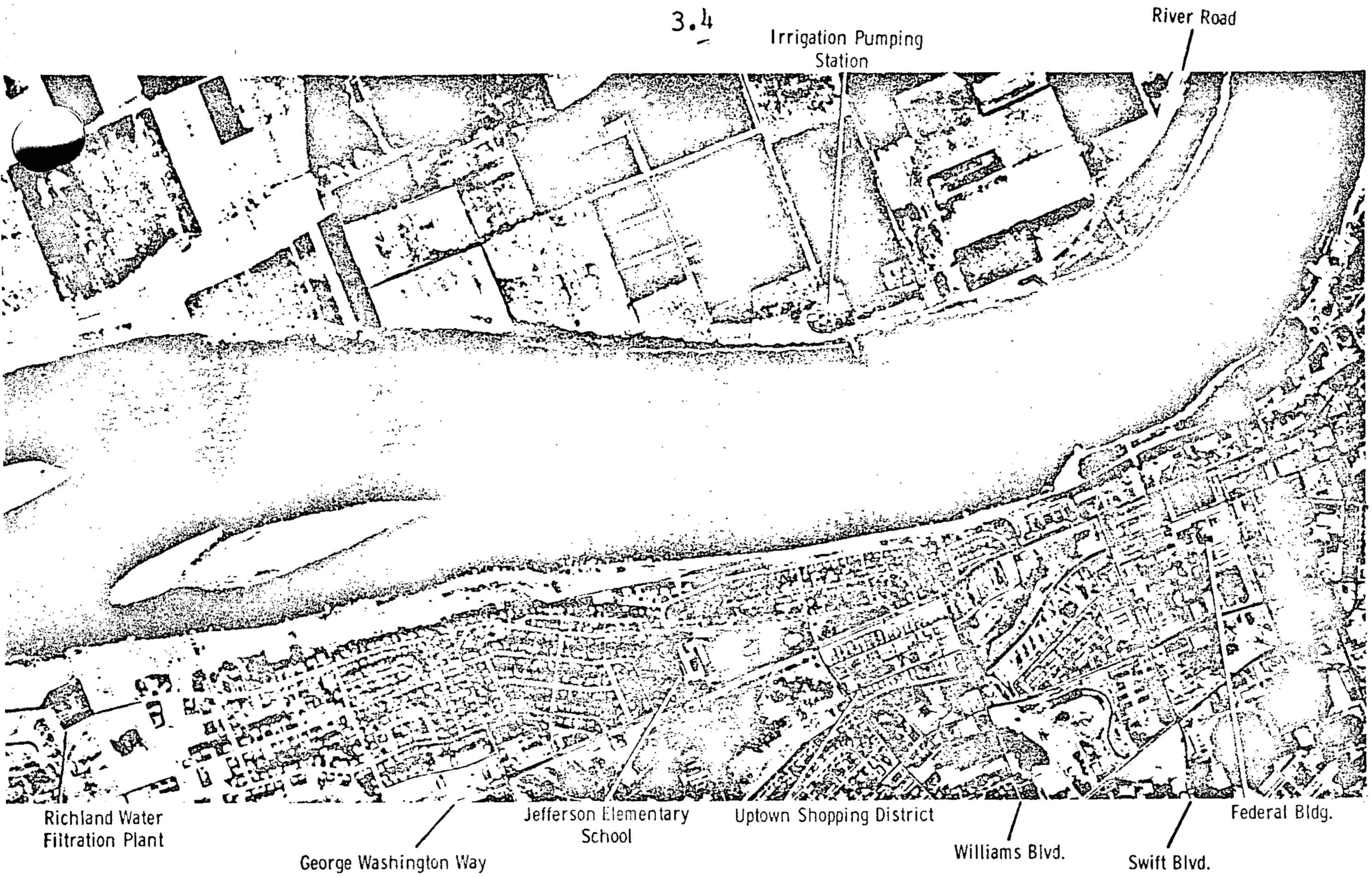


FIGURE III-2

Hanford Plant Layout



one Limit

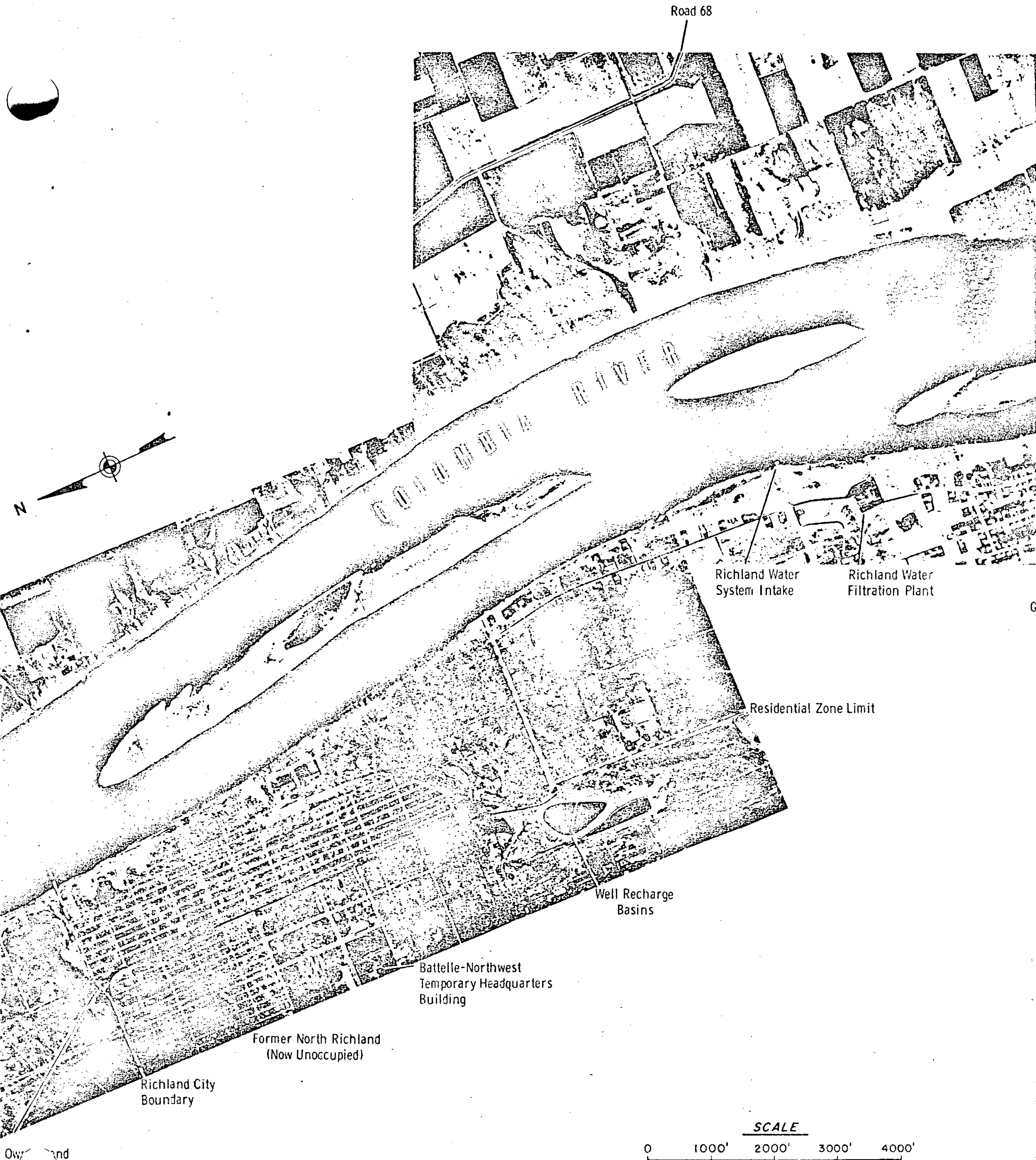
HTLTR Environs

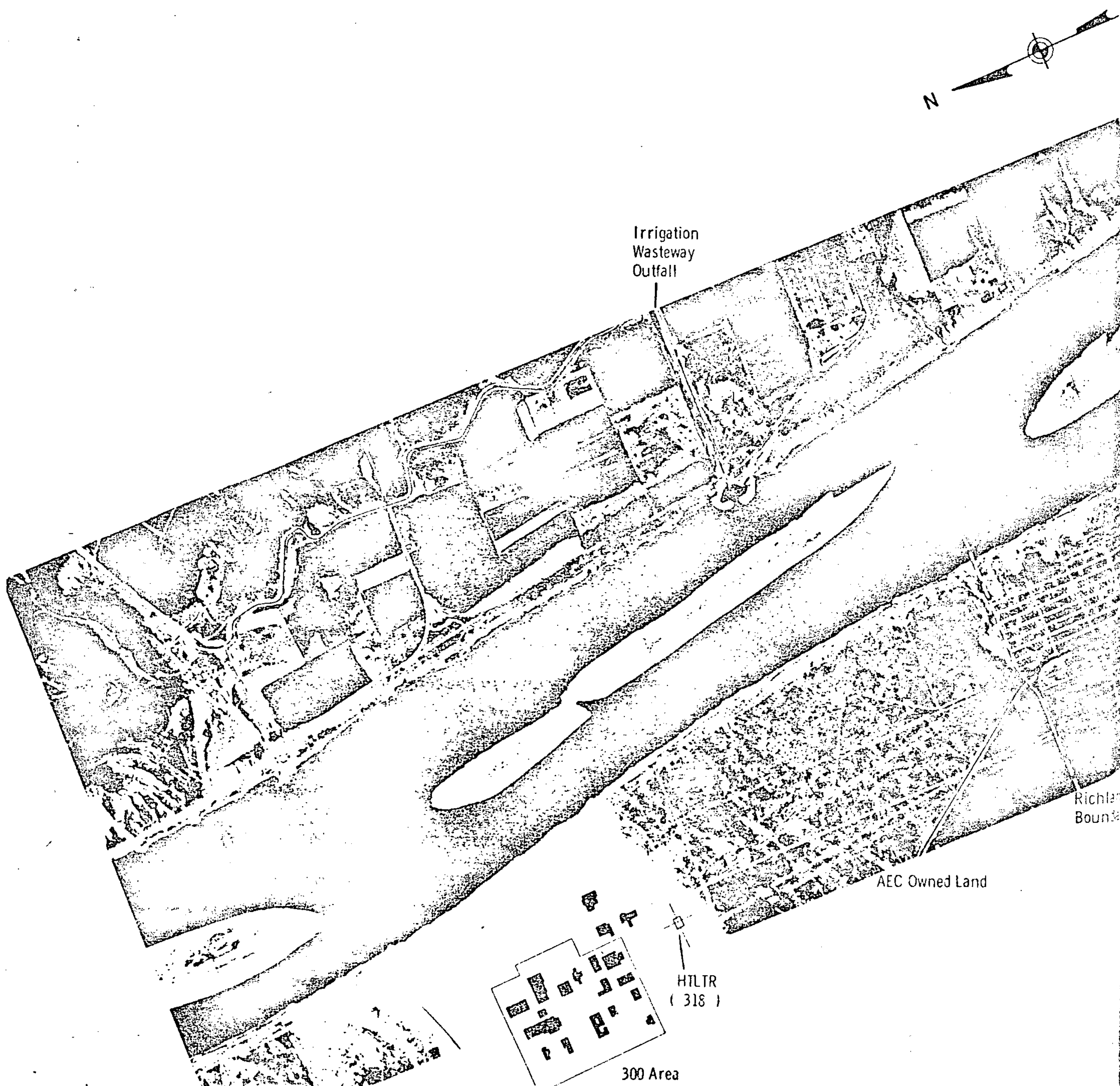
Survey Nov. 1964

000' 4000'

FIGURE III-3

HTLTR Environs


Road 68
COLUMBIA RIVER
N
Richland Water System Intake
Richland Water Filtration Plant
Residential Zone Limit
Well Recharge Basins
Battelle-Northwest Temporary Headquarters Building
Former North Richland (Now Unoccupied)
Richland City Boundary
SCALE
0 1000' 2000' 3000' 4000'


N
Irrigation
Wasteway
Outfall
AEC Owned Land
HTLTR
(318)
300 Area

Unconsolidated sands and gravels of the fluviatile series of sediments underlie the site to a depth of about 20 to 25 feet. Beneath these fluviatile sediments, to a depth of 70 to 100 feet, are semi-consolidated gravels. These are in turn underlain by clays, silts, and fine sands which extend to basalt bedrock at a depth of about 250 feet.

C. Hydrology

The site is about 400 feet above sea level. The water table is normally about 55 feet below the ground surface but may rise to within 35 feet of the surface during high water stage of the Columbia River. The ground water communicates directly with the river; the velocity of flow toward the river is about 15 feet/day through the soil with an infiltration rate of 10-15 gal/sq ft/day.

During the past 15 years, the variation in flow of the Columbia River has ranged from a minimum of 34,000 cfs to a maximum of 659,000 cfs. The estimated 100-year maximum flood stage of 740,000 cfs would give a river level of 365 feet, and the estimated average flood stage of 400,000 cfs would give a level of 351 feet. The site, therefore, affords ample safety from natural Columbia River floods.

Richland, Kennewick, and Pasco all take their municipal water supplies from the Columbia River. The intake of the new Richland filtration plant is about 4 miles south of the 300 Area; effluents discharged from HTLTR are pumped; the sump contents may be analyzed and disposed of in a manner which ensures that any flow to the river is within the required effluent discharge limits.

D. Meteorology

Meteorological and climatological data for the whole Hanford site and environs are obtained routinely by Pacific Northwest Laboratory. The nearest measurement station is located approximately 2000 yards north of the reactor site. The mean wind speed at this station is 10.3 mph. The southwest wind has the highest mean speed (14 mph) and the highest frequency of occurrence. Easterly winds have both the lowest speed and lowest frequency of occurrence. Wind speed and direction are functions of the stability of the atmosphere. In general, the winds at nighttime are representative of stable atmospheric conditions and those in daytime are representative of unstable conditions.

Thunderstorms occur on an average of 13 days per year with 85 per cent of the total occurring during the months of May through August.

About 20 per cent of the storms have high winds (gusts to 40 mph or more), and a lesser percentage are accompanied by blowing dust, rain, or hail. Hurricanes are unknown in this locality. Tornado funnels have been observed twice in 18 years, but no resulting damage has been reported.

The heaviest rain in 18 years amounted to 1.68 inches in six hours in October, 1957, and the heaviest snow totaled eight inches in six and one-half hours in December, 1955. Review of the records shows that storms of this intensity can be expected about five times in 100 years.

Pertinent meteorological data are summarized in Appendix D.

E. Seismology

Hanford facilities are exposed to the possibility of earthquake damage from two sources: 1) the active seismic zones of western Washington, and 2) closer shocks originating in the seismic zone that includes Walla Walla. The underlying sands and gravels in the Hanford reservation provide an almost ideal protection against damage. As far as can be determined, earthquake intensities greater than three on the Modified Mercalli Scale (MM-III) have not occurred in the immediate Hanford area, although intensities as high as MM-V or MM-VI have been observed at surrounding towns.

The strongest shock to occur in western Washington in historic record was the 1949 earthquake originating in the Puget Sound channel just off Steilacoom, a distance of 150 miles from Hanford. At distances of 150 miles, Neumann[2] reports that intensities from MM-IV to MM-VII were experienced. Four shocks between 1932 and 1946 had maximum intensities of MM-VII in western Washington.

The eastern Washington earthquakes occurring in historic times have not been as intense as those in western Washington, nor have they been as frequent. In 1936, the Walla Walla area experienced an MM-VI shock. In 1934 at Ellensburg and in 1957 near Othello, "seismic swarms" of small shocks occurred. A magnitude of MM-VI was reached in some of these, but the shocks were highly localized. The closest recorded shock has occurred at Corfu, 31 miles northwest of the 300 Area. In 1918, it was recorded that the most severe shock at Corfu lasted several seconds and shook goods from shelves and caused landslides.

The effects of the great Alaska earthquake of 1964 and the most recent shock in western Washington (1965) were not felt as strongly at Hanford as in surrounding localities. In addition, Hanford is not located in a historically active seismic zone. These considerations make the area one of the safest in the state.

A map summarizing the seismic history of Washington State is given in Appendix D. The approximate relationships among earthquake intensities, ground accelerations, and building code zones are also given.

For building purposes, the Hanford area has been included in Zone 2 in the seismic probability map adopted as part of the Uniform Building Code by the International Conference of Building Officials, and the reactor and building have been designed in accordance with the Code.

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V. REACTOR

IV. BUILDING

The facility comprises a concrete structure housing the reactor, a steel panel service building abutting on the west, a neutron time of flight detector building of pre-fabricated type construction located to the south, and electrical and other utility stations nearby. Refer to Figure IV-1.

A. Reactor Enclosure

The concrete structure includes two levels, the reactor room which is 33 x 54 feet in plan with a clear height of 29-1/2 feet and the reactor basement immediately below with a height of 14-1/2 feet.

The building walls are of ordinary reinforced concrete, 4 feet 8 inches thick on the west (the office side) and 4 feet 3 inches thick on the other three sides. The roof is 3 feet thick and the floor 2 feet thick. The floor is recessed 2 feet over the region occupied by the reactor and gas containment shell. This sets the bottom face of the reactor at the same level as the work area floor outside the shell.

A shield door to the reactor room covers an opening 11 feet wide by 10 feet high on the axial center line of the reactor on the west. A second shield door, 7 x 7 feet, covers the opening into the reactor basement from the west. Stepped concrete plugs are provided in the north and east walls of the concrete structure at the axial and lateral center lines of the reactor. A circular port in the south wall at the lateral center line of the reactor provides the exit for the flight tube for the neutron energy analyzer. Also in the south wall is a 4' x 4' nonreinforced section, which is provided for easy modification to allow the addition of a fuel storage area. The walls are enamelled and the floor is finished with a coating which is resistant to chemicals and radiation. A 10-ton bridge crane serves the reactor room. These features are shown in Figure IV-2.

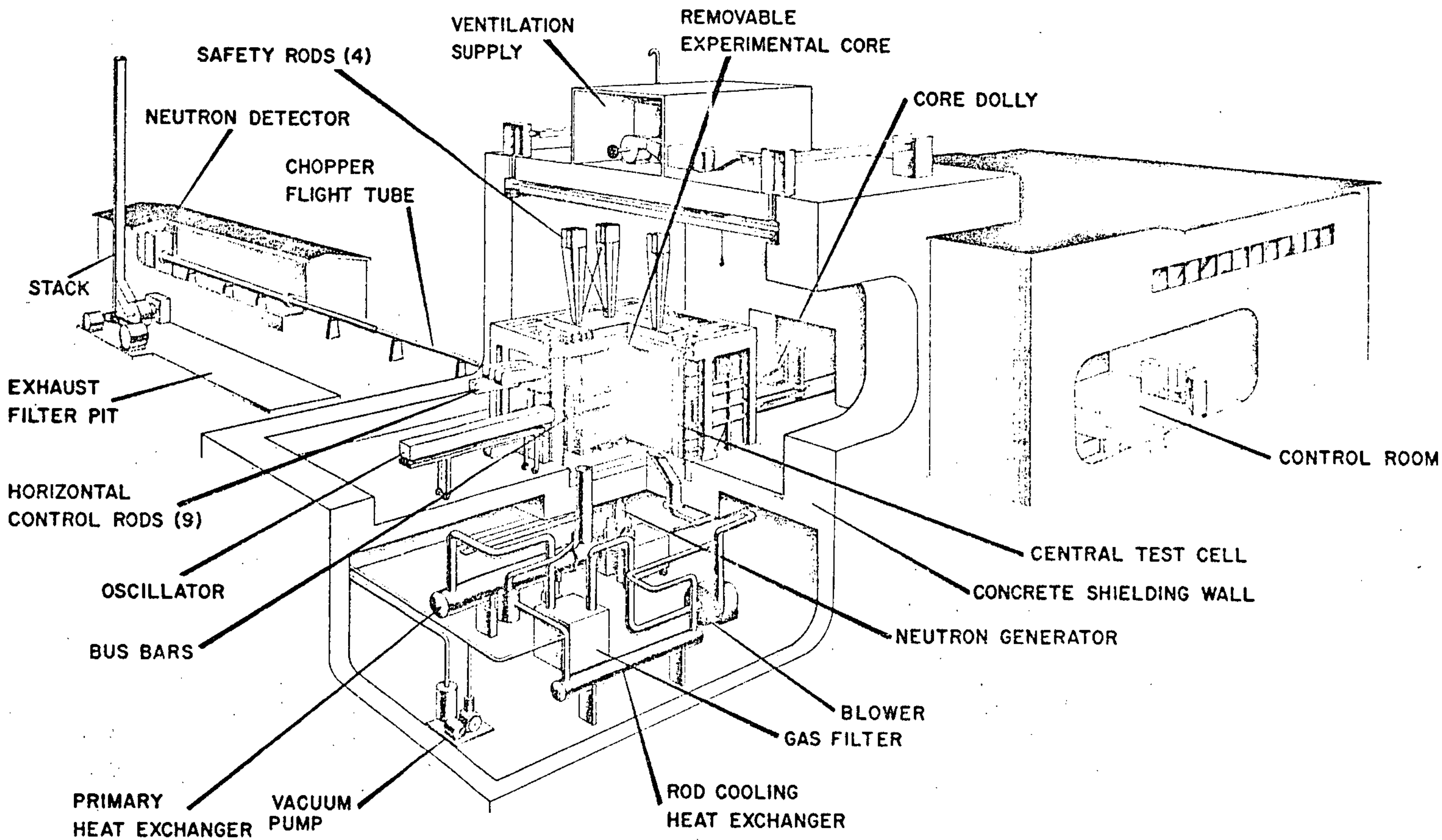


FIGURE IV-1

Cutaway View of Reactor and Building

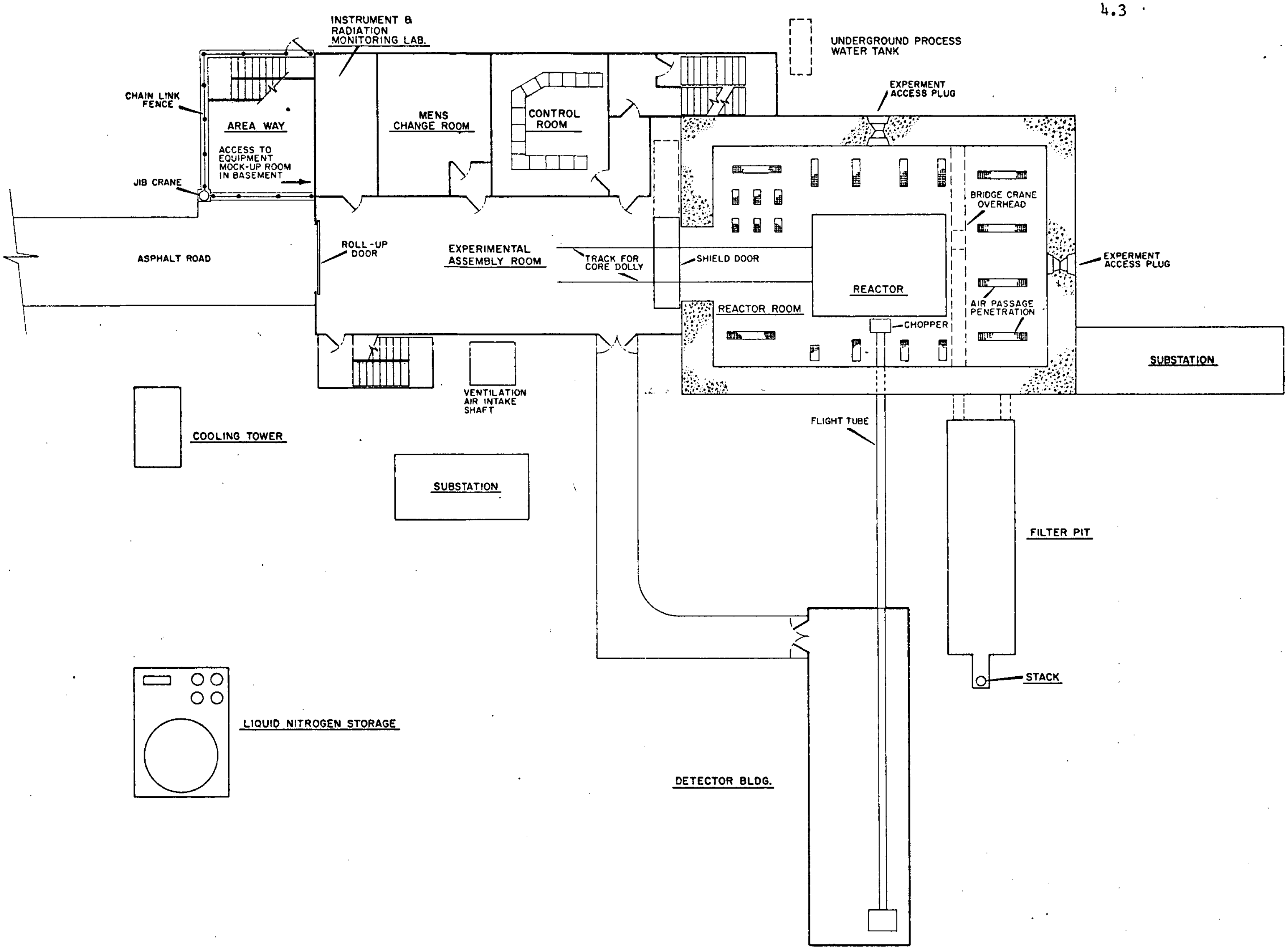


FIGURE IV-2

HTLTR Building Plan

In addition to the above mentioned openings, the reactor room and basement have the following penetrations:

- six shielded ventilation supply openings in the roof to the reactor room
- a large ventilation exhaust opening in the basement to the underground vault
- wiring and piping penetrations, all below the level of the reactor

The doors and penetrations in the enclosure (reactor room and reactor basement) are not sealed, nor designed to withstand internal pressures. However, the reactor enclosure is required to be sufficiently air-tight that the ventilation equipment can maintain the design negative pressure of 0.12 inches of water relative to atmosphere. The reactor enclosure provides the secondary confinement barrier, backing up the reactor shell and piping.

B. Service Area

Abutting the concrete structure on the west is a two story steel panel building with a full basement. The basement space is divided equally between a heating and ventilating machinery room and an experimental equipment mockup room. A jib crane is available for equipment transfer from an access drive at ground level down to a service area at the end of the mockup room. A concrete-walled counting room is also located at this level, adjacent to the mockup area.

On the first floor of the service building are located the experiment assembly room where unfueled test cores are assembled before they are charged into the reactor, a clothing change room with lavatory, an instrument shop, and the reactor control room.

The control room, 21-1/2 feet x 17-1/2 feet, contains the programmed measurement and control system, (PMACS) which includes a small control console with video tube display of process variables, input and output typewriters, and several instrument cabinets and logic systems. A building and ventilating control cabinet is also located in this room. Signal cables to the computer-controller are brought in through floor penetrations.

The second floor of the service building contains offices, a lunch room, women's rest room and corridors. Two stairways to this level are provided, one of which is outside the building. The latter leads to the roofs of the service and reactor buildings and much of the ventilation equipment.

C. Ventilation

Three separate ventilation systems are provided. The first provides about 3700 cfm to those areas in which radioactive and fissile material will not be permitted, such as the lunch room, offices, and upstairs corridors. The ventilation system maintains these areas at a slightly positive pressure relative to the atmosphere and to the pressure in the adjoining work area.

The second ventilation system provides 7600 cfm to the work and mockup areas, instrument shop, control room, counting room, and change room and maintains these areas essentially at atmospheric pressure. Systems one and two have a common intake at the southwest corner of the facility; both exhaust via separate blowers above the roof of the service area.

The third ventilation system has an entirely separate intake system on the top of the reactor building and exhausts via a forty-foot stack. This system has a capacity of 11,200 cfm, and serves the reactor enclosure only, maintaining it slightly below atmospheric pressure. The supply equipment comprises an intake air filter, two dampers, preheat steam coils, cooling coils, a humidifier, a reheat coil, and an intake blower. One damper is for normal service and the other for emergency use during power outages. The equipment is well drained and is enclosed in a heated casing to avoid the possibility of the drains freezing. Accidental flooding of the reactor room by water from the ventilation supply equipment is thus made improbable. The conditioned air is supplied at six shielded jet diffusers at the reactor room ceiling and flows from the reactor room via 20 floor gratings into the basement area.

From the south side of the reactor room basement, the ventilation air passes via a tunnel through a remotely controlled valve, a bank of dust filters, dual banks of high-efficiency filters (99.97% retention of standard 3-micron dioctyl phthalate aerosol), a manually operated valve, an exhaust fan, and finally discharges through the 40-foot stack. The filter pit is a continuous reinforced concrete structure up to the intake of the exhaust fan, is entirely underground, and has sealed manholes at necessary service points.

A gasoline engine is directly coupled to the exhaust fan at the base of the stack. This provides the necessary back-up capability for continuous ventilation of the reactor structure in the event of motor failure or electrical outages.

The 40-foot exhaust stack is tall enough, is located sufficiently distant from the intake louvres of the ventilation system, and has adequate release velocity (4000 feet per minute in a 24 inch diameter pipe) to ensure that short-circuiting of the exhaust air from the stack to the intakes is improbable.

D. Services and Utilities

Power is supplied to the building from an existing 2400 V substation near the building, via underground cable, to pad-mounted transformers, also outside the building. Since the safety of the reactor does not depend upon an uninterrupted electric power supply, only a single source is provided. Battery-powered emergency lights are provided to allow safe evacuation of personnel from the building in the event of power failure.

The following 300 Area service systems are piped into the building:

Compressed air (90 psig)
Sanitary water (also used for process water) 115 psig
Fire protection water (separate connection to sanitary water) 115 psig
Steam (175 psig)
Condensate return (125 psig)
Sanitary sewer (gravity)
Process sewer (to waste disposal ponds; sump pump discharge)

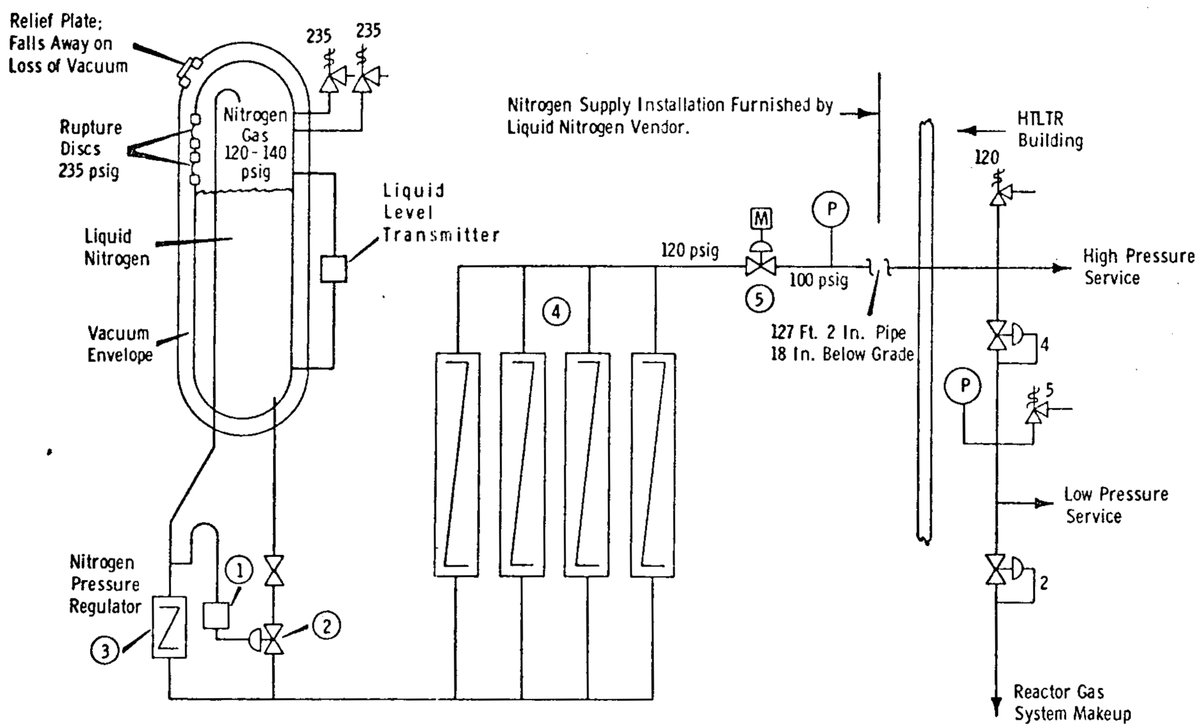
Within the building, process cooling water is provided by two pumps (115 psig discharge head) whose supply is a 5000-gallon break tank fed by a line from the 300 Area sanitary water system.

E. Detector Building

A drift tube, up to 18 inches in diameter and about 25 meters long, extends from the reactor into the detector building, terminating at the target area. The detector building is a long, narrow (50-1/2 ft x 16 ft) pre-fabricated galvanized steel panel unit, housing part of the drift tube and the neutron detector assembly. It is located about 34 feet to the south of the reactor building. Separate electrical heating and an evaporative cooler ventilation system which exhausts to the atmosphere are provided for this building.

F. Nitrogen Supply (Figure IV-3)

Nitrogen gas is supplied at 100 psig from an 8000-gallon liquid nitrogen tank designed to the ASME Unfired Pressure Vessel Code, located on a 14 ft x 18 ft concrete pad about 60 ft to the southwest of the service building. The storage tank is provided with dual rupture disks, dual safety relief valves, a pressure indicator, and liquid level indication. An electrical contact for low liquid level in the storage tank is provided for alarm annunciation in the control room.



SET PRESSURE

Pressure Regulator

SET PRESSURE

Safety Relief Valve

Pressure Differential Valve, (Manually Loaded)

Pressure Gage

Nitrogen Evaporator Panel

1. Pilot Regulator; Set Point Delivery Pressure Plus 10 psig
2. Bailey Regulator Model A30E-1; C_V For Liquid Nitrogen = 7.9
3. Evaporator For Tank Pressure Regulation.
4. Main Evaporator Bank; Capacity 100 CFM.
5. Bailey Regulator Model A30E - 1; Capacity At Stated Differential 224 CFM

FIGURE IV-3

Flow Diagram, Nitrogen Supply System

Four vaporizers (heat exchangers from liquid nitrogen to ambient air temperatures) are provided, each with a normal load capacity of 1500 ft^3/hr, one of which would satisfy the normal demand and two of which would meet any conceivable emergency.

The gas discharge line from the station is equipped with a safety relief valve. A two-inch line about 132 feet long at 18 in. below grade provides about 70 square feet of additional heat exchanger surface before the first control valve within the facility.

The design of the nitrogen piping and heat exchange system makes the entry of liquid nitrogen into the reactor gas system a physical impossibility. The designers deliberately rejected the idea of installing an automatic, thermostatically controlled valve in the nitrogen supply line (which would close on contact with cold liquid nitrogen) since they regarded it as entirely unnecessary and wished to avoid spurious operation of the valve and starvation of the reactor gas system.

Within the facility, the high and low pressure nitrogen service is divided. A pressure regulating valve and flow control valve supply low pressure gas flow to the reactor. The high pressure service is provided to the reactor auxiliaries such as the oscillators, flux wire manipulators, safety rod brake cylinders, and neutron spectrometer collimator.

V. REACTOR

V. REACTOR

A. General

The HTLTR moderator block is a graphite cube 10 feet on a side, made up of cored, nuclear grade graphite bars, 4-3/16 in. square, 4 ft and 2 ft in length. The bars are stacked, keyed, and doweled together to form a solid unit. The 5 ft x 5 ft x 10 ft long central section is entirely removable.

This central section may thus be readily adapted to any size of test core up to a 5 ft cube. For smaller test cores it will be lined with graphite blocks which can be loaded with driver fuel. The central unit of the test core, the test cell, can be removed or oscillated during experiments. Maximum nuclear thermal power is 2 kilowatts; this, however, will be infrequently scheduled. About 475 kilowatts of electrical power are available for heating the moderator to the design temperature of 1000 C.

Nitrogen gas provides an inert gas atmosphere at elevated temperatures. The gas is contained in an insulated carbon steel shell.

The reactor is controlled by a system of 8 shutter-type control rods and 4 gravity-drop blade-type vertical safety rods (VSR). The neutron flux in the reactor is sensed by four channels of nuclear detection instrumentation. Two channels provide input to the Programmed Measurement and Control System (PMACS), and two channels provide period, flux-level, and on-scale input to the safety circuit.

Radiation shielding is provided by a Boral liner within the steel shell and by the outer concrete walls of the building.

B. Reactor Loading

Driver fuel will be loaded into the moderator on a 4-3/16 to 8-3/8 inch spacing. The driver fuel elements will be 1.06 in. diameter enriched (5% U-235) UO_2 ceramic pellets in graphite cladding. The cladding diameter will be about

1.6 inches o.d. Various fuel element lengths will be used. Flux leveling slugs 4 inches and 8 inches long are specified, and main drivers may be 2.5 feet or 5 feet long. The longer lengths of driver fuel will be loaded in a ring around the test core, which may be moderator only or a test fuel and moderator assembly. The shorter lengths will be loaded on the ends of the test core. Procedural control and the safety circuit are relied on to prevent startup accidents with this reactor. Such controls have been in use for ten years in the operation of PCTR. This is discussed in detail in Section IX. The reactors are compared in Appendix E.

Poison shim rods will be loaded in certain channels of the core. Their function is to reduce the negative moderator temperature coefficient of the reactor (such that it remains negative but less than 0.1¢/°C) to avoid the necessity for controlling large amounts of excess reactivity at room temperature and to reduce the spurious reactivity signals caused by fluctuations in temperature during experimental measurements. Gadolinium oxide has been chosen as the major material for this temperature-dependent poison shim but it may be modified slightly with iridium or with hafnium oxide to ensure that the coefficient will remain negative above the operating temperature.

The test core will have many different arrangements of moderator and fuel. However, by procedure it will always be loaded so as to be subcritical by itself; also, the ends will usually be loaded with flux leveling slugs, which are composed of driver fuel as in the rest of the driver region. Since, in experiments of this kind, most of the power is developed in the driver fuel, the effect of the test core on the overall core kinetics will not be controlling. Up to about six kilograms of fissionable material may be contained in the test core. Since this compares with about 40 kg of U^{235} in the driver region, the effect of the test core fuel on the reactor kinetics will be small, even when the test fuel contains U-233 or Pu-239.

Those channels not in use as fuel or poison channels, or for gas circulation, control, or heating, will be plugged with graphite.

C. Gas Envelope and Thermal Insulation

The reactor, and its thermal insulation, are enclosed in a gas-tight steel container. This has a removable door on the front (east) west face, for complete access to the reactor core for alterations, and a smaller door on the rear (west) east face for access to the test core. Both doors are designed to be gas-tight. Other penetrations, for control and safety rods, gas system piping, experimental openings, electrical leads, and instrumentation are sealed or capped as necessary. The envelope can be evacuated to aid in purging air from the system before heating. The gas envelope is designed to meet a requirement of leakage no greater than one-half percent of the gross volume per hour, at a pressure differential of one psi. The gas flow is directed from a three-foot plenum at the front face through the channels in the moderator block to a three-foot plenum at the rear face by suitably disposed gas diversion blocks at the top and sides of the reactor.

The reactor is designed for continuous operation at 1000 C. Specially insulated test cores may be heated and operated at temperatures above 1000 C as long as the rest of the reactor remains below 1000 C. The thermal insulation exterior to the reactor is about two feet thick, consisting of successive layers of insulating firebrick, stagnant nitrogen, and fibrous mineral blanket.

D. Shielding

Since the reactor will be operated at only nominal power levels, there is no massive shielding immediately adjacent to the core or gas envelope. The reactor is operated remotely, and the walls of the reactor room provide shielding for personnel as discussed above in the description of the building. A Boral shield 1/8 inch thick is attached to the inside of the gas envelope, with openings as required for admission of neutrons from the source and efflux to the flux detectors. This shield is provided to minimize the generation of capture gamma rays in the steel shell.

E. Heating, Cooling, and Gas Circulation

The reactor is heated electrically, by low voltage, high current graphite heating bars, traversing the reactor. The heater surface design temperature is 1600 C. Heating current is controlled by saturable reactors and is supplied through copper bus bars to large graphite buses into which the heater bars are fastened with screw plugs and tapered sleeves. The graphite-to-copper bus bar connections are made in a relatively low-temperature zone, but water-cooling of the joints is necessary.

The reactor is blanketed in a recirculating nitrogen atmosphere at low pressure. The purity of the reactor atmosphere is maintained by a water removal system and by continuous purge and makeup of the recirculating system. Both the gas purged from the system and the recirculating gas are analyzed for O_2, CO, CO_2 and H_2 by gas chromatographs. A complete analysis requires 8 minutes and the two streams are sampled alternately; thus each stream is analyzed about 4 times/hr. The analyses are used to determine the gas purity and as an indication of small leaks and gas-graphite reactions. In addition, both streams are continuously monitored for alpha-radiating particulate matter as an indication of fuel element rupture. The purge stream is filtered through dual high-efficiency filters and then discharged to the stack. The recirculating stream is filtered by a single high efficiency filter. During normal operation most of the flow is returned to the inlet of the heat exchanger. A side stream is further cooled and supplied to the control and safety rod actuator housings. Moisture monitors are included in the recirculating gas lines immediately following the heat exchangers. These provide rapid indication of any water seepage into the system.

For cooling the reactor, the main stream of the recirculating nitrogen is permitted to flow through the reactor. A flow-sheet of the circulating gas system is shown in Figure V-1.

F. Control Rods

Each of the eight horizontal shutter-type control rods can insert its full change of reactivity with a movement of only six inches. Rod movement may be ganged, or single rods may be moved as criticality is approached. The design of the rod is shown in Figure V-2; details of its operation follow:

The active section of the control element is made up of six-inch long cylinders of two kinds; namely, graphite and UO_2-graphite. The outer row of cylinders, alternately graphite and UO_2-graphite, seven of each for a total length of seven feet, is housed in a graphite sleeve which is anchored to the reactor shell at one end. The inner row of cylinders, alternately UO_2-graphite and graphite, eight and seven in number, respectively, for a total length of seven and one-half feet, is assembled on a 1/2-inch diameter thoria-dispersed nickel (TD-nickel) rod. Graphite cylinders about the same size

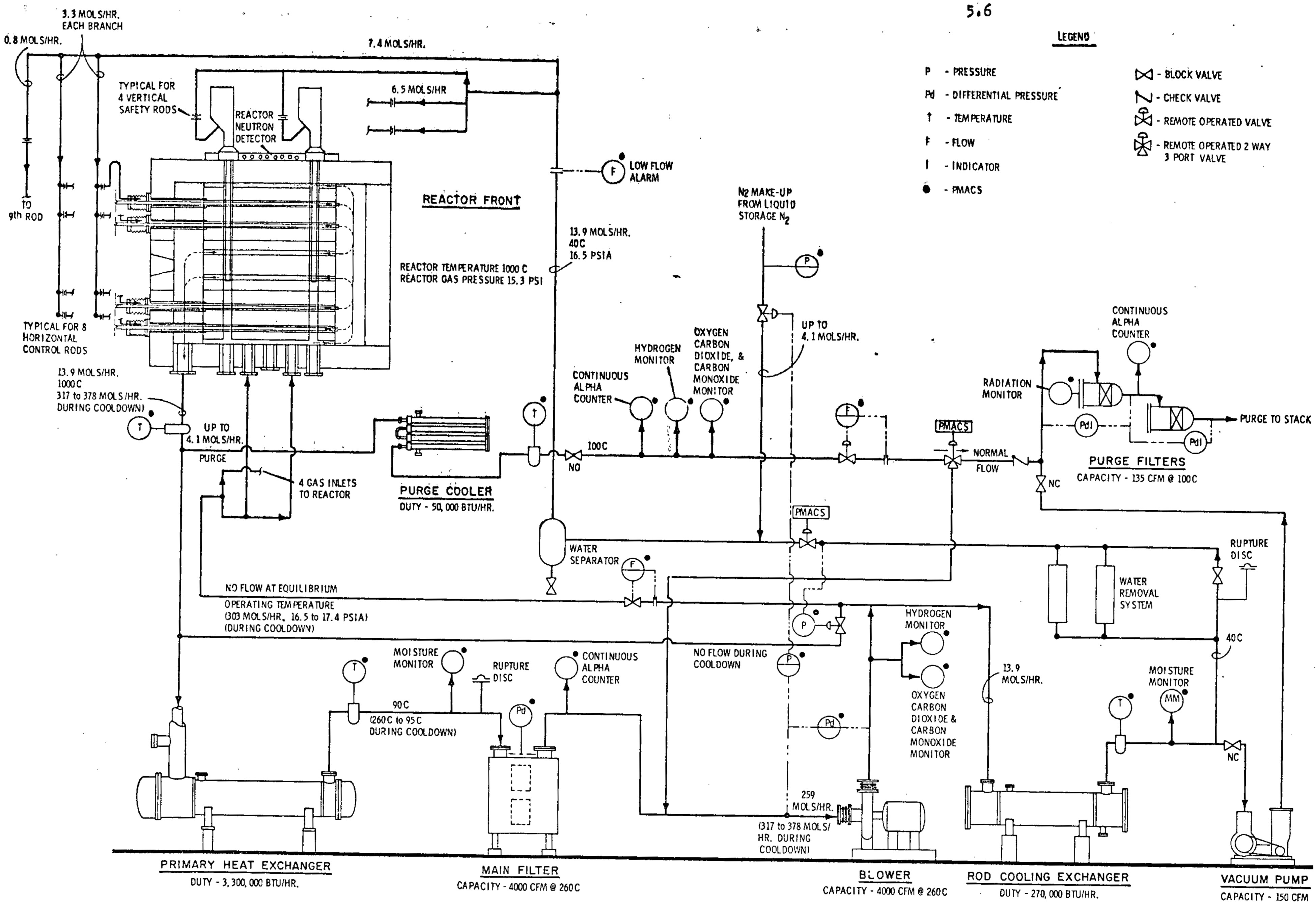


FIGURE V-1

Flow Diagram
Recirculating Gas System

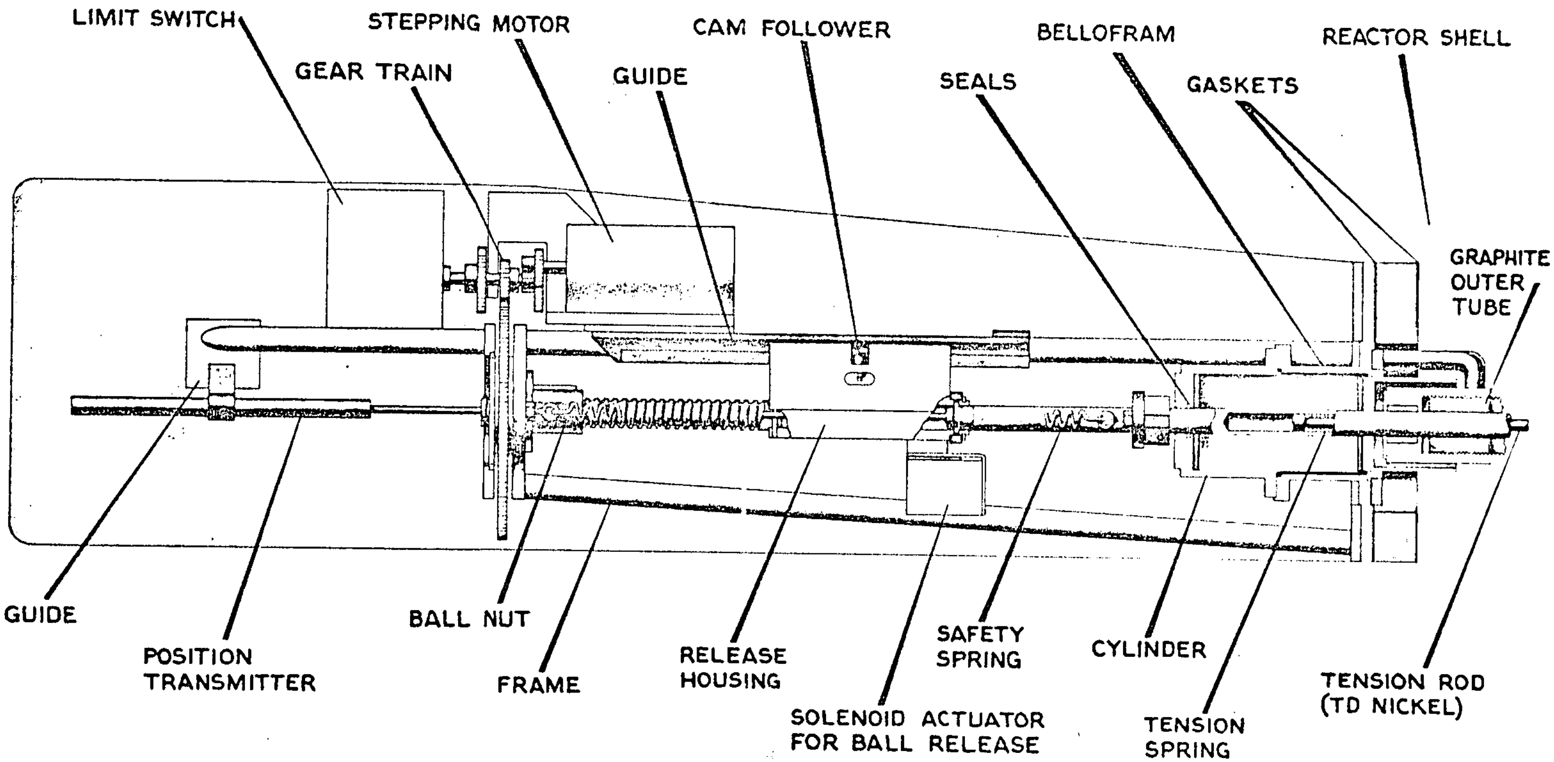


FIGURE V-2a

Control Rod: Drive Mechanism

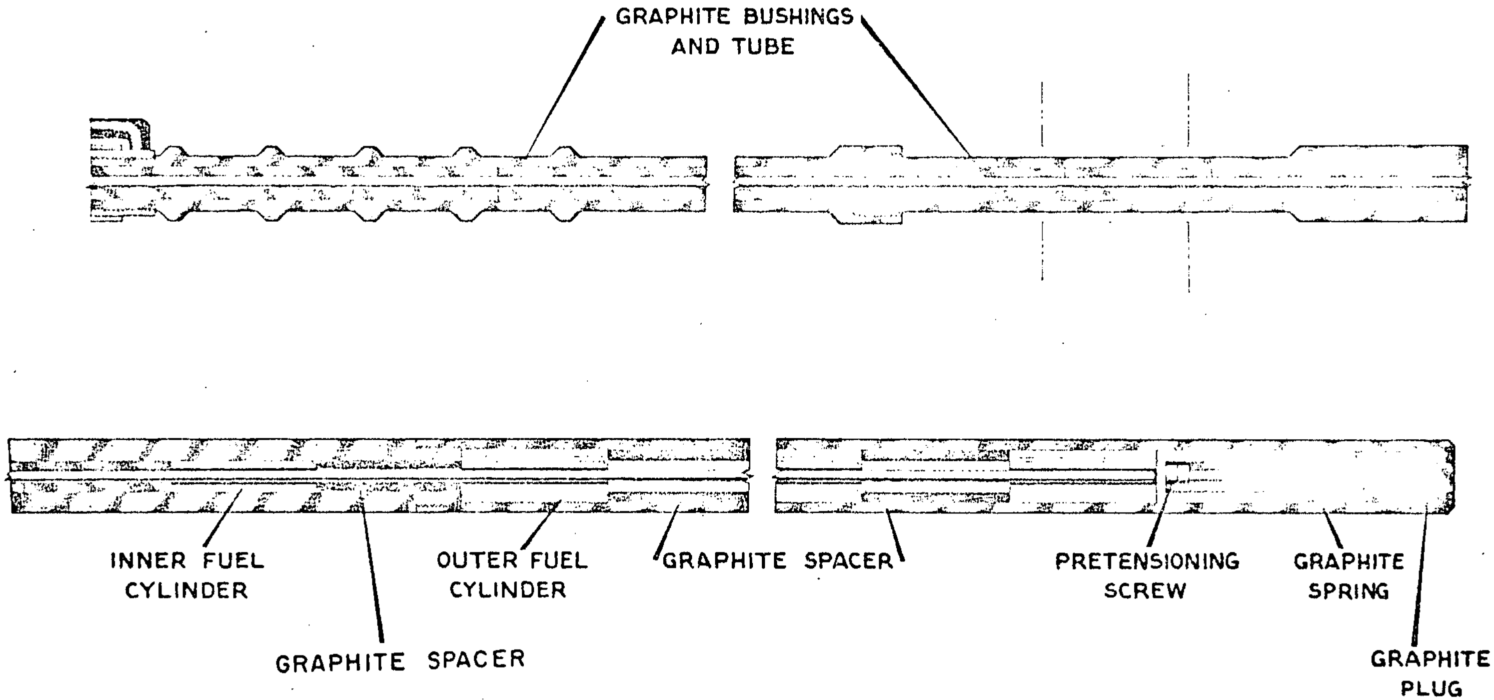


FIGURE V-2b

Control Rod: In-Reactor Portion

as the graphite and fueled cylinders are assembled on the remaining portion of the rod which is in the high temperature zone. When installed in the assembly, all the inner cylinders are held in compression (about 300 pounds, cold) on the TD-nickel rod by a heavy compression spring in the actuating mechanism. At 1000C, this compression is reduced to about 150 pounds because of linear expansion in the TD-nickel rod. The entire inner row of cylinders is moved in both directions by the actuating mechanism, which is directly coupled to the TD-nickel rod. The outer sleeves are held in compression by a graphite spring. By constructing all parts of the control rod assembly except the TD-nickel center shaft of graphite or graphite base materials, differential thermal expansion, which could affect the worth of the rod, has been minimized.

The actuating mechanism of each rod consists of a stepping motor connected through suitable gearing to a ball nut which rotates on a nonrotating lead screw. The stepping motor is driven by pulses originating in the reactor computer-controller (PMACS). The motion of the control rod is limited by limit switches. The control rod position is determined in two ways. The coarse position indication is obtained with a direct current differential transformer (DCDT), which converts linear motion into a proportional dc signal. The DCDT core is connected to the operating rod so that it always reads control rod position, even during a scram. The output of the DCDT goes to the PMACS computer. The fine position indicator is a motion transducer, incorporated in the stepping motor, which puts out two pulses for each step the motor makes. The sequence of the output pulses indicates the direction of rotation of the motor. One step of the motor (two pulses from the transducer) equals 0.00025 inch linear travel of the actuating mechanism and operating rod when they are connected by the latch. The pulses are counted and converted to exact rod position by the computer.

The latch is a rotating-ring ball-detent type, operated by a solenoid. When the solenoid is energized, the ring rotates against spring tension, causing four balls to be cammed into a groove in the operating rod. This locks the actuating mechanism and the operating rod together. Actuation of the stepping motor causes the control rod to move against the tension of the two scram springs. De-energizing the solenoid allows the ring to rotate in the opposite direction, releases the balls, and thus disconnects the control rod from the operating mechanism. The scram springs then pull the operating rod rapidly to the outer limit of travel. Before the latch can be re-engaged, the operating mechanism must be returned to its outer limit of travel.

During a scram about 40% of the full rod travel occurs in the first 0.15 seconds. A deceleration cylinder, incorporated into the assembly, cushions the impact of the rod at the end of its travel. Replacement of the discharge orifice in the deceleration cylinder with a different size permits some adjustment in the time required for total rod travel. Full rod travel, as measured on the prototype control rod, requires less than 0.72 seconds. Failure of a single spring has been simulated in tests of the assembly; full rod closure in less than 1.15 sec resulted.

A fine control rod of the same design and construction as the other eight is also provided. This rod has only a small fraction of the reactivity strength of one of the primary control rods and is not a part of the shutdown system or safety circuit. Its only purpose is to obtain precise measurements of small changes of reactivity.

G. Vertical Safety Rods (VSR)

A system of four vertical rods (flat plates) is provided. These rods drop into the core on a reactor scram. The rods are usually moved in pairs, though single rods may be dropped on tests. The design of the rod is shown in Figure V-3; details of its operation follow:

The control element is a blade, 7 inches wide, about 138 inches long and 1/2-inch thick. A mixture of gadolinia and europia in a suitable matrix in
gadolinia - europia - dispersed nickel
pieces 6 inches square and 3/8 inch thick, makes up the active rod length of 120 inches. The nuclear poison squares are mounted in a ladder frame made up of 1/2-inch TD-nickel rod, covered with 0.030 inch TD-nickel sheet and assembled into one unit with TD-nickel rivets. A stainless steel support assembly, clamped to the top of the frame, provides the coupling to the lifting cable. The four safety rods move in rectangular slots in the moderator block. The flat, rigid blade assemblies are capable of intermittent or continuous service at 1000 C.

Each safety rod is supported by a 1/8-inch stainless steel aircraft cable, which winds on a grooved drum to prevent overlap. The drum is mounted on the main drive shaft, to which the armature half of a magnetic clutch is keyed. The rotor of the clutch is driven by a stepping motor through reduction gears. A disk brake in the drum provides the necessary deceleration to prevent damage to the rod or cable at the bottom of their fall. The brake is held off by mechanical interference of a cam until, 18 inches above the full-in-position, the brake is applied by spring action. The brake remains applied until it is pneumatically released by nitrogen under pressure when the rods are to be raised. The brake functions effectively regardless of the position of the rod when the scram signal is received. It remains applied

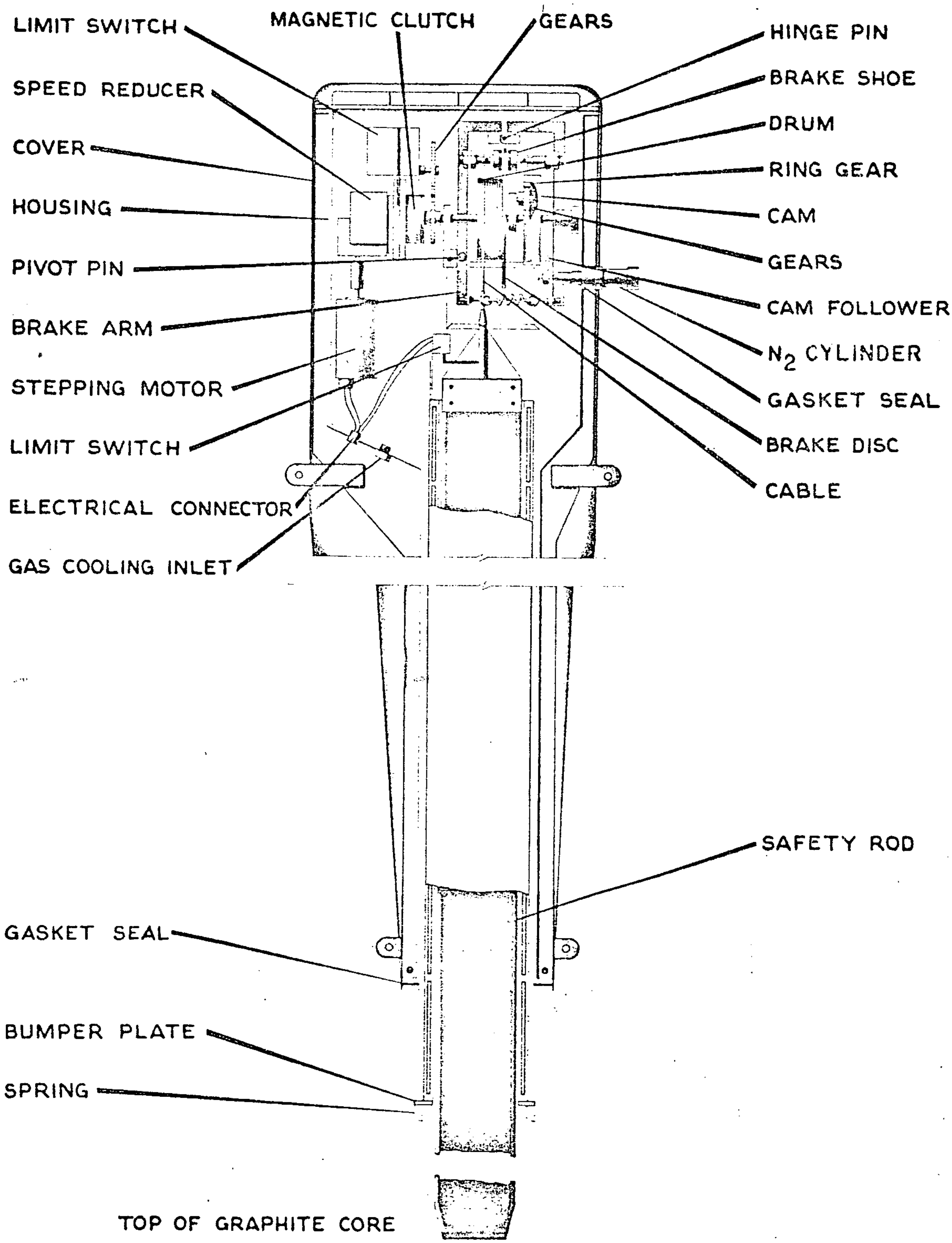


FIGURE V-3

Vertical Safety Rod

at the full-in position of the rod until the safety circuit is made up, preventing drum movement (and rod movement) until reactor startup is planned. The time required for full insertion of the rods including the brake action, instrument delays, and magnet release time is less than 1.25 sec.

All of the parts described, and limit switches which provide indications when the rods are fully in, fully out, (and at the 40 inch level) are enclosed as an assembly in a housing which is sealed to the reactor shell and contains the reactor gas atmosphere. The only penetrations into the housing are the electrical leads (bulkhead fittings), the brake release cylinder shaft, and the cooling gas inlet. Nitrogen is used as the motive power to the cylinder to avoid air contamination of the reactor atmosphere.

The maximum rate at which the safety rods can be withdrawn is set by the pulsing rate from PMACS. Rod movement is 0.003 inch/pulse and PMACS is limited to 125 pulses/sec, thus the maximum withdrawal rate is 22.5 inches/min. If PMACS should attempt to deliver a higher pulse rate, a further limitation is set by the usable rate of the stepping motor, i.e. the maximum rate at which the motor can accept pulses and still develop enough torque to operate the drive. In this design this maximum rate is approximately 140 pulses/sec.

Manual scram buttons are provided so that the electric power to the dc power supply for the holding magnets may be cut, independently of the safety circuit. Thus a scram may be initiated even in the event of a failure of the safety circuit.

H. Safety Circuit (See Figure XI-4, page 11.24)

The safety circuit is a separate dc loop, consisting of a power supply, a coupling transformer, solid state diodes, and the holding magnet coils for the 8 control rods and the 4 safety rods.

The reliability of the circuit logic and of the components used, which are either passive or solid state devices, is discussed in Appendix F. Instantaneous, automatic shutdown scram will be initiated by any of the following:

1. Neutron flux channel down scale (low level trip)
2. Neutron flux channel up scale (high level trip)
3. Short reactor period
4. PMACS self-audit signal failure (10 audits per second)

In addition some off standard process variables or equipment interlocks may be programmed in PMACS to cause an automatic shutdown, such as the (nuclear incident monitor) or movement of an access door to the reactor room or reactor basement.

Each of the first three shutdown functions is provided by two duplicate channels set up in one out of two failure logic. Each channel consists of a fission detector ion chamber, an amplifier, a period circuit, and a power supply. If either channel produces an off-normal signal for any of the three functions, the reactor is scrammed automatically. The system is designed so that component failures also result in a reactor scram.

In addition to the above, a separate set of neutron flux chambers, which also provide the signals for nuclear experimentation, are arranged in dual channel configuration to provide two signal channels to PMACS. In response to off-normal signal levels, PMACS will trip the safety circuit on a one out of two logic basis at the levels of downscale (low level), upscale (high level), and short period corresponding to those of the fission chambers. Thus two entirely separate one out of two systems of different design have been provided for monitoring and shut down control of the neutron flux level.

The PMACS self-audit signal results from an internal performance check every tenth of a second. The self-auditing includes a random memory check (programmed) and two independent neutron flux digitizing checks (hardware). This audit also provides a check on the internal clock and timing signal.

no it doesn't

The nuclear incident monitor is the standard Hanford nuclear criticality alarm. The chamber is located on the reactor building roof and will be set at an appropriate tripping point between 5 and 20 mrem/hr.

In addition to automatic shutdowns, manual scram buttons are provided at the operating console, inside the reactor room, and in the reactor basement.

Overall supervisory control of the safety circuit and reactor operation is provided with two key lock switches on the control console which control power to the safety circuit and to the computer, respectively.

J. Annunciators

An alarm will sound and visual annunciation via PMACS will occur for any of the following off-standard conditions which do not require immediate automatic reactor shutdown:

- Building radiation monitor high level
- Unscheduled high temperatures in the reactor*
- Loss of cooling water flow to the cooling coils in floor under reactor*

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* These signals also shut down the electrical heaters automatically.

- Loss of cooling water flow to the reactor heater terminals*
- Loss of cooling water flow to the low voltage transformers supplying the heater busbars*
- Basement sump high level
- Unscheduled high heater power*
- Off-standard conditions in the reactor gas circulating system, such as:
    - High moisture content**
    - Low gas pressure to gas blower seals
    - Carbon monoxide, carbon dioxide, oxygen or hydrogen significantly above tolerance
    - Fission product activity in purge gas filter
    - Off-normal temperatures in gas blower bearings
    - High temperature, main heat exchanger outlet
    - High temperature, rod heat exchanger outlet
    - High differential pressure, main filter
    - Low purge gas flow
    - Alpha activity in purge gas flow
    - Off-normal liquid nitrogen supply
- Safety circuit ground
- High or low water level in the process water break tank

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* These signals also shut down the electrical heaters automatically.

** This signal also automatically shuts down the gas blower and the electrical heaters.

Alarm annunciation is also separately provided on the heating and ventilating control panel located in the control room for seven variables of the high capacity ventilation system which serves only the reactor building:

- Input temperatures--four monitored
- High exhaust temperature
- Off-normal exhaust flow
- Failure of exhaust fan

Off-normal alarms (16 provided) for the other two heating and ventilation systems are located on the control panel in the heating and ventilating equipment room.

Any transducer for temperature, flow, pressure, pressure differential, etc., in the process flow stream or in a service or utility system can be programmed by PMACS for off-standard alarm, using one of the analog input channels up to a total of 128, or one of the digital channels, also up to 128.

K. Core Nuclear Characteristics

The HTLTR test core will usually be loaded either with graphite alone or with graphite plus fuel and other materials in a regular lattice array. The driver region will be loaded with one or more rings of fuel rods, and the ends of the test core will be loaded with layers of the short flux leveling slugs.

The nuclear characteristics of the HTLTR are not greatly affected by the loading of the test core. Since the calculation of these characteristics must be based on some assumed loading of the test core, the values given in this section are for the driver lattice required for a test core containing graphite and natural uranium, and with a typical array of flux leveling slugs in the ends.

Four-energy-group neutron calculations have been made for a typical driver rod configuration to determine the critical mass of fuel for HTLTR and its distribution between radial driver rods and end flux leveling slugs. A number of calculation methods were tested against a known graphite-moderated natural uranium lattice to determine which model best approximates HTLTR.

Values of the Doppler coefficient were calculated using Doppler broadening of a Breit-Wigner single-level resonance shape. This calculation gives a value of $\frac{1}{k}\frac{dk}{dt} = 9.93 \times 10^{-6}/^{o}C$ at room temperature. This gives a $\Delta k(T)$ from room temperature to a fuel temperature T(K) of $4.95 \times 10^{-4}(T^{1/2}-293^{1/2})$; from room temperature to 1000 C Δk is -0.92% and from 1000 C to 2000 C it is -0.59%. Thus there is available a prompt negative temperature coefficient to limit a reactor excursion at all temperatures.

The temperature coefficients calculated are the best estimates available. Admittedly, however, the methods are not very precise. The actual temperature coefficients will be measured as a part of the initial startup program. This is, in fact, one of the main fields of usefulness of HTLTR: to obtain experimental temperature coefficient data for high temperature reactors, for which calculated coefficients are too imprecise.

All of these calculated parameters are presented in Table V-I.

The results of calculations to determine the effect of adjusting the slow temperature coefficient of the reactor with gadolinium oxide shims are shown in Figure V-4. For these calculations it was assumed that a ring of 76 graphite shim rods, each 1-1/4 inches in diameter and 8 feet long, containing the weight percent of Gd_2O_3 indicated, was placed in the reactor immediately outside the driver ring. The amount of this poison required to reduce the temperature coefficient

TABLE V-I

HTLTR DRIVER LOADING

Calculated fuel requirements and nuclear characteristics for a 54-inch unfueled test core; no gadolinium oxide shim rods.

Number of 5 ft drivers	68
Number of 15 in. end drivers	240
Critical mass, U^{235}	47 kg
Total fuel mass (5% $U^{235}O_2$)	1067 kg
Worth of one 5 ft driver	25¢
U^{235} to carbon atomic ratio	1.92×10^{-4}
U^{238} to carbon atomic ratio	3.64×10^{-3}

Temperature coefficients (¢/C):

Temperature range, C	20-425	425-700	700-1000
Nitrogen coefficient	+ 0.039	+ 0.014	+ 0.009
Doppler coefficient	- 0.155	- 0.117	- 0.093
Moderator coefficient	- 0.631	- 0.605	- 0.451
Total temp. coefficient	- 0.747	- 0.708	- 0.535

Reactor kinetics parameters:

Temperature, C	20	425	700	1000
Prompt neutron lifetime, msec	1.18	0.92	0.85	0.80
Migration area, cm^2	7.09×10^2	7.66×10^2	7.95×10^2	8.22×10^2
Nitrogen pressure coefficient (¢/mbar)	-0.062	-0.045	-0.043	-0.040

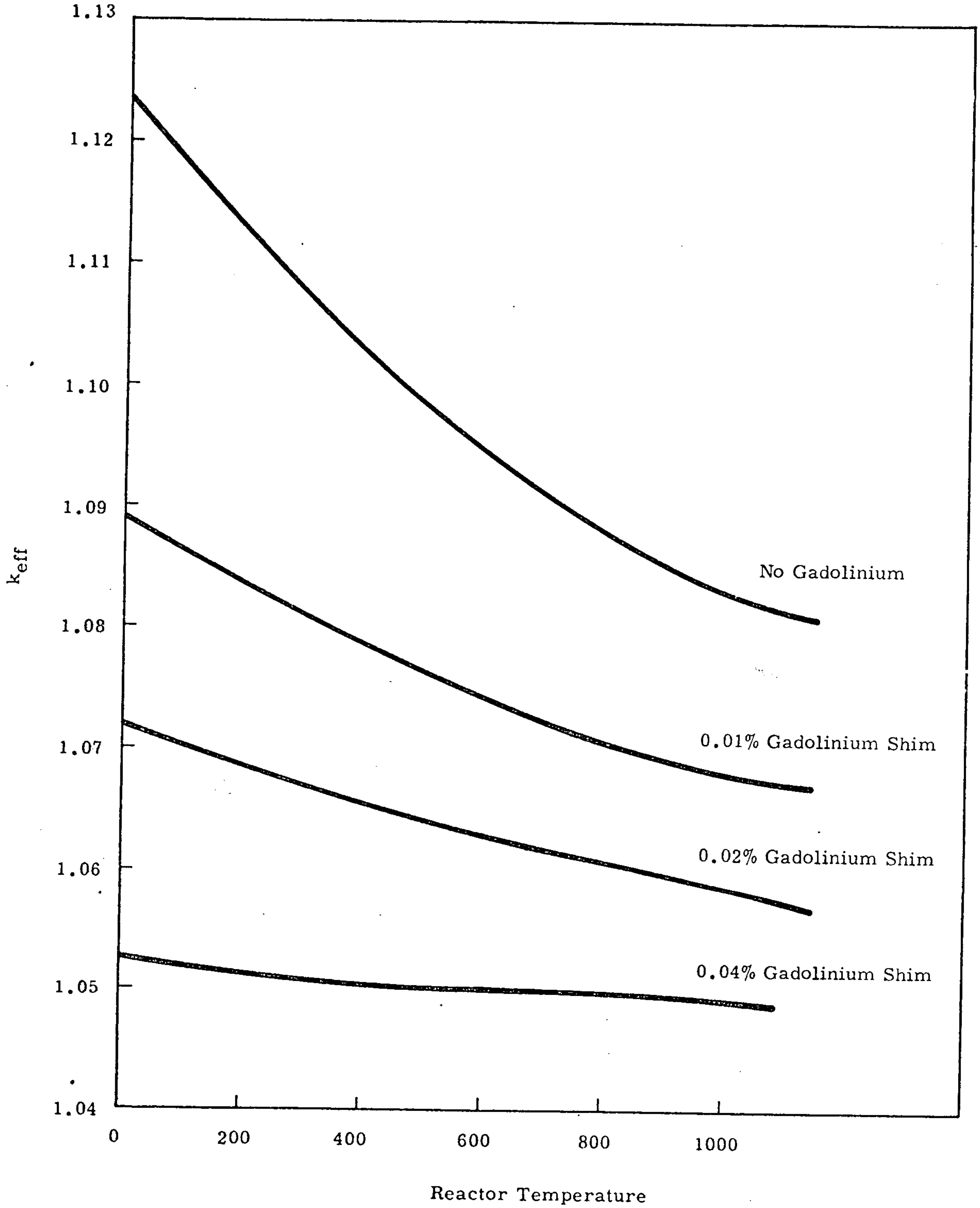

FIGURE V-4

Effect of Gadolinium Shim in
Reducing the Slow Temperature Coefficient of HTLTR

2

sufficiently for proper operation will be determined during initial startup tests. If the slow temperature coefficient shows a trend toward becoming positive above 1000 C, iridium or hafnium shims may be used in addition to gadolinium.

Even with the shims in place, it is possible that a test core might be loaded having a greater negative temperature coefficient than expected, because of the uncertainties in these calculations. This would cause the margin of excess reactivity at the desired operating temperature to be too small for efficient operation. If no compensation were provided, serious loss of experimental time would ensue, since the time to cool and reheat the reactor is so long. Therefore, for certain test cores of unfamiliar materials, it is planned to load excess reactivity equal to the worth of two control rods in excess of the normal limit of $1.90. The two control rods will be locked at the reactor face in their least reactive position and will be unlocked only if the extra margin of reactivity is needed for experiments at the operating temperature. The two rods will be returned to their least reactive positions and relocked before the reactor is cooled.

Figures V-5 and V-6 show the flux and adjoint flux distributions calculated for HTLTR with a natural uranium lattice in the test core, 68 fuel rods in the driver region, and with no gadolinium shims. The calculations were made with the HFN neutron diffusion code (one-dimensional, multi-region, multi-group code) for the following neutron energy groups:

Group 1	150 keV and up
Group 2	3 eV to 150 keV
Group 3	0.6 eV to 3 eV
Group 4	0 to 0.6 eV

Only the group 4 flux shows much variation with temperature, increasing with

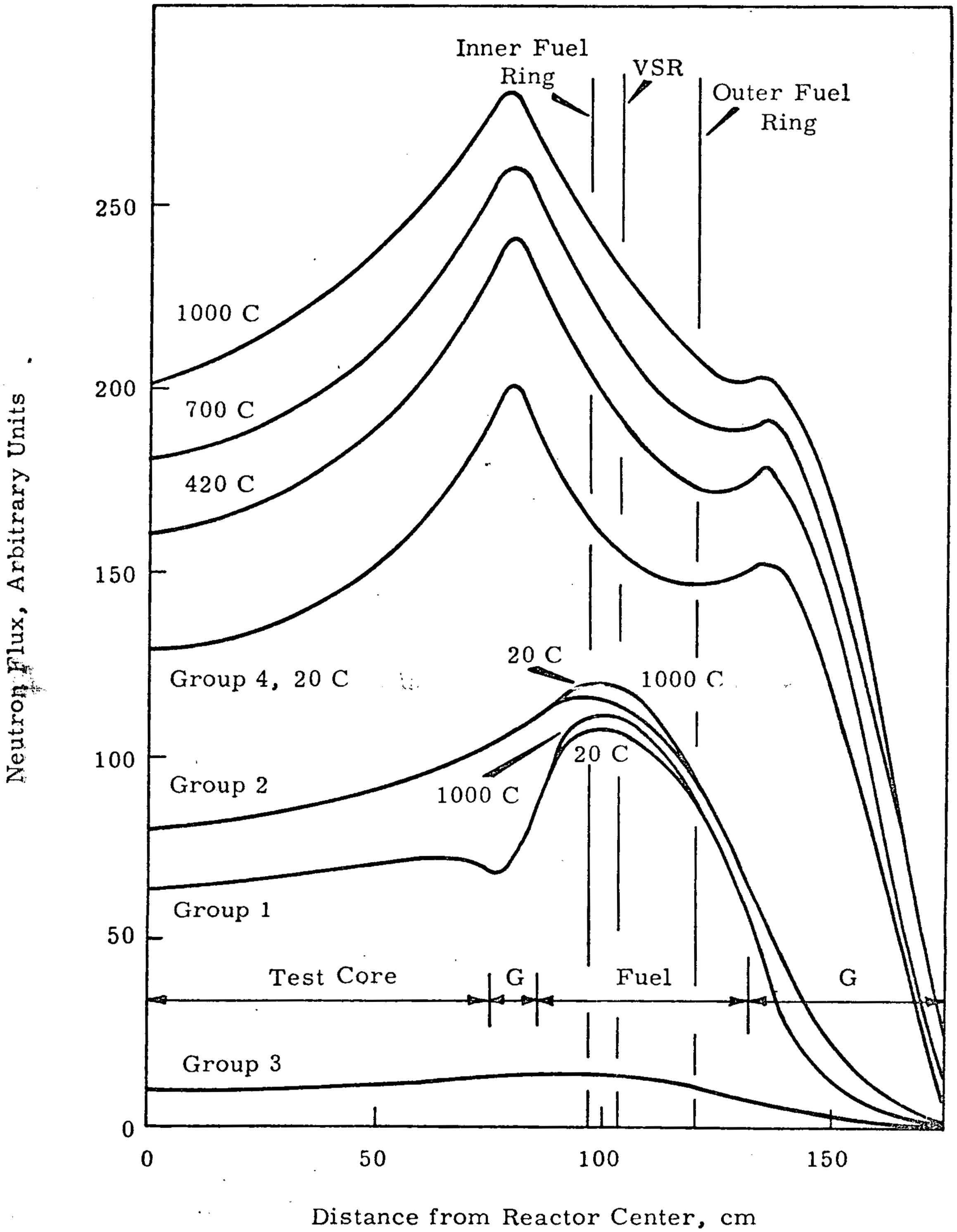

FIGURE V-5

Flux Distribution, HTLTR Core

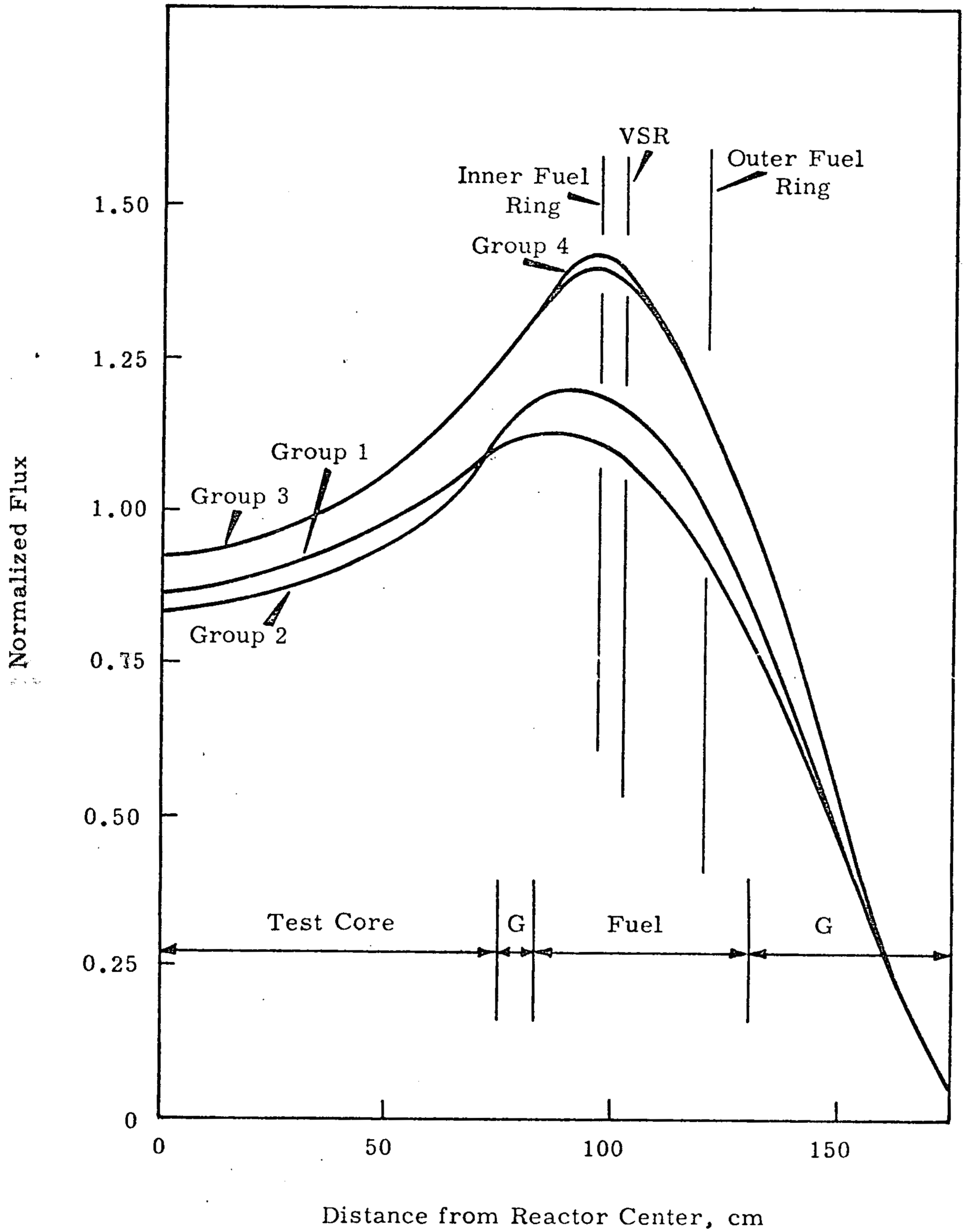

FIGURE V-6

Adjoint Flux, HTLTR Core

temperature for equal fission rate as expected, considering the fairly large negative moderator temperature coefficient of the reactor. The group 1 and 2 fluxes show a slight increase with temperature in the driver fuel region. The group 3 flux and the adjoint flux show negligible variations with temperature. The figures show normalized flux level versus distance from the reactor center in cm. The boundaries of the test core, a graphite buffer region, the fueled driver region, and the graphite reflector are shown, as are the positions of the inner and outer fuel rings and the vertical safety rods.

The calculated flux distributions were used in estimating the worth of the control and safety rods. Considering only thermal neutron absorptions, these calculations predict worths of about 50¢ for each control rod and 1.50$ for each safety rod. Epithermal absorptions will probably increase these worths somewhat. The flux distribution will be considerably changed by variations in the loading of the test lattice and by the adjustments which will be required in the buffer region. It is anticipated that the control rods will never be worth less than 35¢ each and the safety rods 1.00$ each.

Depending entirely upon the loading, for example, a shutdown margin for the reactor could be (with two control rods locked) 6 control rods at 35¢ = 2.10$ plus 4 safety rods at 1.00$ = 4.00$ for a total of 6.10$ against 1.90$ excess reactivity loaded, a margin of 4.20$. Or, when the reactor is heated to operating temperature, the minimum shutdown margin could be 6 control rods at 35¢ = 2.10$ against about 50¢ excess reactivity at operating temperature, leaving 1.60$ margin in the control rods, and 4.00$ also available from the VSR. Thus, when the reactor is cold, only two VSR would have to drop, or all control rods and no VSR's, for a shutdown. When the reactor is at temperature, any two control rods or any one VSR would shut the reactor down. The above discussion is for the partially shimmed reactor. With the full complement of the Gd_2O_3 shims in place, the cold reactor would be shut down by only any three control rods or any one VSR. It is not intended to start up the reactor (nuclearly) unless all control and safety rods are operable and of at least the minimum worths listed.

L. Core Thermal Characteristics

The weight of the graphite in the core is approximately 100,000 lb. The specific heat over the range from 260 C to 1000 C averages 0.4 Btu/(lb)(F). The heat input to heat up the reactor through this temperature range is therefore $(1000 - 260)(1.8)(0.4)(100,000) = 53 \times 10^6$ Btu or about 15,000 kW-hr assuming no losses from the graphite. The time required to heat the reactor will therefore be more than 30 hours.

The initial cooldown rate with a flow of 9000 lb/hr of nitrogen having an average specific heat of 0.26 Btu/(lb)(F) will be $(9000)(0.26)(1000 - 260)(1.8) = 3.1 \times 10^6$ Btu/hr. The final cooldown rate with a flow of 10700 lb/hr of nitrogen having a specific heat of 0.25 Btu/(lb)(F) will be $(10700)(0.25)(260 - 95)(1.8) = 0.8 \times 10^6$ Btu/hr. The average cooldown rate will be about 2×10^6 Btu/hr, assuming no other heat losses. The cooling time from 1000 C to 260 C will therefore be about 26 hours. The normal maximum cold excess reactivity to be charged to the core is 1.90$, with the possible addition of 1.00$ in locked control rods. Of this, 50¢ will remain as operating margin at operating temperature. The maximum possible rate of reactivity addition by normal cooling is therefore $(3.1/2)(240)/(26)(3600) = 0.0040$¢/sec. For abnormal conditions the cooling rates are still slower, as shown in Section XI.

VI. EXPERIMENTAL AND TEST FACILITIES

VI. EXPERIMENTAL AND TEST FACILITIES

A. Test Lattice Oscillators

Two oscillators are provided. One of these, on the rear face of the reactor, is capable of remotely moving either the test cell or the adjacent cell into and out of the test core. These cells may be up to 15 x 15 x 84 inches and weigh up to 1200 lb. The stroke is adjustable up to 36 inches with any time cycle greater than 5 sec. This permits making the substitution experiment from the control room. This oscillator may also be used to move a fuel element train to permit comparison of test fuels with a standard reference.

The other oscillator is on the front face of the reactor and will manipulate small specimens anywhere in the central 15- by 15-inch region of the test core. It is used to insert neutron detection foils for irradiations of short duration and to insert standard neutron absorbers as a calibration of the reactivity of the test cell. It has a capacity of five pounds and a maximum stroke of nine feet in a minimum of 2.5 seconds. The period of oscillation is adjustable. Cooling chambers and isolation valves are included on both oscillators so that specimens may be removed from the reactor and replaced with others without the necessity of cooling the entire reactor. The specimens can be cooled from 1000 C to 100 C in one hour and removed through ports in the chambers.

B. Neutron Time-of-Flight Spectrometer

A neutron velocity spectrum analyzer is mounted on a side face of the reactor at the lateral center line. A removable graphite plug permits the installation of a collimator allowing a beam of neutrons to emerge from the test lattice. A rotating chopper is mounted in the beam just outside the reactor gas envelope.

An evacuated flight tube permits the neutron pulses from the chopper to reach the target 25 meters from the center line of the chopper rotor. A shutter stops the beam when not in use.

The target consists of a bank of neutron detectors (enriched BF_3 proportional counters) arranged to present a 12-inch square target of about four to five inches depth, all feeding a multi-channel time-of-flight analyzer system.

This equipment provides a measurement of the velocity spectrum of the thermal and near-epithermal components of the reactor flux.

C. Core Dolly

A wheeled vehicle is provided on which test cores can be stacked in advance of experimental work and then transferred into the reactor as a single unit when the experiments are to start. The machine has a load capacity of 28,000 lb. After the vehicle is anchored to the floor of the reactor hall, the transfer of the core into position in the reactor is accomplished by a ram mechanism having a thrust of 10,000 lb. Only unfueled moderator will be transferred in this manner. Fuel is added to the reactor by the loading procedures described in Section IX.

D. Wire Traverse Mechanism

Two mechanisms are provided for flux determinations from front to rear and from side to side close to the reactor center lines. Test wires are loaded in the storage units and the reactor core while the reactor is at room temperature. Immediate removal of the flux wires following a traverse is possible following closure of compression valves near the shell penetrations. The valves also provide gas-tight closure for the four penetrations when the equipment is not in service. The housing shells of the storage and receiver units for the wire become part of the gas shell enclosure during routine use of the tool.

VII. INSTRUMENTATION AND CONTROL

VII. INSTRUMENTATION AND CONTROL

A. Neutron Flux

Uncompensated ionization chambers provide the neutron flux signal for the high precision experimental measurements. The chambers are located at the top center of the reactor, outside the gas envelope, beyond the high temperature zone and over openings in the Boral shield. An inch of lead on the bottom side of the chambers shields these units from the reactor gamma flux, and two inches of polyethylene above provides a reflector. The chambers are paralleled electrically into two groups with the more sensitive group (one or two more chambers) providing a signal of 5×10^{-11} amps at a flux of 30 n/cm^2 sec. The analog signal from each group is digitized in PMACS and covers the entire range, from source level to maximum permissible power, in 7 decades. A signal to trip the safety circuit is generated in PMACS from these channels for off-normal flux levels and periods.

Two independent channels using fission counters provide level and period signals to trip the safety circuit directly from the appropriate response in either or both. The channels are designed for high reliablity and speed of response. Accuracies in neutron flux measurement of 20 per cent of a decade and variations in period measurement up to $\pm$ 10 percent over the range from 10 to 100 seconds are acceptable with this system.

A BF_3 proportional counter located near the ion chambers at the top of the reactor gives a signal which, when amplified and supplied to loudspeakers in the reactor room, reactor room basement, and control room, gives a continuous and audible indication of flux level. A step switch is provided to adjust the scale factor of the amplifier to the existing flux level.

A nuclear incident monitor which is sensitive to neutrons only is located on the roof of the concrete reactor building to provide a criticality alarm from an incident that might occur anywhere in the facility. A signal via telephone lines is provided to 300 Area Security Patrol headquarters to initiate the appropriate emergency procedures for the area.

B. Radioactivity

Zone radiation monitors sensitive to gamma radiation with nonsaturable count rate meters having a range from 1 mR to 1000 R per hour are located in the assembly room, the reactor room, and the reactor basement. An alarm and identification is provided in the control room via PMACS.

A beta-gamma detector is located at the purge nitrogen filters. In addition, downstream of the main filter and at the input to the purge filters two continuous samplers are provided for alpha monitoring of the nitrogen gas. The alpha monitors provide an alarm within five minutes at a contamination level of 10^{-4} $\mu Ci/ft^3$. These detectors are coupled to PMACS for continuous surveillance and off-normal indication.

An alpha-beta-gamma hand and foot counter is provided for routine personnel monitoring.

C. Reactor Temperature

Reactor temperature measurement for experimental information is obtained by:

- 16 sheathed thermocouples entering the driver section from eight penetrations on each side

- 16 sheathed thermocouples entering four penetrations in the test core section from the rear face
- two movable resistance temperature detectors (RTD's) entering the test core from two penetrations on the rear face.

The RTD temperature scanning will be restricted to precision measurements below core temperatures of 400 C. The above inputs to PMACS will provide both test data and information for control of reactor temperature.

The outputs of the same 32 thermocouples are processed by PMACS into four zone temperatures. Power to each of the four groups of heater rods is controlled by the appropriate zone temperature signal which controls the output of one of four transformers with saturable cores. In addition, each heater rod is individually protected from over-temperature as the surface temperature of every rod is monitored by a thermocouple whose output PMACS records and compares to a programmed set-point. For as long as the measured surface temperature of any single rod exceeds its set point, electrical power to it is automatically removed. Each of the 24 heater rods is also equipped with a second thermocouple for a spare.

D. Gas System

Gas pressures and temperatures at various points, rates of gas flow, and the composition of the gas stream are sensed by transducers of standard design whose outputs are scanned at a high rate by PMACS. The signals are processed by PMACS to display a temperature or control a flow rate or indicate an off-normal condition depending upon the program within PMACS.

The make-up nitrogen admission valve, the exit purge flow control valve and the cooling water supply valves to the heat exchangers are typical of the process valves operated by PMACS to maintain set point control of temperatures and flow rates for the gas system. No direct pneumatic lines to the control room are used either for measurements or control.

E. Programmed Measurement and Control System (PMACS)

PMACS is a multipoint data handling device, with on-line computing capability to provide closed loop control on both nuclear and non-nuclear systems. Also, it can be used as an off-line computer, following an experiment, to process the accumulated data. The unit consists of solid state logic circuits designed for this application, coupled to a commercially available digital computer.

All data handling and control functions are performed by the PMACS according to stored programmed instructions and in response to instructions from the operator. The operator, using the input typewriter or a pre-programmed tape, instructs the computer to perform its functions. The computer displays nuclear and process information on a 19-inch cathode ray tube. The display is a combination of letters and numbers, arranged to identify the point being monitored, followed by a number indicating the quantitative measurement. This is referred to as an alpha-numeric display. To assist operator comprehension, normal operational data are displayed in blue or green and off-standard process conditions are indicated in red. The output typewriter also records off-standard conditions, and logs process information for later reference and a historical file. A second five-inch cathode ray tube provides a display of curves of process trends on command, as a further assistance to the operator.

The computer itself has a high speed magnetic core memory of 8,192 18-bit words. If power is lost to the computer, the information stored in the memory is not degraded nor lost, and is intact for recording and control when power is restored. There are two low-speed, three-million-bit magnetic tape memory units. One of these magnetic tape units is used by the computer to store historical process information for display or logging. The other magnetic tape unit stores the programmed instructions loaded into the computer by means of punched paper tape; this tape unit has a write lock-out feature to prevent inadvertent alterations to the programmed instructions.

Since the operator uses the computer to direct process control and to display system information, it must be capable of completing its functions at speeds responsive to the process and to the operator needs. Instructions or data words are retrieved from the memory for use in calculations or logical decisions in 1.75 microseconds. Addition is performed in 3.5 microseconds. Because of this speed, the computer can spend a large fraction of its time performing diagnostic functional tests to check its own responses to process situations. For example, the computer checks the neutron flux level digitizing system 10 times per second and it will trip the safety circuit if the check indicates an off-normal result. Only the signal to automatically shut down the reactor has a priority associated with it that is higher than the priority of the requirement to check the flux digitizers every tenth of a second. The system is further protected in that if an out-of-limit signal occurs simultaneously with a command from the operator to display some data, the computer first responds to the out-of-limit signal and then handles the operator data display request.

This integrated reactor control system, in which the computer handles functions of the highest priority first, may be considered to be five subsystems, each sharing the time of the computer for the control of its part of the reactor and process variables, for the display of its raw or processed data, and as a warning to the operator of off-standard conditions. The five subsystems are flux monitoring, control rod positioning, gas circulation, reactor heater control, and experiment recording and analysis. The reactor safety system also receives signals for monitoring by PMACS, but since the safety system is separate from PMACS, it is discussed elsewhere.

The flux monitor subsystem provides a signal proportional to neutron flux level. Two banks of ion chambers each feed preamplifiers giving variable current signals proportional to the instantaneous neutron flux. The flux level signals are digitized, stored in the high speed memory, and continuously updated. Changes in flux level over brief intervals are used to calculate the reactor period. The calculations are perfomed ten times a second, and the results are compared with prestored trip points. If the results are out of limit, the reactor is shut down automatically. Precise flux level data is experimental information. When compared with rod position data from the control rod position subsystem, reactivity is computed by PMACS by the inverse multiplication method for information for the operator.

The control rod positioning subsystem monitors rod movement using limit switch signals, linear position transducer signals and feedback pulse signals from the rod positioning stepping motors. The computer processes the rod position information, stores it and directs its display in alpha-numeric notation on the large cathode ray tube. Pulses to drive the rod positioning stepping motors are produced by the computer, but additional operator action will be required to move the rods. The operator will select the rods to be moved and the speed of rod movement using the typewriter, and then must activate a control panel switch to enable the computer to produce the drive pulses.

The gas circulation subsystem provides the computer with temperature, pressure, flow, purity, and valve position information. The computer processes the information, stores it, and uses it to generate control signals for the gas circulation loop. The loop control signals drive stepping motors to position throttling valves or operate solenoid-actuated valves. If any of the gas circulation subsystem parameters deviates from its normal or pre-set value, the digital control computer adjusts process equipment to restore the system to equilibrium operation. The action to be taken is programmable and depends upon which paramater deviates and how large a deviation occurs.

Use of the gas system is optional when the reactor temperature is below 260 C. When the gas system is in use, the gas in the reactor envelope is maintained at a pressure slightly above atmospheric pressure, and is measured precisely to permit correction for changes in the neutron poisoning effect of the nitrogen. The measurement includes a span from 465 to 840 mm mercury absolute, to an accuracy of about 2 mm and with a repeatability of about 0.5 mm.

Reactor temperature and heater current signals from the reactor heater subsystem are used by the computer for display, to generate signals to drive the saturable reactors which control power to the heater elements to maintain the programmed operating temperature, and to produce a historical record.

The experiment subsystem will be used primarily by the experimenter to analyze changes in reactivity, neutron flux distribution and the velocity distribution of a chopped neutron beam, and possibly neutron flux decay under a neutron pulse excitation mode. However, reactivity determination by the computer using experimental subsystem data can also be used by the reactor operator for routine operation. The operation and control of the experimental equipment such as the neutron source, neutron chopper, light duty oscillator, and heavy duty oscillator will be computer-controlled at the direction of the operator; and the experimental data received by the computer will be processed for display or for logging and subsequent analysis.

The reactor operator thus has at his command an integrated reactor control system capable of generating and logging extensive experimental data while continually monitoring the entire process, pinpointing by alarm any process variable whose behavior is questionable, logging process variables for historical record, displaying their instantaneous value or trends, generating signals to control the process, and requiring the reactor safety system to shut down the reactor if that becomes necessary. The computer also monitors its own operation ten times per second. If a single failure of the self-audit test pulse occurs, the reactor is shut down automatically and the electric heater power is set back.

F. Neutron Source

75 μa beam at 2.0 Mev

The neutron source is a positive ion accelerator which is located on the level below the reactor. The maximum output is about 8 x 10^{10} neutrons per second from a water-cooled beryllium target bombarded by high energy deuterons. The target is positioned about two feet below the center of the bottom of the graphite of the reactor. An alternate position for the accelerator at the side of the reactor has been provided.

The accelerator is operated from the control room. Electrical interlocks on access doors prevent operation of the accelerator until personnel have left the vicinity and the area has been secured.

G. Console Controls

Controls at the reactor console are as follows:

Safety circuit power supply key lock switch. This ensures that reactor startups can be made only by authorized personnel with the knowledge of the reactor supervisor.

- Computer output power supply key lock switch. This disables the computer's control outputs, permitting operation of the computer with the reactor shut down and the safety circuit inoperable.
- Nuclear safety channel bypass switch. This permits bypassing one independent safety circuit at a time for testing or maintenance.
- Reactor room door and control room door permissive switches. These prevent unauthorized entry into the reactor room or control room.
- Control rod drive enable switch. Operation of the control rods is programmed in the PMACS. However, movement of the control rod requires that the operator hold this switch closed.
- Manual scram button.
- Two safety circuit function monitor lamps.

VIII. MAINTENANCE AND MODIFICATION

VIII. MAINTENANCE AND MODIFICATION

Maintenance procedures will be prepared to ensure that maintenance work on equipment and components essential to reactor safety is satisfactory. Maintenance work that deactivates a system, renders it inoperative, or which can alter its performance as a result of the maintenance effort will be covered by written authorization approved by the reactor supervisor or systems engineer. The written authorization will describe the work to be performed, list any special conditions to be met, such as deactivation of the system, and prescribe any tests or checks to be made when the work is complete. Upon work completion the authorization will be closed out. All outstanding authorizations will be reviewed prior to the initial reactor startup of a test or experiment.

Modifications to the HTLTR may be made, provided the modification does not involve either a conflict with the Operating Safety Limits or an unreviewed safety question. Modifications which do not affect the nuclear characteristics, control systems, or instrumentation systems of the reactor require the approval of the Supervisor of the HTLTR. Modification of critical components requires review and prior approval by Laboratory personnel not having direct responsibility for reactor performance, in addition to the Supervisor, HTLTR.

All modifications of critical components will be processed as design changes. A design change will describe the changes to be made in detail, list any hazards asociated with the work, note the precautions to be taken, provide for updating the appropriate drawings, and prescribe tests or checks to be made prior to activation of the involved system or equipment.

The status of maintenance work on the reactor and its auxiliaries is subject to audit by an independent organizational component not having responsibility for reactor operation or experimental planning.

IX. INITIAL TESTS AND OPERATING PROCEDURES

IX. INITIAL TESTS AND OPERATING PROCEDURES

A. Tests with No Fuel Present

Three types of testing are being completed before the reactor is ready for experimental work.

1. Mockup and materials testing provide the necessary information about prototype reactor components to ensure the design and construction of an operable and safe reactor system. Mockup and materials testing are being performed by plant forces and have included: a furnace containing a heavy-walled capsule into which samples of materials have been placed for exposure to a nitrogen atmosphere for long periods of time and at temperatures up to 1200 C; a small mockup of the reactor and thermal insulation containing a short section of a heating element, which have been operated at temperatures up to 1500 C; and a large mockup of the HTLTR graphite and insulating material, into which full-length prototype heating elements, control and safety rods, and other components have been placed and exposed to an environment very similar to that expected in the reactor. Natural uranium was used in place of enriched uranium in the prototype control rod. In the large mockup, temperatures over 1000 C have been attained in the graphite, and over 1200 C on the heater surface. Equipment whose performance cannot be predicted with confidence will be tested prior to installation. The following tests have been or are being conducted:

   - Properties of metals, alloys, and ceramics after exposure to nitrogen, graphite, and each other at temperatures up to 1200 C over periods of hundreds of hours. All materials considered for use in the HTLTR will be studied.

- Full-size prototype heater, control, and safety rods have been tested for mechanical integrity and operability up to 1000 C in the reactor environment. In these tests the control rods have contained non-fissile material.
- Behavior in the atmosphere of the mockup at high temperature, of samples of nickel, Inconel, thoria-dispersed nickel, and Hastelloy-B in contact with graphite, and in contact with insulating brick.
- The behavior of insulating materials, analyses of the gases they give off, and the temperature distributions in them.
- Procedures for evacuating and purging the system to ensure adequate gas purity at high temperature.
- Thermocouple properties at high temperature in the reactor environment.
- Prototype fuel element jackets.
- Conditions to be met by the reactor oscillators during the movement of large pieces of graphite and small samples of poison materials.
- The Programmed Measurement and Control System (PMACS) pretested and programmed.
- The response, linearity, and saturation characteristics of the ionization chambers as a function of ambient temperature.

2. Acceptance test procedures (ATP's) demonstrate that the equipment and systems of a new installation function correctly, as described by the drawings and specifications. Acceptance tests are prepared by the Architect-Engineer and are performed by the construction contractor. HTLTR Operations personnel will operate PMACS when needed in any of the ATP's. These tests are observed and approved

by BNW prior to acceptance of the facility. The following ATP's are presently listed:

- Utilities and services including heating and ventilation, electrical, fire alarm and sprinklers, nuclear and civil defense alarms, and the process water systems.
- Nitrogen gas system including the gas loop, supply, gas analyzers, and the evacuation and purge systems.
- Reactor case, insulation, moderator, control and safety rods, including performance at temperature and following cool-down.
- Tests of reactor-associated equipment including oscillators, neutron spectrometer, neutron generator, and the flux wire manipulator.
- Control and instrumentation (PMACS).

3. Design tests provide much more detailed information on the operation of reactor components and equipment than is given in the ATP's. Calibrations, measurement of sensitivities, timing of actuating systems, and other quantitative data will be obtained with the precision and thoroughness required for operational use. All of this work will be done before the initial fueling and nuclear startup of the reactor with the possible exception of performing the design tests at elevated temperatures. However the reactor will not be fueled or taken critical at an elevated temperature without first performing the following tests at that temperature or above. Omitting fuel in these tests includes the omission of the fueled parts of the control rods. Dummy fuel cylinders may or may not be used in the control rod tests.

Tests planned under this heading are:

- Functions of PMACS not dependent upon fuel being in the reactor.
- The reactor control and safety system.
- The neutron startup source and detectors, with safety rods in and out.
- The circulating gas system in evacuation, purging, heating and cooling.
- The functioning of all of the above systems at various temperatures.
- Functioning of the temperature control.
- Utilities and services, including effects of their loss or abnormal operation.
- Operation of the chopper, oscillators, flux wire manipulator, temperature sensors, pressure transducers, gas analyzers, etc.
- Alarm systems functional tests.
- Emergency cooling provisions.
- Exhaust filter integrity.

B. Initial Criticality Tests; Zero and Low Power Physics Tests

The initial approach to criticality will be made at room temperature with graphite in the test cavity. Enriched fuel including that contained in the control rods will be inserted in the driver region in accordance with standard procedures. The control rods will be the first fuel bearing components to be inserted. At this time measurement of control rod scram times, and any other quantities not covered in the unfueled tests, will be done. The control and safety rod worths will be measured by the inverse multiplication method as the initial approach-to-critical is made. These will be compared with the design estimates. After achieving criticality the following zero-power physics measurements will be made:

- Rods will be calibrated and the total safety system worths and response times will be measured. Flux monitors will be calibrated (approximately) in power level units. The neutron lifetime and various reactivity coefficients will be measured.
- After a second approach to critical with a natural uranium lattice in the core, measurements of k_∞ will be performed with the oscillator. The nitrogen correction will be measured. The pressure coefficient will be measured.
- Measurements of k_∞ at slightly elevated temperatures--100 to 200 C--will be made. Rod calibration and excess reactivity will be rechecked. The reactor will then be cooled.
- From the measured change in core reactivity with temperature, the amount of gadolinium and other shim material needed to compensate the slow reactor temperature coefficient to about 500 C will be estimated. The materials will be loaded and the reactor heated in steps to about 450 C. Measurements of k_∞ will be made at each step.

C. Experimental Program

The experimental program in the HTLTR will include the following types of measurements:

- The value of k_∞ of a lattice or medium, and its variation with temperature.
- The temperature coefficient of k_∞ when only a fuel element is heated.
- The worth of a lattice heterogeneity in a supercell as a function of temperature.

- Activated foil measurements of various lattice parameters, integral spectra, or effective lattice cross sections.
- Differential neutron spectra in multiplying or non-multiplying media, using the neutron time-of-flight spectrometer (chopper system).
- Neutron spectra in moderators near property discontinuities, as a function of temperature.
- Effective cross sections of materials as a function of temperature.
- Other types of measurements may be devised in response to the needs of sponsored research programs.

The program, to a large extent, will consist of the same types of measurements now made in the Physical Constants Testing Reactor (PCTR), with the additional capability for controlled temperatures to 1000 C and above as the only difference.

The test cavity, during the initial experimental programs, will be filled with graphite, various lattices fueled with U-233 and thorium or their compounds, various combinations and compounds of uranium at different enrichment levels, and plutonium of various isotopic compositions. The fuel may be metallic, ceramic, or cermetic in form depending upon the temperature to be reached. The cladding will be of nickel, thoria-dispersed nickel, Inconel or graphite, as required and appropriate from a review of the hazards involved.

The driver section of the reactor will be fueled with enriched uranium oxide, packed to about 95% of the theoretical density, and suitably clad. The total critical mass may vary by a factor of two depending upon the size of the experimental core. Some natural or slightly enriched uranium may be added to adjust the spectrum of neutrons entering the test core.

D. Operating Procedures

1. Startup

All changes in the test core loadings will be preceded by approach-to-critical measurements. Driver fuel from any previous run will have been removed to leave no more than half the number of fuel pieces predicted for criticality before fuel is added to the test region.

Pre-startup safety circuit and equipment checks will precede nuclear operation or fuel manipulation in the reactor. They will be repeated at suitable intervals, and following maintenance or design changes of major reactor components. The checks will include testing the actual movement of the control and safety rods. The neutron source will be in operation to generate the on-scale signal required to energize the safety circuit and to activate the audible monitor. Before any reactivity adjustments are made to the reactor core, the safety circuit must be in operation, two vertical safety rods must be pulled and left cocked, and a neutron source will be in operation to generate the on-scale signal.

The test core will first be loaded. Following this, the flux-leveling slugs in the regions of the center section of the reactor will be adjusted to the best predicted settings. Driver fuel will then be added. All fuel will be added in increments, with multiplication data taken after each increment. The first fuel additions will be limited to increments not exceeding one-fourth of the predicted remainder to critical, as determined by plotted multiplication data.

Reactivity will be monitored by the position of the control rods at critical. When a critical configuration has been reached, the axial neutron distribution will be measured and fuel positions adjusted as needed. These adjustments, as for any fuel positioning, must be done manually and will be permitted only with the eight control rods and two vertical safety rods fully inserted. The remaining two vertical safety rods will be raised and cocked and responsive to any unwanted short period or high neutron flux level.

The control rods will be calibrated when the required axial flattening has been achieved. Once calibrated relative to position, the calibration and relative worths of the individual control rods do not change markedly, as long as symmetrical driver fuel loadings are used. Thus, after initial calibration, it will be necessary only to establish the absolute value of one section of one control rod to evaluate the worth of the system.

2. Heatup

Initial critical and experimental operation from ambient temperature up to 260 C may be done with the reactor doors removed and with air in the core. For operation above 260 C, the core envelope will be sealed and the nitrogen atmosphere established, with a purge flow past the gas activity monitors. Evacuation and purging, as well as temperature changes, will normally be carried out with the system subcritical. Control rods will be recalibrated and excess reactivity checked at appropriate times during the rise to temperature. However, the system may be made critical while under partial vacuum to measure the nitrogen correction in some experiments.

3. Experimental Operations

After the reactor has been made critical at any test temperature, a variety of experiments may be done. All tests will become more or less routine, independent of the loading of the test region and of the reactor. Many experiments will utilize the two oscillators, as described briefly below:

a. The central cell oscillator will be operated to obtain period measurements as a function of central cell or buffer cell position, or to measure the reactivity coefficients of these cells by oscillating them from the fully-in position to a partially-out position.

Typical values for this test might be with a cell having an excess multipilcation of ten percent, and a temperature coefficient of -4×10^{-5} k/°C. At 1000 C the k_∞ of the cell may be reduced to 1.08. A three foot or greater stroke of the central cell would be expected to decrease the reactivity of the entire reactor by 16.5¢ in 5 sec at 20 C, or by 13.2¢ in 5 sec at 1000 C.

b. The sample oscillator will be operated to obtain period measurements as a function of the position of the sample in the central cell or a buffer cell,or to measure the reactivity coefficients of the samples by oscillating them from the fully-in position to a partially-out or fully-out position. These samples would normally be worth from 2 to 10¢, i.e., large enough to measure to an accuracy of a fraction of 1%.

In a test involving addition of poison to calibrate the change in reactivity of the central cell, a piece of 40-mil sheet material, an inch wide and 25 inches long, weighing 150 g would be typical. Such a piece has a worth of about 3.3¢, or 45 g per one cent reactivity change. Inadvertent removal of the poison sheet entirely from the reactor would contribute only an additional 2.5 to 3.5 cents.

c. The wire traverse mechanism will be operated with wires of various materials; or the oscillator will be used to position foils or samples for irradiation. Irradiation will be typically between 0.1 kW-min and 6.0 kW-min.

d. The neutron beam chopper and neutron spectrometer will be operated to measure the thermal and near-epithermal neutron spectrum of the test region at the operating temperature. A typical experiment would require operation of the reactor at a constant power level of 500 W for 4 hr, or 120 kW-min.

e. The fine control rod will be used, either alone or in conjunction with one or more of the eight main control rods, to maintain operation at critical as

- the whole reactor, the central cell, a fuel element, or a sample is heated;
- the pressure of all or a part of the reactor enclosure is varied;
- the composition of the circulating gas stream is varied;
- a sample is withdrawn or oscillated.

f. The flux instrumentation will be used to obtain reactivity coefficients of changes in the composition or position of core or cell components. The changes will be made repetitively with the oscillators. Examples are:

- small changes in pressure
- small changes in gas composition
- small changes in position of one control rod

g. The pulsed neutron source will be operated while the reactor is slightly subcritical or just critical to provide repetitive bursts of nonfission (extraneous) neutrons into the reactor. The change in the neutron density following each burst will be measured by the flux monitors and the data will be analyzed by PMACS.

h. Some of these experiments may be performed simultaneously. For example, the cell and sample oscillators may be operated in phase or out of phase to give an overall null response of differential flux (or power) as a function of time.

4. Shutdown and Deactivation

Nuclear shutdown of the reactor is simple and flexible. The safety and control rods can be run in individually at controlled speeds, or the entire rod system can be inserted at high speeds. Both of the above modes can be done automatically or manually.

Recirculating cooled nitrogen gas will cool the reactor at a controlled rate. Gas purity will be maintained until the desired terminal temperature has been reached. The reactor shell door can be removed, exposing the system to ambient air, whenever the maximum graphite temperature is below 260 C.

A shutdown precipitated by a power or cooling system failure will create no problems to any part of the reactor or fuel but would lengthen the time required to cool down the reactor. The cooled reactor will be deactivated by defueling and/or by inserting the rods and locking out the power to the safety circuit, the rod drives, the neutron source, and the reactor heaters.

While deactivated but still fueled, the reactor would be under constant surveillance through an alarm circuit to a neighboring constantly-manned facility.

5. Charge-Discharge of the Test Region

The removable core will be assembled outside the reactor, checked for alignment, and the test cells functionally checked with an oscillator. The experimental core will not be fueled outside the reactor.

Typically, with the driver region defueled to less than one-half of that required for predicted critical, the removable core will be placed in the test cavity, as a single large stack if the equipment (core dolly) is available, or manually, block by block, if necessary. The reactor will then be fueled as previously described, and the heaters, traverse wires, thermocouples, and oscillator actuators will be connected.

Upon shutdown, the procedure is exactly the reverse. After sufficient cooldown, the required fraction of the driver fuel will be removed, the experimental and test cells will be decoupled from the oscillators, and the heaters will be removed. The experimental core will be defueled and then removed.

The capability of performing experiments with water in the test lattice is not excluded. The above procedure will be extended to include leak testing of the in-reactor section, reliability and back-up status of the out-of-reactor equipment, a check for reactivity worth for loss of H_2O, and special procedures to cool, depressurize and vent the in-reactor loop during cooldown. The limiting temperature for water experiments will, of course, depend upon the strength of the container at the desired temperature.

X. ADMINISTRATION AND PROCEDURAL SAFEGUARDS

X. ADMINISTRATION AND PROCEDURAL SAFEGUARDS

A. Organization

The HTLTR will be operated by the High Temperature Reactor Physics Section of the Reactor Physics Department of the Battelle-Northwest Laboratories.

The manager of the High Temperature Reactor Physics Section will be directly responsible for the ultimate safety of the reactor. The staff will include a senior engineer, responsible for all start-up activities, reactor operation, and maintenance; a systems engineer, one or more senior working leaders, and two or more reactor operators. Qualified reactor operators from the PCTR, TTR, and PRCF will be available for training at HTLTR for initial and subsequent relief operations. An operator will be qualified upon successful completion of the requirements described below.

Although the reactor will usually be operated only on day shift, it will sometimes be operated continuously. Two qualified operators will be in the control room during the periods of nuclear operation. The reactor will either be attended or connected through an alarm circuit to a constantly manned station when its temperature is above 260 C.

Operation and maintenance of the reactor and storage of its fuel inventory are subject to audit by the Nuclear Safeguards and Engineering Section of the Environmental Health and Engineering Department of Battelle-Northwest.

B. Training of Personnel

1. Reactor Operators

Canidates for training must have a minimum of high school education or equivalent and a demonstrated interest and ability in technical work. They must be alert and reliable; and they must display a mature attitude in matters of safety and teamwork.

The training program to qualify reactor operators is described in Appendix C. Classroom instruction will be given to the first group of trainees. Some will spend time in the facility during construction. Some will observe the acceptance tests and all will take part in the pre-operational design tests performed after acceptance of the facility.

2. Reactor Supervisor and Systems Engineer

The Reactor Supervisor must have completed a four-year college program in physics, engineering or chemistry with courses in reactor physics or equivalent experience. Prior experience in reactor operation is required. The Supervisor will become familiar with the background of the project, the experimental physics program, and the design criteria; will be trained on-the-job; and will pass written and oral tests on all phases of reactor operation.

The object is to become familiar with all equipment and to understand it completely but not necessarily to attain dexterity. A thorough knowledge of the safeguards analysis, operating limits, and operating procedures is necessary. The approval procedure is similar to, and the authorization review period is identical with, that given for the reactor operator candidates.

The Systems Engineer must have completed a four-year college program in one of the fields of engineering or science, or attained the equivalent in experience. His main responsibility will be to maintain all systems in proper operating condition. On-the-job training in reactor operation will be provided until the incumbent is deemed capable of replacing the Reactor Supervisor in an emergency or for short periods of time under defined restrictions. The approved procedure is similar to, and the authorization review period identical with, that given for the Reactor Supervisor.

C. Initiation and Control of Experiments

Experimental programs may originate within the Hanford complex or may be undertaken at the request of the Atomic Energy Commission. After considering a proposal, the Manager, High Temperature Reactor Physics Section, selects an experimenter-in-charge and tentatively schedules the experiments. In practice, most of the experiments are designed by members of the Reactor Physics Department of Battelle-Northwest and most of the experimenters will be members of the High Temperature Reactor Physics Section.

All experiments must satisfy the requirements of the Operating Safety Limits. A written procedure is prepared describing the experimental steps to be followed and the hardware items to be used. In addition, the procedure describes possible hazardous conditions that could occur during reactor loading, operation, or unloading, and any special fuel- or material-handling precautions that may be required. The procedure is approved by the Reactor Supervisor and the Manager of the High Temperature Reactor Physics Section. A permanent file is retained. If an experiment is a minor variation of previously completed work, a less comprehensive procedure may be prepared, but the section describing potential hazards is mandatory.

The experimenter is responsible for procurement of any special equipment or fuel; for providing guidance to operations personnel while the experiment is in progress, when necessary; for recognition of any unsafe condition and initiation of a shutdown of the reactor, when necessary; and the data and reporting of the results obtained.

XI. ACCIDENT ANALYSIS

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A. General Safety Features

A critical facility such as the HTLTR, purposely built to permit wide variation in core configuration, must depend for safety in large part upon careful attention to procedure by the operators. Some of these procedures, described in detail above, are

- control of incremental fuel loading
- control of excess reactivity
- maintenance of worth of shutdown mechanisms
- control of reactivity addition rate
- routine functional testing of safety system components
- safety analysis of experiments

A considerable degree of inherent safety is also provided. The relatively long neutron lifetime in the driver region makes all reactor transients comparatively long in duration. Also the prompt negative temperature coefficient of the fuel, i.e., the Doppler coefficient, would terminate any nuclear excursion in the event that a scram were delayed longer than normal. The effects of the concurrent chemical reactions are discussed in Appendix A.

There are three barriers to the escape of plutonium or fission products. The first is the fuel element matrix and jacket, the second the gas envelope, and the third is the reactor room with its directed ventilation and exhaust filters. The effectiveness of barriers in preventing the escape of hazardous amounts of radioactive materials is discussed in later portions of this section.

The reactor control and safety rods are mechanically and electrically simple, and are of proven types. The shutter-type control rods add their full worth of negative reactivity in less than one second with a movement of only six inches. The four vertical safety rods drop by gravity and are completely effective in less than 1.25 seconds.

Both the control and safety rods are designed for fail-safe operation. The worst mishap that could befall either system would be loss of cooling gas to the rods and rod housings. The temperature rise would not be rapid enough following the loss of coolant to affect the mechanical integrity of the driver systems before the rods could be inserted into the reactor. However, if the temperatures did become excessive because of failure to take proper action, it would not prevent normal movement of the vertical safety rods and they would drop into the reactor. The response of the horizontal control rods at this time would be determined by the remaining strength of the drive springs after the latching solenoids failed. However, the spring size is such that the rods would rapidly travel over most of the six inch distance if they were in the most reactive position at this time.

The safety circuit is composed of five basic functional blocks; the process sensors, comparators, logic circuits, a power supply and the electromagnetic clutches. A desirable level of reliability is achieved by either making the functional section fail safe; that is, the reactor will automatically go subcritical in case of any component failure in that function; or by including at least one redundant, identical channel to perform the same operation. In some cases more than one redundant channel is provided; for example, four neutron flux monitoring channels are used. Diversity as well as redundancy is also achieved in this example by equipping two of the four channels with fission chamber detectors and the other two channels with ion chambers.

Interlocks are provided to prevent the withdrawal of control or safety rods unless the safety circuit is in operation. The downscale trips in the neutron flux monitoring and measurement systems prevent making up the safety circuit unless sufficient neutron flux is present at the sensors to cause an on-scale reading of the instruments. This arrangement positively prevents startup of the reactor without the startup neutron source in operation.

In addition to the special design that has been provided, all hardware and systems were obtained from specifications written to satisfy the reliability required in these particular circuits.

B. Accidents Considered

The accidents considered in these analyses are tabulated below:

- nuclear excursions
  - ramp addition of reactivity at various rates
  - effects of core temperature change
  - startup accident
  - loading accident
- mechanical and utility failures
  - leakage of water into the gas system
  - leakage of water directly into the core
  - breach of the reactor gas envelope
  - breaches of gas circulation system
  - failure of gas blower
  - loss of nitrogen supply

- electric power interruption
- service water supply interruption
- failures of reactor components by interaction of materials

- instrument failures
  - safety circuit reliability analysis
  - other instrument failures
  - effect of temperature on instruments
- escape of contamination from the fuel
  - particulate alpha contamination
  - iodine-131 and other iodine isotopes
  - other radioactive isotopes
- critical assembly outside the reactor
- spread of contamination in disposal of wastes
- interaction with other facilities
  - nuclear
  - contamination
- non-nuclear incidents
  - sabotage
  - bombing
  - earthquake
  - wind and storm
  - flood

The probability of some possible variations of these accidents is thought to be negligible, for the reasons noted in the discussion which follows. The maximum credible accidents are postulated to arise from startup errors and loading errors. Two typical errors are considered in detail. The safeguards against these accidents are sufficient to make them highly improbable, and their consequences are not worse than a tolerable exposure of one or two employees in the building to neutrons.

C. Nuclear Excursions

Calculations of power excursions resulting from a continuous ramp addition of reactivity are shown in Figure XI-1 for the following cases:

Initial reactivity	- 0.90$
Available excess reactivity	1.90$
Temperature	20 C
Rate of reactivity input	variable ramp

These transients were computed using the TRIP[3] code, with assumed failure of the log period trips so that the reactor is shut down by the log high level trips. It should be noted that these calculations, with the exception of the 4¢/sec and 8¢/sec ramp rates, are for entirely hypothetical cases and serve only to illustrate the favorably slow kinetics of HTLTR. There is no mechanism whereby reactivity can be added at a rate greater than about 8¢/sec. In addition, only rarely will as much as 1.90$ be available from the operating console. Since rods containing a neutron poison will be used to partially compensate the slow temperature coefficient, the necessity for large amounts of available excess reactivity at room temperature is removed.

Only with ramp addition rates greater than about 25¢/sec, which is six times the normal reactivity insertion rate, could the entire 1.90$ of excess reactivity postulated be added before the safety rods began to terminate the excursions. In these cases the shortest period was 163 milli-sec, independent of

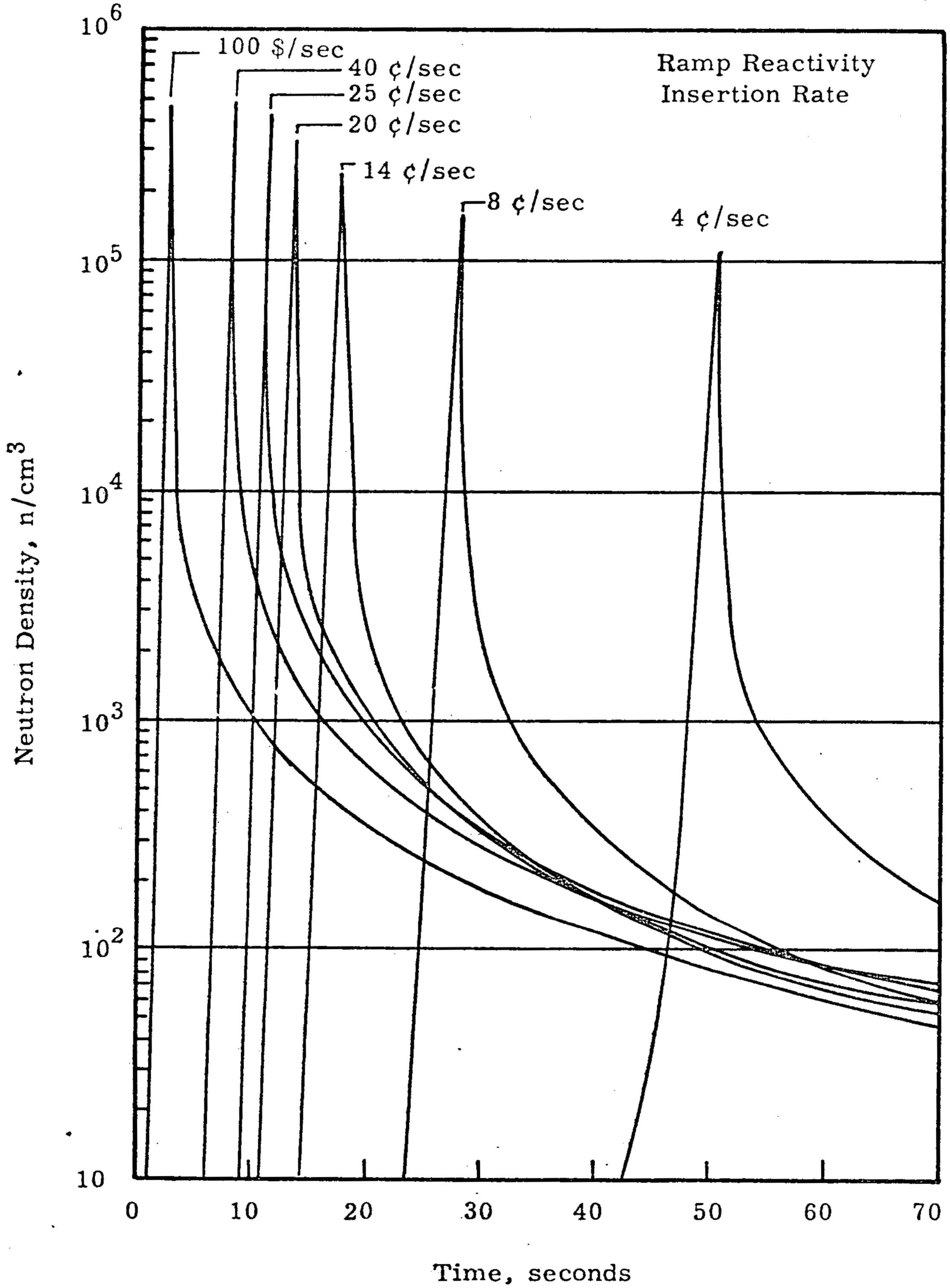

FIGURE XI-1

Power Excursions Terminated by Scram

the ramp rate up to 100$/sec, which rate was used to simulate a step input. For ramp rates slower than 25¢/sec, when the entire 1.90$ could not be added, the shortest period was greater than 163 milli-sec.

The variation with rate of reactivity insertion of several parameters of interest is shown in Figures XI-2 and XI-3. The assumed high flux level trip point was 3 kW. The variation of these parameters with trip point with a 4¢/sec ramp reactivity input is shown in Table XI-I.

The results of calculations similar to those shown in Figures XI-2 and XI-3 but for the case in which delayed neutron fractions for plutonium were used are shown in Table XI-II. The data show that even in this case, the neutron kinetics of HTLTR are favorably slow.

Although the low-enrichment fuel has a prompt negative temperature coefficient which is capable of terminating a reactor transient, this is not relied on for the safety of the reactor. It is expected that the safety circuit and control and safety rods will never fail to shut the reactor down in the event of an inadvertent supercritical condition. The effect of this negative Doppler coefficient on the termination of excursions is shown in Table XI-III and in Appendix A, assuming the absence of the control and safety rods.

The response of the reactor to a change in temperature can be predicted from the nuclear parameters quoted in Section V-K. With gadolinium shim rods worth 1.5% $\Delta k/k$ at 1000 C, the excess reactivity needed from the console at room temperature is about 90¢, while still leaving excess reactivity of about 50¢ at operating temperature. This is sufficient to give a period in the 20 to 90 sec range after removal of the central cell by the oscillator. The precision of the experimental data is expected to be best for periods in this range.

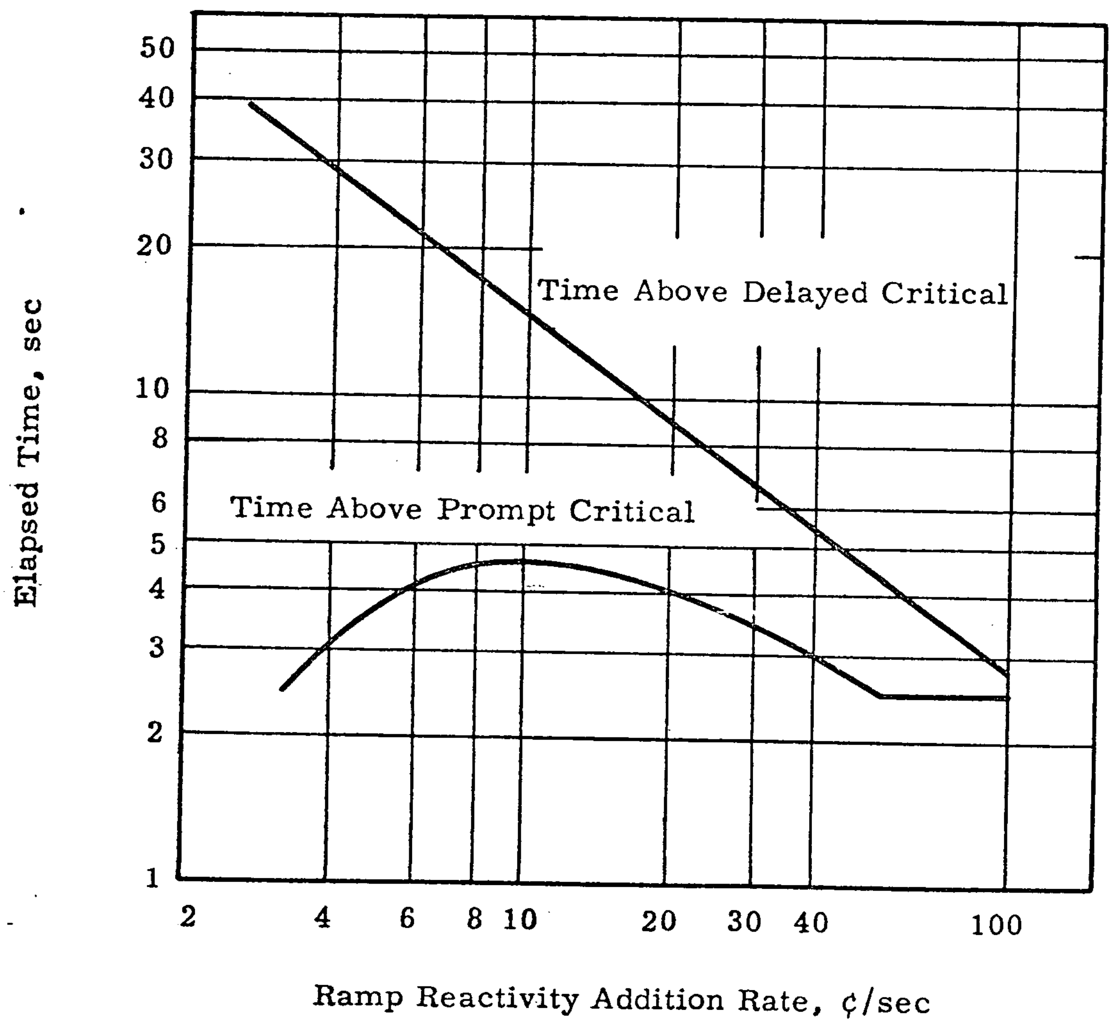

FIGURE XI-2

Response of HTLTR to Various Rates of Reactivity Input; Time Above Critical

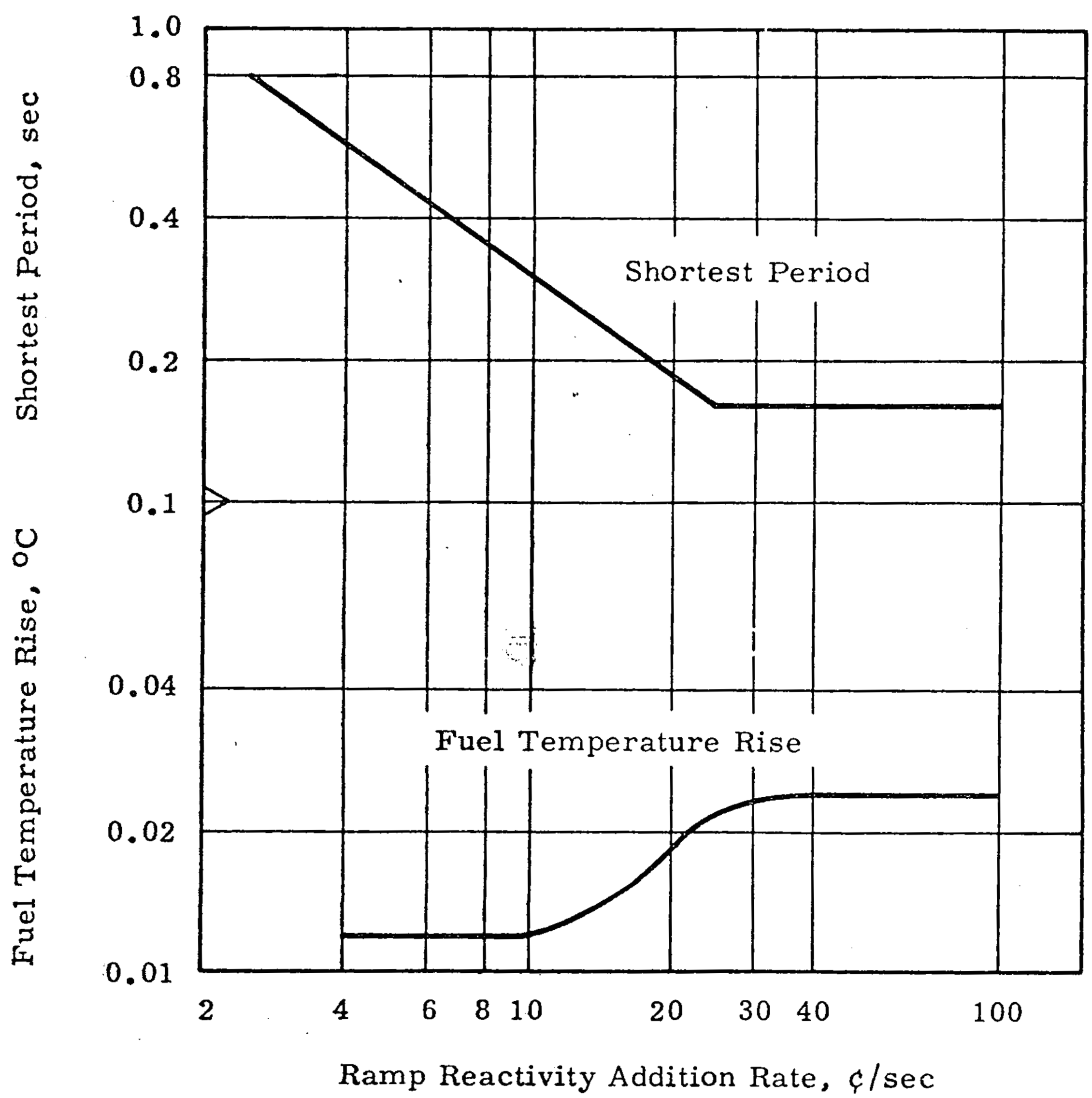

FIGURE XI-3

Response of HTLTR to Various Rates of Reactivity Input; Period and Fuel Temperature

TABLE XI-I

RESPONSE OF HTLTR TO A 4¢/SEC REACTIVITY INPUT

VARIATION WITH HIGH FLUX LEVEL TRIP POINT

Trip Point, kW	3	30	300	0.100
Maximum Neutron Density, n/cm^3	1.09×10^5	1.17×10^6	1.26×10^7	3.24×10^3
Fuel Temperature Rise, C	0.012	0.115	1.14	0.00
Time Above Delayed Critical, sec	28.30	29.56	30.67	26.05
Time Above Prompt Critical, sec	3.16	4.43	5.55	0.86
Shortest Period, sec	0.58	0.51	0.46	0.74

TABLE XI-II

RESPONSE OF HTLTR TO VARIOUS RATES OF REACTIVITY INPUT
PLUTONIUM FUEL KINETICS CONTROLLING

Ramp Rate of Reactivity Addition, ¢/sec	4	14	25	40	100
Maximum Neutron Density, n/cm^3	1.01×10^5	1.35×10^5	1.35×10^5	1.35×10^5	1.35×10^5
Fuel Temperature Rise, C	0.036	0.035	0.035	0.035	0.035
Time Above Delayed Critical, sec	37.78	15.95	11.88	10.11	7.17
Time Above Prompt Critical, sec	12.68	8.72	7.79	7.52	7.07
Shortest Period, sec	0.792	0.487	0.485	0.480	0.190

TABLE XI-III

INHERENT SHUTDOWN CAPABILITY
OF THE HIGH TEMPERATURE LATTICE TEST REACTOR
BY PROMPT NEGATIVE FUEL TEMPERTURE COEFFICIENT

Computer Case Number[§]	20	11	12	13	14	21	15-16	1
Fuel Enrichment, percent uranium-235	5	5	5	5	5	5	5	5
Number of Drivers	32	80	80	80	80	32	80-32	80
Initial Reactor Conditions:								
Temperature, °C	20	20	20	20	20	1000	1000	1000
Reactivity, $	- 0.90	- 0.90	- 0.90	- 0.90	- 0.90	- 0.90	- 0.90	0.00*
Available Excess Reactivity, $	1.00	1.90	1.50	1.00	0.50	0.50	0.25	0.50
Maximum Reactivity Added, $	1.00	1.25†	1.25†	1.00	0.50	0.50	0.25	0.50
Shortest Period, sec	0.99	0.46	0.46	0.85	6.6	7.2	20.6	6.7
Elapsed Time Super-critical, sec	45	39	38	47	105	99	> 500	106
Elapsed Time Prompt Critical, sec	8	9	7.5	8	0	0	0	0

§These cases are illustrated graphically in Appendix A.

*Initial power level 2 kW

†Power rise terminated by prompt negative fuel temperature coefficient before all available reactivity could be added at the 4¢/sec rate.

TABLE XI-III (Continued)

INHERENT SHUTDOWN CAPABILITY
OF THE HIGH TEMPERATURE LATTICE TEST REACTOR
BY PROMPT NEGATIVE FUEL TEMPERATURE COEFFICIENT

Computer Case Number§	20	11	12	13	14	21	15-16	1
Maximum Power Level, MW	36	1831	183	47	4.3	6.0	-*	8.2
Final Fuel Temperature, °C	4050	3820	3120	1870	880	3650	1000	2400

Input data to TRIP[3] code for the above calculations:

Total delayed neutron fraction = 0.0064 (uranium-235)
Prompt neutron lifetime = 0.001 sec (graphite)
Ramp rate of reactivity input = 4¢/sec
Fuel temperature coefficient of reactivity = $-0.05945\$/K^{1/2}$ (80 drivers)

= $-0.03448\$/K^{1/2}$ (32 drivers)

Specific heat of oxide fuel = 0.072 cal/(g)(C)

§These cases are illustrated graphically in Appendix A.

*Power rise not complete at 350 sec, actual power reached ∿ 2 kW.

Cooling of the core by the circulating nitrogen gas system is so slow as to add reactivity at a negligible rate, as was shown in Section V-L. Direct exposure of the core to the atmosphere by some accident could not cause more rapid cooling. Although the most rapid cooling of the core would be caused by the introduction of large amounts of water into the circulating gas stream, the pipe size and hardware available are too small to add reactivity at a rate of 4¢/second even with the worst assumptions on the size of the temperature coefficient. (See Appendix B).

Calculations show that the replacement of the nitrogen by water in the driver region of the cold reactor causes a decrease in reactivity. Thus the shut-down reactor would not become critical from flooding with water.

The examples of nuclear excursions given above are hypothetical. For a nuclear excursion to occur in the HTLTR, standard operational procedures would have to be violated, multiple control circuits would have to fail simultaneously, and the reactor operators would have to be assumed to have their attention elsewhere.

Operational procedures call for loading the reactor by increments with neutron multiplication measurements after each increment of fuel is loaded, so that the critical loading and the amount of fuel for the necessary excess reactivity can be predicted. The strength of the control rods is determined for each new loading before the reactor is taken critical to ensure that the strength of the system is sufficient for complete control of the final loading. Final approaches to critical are made at conservative rates and eventually by withdrawing one rod at a time, so that criticality is approached slowly.

During reactor operation, manipulations with the oscillator are designed to give positive periods in the range 20 to 90 sec. The power rise on periods as long as these is easily terminated by manually closing the control rods after the measurement of the period is complete.

If foils are to be irradiated or a neutron spectrum determined with the chopper, the reactor power level is brought from critical to the desired level (from a few tens of watts to possibly 2 kW) on a safe and acceptable period. It is held level for the desired time; then the reactor is shut down by closing the control rods.

In short, inadvertent nuclear excursions are not expected to occur in HTLTR. But if excursions do occur, the control of reactivity additions would make them inconsequential; and they are inconsequential even under the assumptions of gross and multiple failures of the control circuits and the operators.

D. Mechanical and Utility Failures

1. Leakage of Water into the Gas System

The leakage of water into the gas system at rates so high that liquid water is carried into the heated reactor core has been precluded by design, i.e., the size and the type of heat exchangers, the type of blower and the barrier presented by the filter. This section considers leakage rates such that all liquid water is vaporized before it reaches the reactor core.

In the event of slow leakage at either heat exchanger, the moisture would be vaporized by the gas and piping and be detected by the monitors downstream of the heat exchangers. If these detectors should fail, the carbon monoxide and hydrogen from the water-gas reaction w/graphite would be detected by the chromatographs. An alarm would be sounded in either case. The reactor would be shut down and the gas circulating blower would be turned off. The electrical heaters would be de-energized and the reactor would be allowed to cool gradually.

A gross failure could occur in either the main heat exchanger or the rod cooling exchanger. In the former, water would flow into the gas stream at a maximum differential pressure of 2 psig. The water entering the hot gas would be quickly vaporized; any liquid water carried by the gas would be stopped at the filter. The heat exchangers are so designed (water-cooled inlet gas channels) that there is little possibility for heat storage in the metal of the hot end of the exchangers. Therefore, a simple thermal analysis by the method of mixtures is possible, from which it is shown (Appendix B) that a reduction in system pressure and not a pressure surge would occur.

In the event of a gross failure of the rod-cooling exchanger, liquid water would enter the exit gas stream at 40 C. The velocity in the exit gas line is 7.0 ft/sec. The liquid water carried at this velocity would be removed by the water vapor removal system, (or if it were bypassed by a water separator in the line,) which has a retention capacity of 25 gallons of liquid. Because of the low available head tending to drive water into the gas stream, water from any likely leak, such as complete severance of an exchanger tube, would be simply accumulated during the response time of the moisture monitor and the shut-down of the gas blower.

2. Leakage of Water Directly into the Core

Liquid water leaking directly into the core would cool the core at the point of introduction of the water. For each mole of water entering the core there would be formed a mole each of hydrogen and carbon monoxide by the water-gas w/ graphite

reaction. The pound molal volume at maximum temperature conditions is $(359)\left(\frac{1273}{273}\right)\left(\frac{15}{17}\right) = 1477\ ft^3$. The rate of gas formation is therefore $(2)(1477)/18 = 164\ ft^3/lb$ water entering the core. The maximum purge flow rate (and nitrogen makeup rate) is only 60 ft^3/min. Pressure increases in the gas system would therefore result from any influx of liquid water into the reactor core at a rate greater than 0.4 lb/min. Pressure surges in the reactor shell could occur only if the rate of water influx were sufficient to increase the pressure drop in the large, direct pipe line between the reactor and the main heat exchanger, the filter, and the blower intake line. Since during cooldown this line carries up to nearly 400 moles/hr with a pressure rise of only 0.7 psi, it follows that water influx rates greater than 150 moles/hr or 45 lb/min would be required to cause a noticeable pressure surge in the reactor shell. Flow of gases in such a path is approximately proportional to the square root of the pressure drop. A water influx rate of about 140 lb/min would be required to cause a pressure surge of 5 psi in the reactor shell, its design pressure. The only points at which water could enter the reactor core are via the rod-cooling gas circuit into the VSR channels (filling the water separator) and via leakage from the copper-to-graphite heater connections, which are water cooled. Neither of these channels of water influx is capable of admitting as much as 140 lb/min of liquid water to the core even under wide open failure.

3. Breach of the Reactor Gas Envelope

The reactor gas envelope would not be expected to fail from internal or external pressures resulting from any foreseen abnormal conditions. The envelope is designed to withstand a vacuum and 5 psig internal pressure. The admission of liquid nitrogen into the recirculating gas stream has been made impossible by the design of the system, as described in Section IV F. The anticipated maximum internal pressures will not exceed 2 to 3 psig. There is no mechanism whereby failures of process instrumentation or control valves could cause large and rapid pressure increases, since the nitrogen makeup and purge lines are small and are connected to a system having a large free volume.

A sudden lowering of the nitrogen pressure in the reactor from its normal maximum of about 3 psig to atmospheric pressure, from whatever cause, would add 10 cents or less of reactivity in a step or fast ramp. If the reactor were supercritical on a 20 sec period before the step, the new period would not be less than 10 seconds, and easily controlled by the safety systems.

A large rupture of the reactor envelope could result only from an unusual accident in which violent damage might be done to one of the shell penetrations by an operator mishandling a crane or fork-lift. In such an event, since the pressure in the reactor envelope is normally kept slightly positive relative to the reactor enclosure pressure, there would be an initial surge of nitrogen out of the breach. If the breach were so large that the nitrogen supply at full capacity could not maintain the gas system pressure, air could enter the breach by convection caused by turbulence. This would cause oxidation of the graphite in the local area of the reflector adjacent to the break. The oxygen and carbon monoxide monitors would soon respond and the gas circulating blower and the heaters would be turned off. (It is presumed that the reactor would be at temperature but not operating, since if it were operating there would be no personnel in the reactor room to cause the postulated accident.) The rate of heat production from the burning graphite under this condition would be very low compared to the normal production of heat by the electrical heaters. Personnel would have access to the reactor shell and could make temporary repairs which would halt the reaction. No damage to fuel, instrumentation, shutdown devices or heaters would be anticipated. All damage would be confined to the reflector graphite, and would probably not require repair.

4. Breaches of the Gas Circulation System

A breach of the gas circulation system under normal operating conditions would permit a leak of nitrogen into the reactor room basement, since the system pressure is normally slightly positive relative to the reactor enclosure pressure at all points. The pressure control would compensate for this by the admission of more nitrogen from storage, up to the maximum rate available from the system, about 60 cfm. In the event of a large breach near the intake of the circulating blower, air would eventually be drawn in and would flow to all of the control rod and VSR drive enclosures and thence to the reactor. The oxygen and carbon monoxide monitors would act to shut down the circulating blower within 15 minutes at the longest, and the reaction would soon stop. The reactor would be allowed to cool down slowly without gas circulation until temporary repairs to the system could be made, the system purged, and the reactor cooled by normal circulation.

After such an incident it would be only prudent to unload fuel from the reactor, remove all control and safety rods, and to inspect all components to determine whether repairs were required.

Breach of the gas system would not be hazardous to employees in the reactor enclosure, because the nitrogen escaping would be removed quickly by the ventilation exhaust.

5. Failure of the Gas Blower

On failure of the gas blower, the reactor would be shut down manually, and the power to the heaters would be turned off. Reliable monitoring of the gas system purity would be lost, but no damage would occur. The temperature of the reactor would drop slowly by leakage of heat through the insulation, requiring several days to cool from maximum temperature to 260 C.

6. Failure of the Ventilation System

If the ventilation system were to fail, the air in the reactor room would slowly rise in temperature because of the leakage of heat from the reactor shell. The efficiency of the insulating material is such that at reactor temperatures of 1000 C, the temperature of the reactor room air under stagnant conditions would not exceed the design limit for the reactor flux instrumentation for about 40 minutes. To avoid the necessity for constant attendance of the reactor when the core is heated, an auxiliary gasoline engine drive for the ventilation exhaust fan has been provided. The engine is connected to the drive by an over-running clutch. The engine will be started and left running at all times when the core temperature is above 750 C and the reactor unattended. In the event of failure of the electric motor, the gasoline engine will then drive the exhaust fan to provide uninterruped ventilating air flow.

7. Loss of Nitrogen Supply

On failure of the nitrogen supply, the gas system pressure would drop to atmospheric at a rate depending on the leakage of the gas system. If the loss were prolonged, and if the system temperature were being lowered, intake of air by diffusion would lead to very slow oxidation of the graphite. Assuming that the reactor case and gas system remain intact, reaction would be negligible compared to the incidents previously described.

If the nitrogen supply were to be lost concurrently with a blower failure or input leak, temporary repairs would be made as described under Breaches of the Gas Circulation System.

8. Electric Power Interruption

On interruption of the electric power supply, the reactor would be shut down by failure of the power supply for the rod holding magnets. The heaters, the gas circulation blowers, and the ventilation supply and exhaust fans would not operate. The reactor would cool, as described above. Emergency cooling of the reactor room would be provided by the auxiliary gasoline engine driving the exhaust fan. Reactor instrumentation would not be damaged, nor would dehydration or thermal stress damage to the structural concrete occur.

The reactor instrumentation would cease to respond, but this would present no hazard in the shut-down reactor. No permanent damage to the instruments or to the PMACS would occur.

A long power outage could result in loss of the instrument air supply. All valves and dampers would already have moved by spring action to their safe positions, so no adverse effect on the reactor or building would result.

Power outages at 300 Area are very infrequent, and only one long unscheduled power outage has occurred in 15 years.

9. Service Water Supply Interruption

On failure of the water supply to the heat exchangers, the reactor would be shut down manually, and the power to the electric heaters and the gas blower would be turned off. The heat exchangers would be undamaged if the blower was turned off before the water could boil out of the gas inlet channel jackets. The heat exchangers will be protected by operating procedures.

The copper-to-graphite heater power connections would heat up fairly quickly and the water would boil out of the cooling tubes. The joints would then fail. The copper bus bars would have to be replaced after the incident.

See Design Review

If the interruption to the water supply were to last more than about an hour, structural damage to the water-cooled floor under the reactor would be expected. Loss of structural strength and collapse of the base under the reactor would likely not occur for several hours. However, no unscheduled interruptions have occurred to the normal service water supply at the 300 Area in 20 years. In addition, the 300 Area has extensive capabilities to supply service water during emergencies that occur within the system. An emergency water supply from a pumper truck could be obtained in the unlikely absence of all other supply.

Damage to the control and safety rod drives by heat conducted from the reactor core would occur after about an hour, and these mechanisms would have to be repaired or replaced before the reactor could be returned to service. This would not affect the shut down reactor, however.

No damage to the reactor or building instrumentation would occur.

10. Failures of Reactor Components by High Temperature Chemical Reactions

Materials suitable for reactor use at the operating temperature are few. Graphite, of course, properly used, is quite satisfactory. Impurities in the graphite which could be released at temperatures up to 1000 C and which would be harmful to other materials must be avoided, however.

Metallic materials for the reactor have been tested in small assemblies and in a large mock-up of the reactor containing graphite, insulation material, metal structural parts, a protctype control rod and a vertical safety rod (VSR), thermocouples designed for high temperature service, and other parts prototypical of the reactor design. The mock-up is heated with graphite heater rods of identical construction to those in the reactor. All of the materials discussed have been shown to be capable of withstanding the reactor conditions for several hundred hours. A 1000 hr test at 1000 C has been completed and the results are being reviewed. This is comparable to the time that the reactor is expected to be above 800 C over a period of one or two years. Materials which must last for the 10 year lifetime of the reactor will have been given a sufficiently large fraction of a lifetime test to allow prediction of lifetime integrity with some assurance. The steel shell of the reactor and its external appurtenances are, of course, not heated to the operating temperature. The design temperature of the steel shell is 175 F for safety of the operating personnel and for the protection of the instruments. The graphite core is entirely free-standing and self supporting. The firebrick insulation is suspended from the steel shell by Hastelloy-B bars. The design is to be such that the bars will not experience temperatures in excess of 860 C. The tests will have shown that these important structural members will perform without failure before installation of the insulating material starts. Only minor surface effects have been noted in specimens examined by metallography[4] after several hundred hours exposure to dry nitrogen and in contact with insulating materials and graphite in a small assembly. These tests were run at 1000 C and 1200 C. In the large mockup, where water is present as a contaminant in the nitrogen atmosphere, corrosion has been noted in samples of Inconel and Hastelloy-B placed at locations in the brick insulation only where the temperature was above 800 C.

If several hangers were to fail and release a section of the insulating brick above the gas plenum at the rear face of the reactor, some, but not all of the control rod sheaths would probably be broken. Some of these could not return to their shutdown position. If several hangers were to fail adjacent to a VSR position, it would possibly not drop. However, the shutdown margin of the two control systems is so large that multiple failures could occur without loss of shutdown capability. (See page 5.23.)

E. Instrument Failures

1. Safety Circuit

Safety system failures have been discussed by J. E. Binns[7] as falling into two categories, which he simply calls failures of the first kind and failures of the second kind. Failures of the first kind render the system incapable of taking appropriate action should action be necessary. Failures of the second kind produce the same action, as a result of the failures, that would be taken following an actual unsafe process condition. The HTLTR safety system, which is diagrammed in Figure XI-4, is designed to avoid failures of the first kind.

To start up and operate the reactor, the vertical safety rods (described above) are withdrawn from the reactor by drives incorporating electromagnetic clutches. Power is delivered to the clutches only if the power amplifier is energized through a key lock switch at the operating console and thereafter by two 1000 pulses per second system-normal signals, internally supplied through

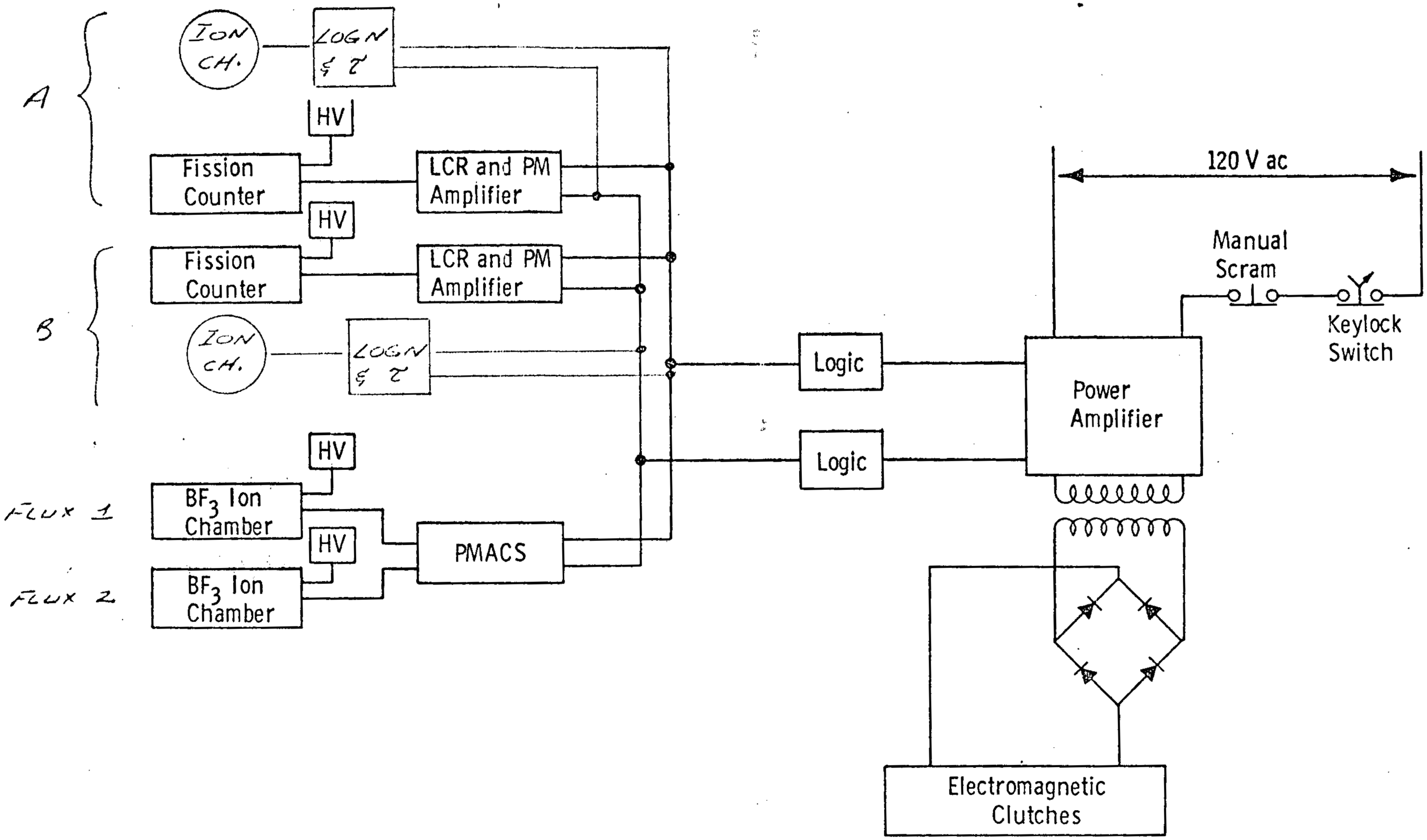

FIGURE XI-4

Safety Circuit Block Diagram

the two safety logic circuits. These system-normal signals energize and de-energize the power amplifier 1000 times per second to produce a-c power at 28 volts and 1000 cycles per second. This power is inductively coupled to a rectifier to produce the direct current power supplied to the electromagnetic clutches. If either system-normal signal disappears, the power amplifier output at 28 volts is lost and the clutches release the rods.

Each logic circuit stops the system-normal signal from passing through it whenever an out-of-limit condition occurs. Although the logic circuit functions are identical, their designs are different. One logic circuit recognizes an out-of-limit condition as the presence of a signal, the other as the absence of a signal. Thus not only are the logic circuits active rather than passive, furnishing the oscillation drive at 1000 H_z for the power amplifier, they are redundant functionally and diverse in design, which provides maximum reliability as shown by Binns. The logic circuitry is automatically locked out for any single system-normal signal loss and both logic circuits must be reset simultaneously before power may be applied to the electromagnetic clutches.

Each logic circuit receives its appropriate signal from [a] two log count rate and period channels [b] from PMACS and [c] from manual shutdown switches. Each log count rate and period channel includes a fission counter, high voltage power supply and signal conditioning circuits, which present in-limit or out-of-limit information for flux level, period and on-scale indication to each logic circuit. One of these channels may be bypassed at any time for maintenance or testing. Each on-scale trip, flux-level trip and period trip indicates either an internal system or component failure or a valid flux level trip.

To provide an entirely different method of monitoring neutron flux level, which would protect against the simultaneous failure of the two log count rate channels, signals from two banks of BF_3 ion chambers are monitored by PMACS. Each tenth of a second PMACS is programmed to read and compare the signals from the two banks of ion chambers with set points for low and high flux levels and a period established within the computer. If a limit is exceeded in either channel, PMACS fails to send a pulse to the safety circuit. If the pulse indicating that the nuclear data from the ion chambers have been monitored and are within limits does not reach the safety circuit every tenth of a second, the logic circuit stops the system-normal signal, shutting down the reactor. Both logic channels receive two signals from the PMACS which originate at the ion chambers.

Note that the fission chambers and log count rate meters described above are directly coupled to the safety logic circuits, and thus are independent of PMACS.

The HTLTR is protected against a complete failure of neutron flux monitoring capability by having two separate, independent systems of entirely different design each of which contains two channels. These are followed by two active, separate logic circuits of similar design, each capable of initiating a reactor shutdown. Power is inductively coupled to the electromagnetic clutches. No relay contact nor solid state switches are used, eliminating the consequences of shorted contacts and shorted or grounded wires.

The dual monitoring of switch contacts by the logic circuits is extended to each of the three manual scram buttons located in the control room, reactor hall and reactor basement.

A detailed analysis of reliability incorporated in the systems appears in Appendix F.

2. Other Instrument Failures

Reactor Temperatures

As described in Section VII-D, there are two thermocouple systems in the reactor, one to limit the heater rod temperatures to below the desired limit and the other to control the core temperature to the desired set point. Both systems provide measurements from several places. The temperature readings are correlated with the energy input to the electric heaters by the PMACS. The thermocouples proposed are chromel/alumel and platinum/platinum-rhodium, sheathed in Inconel. They have been shown to be relatively free from failure over the period of an extended run in the large test assembly.

The temperature of each heater rod normally is monitored with two thermocouples, one of which is a spare. A failure of the active thermocouple would be signaled by PMACS to the operator for replacement. Essentially continuous temperature monitoring of all heater rods is provided.

Similarly, a failure of one or more of the core temperature thermocouples would not significantly affect the resultant measurement. Initially, the reactor will be sectioned into four zones with several thermocouples providing the average for each zone. PMACS will automatically ignore any failed element, signal the failure, and compute the average from among those remaining.

A check on the validity of the rod temperature measurements is possible by comparing the electrical power input as determined by the watt-meter and power as computed by PMACS using the active thermocouple on each rod. When wanted, a comparative check between the two thermocouple systems is also available.

Thermocouple failures would not cause a hazardous condition. The reactor operators could decide that the data being obtained was sufficiently precise and to continue the run, or that the experiment was of no value and to shut down for repairs.

Gas Analysis System

The nitrogen system will be sampled and analyzed at two points--the recirculating gas stream and the purge stream-- for contaminants. The analysis is made by gas chromatographs, which analyze for the various contaminants serially. The chromatographic analysis for hydrogen is a back-up to the moisture monitors, since detectable amounts of hydrogen in the system could arise only from the water-gas reaction with graphite. The chromatographic analyses for oxygen and carbon monoxide are redundant also, since oxygen is quickly converted to carbon monoxide by reaction with graphite. The chromatographic analysis for carbon monoxide backs up the moisture monitors.

Failure of a moisture monitor would lead to delayed knowledge of moisture in the system, because the response of the chromatographs is relatively slow. This would not be likely to result in hazard to the reactor, however, as some reaction of steam and graphite can be tolerated. The likelihood of hydrogen and oxygen ever being simultaneously present in the explosive range is nil, since the explosive range for these gases is greatly narrowed by the presence of nitrogen, and oxygen would be continuously removed by reaction with graphite.

Gas System Controls

Failure of the gas system controls could result in over- or under-pressurization of the system, reduced circulation rates (excessive circulation rates are not possible because of the limited capacity of the blower), or reduced or excessive purge flow rates. None of these failures could cause any serious hazards. The gas system is protected against over-pressure by a rupture disc. Under-pressure could lead to abnormal infiltration of air, but after shut-down of the reactor and the blower no damage would be done. Reduced circulation rate would simply cause slow cool-down of the reactor. Reduced purge rates would cause build-up of contaminants in the gas stream, but this is not hazardous of itself nor would it cause other hazards. The reactor and blower would be shut down until the condition could be corrected. Excessive purge rates would cause no difficulty.

Failures of any of these controls would be quickly detected by the response of other instrumentation, and repairs could be made in most cases without the necessity of cooling down the reactor.

3. Effects of High Temperature on Instruments

The ion and fission chambers, which measure the neutron flux, require maintenance of a high voltage between an electrode and the chamber shell for their operation. At temperatures above 200 F, the electrical insulation which maintains the high voltage could deteriorate. It is necessary, therefore, to keep these sensors reasonably cool. This is done in HTLTR during loss of normal cooling from nitrogen by the recirculating ventilation air. It is partially because of this that the auxiliary gasoline engine drive has been provided for the ventilation exhaust fan.

F. Escape of Contamination from the Fuel

1. Particulate Alpha Contamination

It is estimated that the 1200 kilograms of driver fuel may contain as much as 20 milligrams of plutonium after 10 years use in the HTLTR. The evaporation of plutonium from the uranium oxide fuel is expected to be so small as to be undetectable, even at 1000 C with vented fuel jackets.

The most serious potential for contamination in the HTLTR occurs when experimental plutonium fuel is in the test core. The test core may contain up to 6 kg of plutonium in any of various compounds or mixtures suitable for use at high temperatures. Such fuel would not be volatile at 1000 C, so movement of plutonium into the gas stream could only be in the form of the original (or some subdivided) particles. Contamination of all inside surfaces of the reactor core and the recirculating gas stream can be assumed if a fuel element failed, in spite of the presence of the filter upstream of the blower. The contamination would be detected by the alpha particulate monitor on the gas system purge stream, the only path by which the contamination might be inadvertently released.

The purge stream passes through dual high efficiency filters. These have a rating of 99.7% retention of 3 micron DOP aerosol particles. The number fraction of 3 micron and smaller particles would be extremely low for any fuel in HTLTR cores. In addition, the weight fraction would be even smaller. The weight retention of any plutonium reaching the purge filters will therefore closely approach 100%. The contamination in the ventilation exhaust flow would be negligible.

Leakage of gas from the reactor shell and gas system would carry such contamination to the reactor room and basement. With the reactor room held at a slight negative pressure by the ventilation system, the contamination would be confined to this space. The exhaust from this area is routed through high efficiency filters also. Thus, escape of plutonium from the facility is not expected. In order to assure the operator and public, a continuous alpha monitor is installed in the nitrogen exhaust stream upstream of two absolute filters. In the unlikely event of an out-of-limits release signal, further emergency action, such as stopping the gas purge flow, shutdown of the ventilation equipment, and temporary sealing of the reactor room penetrations, could be taken. In any event, emission of plutonium in hazardous quantities would be prevented. Plutonium escape from a fuel element while the reactor is open or while the fuel element is being handled will be minimized according to prudent practice already long established. The emergency measures which could be taken for confinement of contamination and for decontamination are similar to those described above.

The plutonium contained in the typical fuel element, if scattered through the reactor and gas, would constitute a diffuse source of about 4 curies of low energy gamma radiation. The reactor room walls are completely adequate for protection against such radiation levels. Employees would, of course, encounter some special problems in the decontamination campaign which would follow, but the direct consequences to the public, and also to employees, would be negligible.

2. Iodine-131 and Other Iodine Isotopes

At the maximum operating temperature of the reactor, iodine is quite volatile, so that any iodine released from the fuel would be carried out of the reactor in the hot gas stream. At least some of the iodine would plate out in the heat exchangers and cool piping, but this fraction cannot be closely estimated. Many of the radioiodine isotopes have short half-lives and do not give serious problems when emitted from a facility. After 10 days, only I-131 and I-132 are present in significant amounts.

Iodine released in an environment of hot graphite may or may not retain its elemental character. Iodine released from fuel has been reported in at least three forms: elemental iodine, organic compounds typified by methyl iodide, and unknown other species, perhaps iodine adsorbed on particulate matter, and usually constituting not more than 1-2% of the total.

The driver fuel for HTLTR will be jacketed in graphite. Some of the noble gases and iodine formed by fissions will therefore escape from the fuel element and enter the circulating gas stream. The effects of such release are considered in the following paragraphs.

The fuel, if in the form of particles, will be no smaller than 1 mm diameter spheres. The diffusion of iodine from UO_2 is given by

$$f = 3.4 \times 10^{-4} \times 10^{-5000/T} \times t^{1/2}/a$$

f = fraction of iodine formed which is released

T = UO_2 temperature, °K

t = time, sec

a = radius of spheres, cm

The amount of iodine formed is calculated for two cases: operation of the reactor at maximum power for experimental runs of maximum duration, and an abnormal excursion in power which is self-terminated. These are absolute maxima in both instances. The highest fission product inventory is accumulated during a chopper run with reactor operation at 2 kW extending over 8 hr. This run of 16 kW-hr generates 1.5 Ci of I-131. Since these highest level runs are infrequent, it is expected that an upper limit of 2 Ci of I-131 in the fuel at any time will be reasonable to handle. The escape of I-131 is negligible except at the maximum temperature, 1000 C, at which, over a period of one half-life, only 0.07% of the inventory (or 1.4 mCi over an 8 day period) is released. The release rate permissible under radiation protection standards for the 300 Area is 10 mCi per week.

The maximum total amount of iodine isotopes formed in a self-terminated excursion reaching a peak power of 200 MW with a half-width of 4 sec is about 5.5×10^3 Ci. Of this only about 16.5 Ci is I-131; the remaining isotopes are short-lived. The 800 MW-sec energy input would raise the fuel temperature an average 2200 C to a fuel temperature of 3200 C under maximum operating conditions. The rate of escape of radioiodine immediately after the excursion would be 0.11 Ci/min, but this would be nearly all short-lived. By 2 hr after the excursion, the fuel temperature would be reduced to about 1700 C, and the rate of release would be 0.0048 Ci/min. The release of about 0.5 Ci of short-lived iodine isotopes would cause minor contamination problems in the facility and no off-site damage. It should be noted that this is a maximum conceivable accident for radioiodine release to the environment. It is postulated

- that the excursion would not be terminated by an automatic scram
- that the operator failed to respond to the power peak with a manual scram, and
- that all of the radioiodine escaping from the fuel particles also escapes from the circulating gas system.

These considerations have led to the design decision not to provide charcoal traps for halogens in either the purge line or the main recirculation line.

3. Other Radioactive Isotopes

Noble Gases

Ten minutes after the excursion postulated in the previous section, there would be about 15,000 curies of krypton and 40,000 curies of xenon present in the driver fuel. These decay to the following activities (Ci) at the times stated:

	Kr	Xe
1 hr	3,260	2,590
2 hr	2,320	630
5 hr	950	510
10 hr	280	510
1 day	15.2	320
5 days	0.004	41.5
10 days		22.0
20 days		5.9

The daughters of xenon decay are cesium isotopes which are considered in the following section. The daughters of krypton decay are isotopes of rubidium, strontium, yttrium, and zirconium. The activities (Ci) of these radioelements deposited from the noble gas cloud at the times stated are:

	Rb	Sr	Y-95	Zr-95	Nb-95
10 min	1.47×10^4	1.44	2.66×10^4	0.10	0
1 hr	9.87×10^2	4.86	9.80×10^2	5.20	0.004
2 hr	53.3	5.23	18.7	5.30	0.009
5 hr	0.03	5.23	0	5.30	0.024
10 hr		5.23		5.28	0.049
1 day		4.92		5.25	0.118
5 days		4.66		5.03	0.542
20 days		3.79		4.29	1.63
100 days		1.26		1.83	2.41
250 days		0.35		0.37	0.73

3

The strontium contains 0.03 curies of Sr-90; the remainder is Sr-89. The other long-lived isotopes present are Zr-Nb-95. The inhalation dose at one mile from the noble gas cloud would be less than 0.17 rem to the bone from Sr-89 and about 0.10 rem from Sr-90. The dose rate to an exposed individual would be 4.1×10^{-3} rem/yr. The ingestion dose rate to the bone via consumption of milk from cattle fed on land at a one mile radius would be less than 0.16 rem/yr from Sr-89 and about 1.5×10^{-2} rem/yr from Sr-90. The other long-lived isotopes do not make important contributions to the dose. Thus contamination of the environs of HTLTR with the daughters of krypton decay would present no biological hazards, since these doses are a factor of 10 or more below the level at which remedial action is necessary.

Volatile Fission Products Other than Iodine and Noble Gases

Cesium, tellurium, and selenium formed in the fuel will also be volatile at reactor temperatures. The following tabulation shows the activity of each present (Ci) after the postulated excursion at the times stated:

	Cs	Te	Se
10 min	3.03×10^{4}	2.58×10^{4}	1.15×10^{3}
1 hr	6.20×10^{3}	1.04×10^{4}	1.72×10^{2}
2 hr	2.00×10^{3}	4.07×10^{3}	30.7
5 hr	45.8	3.41×10^{2}	0.83
10 hr	0.16	1.10×10^{2}	0.02
1 day	0.08	80.6	
5 days	0.07	30.7	
10 days	0.06	11.6	
20 days	0.05	2.9	

The cesium listings include both that volatilizing as such from the fuel and that formed by xenon decay. The daughters of cesium isotopes which contribute to the activity at 10 hr are Ba-139, Ba-140, and La-140. The daughters of tellurium are iodine isotopes, which were considered in a preceding section. The daughters of selenium are bromine isotopes, of which only Br-83 makes any contribution after 10 hr. The activities of these radio-isotopes (Ci) at the times stated are:

	Ba-139 and 140	La-140	Br-83
1 hr	3.82×10^3	0.85	184
2 hr	2.37×10^3	1.28	150
5 hr	5.49×10^2	2.54	64.8
10 hr	68.6	4.48	15.3
1 day	25.1	8.98	0.27
5 days	20.2	19.4	0
10 days	15.4	17.2	
20 days	9.0	10.3	

It is seen that only the Ba-La chain will contribute significant contamination after one day; these are daughters of cesium.

The rate of cesium escape from the fuel and its subsequent fate first in the hot graphite environment of the reactor core and then in the gas system piping at various temperatures cannot be precisely estimated. Miller, Browning et al[5] have shown that half the cesium transported in dry helium and dry air passes through high-efficiency filters. However, Cottrell, Browning et al[6] have shown that only about 1% of the cesium is released from UO_2 at 1400 C. Therefore in the postulated excursion about 0.5% of the activities listed above will be dispersed in the environs of HTLTR. Essentially all activity would decay within one year. The resulting minor contamination incident would present no biological hazard, on or offsite.

G. Critical Mass (Out-of-Reactor) Considerations

The total fuel inventory for the HTLTR, including drivers, flux-leveling elements, and fuels for the test core, will be more than a minimum (optimally moderated and reflected) critical mass. The storage of this material will be divided into safe masses and placed in safe geometric arrangements by administrative controls and the provision of special storage racks. The limits established in the administrative procedures are based on optimum water moderation, optimum reflection, and spherical geometry.

In setting these limits, a safety factor is applied such that at least two simultaneous, independent accidents must have occurred to assemble a mass of material which could be made critical. However, the assembly of such a mass would still not result in a criticality, because the necessary conditions of optimum moderation and reflection would still not be present. Because of the design of the facility, the accumulation of enough water to immerse the storage arrays is impossible.

All transfers of fissile materials into and out of the facility are made under strict inventory control, so that the number and fissile content of each type of fuel piece on hand are known at all times. Transportation, handling and storage of fuel are done under the terms of Critical Mass Safety Specifications, drawn up by the operators of the reactor and reviewed by Battelle-Northwest specialists on critical arrays. Special racks, which will accommodate only one fuel piece of a given type at a particular rack position, are to be provided for the storage of all but special test fuel shapes. The latter are stored in their carrying cases under rigid limits on closeness of spacing. Fuel elements are required to be in their racks when not in the reactor.

H. Spread of Contamination in Disposal Wastes

The HTLTR has few waste disposal problems. Water from the heat exchangers, though normally not contaminated, is run to a potentially contaminated sewer which discharges to the 300 Area process sewer pond, from which it is lost by evaporation and leaching into the soil. The pond is regularly monitored. Liquid wastes from decontamination procedures after an incident would have to receive special handling, depending on their composition and amount.

As previously described, the purge gas from the reactor and the ventilation exhaust are both passed through high efficiency filters before release from the stack.

As has been pointed out in the description of the ventilation system, the locations and height differentials between the ventilation exhaust from the reactor enclosure and the various ventilation air intakes, coupled with the high exhaust velocity, make recycle of the exhaust into the intake very improbable. The exhaust would also contain significant quantities of fission products only in the event of a large power excursion, also highly improbable.

I. Interaction with Other Facilities

The HTLTR is located at such distances from other facilities containing fissile materials that interactions among them leading to chain reactions are not possible. Spread of radioactivity from one facility into another through the intakes of the ventilation system is possible, but this would not increase the hazard to the public. Because of the distances involved, the probability of this occurrence is small and tolerable. Emergency procedures for the area and for the HTLTR facility will provide an orderly and safe response to any incidents originating in other nearby facilities.

J. Non-Nuclear Incidents

During duty hours, access to the reactor, process and control area is controlled by the operating personnel. At least two qualified persons must be present in the control room for reactor startup and operation. At least one qualified person will be on duty when the reactor is shut down but not deactivated, as previously described (p. 9.10). The building will be kept locked during night shifts and on weekends, and during these times will be connected by an alarm circuit to a neighboring facility which is manned around the clock. The building will also have regular surveillance by patrol.

These precautions will protect the reactor from ordinary sabotage. Even a knowledgeable saboteur would have difficulty in causing a self-destructive nuclear excursion in the reactor, because of the various fail-safe and independent channels.

The reactor and building are not designed to be proof against bombing.

The reactor and building are designed to withstand earthquakes of an intensity up to VII on the Modified Mercalli scale. The graphite bars of the test portions of the core are not keyed to the permanent portion of the core. The reactor envelope and its doors are designed to withstand the shock load of the central part of the core under a force of 2500 lb. Any movement of the central part of the core with respect to the stationary part increases neutron leakage and reduces reactivity. The control and safety rods penetrate only the fixed part of the core, so it is expected that shutdown control will be achieved by the functioning of either of the two systems even in the event of an earthquake. The reactor is not provided with a seismoscope trip; it will be manually scrammed on any evidence of earthquake.

The reactor building is designed to withstand 100 mph winds, which are in excess of any which have been observed in the locality. The area is not subject to winds of tornado violence. The building is located well above the maximum flood level of record, and flood damage is not expected.

K. Maximum Credible Accident (MCA) and Consequences Thereof

The HTLTR fuel will never contain a significant inventory of fission products. The radiological consequences of a self-terminated excursion are negligible. It is therefore certain that this reactor will never cause radiological hazards to the general public.

As discussed above, the procedures, instrumentation, interlocks, audible signals and operator proficiency required to prevent a nuclear excursion in HTLTR have been perfected and demonstrated in eight years operation of a similar reactor, the PCTR. New features of HTLTR have been tested at operating temperatures in the mock-up facility. No nuclear or radiological hazards are anticipated specifically arising from the high temperature operation of HTLTR.

The HTLTR safety circuit is expected to be free from failures of a nature which could result in the occurrence of a nuclear excursion without a response from the safety circuit. The control and safety rod actuators are simple, of familiar design, and regularly tested. Therefore, at least half the control elements are expected to act in any circumstances. For these reasons, all nuclear excursions in HTLTR are expected to be terminated by reactor scram.

Other accidents, mechanical and utility failures and the like, have been discussed above, and none leads to consequences of any significance offsite.

The only accidents, having serious consequences, to employees and property only, which could occur in HTLTR would be caused by multiple violations of procedures and serious errors in judgment. These would have to be concurred in by at least two individuals of the HTLTR staff. While proper training, procedures, and administration are expected to prevent this type of accident, a sequence of events involving a number of improbable actions may be combined to yield an accident which may be thought possible. Sequences for two such accidents are given below. In each case, a serious loading error combined with various equipment failures and violations of procedure are postulated.

Startup Accident

1. The reactor cavity is loaded with a cermet plutonium-fueled test and buffer zone containing no U^{238}.

2. The reactor is shut down for adjustment of the position of some of the poison pieces.

3. The neutron flux high level trips are erroneously left at the 3 kW setting instead of being reset to ~ 100 W. The same error is made in programming the PMACS. This is concurred in by both the operator and the supervisor.

4. Driver fuel pieces are removed, and the poison pieces are removed and relocated. About 2.00$ worth of poison is not replaced as required by the experiment plan. This error is not recognized by the operator or the supervisor and the experimenter-in-charge is not in attendance.

5. The fuel pieces are replaced to the original core configuration without employing an incremental loading procedure. This breach of instructions is acquiesced in by both the operator and the supervisor.

6. The reactor room is closed, the reactor purged with nitrogen, and the safety rods are withdrawn. This leaves the reactor barely subcritical. The larger than normal readings of the power level instruments are ignored. This inattention to the reactor instrumentation is committed by both the operator and the supervisor.

7. The ganged control rods are withdrawn at the full preset rate, adding reactivity to the system at 11.6 cents/sec, due to the presence of the plutonium, instead of the normal 4 cents/sec or the maximum 8 cents/sec. (See footnote). Routine startup checks of core reactivity during control rods withdrawal are not made.

8. The control room instrumentation and the audible monitor show the increase in neutron flux, but continue to be ignored by both the operator and supervisor. The reactor goes critical early in the control rod stroke.

9. The period trips fail. The reactor quickly becomes prompt critical on a period of about 0.27 sec (Figure XI-2). Personnel do not react quickly enough to initiate a manual scram.

10. As the reactor power level reaches 3 kW, the safety circuit is opened by the neutron flux high level trips. After 0.1 sec instrument delay and magnet release time, the safety and control rods begin to move.

11. The reactor power level reaches about 10 kW before the falling rods turn the transient completely around (Figure XI-1).

FOOTNOTE: With the core configuration postulated, about 50% of the fissions would take place in plutonium, instead of about 20% as the core loading was intended to perform. The value of the dollar would be about 4.5 mk instead of 6.5 mk. Therefore, there would be available about (2.80$)(6.5)/(4.5) = 4.05$ excess reactivity, instead of 0.80$ as expected. Also, when control and safety rods are withdrawn, the rate of reactivity addition would be (6.5)(8)/(4.5) = 11.6¢/sec.

12. The fuel temperature rises an average 0.01 C (Figure XI-3). The hot spot rise in the central test cell would be no more than about 0.5 C with the most pessimistic assumptions as to flux bowing, and low heat capacity of the fuel. Since accidents have occurred in the past due to the compounding of a series of improbable errors and violations, this sequence may approach credibility. Even so, the accident would have no consequences in terms of radiation exposure, either at the reactor building or offsite.

Loading Accident

1. The reactor is loaded with a typical test configuration. It is shut down for adjustment of the position of some of the poison pieces.

2. The neutron flux high level trips are left at 3 kW. The same error is made in programming the PMACS. These variations from normal procedure are overlooked by both the operator and supervisor.

3. Driver fuel pieces equivalent to twice the expected change in reactivity are removed, and the poison pieces are removed and relocated. About 4.00$ worth of poison is not returned to the core, as required by the experiment plan. This error remains unnoticed by the operator or the supervisor, and the experimenter-in-charge is not in attendance.

4. The operator proceeds to replace the fuel pieces to the original core configuration without employing an incremental loading procedure. This violation of instructions is concurred in by both the operator and the supervisor.

5. The audible neutron flux monitor is ignored or has failed in such a manner that it is giving a normal audible signal but is not responding to the increasing flux level. This failure would be extremely complex and improbable.

6. The operator loads a driver rod worth 25 cents into the reactor, which was subcritical by a few cents. The reactor goes critical on a period of about 26 sec.

7. In about 210 sec the power level reaches 3 kW and the safety circuit is tripped, releasing the two cocked safety rods and shutting down the reactor. The power level reaches a maximum of about 3.2 kW before turnaround is complete. If the operator remains at the loading face for the full 3-1/2 minutes without adding fuel, he would receive an integrated dose of about 60 rem. Others in the reactor room would receive an equal or smaller dose of radiation. Outside the reactor room the dose is inconsequential. If the operator adds another driver worth 25¢ one minute later or leaves the reactor room within 2-1/2 minutes his dose would be reduced to about 12 rem. A manual reactor scram from the control room at any time would also reduce the exposure.

The above doses were obtained from a computer code, which includes factors for relative biological effect. The dose rate is above 1 rem/hr for about 3 minutes before scramming. The calculated gamma and neutron doses are:

Dose Component	Integrated Dose, rem
Fast Neutrons ($E > 0.3$ Mev)	1.96
Intermediate Neutrons (0.3 Mev $> E >$ 0.4 ev)	0.78
Thermal Neutrons ($E < 0.4$ ev)	53.33
Gamma (All)	3.72
TOTAL	59.79

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APPENDIXES

APPENDIX A

Self-Terminated Excursions

These calculations are intended only to illustrate the inherent self-shutdown capability of HTLTR provided by the Doppler effect in the low enrichment drivers. As stated in the main discussion, it is not expected that any excursion will occur which is not terminated by the safety rods.

In these calculations it was assumed that all of the heat generated was retained in the fuel. This assumption is valid for excursions having a shortest period of under one second, but probably not exactly so for excursions having a period over 10 seconds. The latter excursions would be prolonged because escape of heat from the uranium fuel would reduce the negative reactivity available from the Doppler effect. Only the Doppler coefficient of the driver fuel was considered; no contribution from the test core was assumed. The gadolinium shims would make no contribution in any of these excursions since the neutron temperature is changed negligibly. The results of the calculations are shown in Figs. A-1 to A-8.

In an excursion of this type two mechanisms could prevent proper operation of the control rods:

1) The TD-nickel operating rod melts or fails due to loss of strength
2) Sufficient CO pressure is generated in the UO_2-graphite cylinders through the reaction $UO_2 + 4\,C \rightarrow UC_2 + 2\,CO$ to rupture them.

In the following analysis, the nickel rod is assumed to have sufficient strength to operate the control rods for short periods of time at rod temperatures of 1400 C.

The composition and volume of the control rod sections which contain

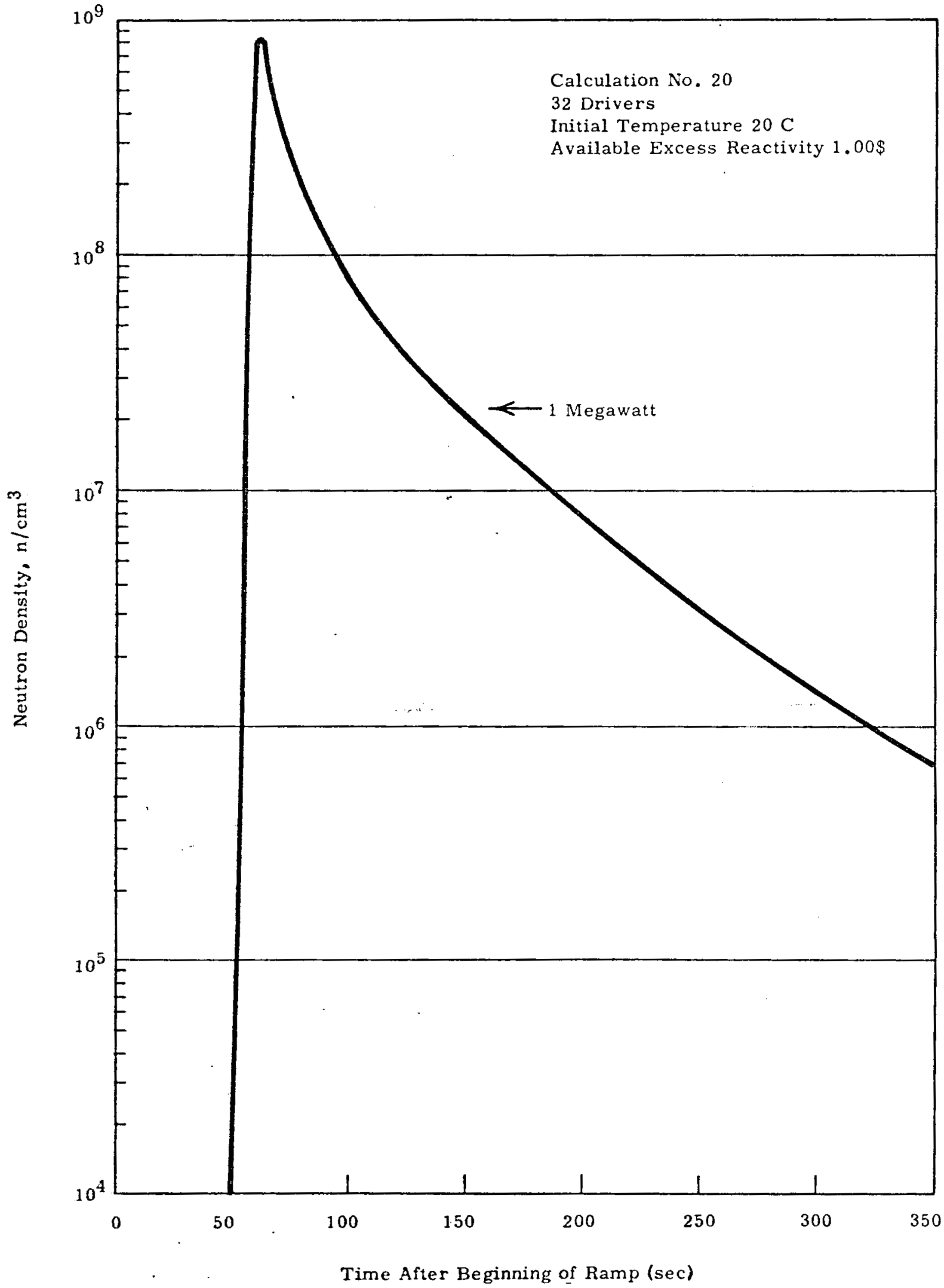


FIGURE A-1a

Flux Level in Self-Terminated Excursion: Calculation No. 20

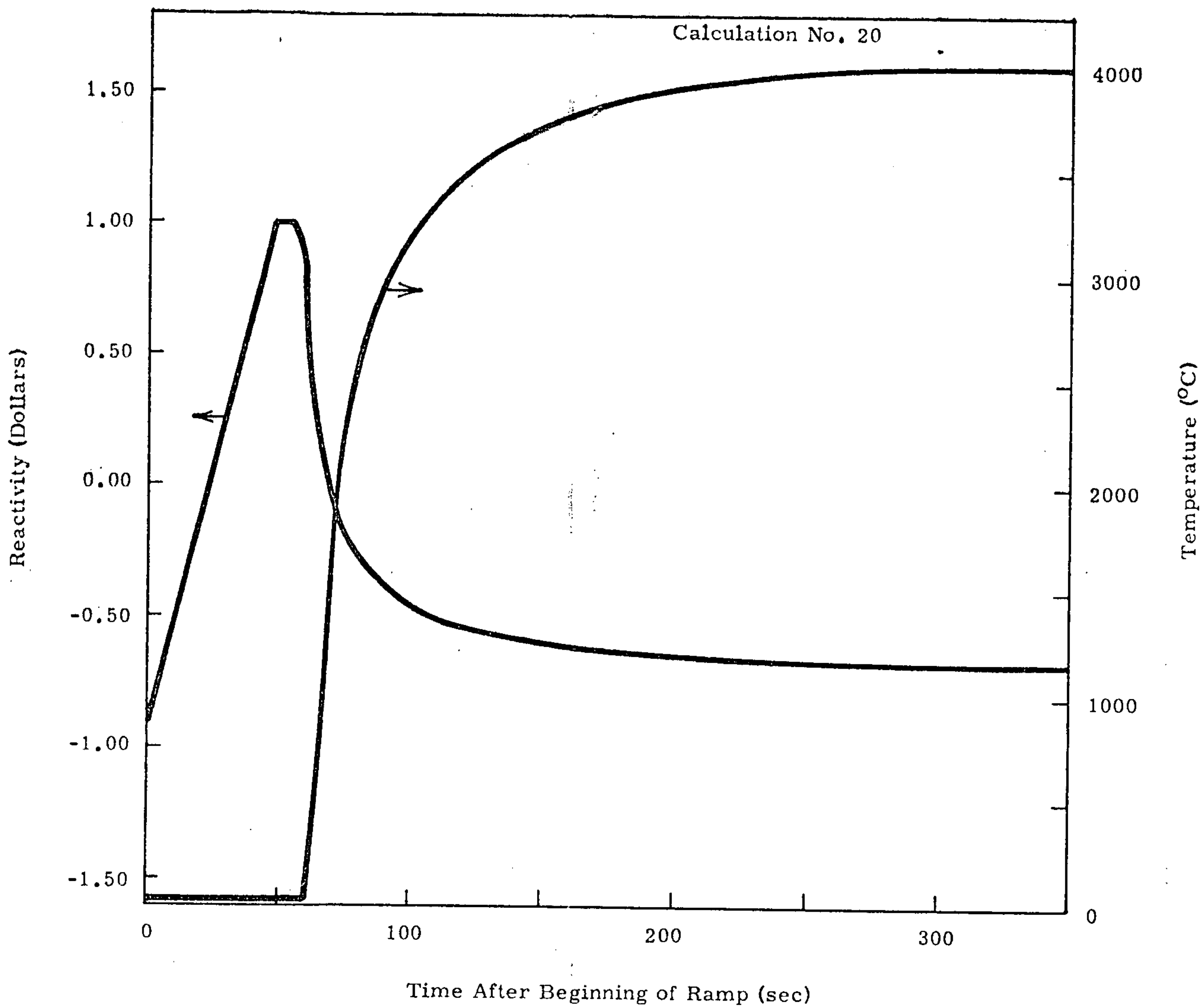


FIGURE A-1b

Reactivity and Fuel Temperature
in Self-Terminated Excursion: Calculation No. 20

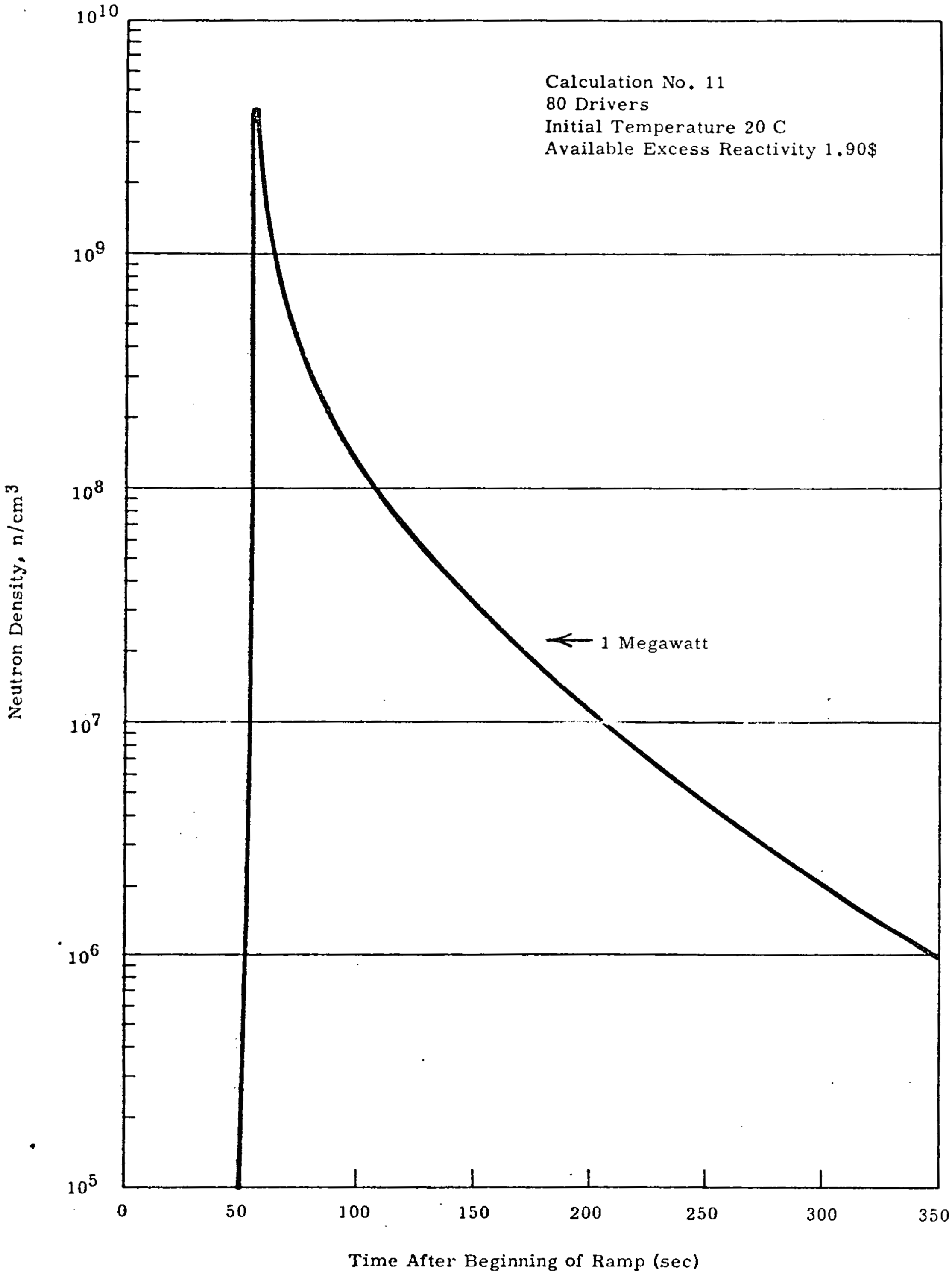


FIGURE A-2a

Flux Level in Self-Terminated
Excursion: Calculation No. 11

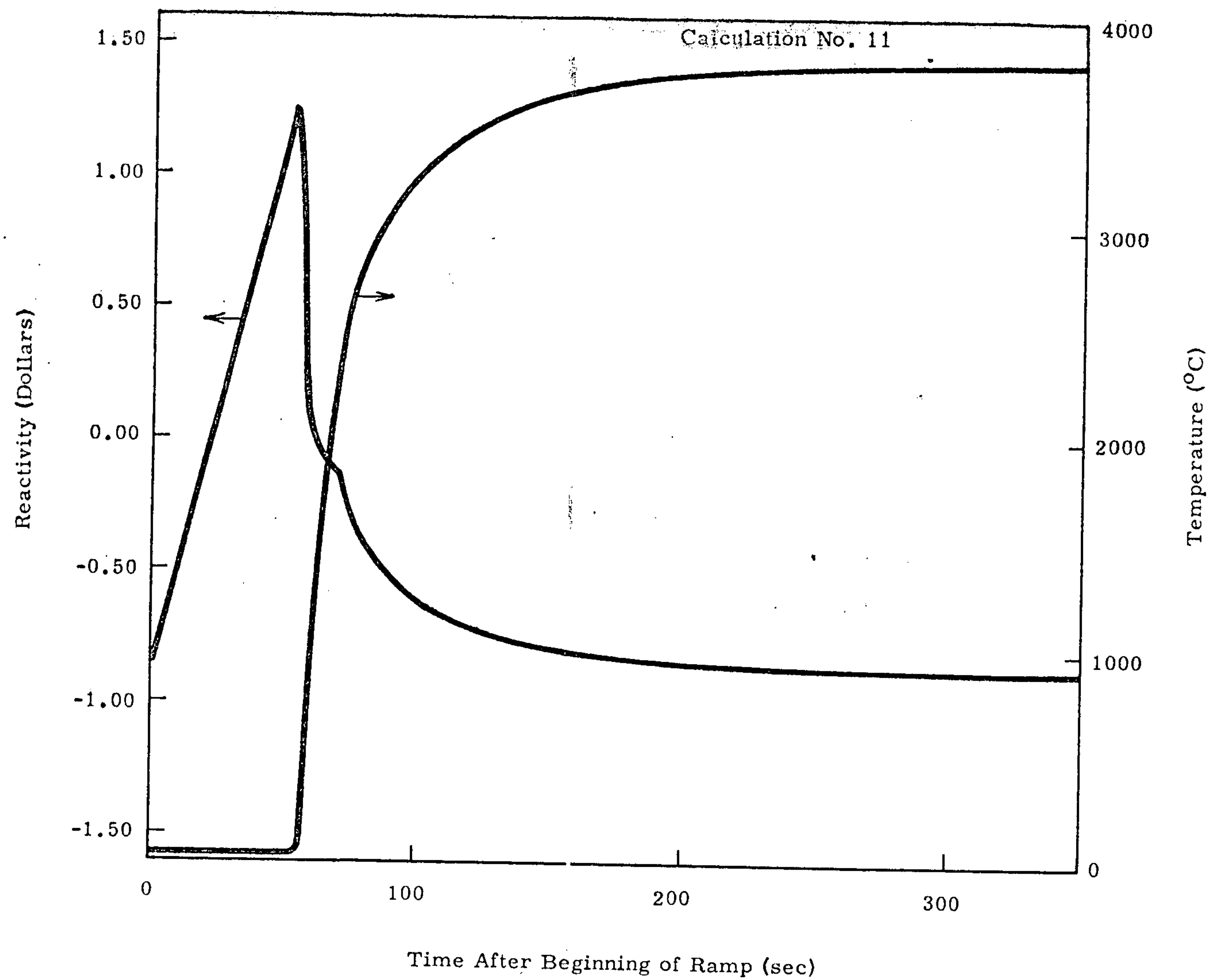


FIGURE A-2b

Reactivity and Fuel Temperature

in Self-Terminated Excursion: Calculation No. 11

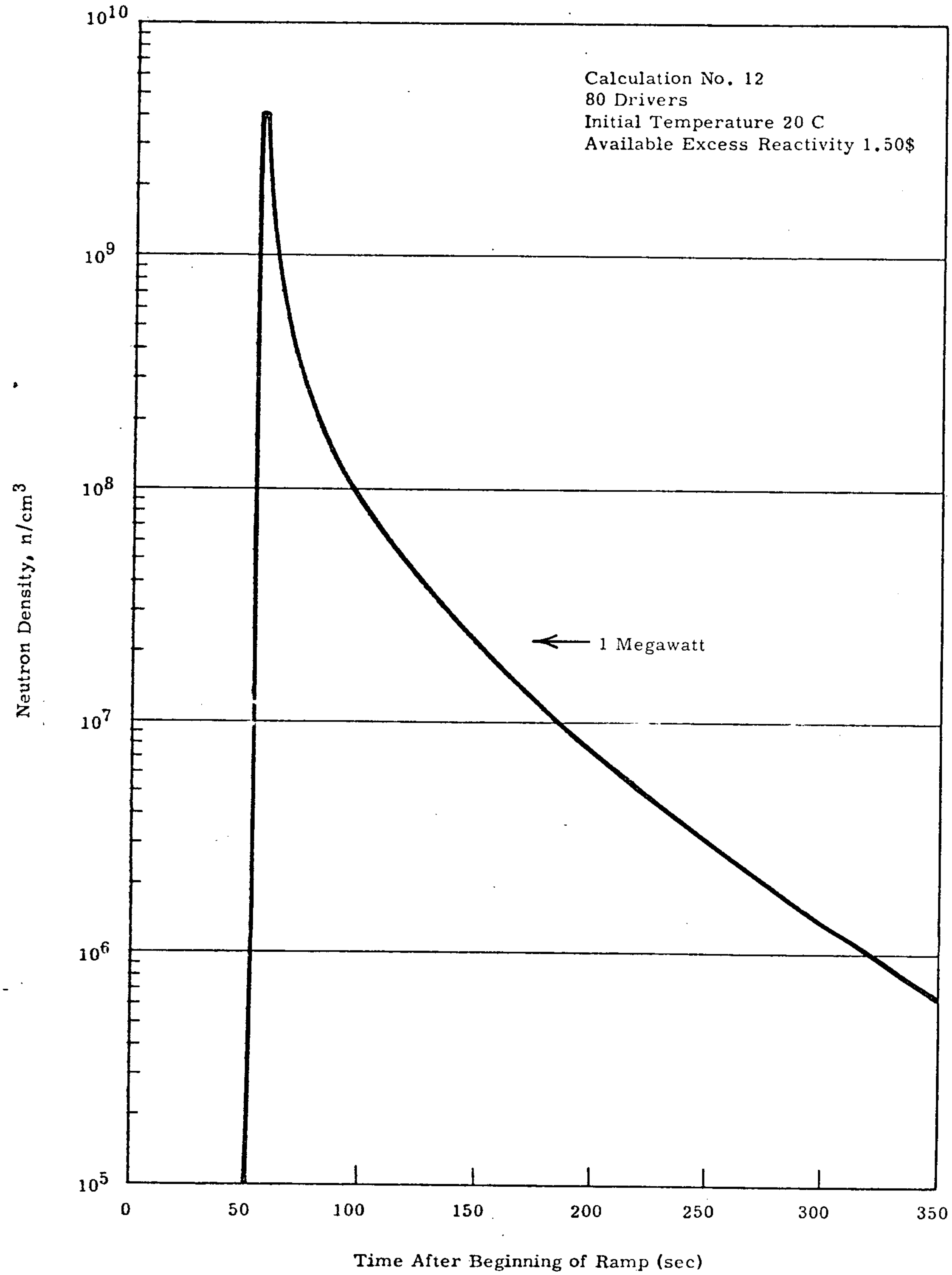
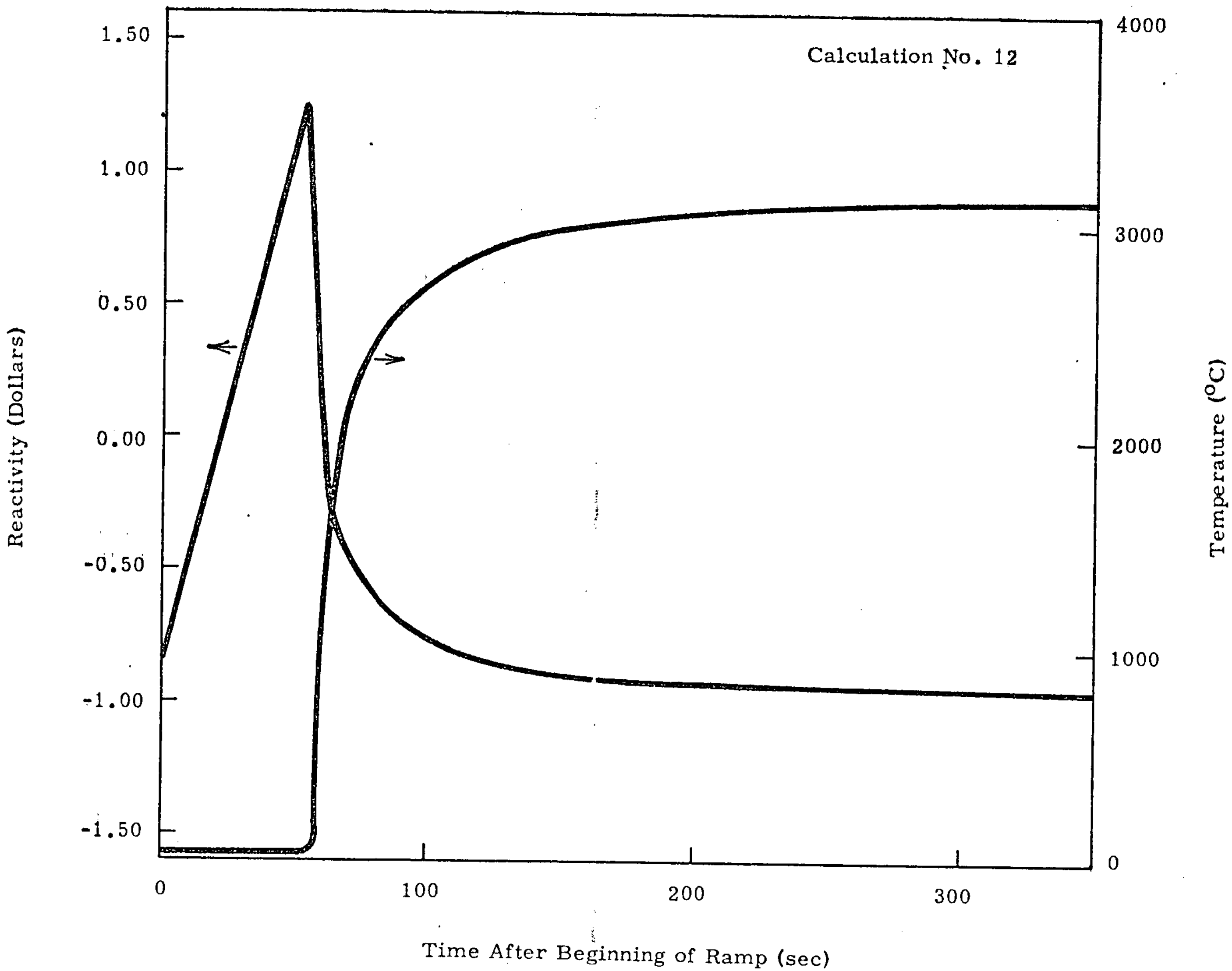


FIGURE A-3a

Flux Level in Self-Terminated
Excursion: Calculation No. 12



<u>FIGURE A-3b</u>

Reactivity and Fuel Temperature
in Self-Terminated Excursion: Calculation No. 12

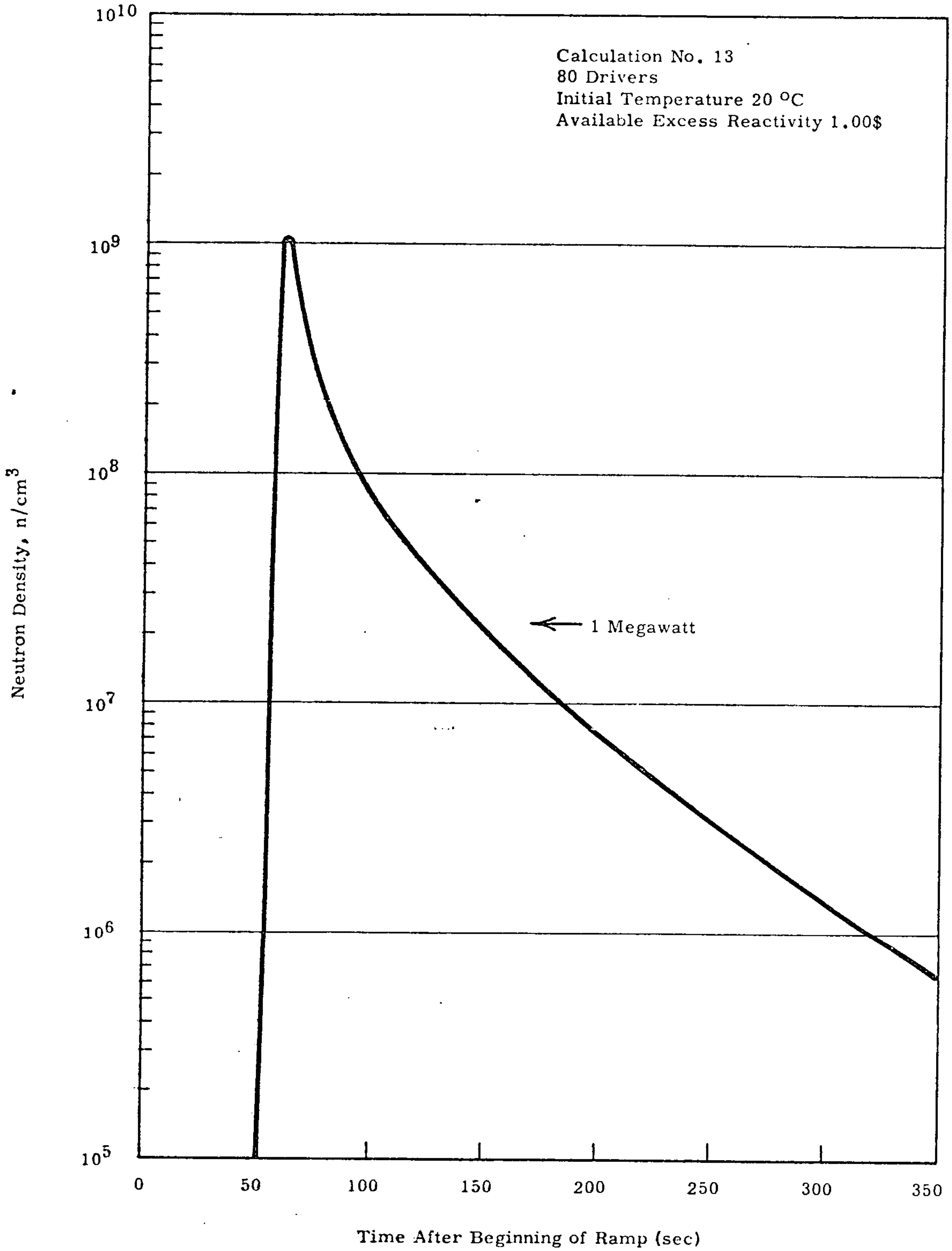


FIGURE A-4a

Flux Level in Self-Terminated Excursion: Calculation No. 13

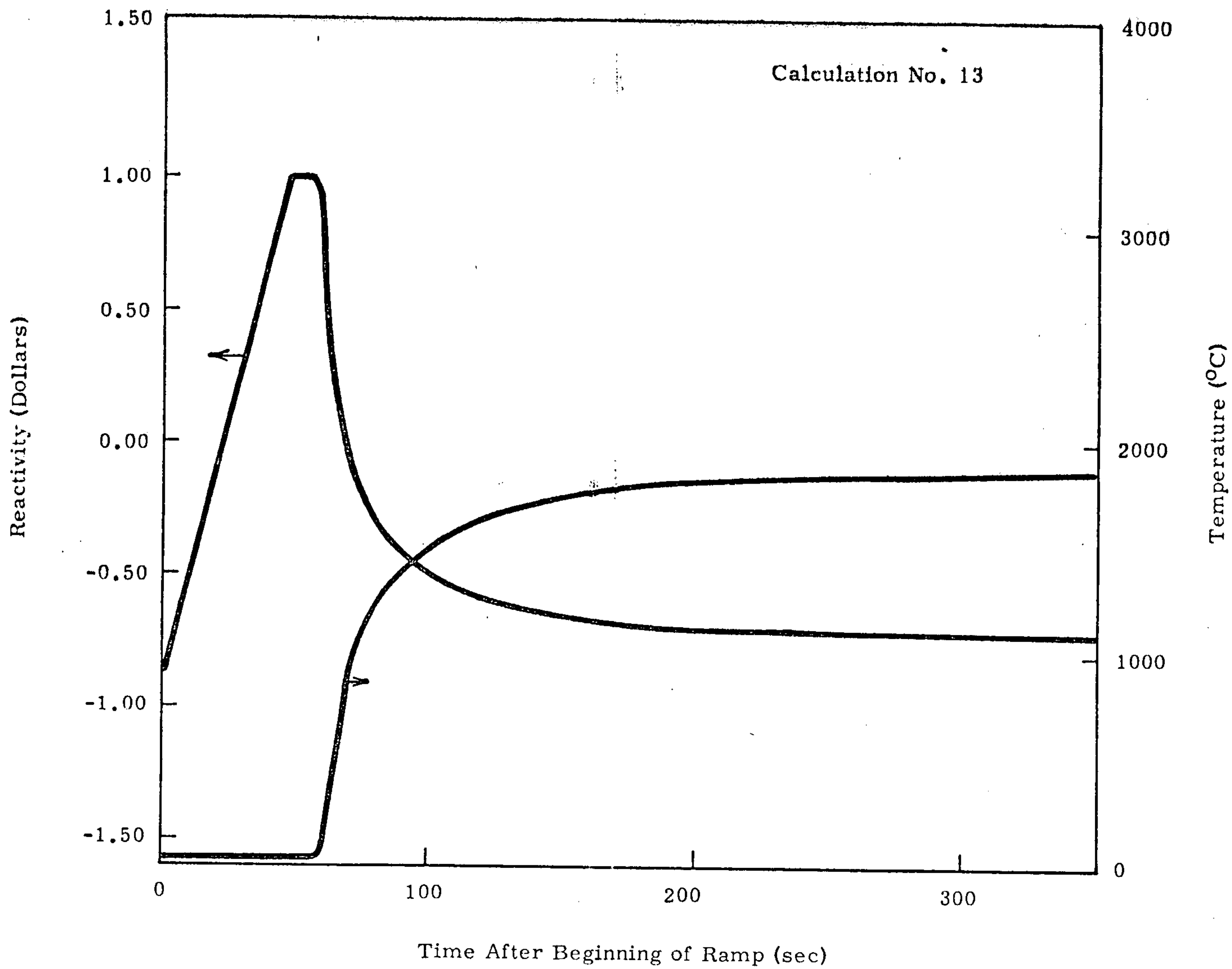


FIGURE A-4b
Reactivity and Fuel Temperature
in Self-Terminated Excursion: Calculation No. 13

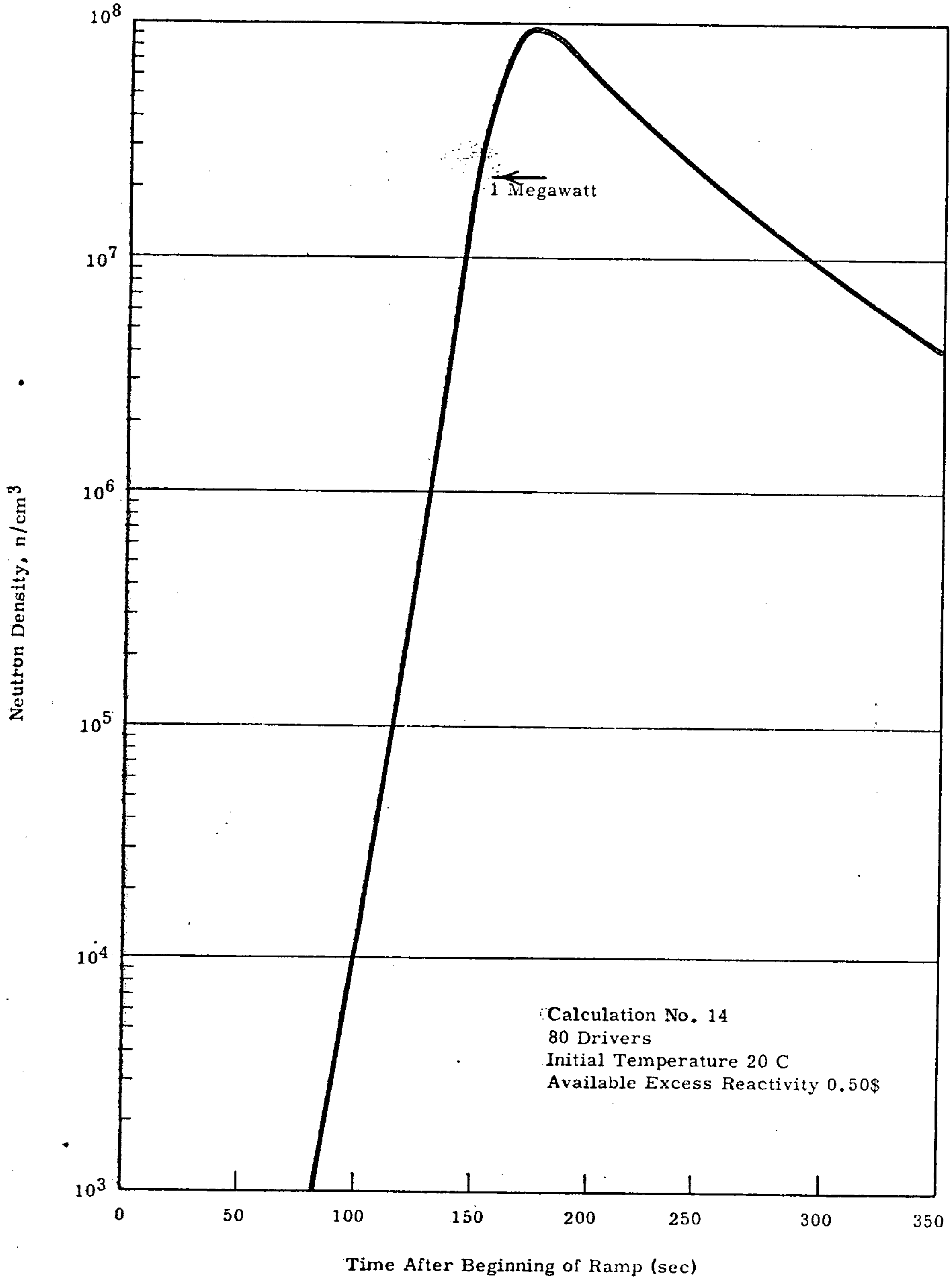


FIGURE A.5a

Flux Level in Self-Terminated Excursion: Calculation No.14

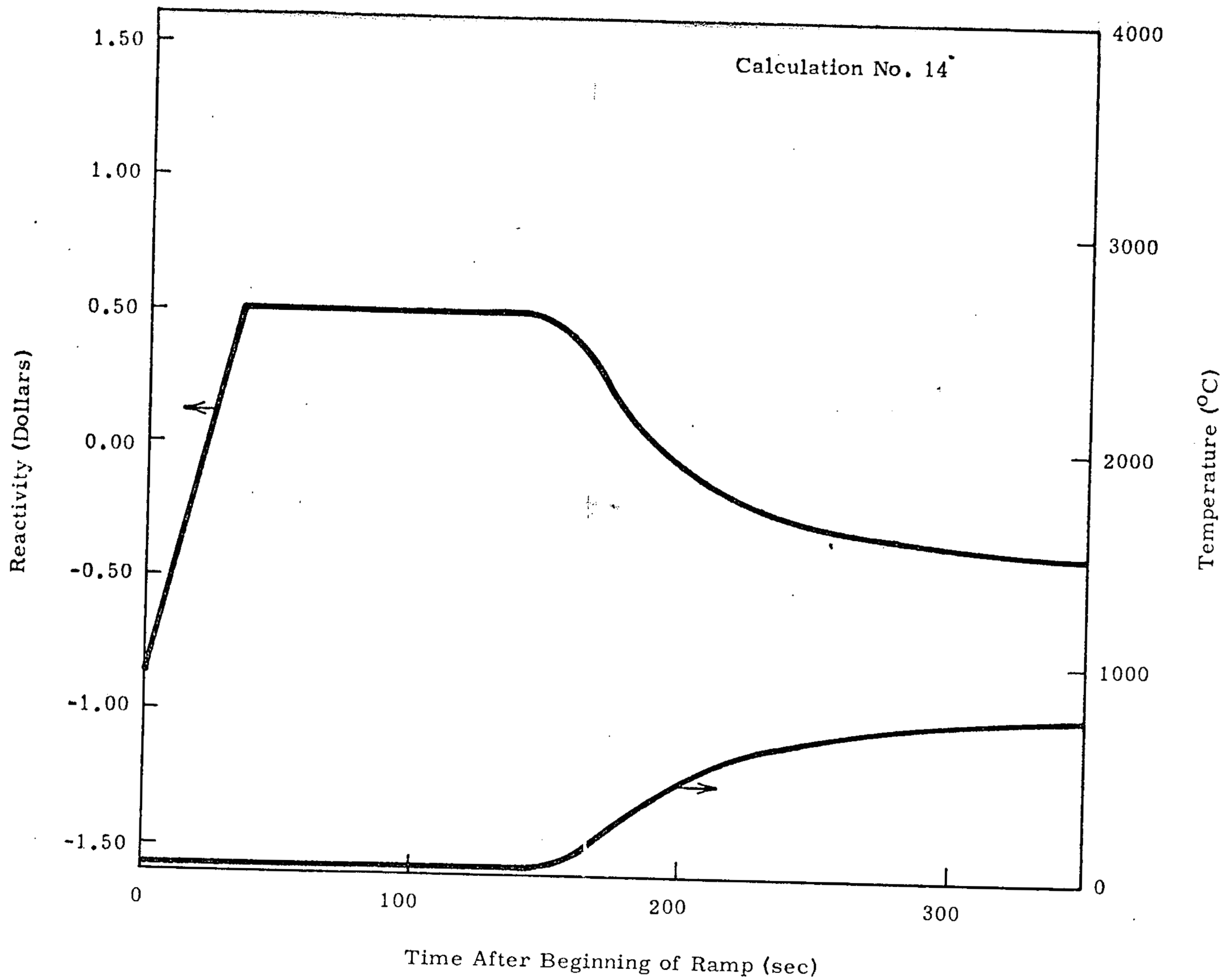


FIGURE A-5b

Reactivity and Fuel Temperature
in Self-Terminated Excursion: Calculation No. 14

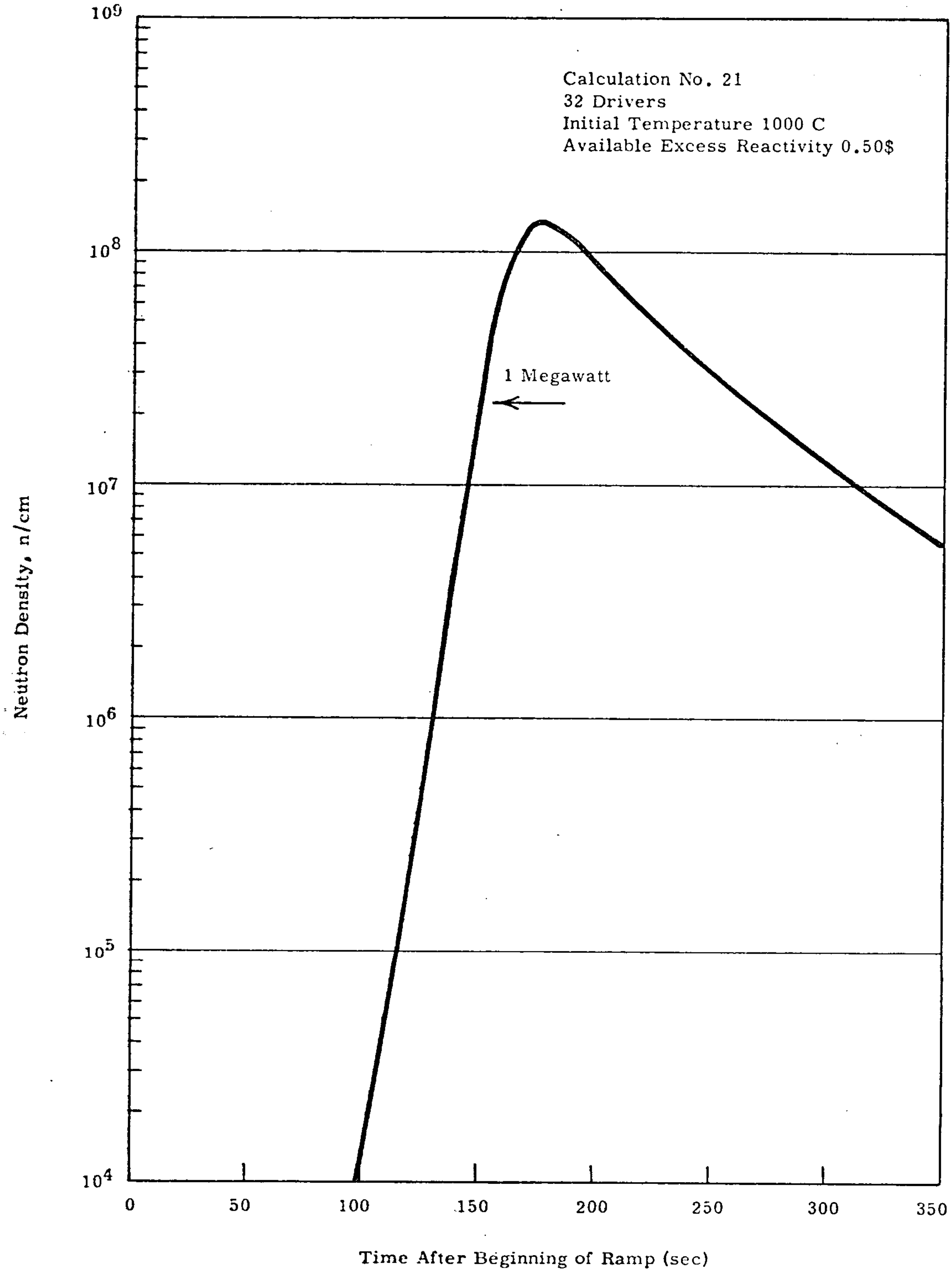


FIGURE A-6a

Flux Level in Self-Terminated Excursion: Calculation No. 21

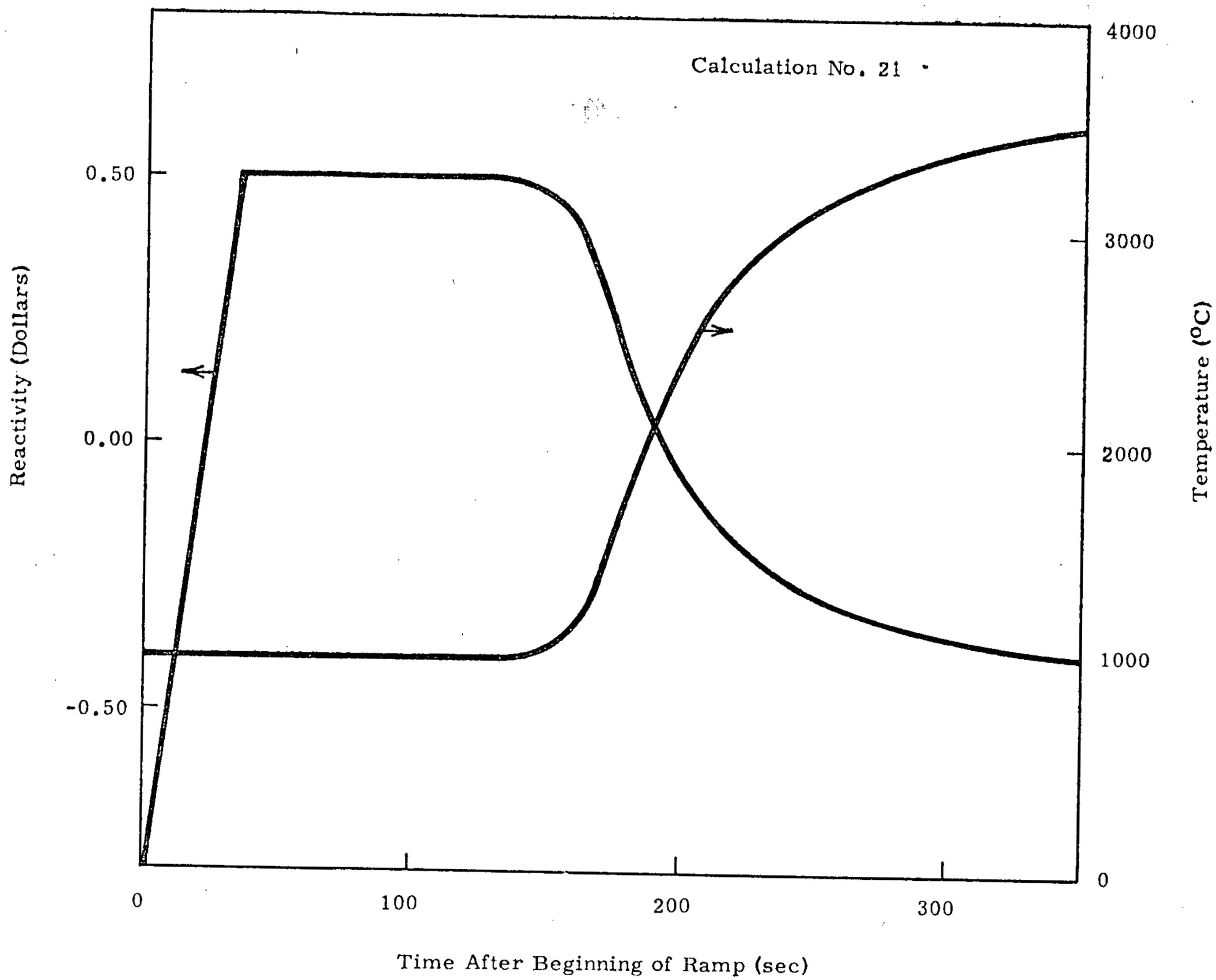


FIGURE A-6b

Reactivity and Fuel Temperature
in Self-Terminated Excursion: Calculation No. 21

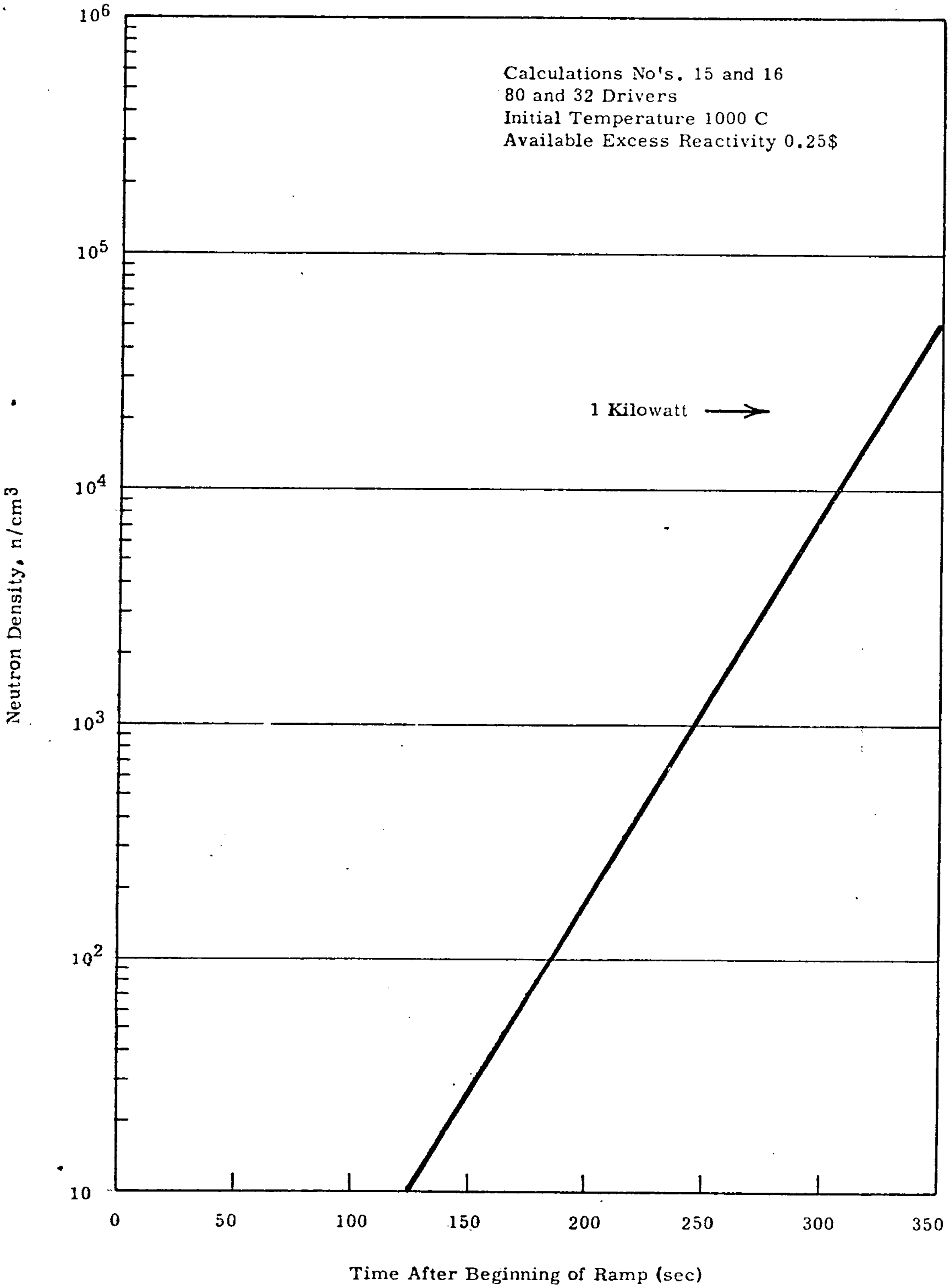


FIGURE A-7a
Flux Level in Self-Terminated
Excursion: Calculation No. 15 and 16

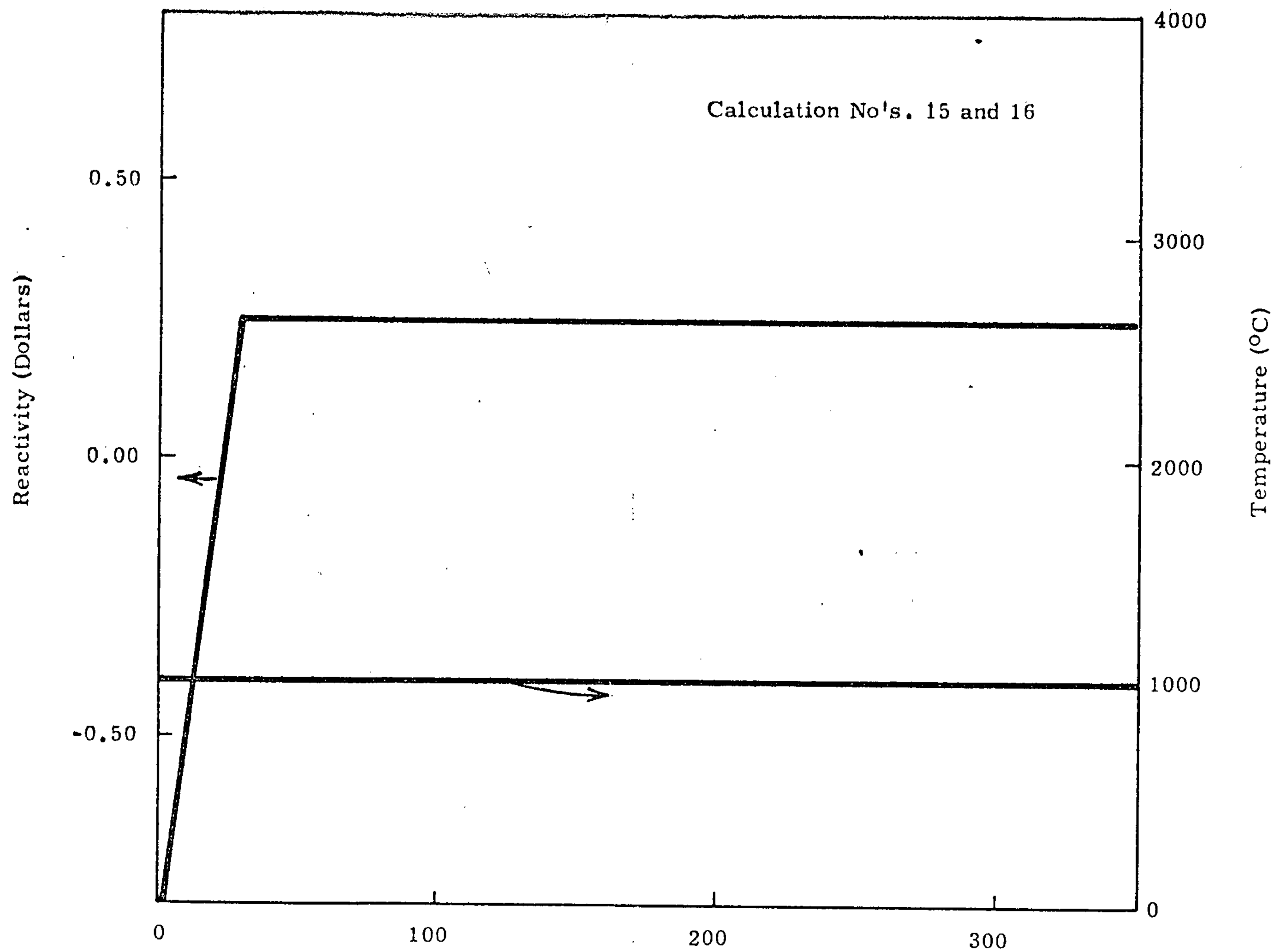


FIGURE A-7b

Reactivity and Fuel Temperature
in Self-Terminated Excursion: Calculation Nos. 15 and 16

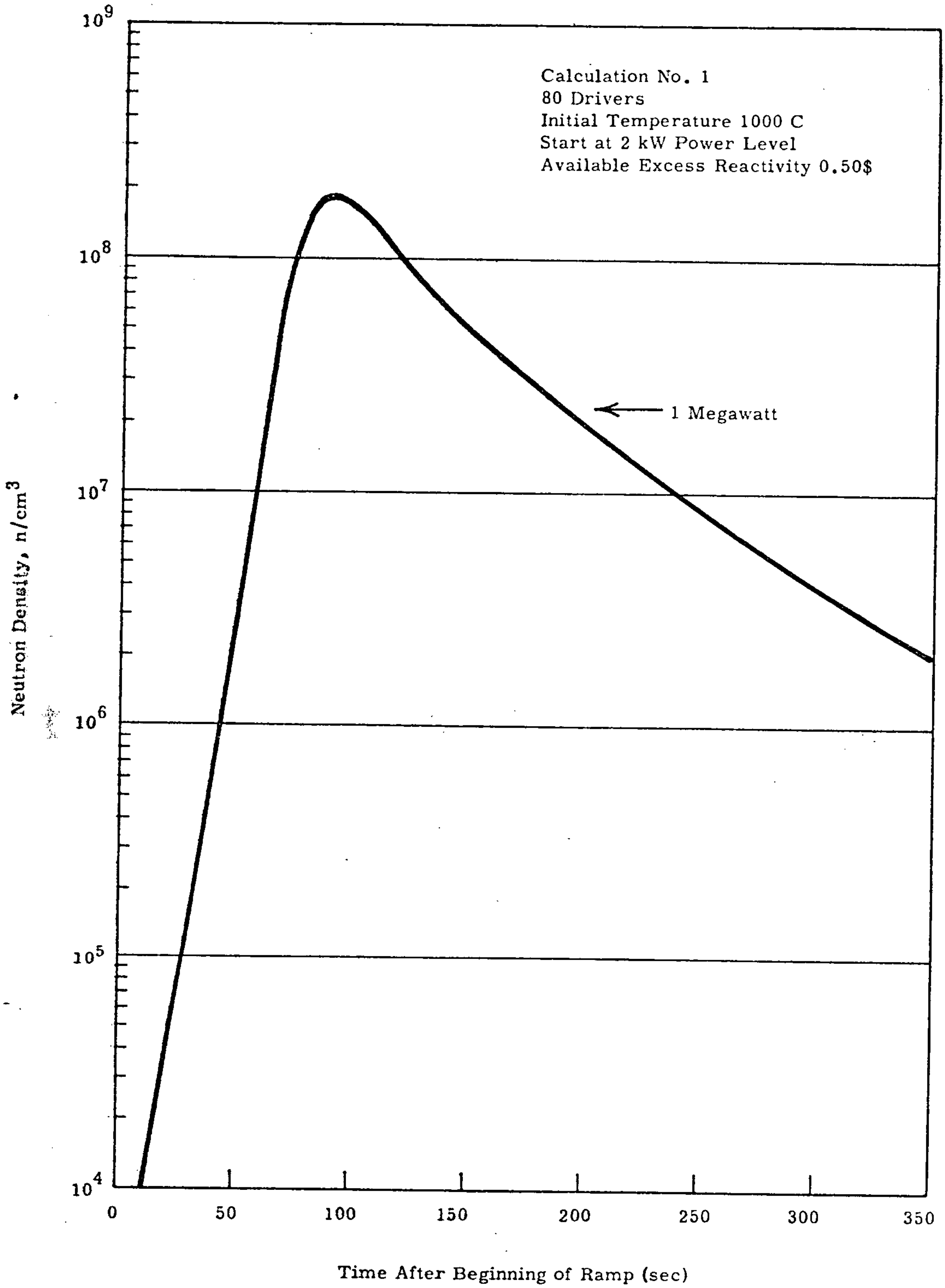


FIGURE A-8a
Flux Level in Self-Terminated
Excursion: Calculation No. 1

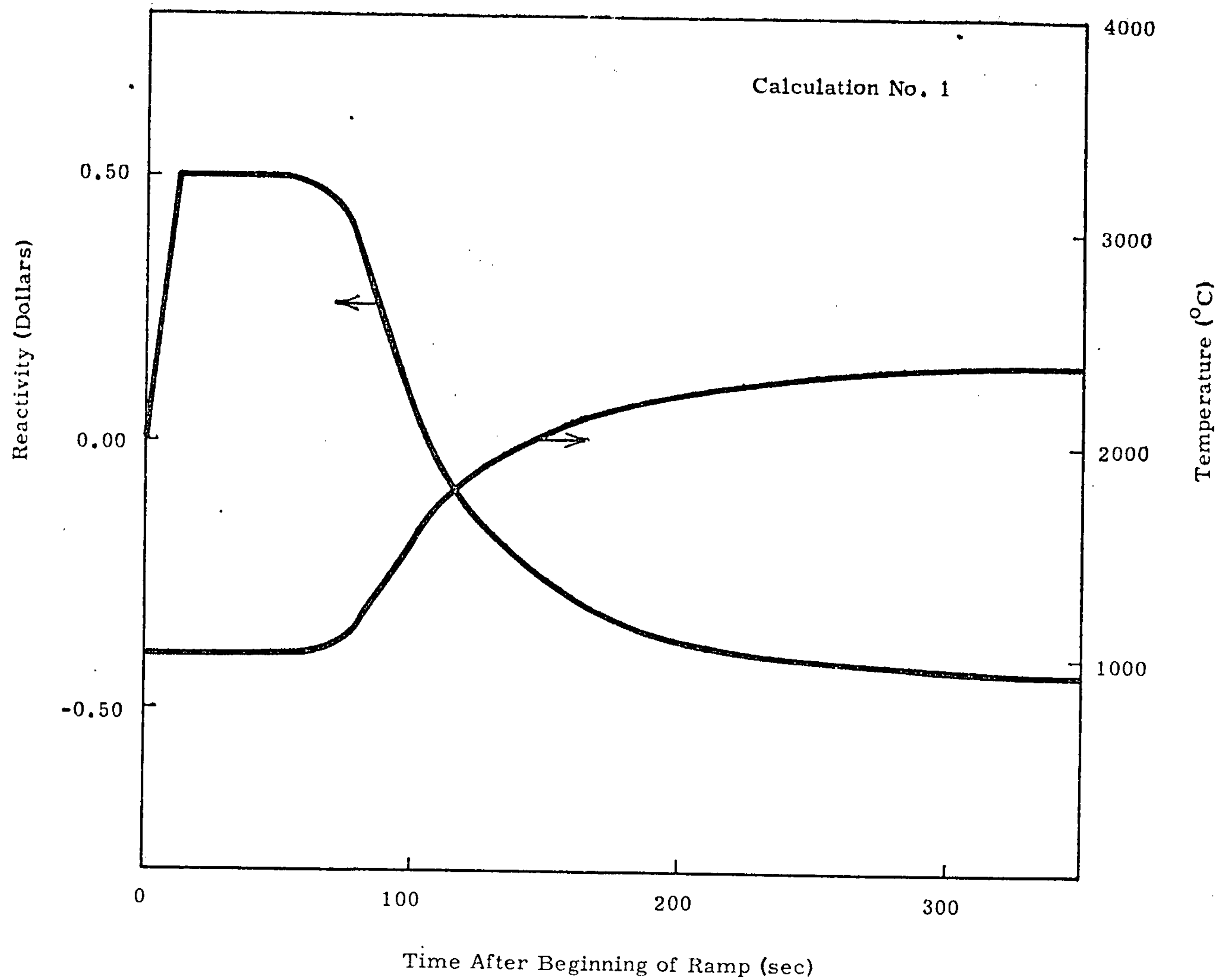


FIGURE A-8b

Reactivity and Fuel Temperature
in Self-Terminated Excursion: Calculation No. 1

fissile material have been adjusted such that if a driver fuel element and a control rod are in the same flux:

- they produce approximately the same total number of fissions per unit length and
- the fissions per unit volume in the control rod is less than in the driver fuel.

If the heating should occur so rapidly that the heat was not shared with the nickel, the critical temperature would be determined by the CO produced in the cylinders by the reaction $UO_2 + 4\ C \rightarrow UC_2 + 2\ CO$. Assuming no diffusion of CO out of the graphite matrix, the cylinders would rupture when the equilibrium pressure of the CO exceeded the ambient external pressure which is one atmosphere. The fueled cylinders would heat more slowly than the driver fuel because of their graphite content. Figure A-10 indicates that the driver fuel, starting at either 20 C or 1000 C, would have attained a temperature of 2000 C when the control rod cylinders reached 1800 C. Since the control rod would of necessity be in or near its most reactive position, this occurrence would tend to reduce the reactivity of the rod either by disassembly of the fuel or by forcing the fueled cylinders to more shielded positions.

If it is assumed that the heat generated in the control rod inner cylinders is shared with the nickel rod, the heat capacity of the combination is even higher than in the previous case. Hence the control rod temperature lags the driver fuel temperature even more. Figure A-10 shows that if the nickel rod reaches a temperature of 1400 C, the driver fuel will have attained a temperature of about 2000 C if the excursion started at 20 C, or 1700 C if it started at 1000 C. Should this occur it would be impossible to move the control rods to a less reactive position; nuclear shutdown could still be provided by the vertical safety rods, however.

If heating continued after the nickel rod had melted, the CO pressure could eventually become greater than one atmosphere. In this event the UO_2-graphite cylinders could be forced out of the center of the control rods. The loss of fuel would cause a decrease in the reactivity contribution of the control rods.

The values of the input parameters used in the calculations are given in Table IX-XII.

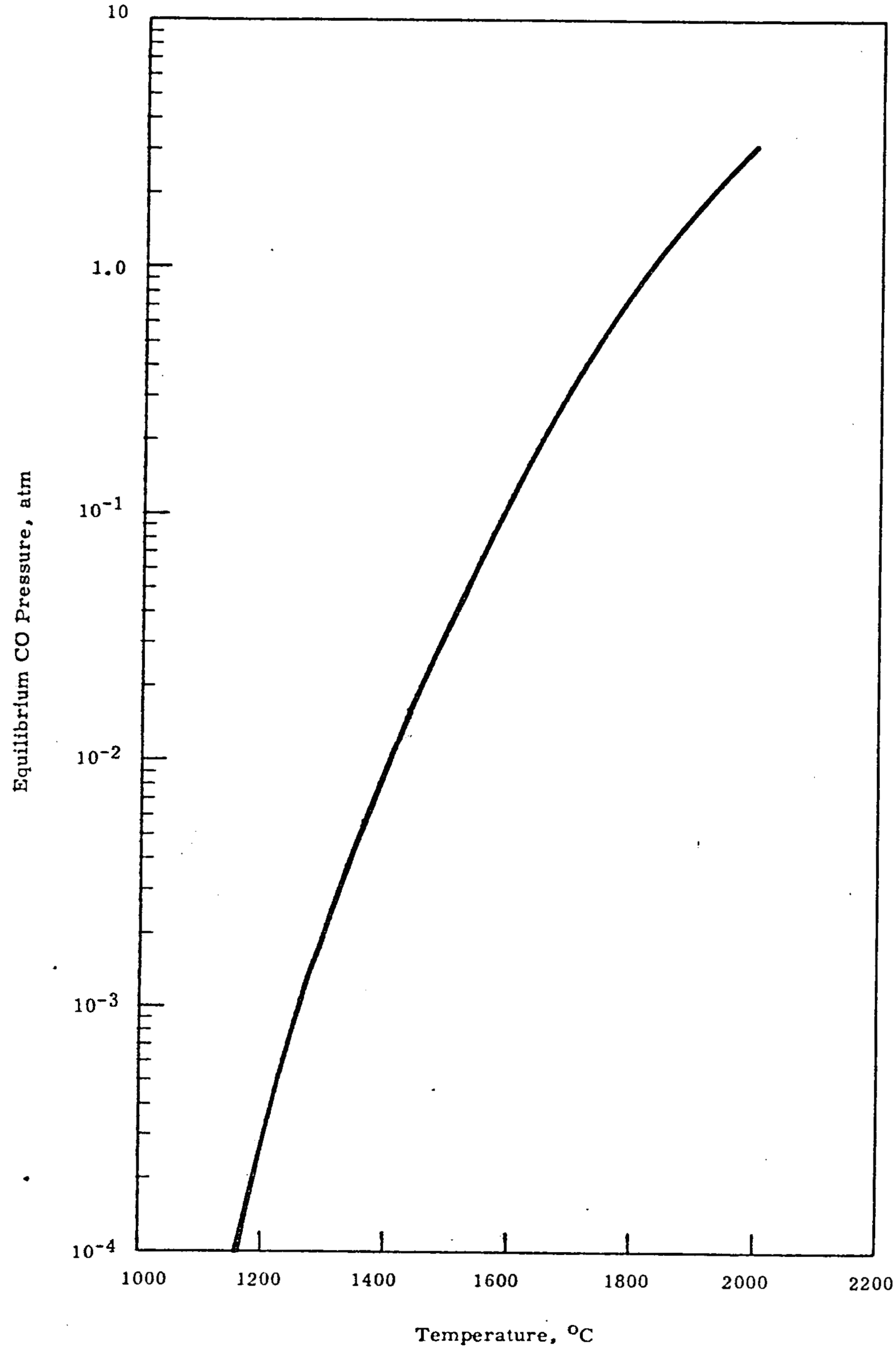


FIGURE A-9

Equilibria in the UO_2-C Reaction

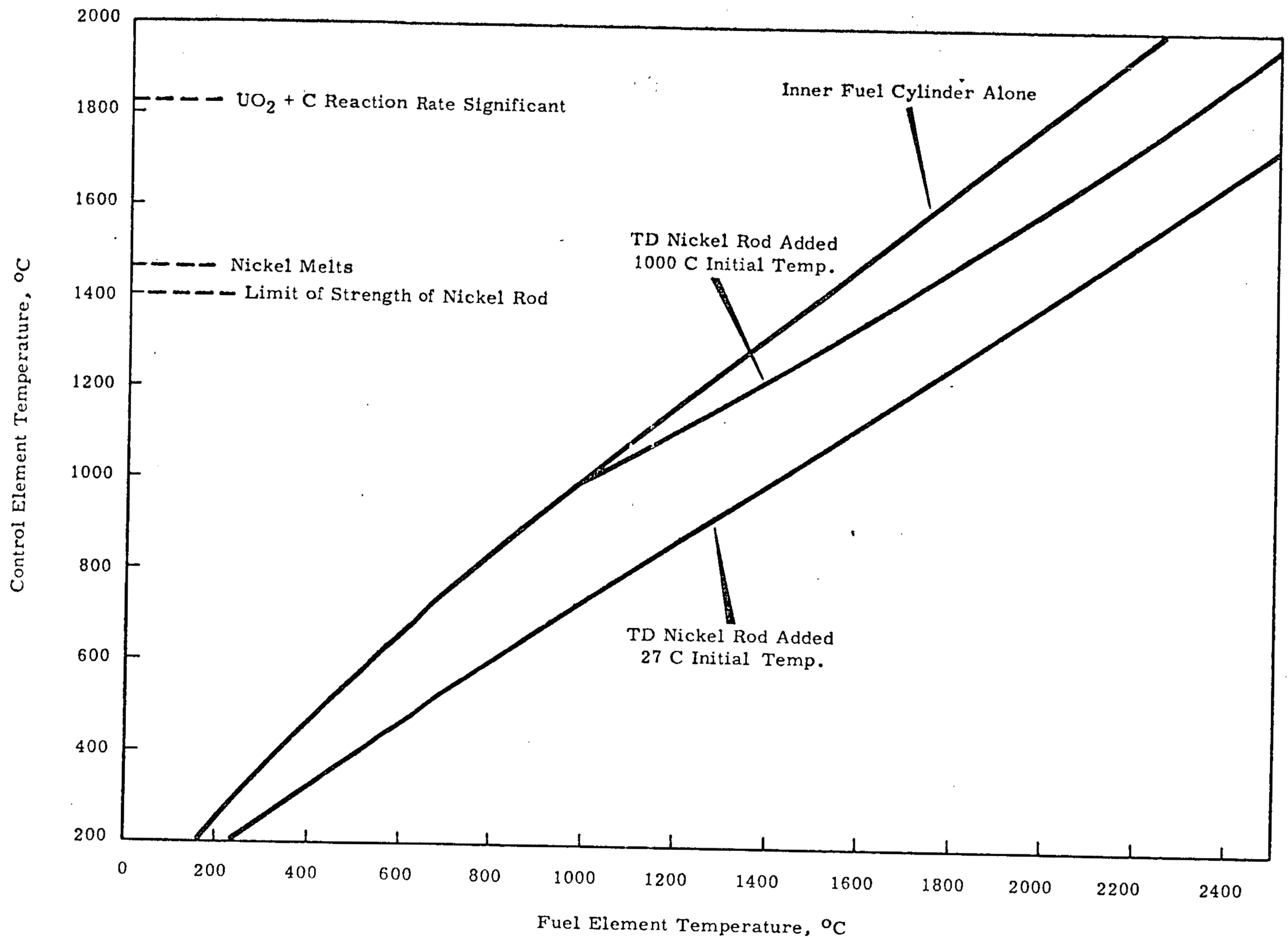


FIGURE A-10

Control Element Temperature in Excursions

APPENDIX B

Main Heat Exchanger Leak

Taking as a basis one gram-mole of nitrogen at 1000 C and adding to it n gram-moles of liquid water at 0 C, the following is obtained, where T is the equilibrium absolute temperature of the mixture:

$$\text{Heat input} = \int_0^{1273} (6.50 + 0.00100T)\, dT + (100.04)(18.016)n$$

$$\text{Heat output} = 9{,}729\, n + \int_0^{T} (6.50 + 0.00100T)\, dT + n \int_0^{T} (8.22 + 1.5 \times 10^{-4}\, T + 1.34 \times 10^{-6} T^2)\, dT$$

Equating these and performing the integrations,

$$9084.76 + 1802.32\, n = 9{,}729\, n + 5 \times 10^{-4}\, T^2 + n\, (8.22\, T + 7.5 \times 10^{-5}\, T^2 + 0.4467 \times 10^{-6}\, T^3) + 6.5\, T$$

Introducing values of T and solving for n:

T, °K	n, moles water	P, fraction of initial pressure
1273	0.000	1.00
1173	0.042	0.96
1073	0.088	0.92
973	0.139	0.87
873	0.196	0.82
773	0.259	0.76
673	0.329	0.70
573	0.408	0.63
473	0.497	0.56

Letting the initial pressure = 1.00, the pressure of the mixture would be

$P = T (1 + n)/1273$. Values of P are tabulated above. This computation shows that the introduction of liquid water into the circulating gas stream gives a quenching effect and a reduction in pressure instead of a pressure surge.

APPENDIX C

Training Program Outline

1. Fundamentals and Background

The candidate is supplied with material concerning the the safety analysis of the reactor, design and operational details, and an indication of the minimum information he must know to become a qualified operator. Included are the Reactor Safeguards Report, Operating Safety Limits and Operating Specifications, reference material on reactor physics and computers, emergency procedures for unusual incidents, blueprints and diagrams of electrical and mechanical components, and the examination questions and answers.

2. On-the-Job Training

Classroom instruction will be given the first group of trainees. They will conduct the pre-operational design tests performed after the facility is turned over to Battelle-Northwest.

Subsequent trainees will be given less formal instruction; since they are added singly, the training is tutorial in nature. The candidate performs all operations under supervision until he is formally qualified.

3. Written Examination

A written examination will be given to all candidates. The examination covers all aspects of the work. There is no fixed time schedule for taking the test because there can be considerable variation in previous background and knowledge. It is expected that most candidates will be in training for about four to six months before formal testing.

4. Oral and Performance Examination

An oral examination and reactor operation demonstration follows successful completion of the written examination. The oral examination allows more detailed testing of any particular aspects that may appear indicated and ensures that the answers to the written questions are truly understood. The candidate must independently perform all phases of standard reactor operation requested and demonstrate familiarity with non-routine experimental and data-taking duties, programming, and emergency procedures.

5. Demonstration Period

Following a satisfactory performance on the oral and performance examination the candidate is formally designated to be provisionally qualified. In this capacity the operator is authorized to perform all normal duties of a qualified operator.

During a probationary period of several months the candidate must demonstrate continued development of his detailed job knowledge plus strong evidence of the usual traits required of responsible reactor personnel. When line management is satisfied that all requirements are met, the probationary period is complete, and the provisional status is lifted.

6. Training Review

Reviews of the operator's knowledge are made continuously as his performance is observed and checked by supervision. More formal reviews are made orally whenever work assignments have resulted in lack of practice in reactor operation. This situation can exist when an operator is engaged in lengthy preparation of experimental equipment. The list of qualified operators is reviewed, and updated on a frequent basis. The authorization lists are retained in a permanent file.

APPENDIX D

HTLTR SITE INFORMATION

Additional meteorological and seismological data are presented in Tables D - I and D - II and in Figures D - 1 and D - 2.

TABLE D-I

WIND SPEED AND DIRECTION FREQUENCY - 300 AREA

Wind Speed in Units of mph. Wind Direction Frequency in Per Cent of Time. Day: 0700-1900 PST. Night: 1900-0700 PST.

	Season														
	Winter			Spring			Summer			Fall			Annual		
	Day	Night	Total	Day	Night	Total	Day	Night	Total	Day	Night	Total	Day	Night	Total
NE	3.5	1.1	2.3	6.1	3.8	4.9	9.8	4.3	7.0	5.2	1.3	3.2	6.1	2.6	4.4
Speed	6.2	3.9	5.0	8.1	8.3	8.2	10.3	8.2	9.2	6.9	7.9	7.4	7.9	7.1	7.5
E	3.0	2.2	2.6	7.7	3.3	5.5	9.5	2.7	6.1	2.0	0.8	1.4	5.6	2.2	3.9
Speed	7.2	8.3	7.8	9.0	10.5	9.8	9.5	7.4	8.4	6.0	6.7	6.3	7.9	8.2	8.0
SE	15.5	17.2	16.4	16.9	16.6	16.8	19.8	16.6	18.2	11.8	17.7	14.8	16.0	17.0	16.5
Speed	8.4	8.8	8.6	9.2	8.0	8.6	10.6	7.4	9.0	7.9	7.8	7.8	9.0	8.0	8.5
S	12.5	13.9	13.2	10.1	9.5	9.8	10.4	6.2	8.3	11.5	9.4	10.4	11.1	9.8	10.4
Speed	14.5	14.0	14.2	17.5	13.4	15.4	14.9	12.9	13.9	12.2	10.2	11.2	14.8	12.6	13.7
SW	22.3	21.0	21.6	25.2	17.1	21.2	30.0	17.5	23.8	15.9	12.1	13.6	23.1	16.9	20.0
Speed	16.5	14.1	15.3	13.6	12.2	12.9	17.5	11.5	14.5	15.9	11.3	13.6	15.9	12.3	14.1
W	5.6	6.7	6.2	8.1	7.1	7.6	5.2	9.4	7.3	5.2	5.4	5.3	6.0	7.2	6.6
Speed	9.6	6.1	7.8	19.4	10.4	14.9	15.5	9.3	12.4	13.7	6.4	10.0	14.6	8.0	13.3
NW	15.0	19.3	17.2	9.7	22.7	16.2	6.8	21.2	14.0	15.2	20.0	17.6	11.7	20.8	16.2
Speed	10.1	8.3	9.2	16.1	12.2	14.1	14.8	11.5	13.2	11.7	8.7	10.2	13.2	10.2	11.7
N	15.5	11.8	13.6	12.6	12.4	12.5	7.7	16.9	12.3	20.8	13.6	17.2	14.1	13.7	13.9
Speed	10.2	9.5	9.8	14.1	11.1	12.6	14.2	10.7	12.4	10.8	9.7	10.2	12.3	10.3	11.3
Variable	6.4	6.5	6.4	2.8	6.0	4.4	0.5	2.0	1.2	9.7	13.1	11.4	4.9	6.9	5.9
Speed	2.2	2.2	2.2	1.9	2.0	2.0	2.0	2.0	2.0	2.2	2.5	2.4	2.1	2.2	2.2
Calm	2.7	0.3	1.5	0.8	1.3	1.0	0	0	0	0.4	0.5	0.4	1.0	0.5	0.8

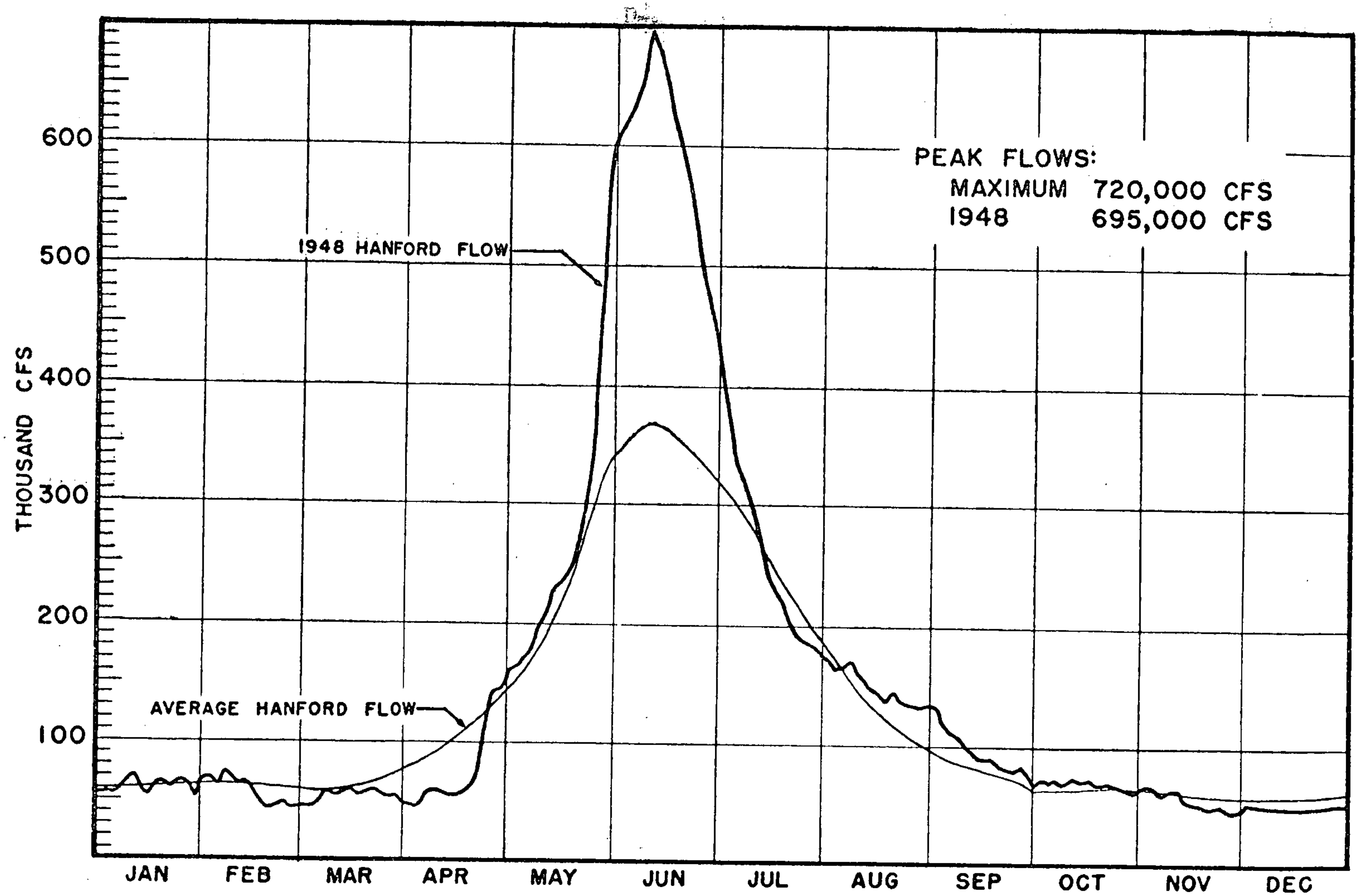


FIGURE D-1
Columbia River Flow

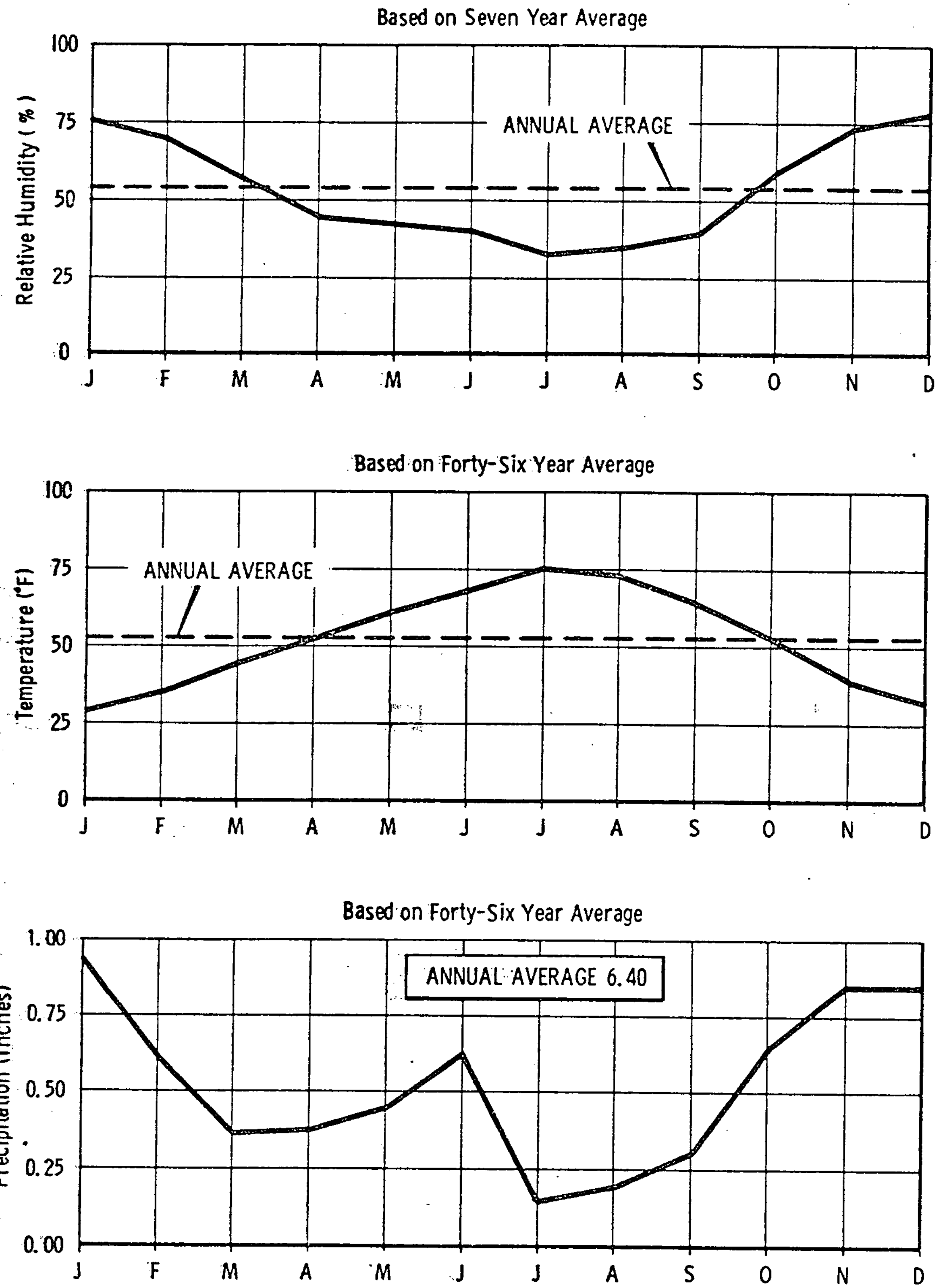
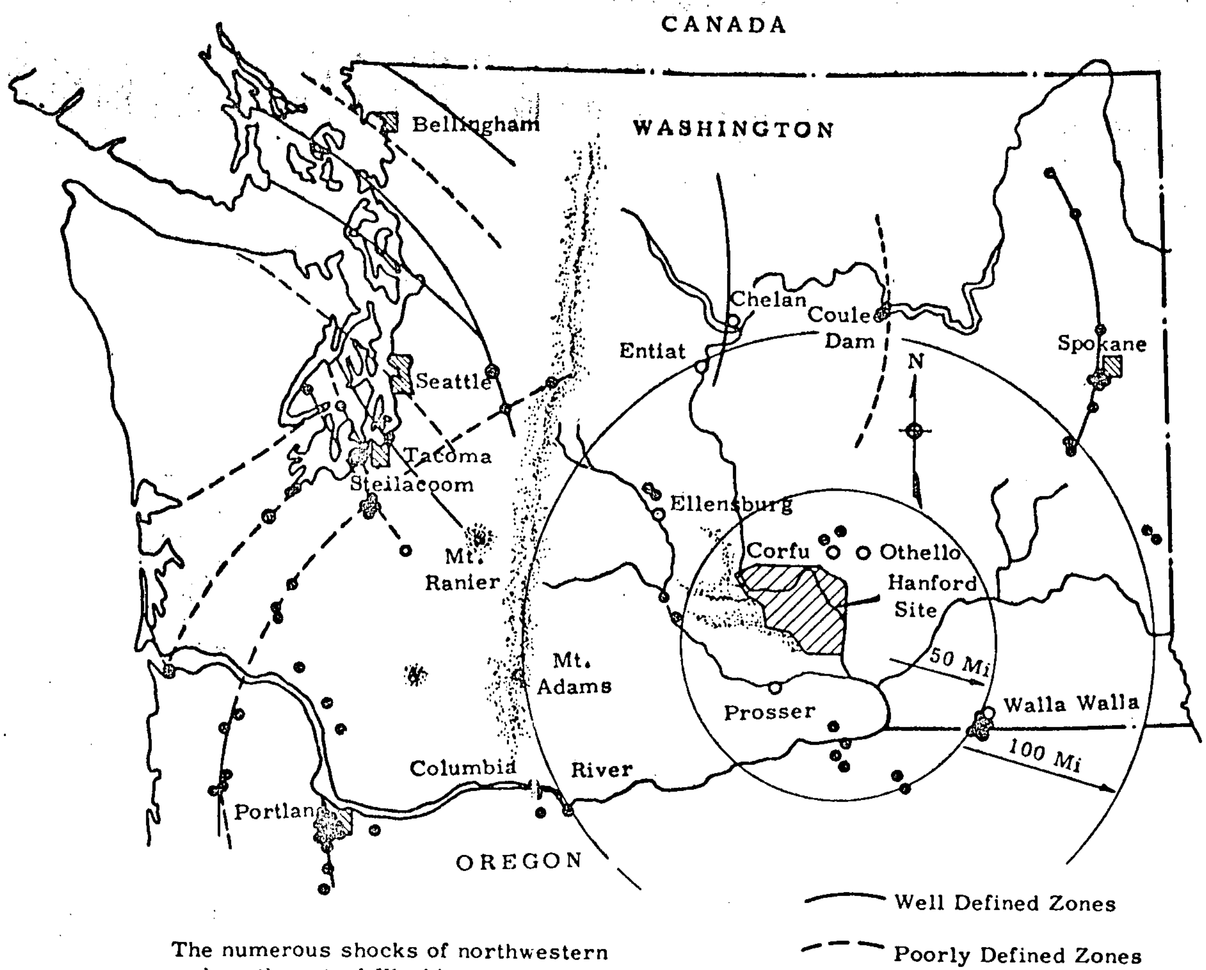


FIGURE D-2

Hanford Meteorological Data



The numerous shocks of northwestern and north central Washington are omitted, their general locations being shown by lines indicating active fault zones.

FIGURE D-3
Earthquake Zones
of Washington State, 1833 to 1958

TABLE D-II

APPROXIMATE RELATIONSHIPS AMONG EARTHQUAKE INTENSITY, GROUND ACCELERATION, AND ZONES

	Modified Mercalli Intensity Scale (1931) Wood & Neumann	a - Ground Acceleration $\frac{cm}{sec^2}$	$\frac{a}{g}$	Building Code Zone
I	Detected only by sensitive instruments.			0
II	Felt by a few persons at rest, especially on upper floors; delicate suspended objects may swing.	2, 3		0
III	Felt noticeably indoors; but not always recognized as a quake; standing autos rock slightly, vibration like passing truck.	4, 5, 6	0.005g	0
IV	Felt indoors by many, outdoors by a few; at night some awaken; dishes, windows, doors disturbed; motor cars rock noticeably.	8, 10	0.01g	0
V	Felt by most people, some breakage of dishes, windows and plaster, disturbance of tall objects.	20, 30		1
VI	Felt by all, many frightened and run outdoors; falling plaster and chimneys, damage small.	40, 50, 60	0.05g	1
VII	Everybody runs outdoors, damage to buildings varies, depending on quality of construction; noticed by drivers of autos.	80, 100	0.1g	2
VIII	Panel walls thrown out of frames; fall of walls, monuments, chimneys; sand and mud ejected; drivers of autos disturbed.	200, 300		2
IX	Buildings shifted off foundations, cracked, thrown out of plumb; ground cracked, underground pipes broken.	400, 500, 600	0.5g	3
X	Most masonry and frame structures destroyed; ground cracked, rails bent, landslides.	800, 1000	1g	3
XI	New structures remain standing; bridges destroyed; fissures in ground, pipes broken; landslides, rails bent.	2000, 3000		3
XII	Damage total; waves seen on ground surface, lines of sight and level distorted; objects thrown up into air.	4000, 5000, 6000	5g	3

APPENDIX E

COMPARISON WITH PCTR

The nuclear characteristics of the HTLTR and their changes with driver fuel and core loadings are expected to be very similar to those for the Physical Constants Test Reactor (PCTR), which is quite similar to HTLTR in its general configuration. The HTLTR differs from the PCTR in the ways listed below:

- Larger overall size
- Thicker reflector
- Larger volume available for test cores
- U-238 in driver fuel
- Shutter-type control rods ~~in which no poison is exposed in the most reactive setting~~
- Gd_2O_3 & Eu_2O_3 ~~Boron~~ instead of cadmium as the poison in ~~control and~~ safety rods
- Graphite fuel cladding
- Gadolinium oxide shim rods to adjust the slow temperature coefficient. These may be modified with small amounts of iridium or hafnium.
- Nitrogen atmosphere
- Heated and enclosed

Although these are significant differences from PCTR, it is obvious that reactor loadings, reactor startup, and procedural controls on reactor safety are very similar. Data are presented in Table E-I on the worth of individual driver rods, control rods, and safety disks, and on critical masses, for the PCTR with graphite cores which have been used extensively in experimental programs.

TABLE E-I

PCTR LOADINGS

Core Size	26-1/4"	33-3/4"	41-1/4"	48-3/4"
No. of driver rods	22	32	46	73
Worth of driver rod (¢)	108	84	54	34
Worth of control rod (¢)	44	56	51	45
Worth of 1 Safety disk ($)	2.5	3.0*	3.4	3.4
Worth of 2 Safety disks ($)	--	--	5.7	--
U-235 in end driver fuel (kg)	5.6	5.6	5.6	5.6
Total U-235 (kg)	8.9	10.4	12.4	16.4
Total temperature coefficient (20 C)			$-0.45¢/^{o}C$	
Pressure coefficient			-0.05¢/mb	

*Estimated from other data

APPENDIX F

DETAILED DESCRIPTION AND RELIABILITY ANALYSIS OF HTLTR SAFETY SYSTEM

INTRODUCTION

Because the HTLTR safety system is composed of solid state devices instead of the customary arrangements of relays, a detailed description and exhaustive reliability analysis has been prepared to ensure that all modes of failure of the system which could lead to an uncontrolled reactor power excursion have been considered and precluded by the design. In the detailed description which follows, the basic functional parts of the system are first outlined. The elaboration of each functional block to ensure fail-safe operation is then discussed in a stepwise fashion. Block or schematic diagrams illustrate each step in the discussion.

In the reliability analysis, interconnection failures (open and short circuit and multiple grounds) at each point of the system are considered, as well as all possible modes of failure of the solid state devices. The probability of an unsafe failure of the system is then computed to give a quantitative measure of performance.

FUNCTIONAL BLOCK DIAGRAMS

The safety system is composed of five basic functional blocks (Figure 1): process sensor, comparator, logic, power supply, and electromagnetic clutch. The process sensor block monitors such variables as neutron flux level, coolant flow, or reactor temperature and, using amplification and signal conditioning, provides an input to the comparator. The comparator determines whether the variable is within or out of limits and transmits all neutron flux comparisons and any necessary comparisons from other process variables to the logic block. Recognizing in-limit information from the comparator, the logic block passes the system-normal signal to the power supply block. If any out-of-limit information appears at the output of the comparator, the logic block does not pass the system-normal signal. The power supply is energized to produce power for the electromagnetic clutches only when the system-normal signal is present. Some of the sensor-comparator combinations have direct lines to the logic block. Others share a common logic block input line. In these cases, another comparator is added to determine whether any of the process variables is out of limit. (Figure 2).

To ensure that the logic de-energizes the power supply if any out-of-limit information appears at its input, two logic blocks are provided, both of which must pass system-normal signals to keep the power supply energized. Although functionally the logic inputs are identical, each sensor-comparator combination provides physically separate inputs to the two logic blocks (Figure 3).

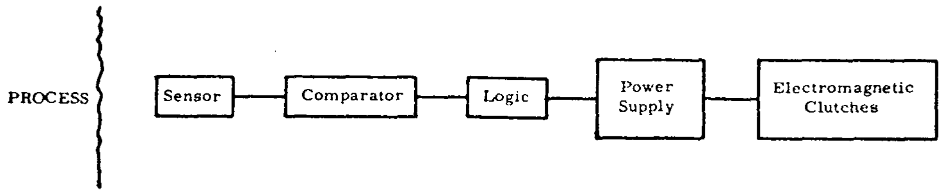


FIGURE F-1

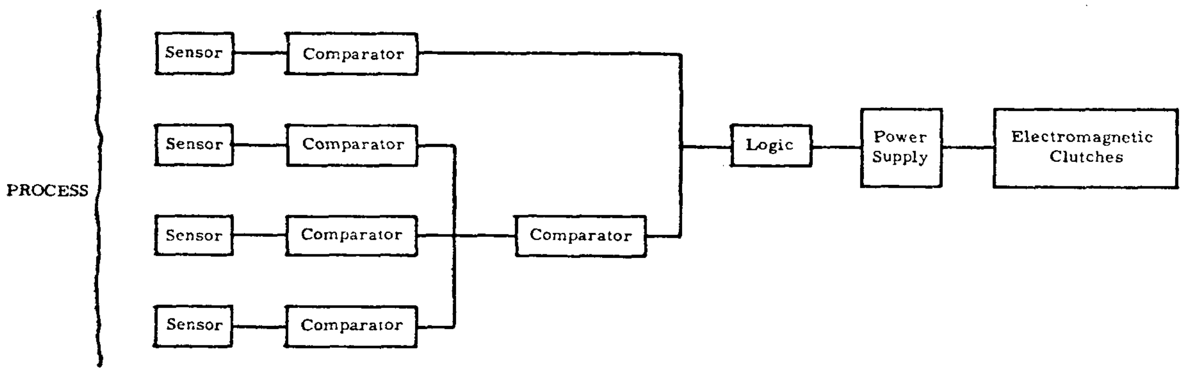


FIGURE F-2

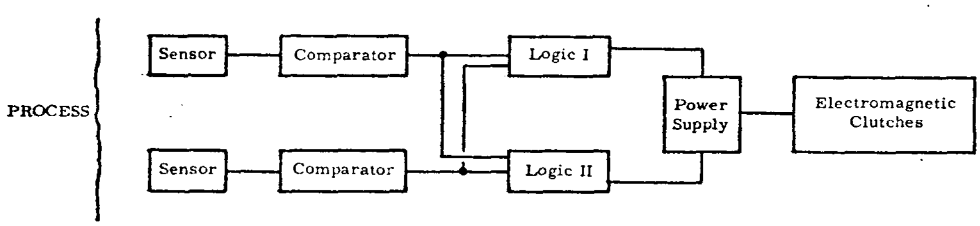


FIGURE F-3

Safety System Block Diagrams

SENSOR AND COMPARATOR BLOCKS

Neutron Flux Monitor Channels (Figure 4).

Neutron flux is monitored from startup through operating levels by two logarithmic count rate and period meters (LCR-PM) receiving pulse signals from fission counter chambers. The meters produce output signals proportional to the logarithm of the flux level and to the period. These meters are conventional in design and have proved highly reliable in service. In accordance with the principle of redundancy, two channels are provided; reactor shutdown occurs if either channel provides an out-of-limit signal for flux level (on-scale and high trips) or for period (short period trip). Since the two channels are completely separate and independent, a failure in one channel would not induce a failure in the other. They are, however, identical, and thus subject to possible identical but independent failures. Such a failure is prevented from permitting a power excursion by the flux measurement channels discussed below.

The comparator receives the level and period signals from the meter and compares them with preset values. As long as they remain within limits, the coil of the typical output relay of Figure 4 remains energized holding the contacts in the position shown. An open circuit, short circuit, or ground of the signal at this point, as well as the trip associated with the particular relay, would reduce or interrupt the current to the coil, allowing the relay contacts to move to the other position. When all relays are in the system-normal position as shown, the system-normal input signal to Logic I is zero volts and to Logic II is negative. When any relay is de-energized, the voltage at both logic block inputs is reversed.

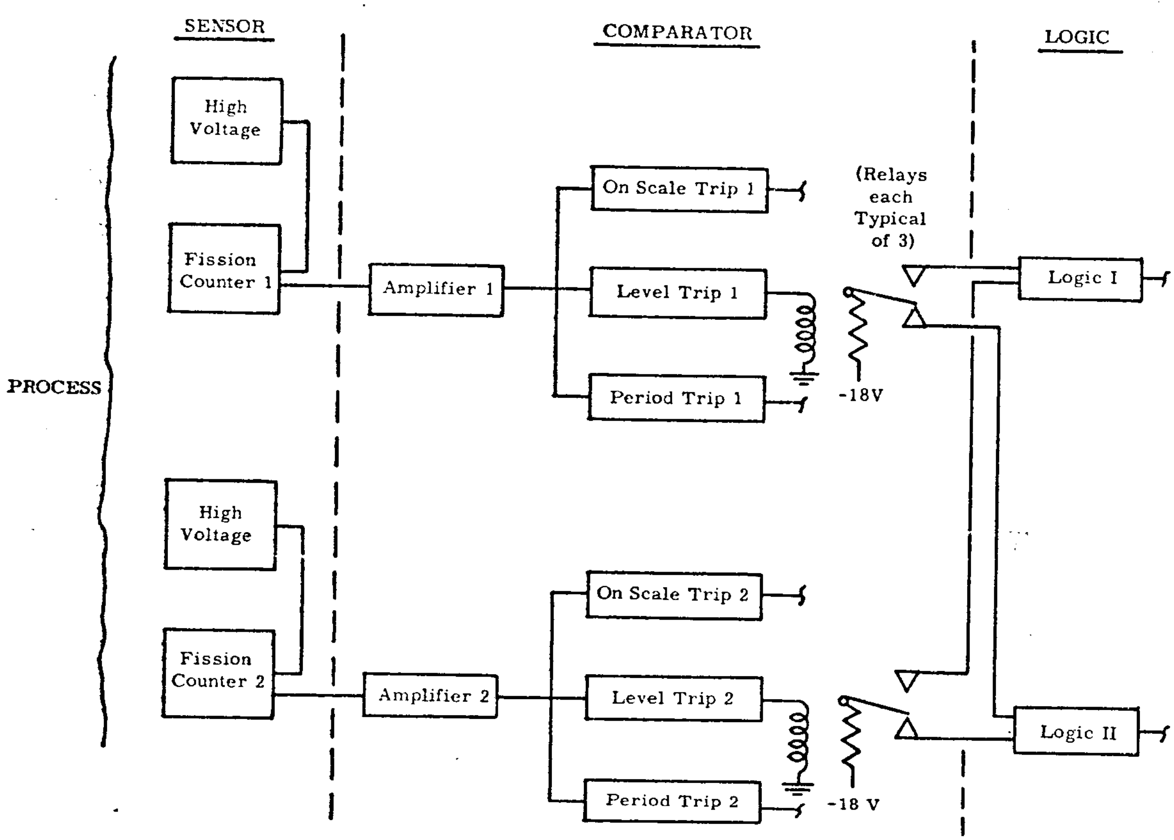


FIGURE F-4

Flux Monitor Channels

Flux Measurement System (Figure 5)

Neutron flux is measured from startup through operating levels by the PMACS, using entirely different equipment from that in the flux monitor channels. Two interlaced banks of BF_3 ionization chambers act as the sensors. The two banks are of different sensitivity, one having more chambers than the other. The output signal from each bank is a current proportional to neutron flux level. The signal is amplified and converted to a pulse signal whose frequency is proportional to input current. A counter receives and registers each pulse, and each 0.1 sec provides an input to the computer of the total count registered during the counting period. ~~The computer compares the count received with the expected count, based upon the last previous count and the mode of operation of the reactor. If the discrepancy between the two counter readings exceeds a programmed limit, the flux-normal signal from the computer is immediately switched to the trip level at both logic blocks. Following the direct comparison check of the raw counter reading, it is~~ These values are used by PMACS to compute flux level and period, and these are compared with programmed limits. Comparisons are also made between the values indicated by the two channels. Discrepancies in any of these comparisons result in switching of the flux-normal signal. Range changes and calibration checks are made ~~automatically.~~ by program This program of rapid and continuous sampling, testing and comparison of data and computer results provides the following monitoring features in addition to furnishing the precise experimental data:

immediate recognition of and action upon an off-standard ~~process condition (high priority comparison of raw counter readings),~~

- short interval testing of channels for freedom from component failure (comparison between non-identical channels).
- freedom from drift or surge of high voltage supplies (comparison between channels and with expected values).
- freedom from amplifier drift (comparison between channels; acceptable zero offset check).
- correct automatic by program ranging of amplifiers (comparison with expected values).
- correct amplifier calibration (zero offset and calibration checks acceptable).
- proper functioning of current-to-pulse converter, counters and computer (comparison between channels and with expected values).

That this testing and monitoring program is actually being accomplished is ensured by the last step of the programmed sequence, the emission of a nuclear check normal pulse. This pulse must be received by a detector each 0.1 sec or a separate nuclear check normal signal to the two logic blocks is switched.

As remarked above, the provision of the two similar but non-identical measurement channels, entirely dissimilar from the two monitoring channels, precludes the possibility that identical failures of the latter could permit a reactor power excursion.

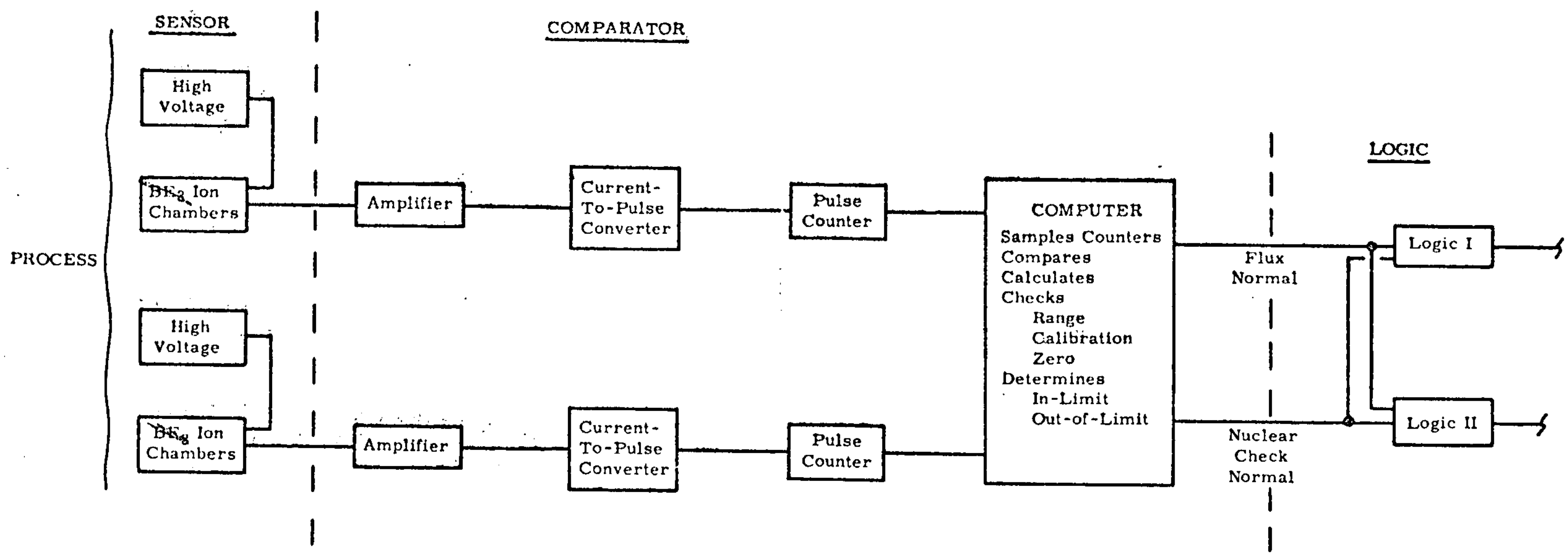


FIGURE F-5

Flux Measurement Channels

Process Environment System (Figure 6)

Process variables such as temperatures, pressures, flows, and gas quality are measured as frequently as required by PMACS, which functions also as a controller for the gas flow system, electrical heater system, and other systems. Depending upon the nature of the experiment, some variables that exceed pre-determined limits may be programmed to shut down the reactor by interrupting the environment-normal input to the logic blocks. Frequent sampling and computer evaluation of process trends give close process control. Extended discussion of this system is unnecessary here.

Logic Blocks

As has been shown, each logic block has inputs of several system-normal signals from the reactor process. Each also has a manual scram input and a lock-out channel which requires manual resetting of the safety circuit after even a momentary trip. Finally, each has an input of a pulsed signal, 1000 pulses per second. This signal must pass through both logic blocks and occur in their outputs to maintain power on the electromagnetic clutches of the safety and control rods.

The functioning of the logic blocks is more clearly understood if they are separated into two portions, Stage A and Stage B, as shown in Figure 7. The high level, period, and on-scale outputs of both flux monitor channels, and the 1000 pps signal are the inputs to Stage A of the logic block. As long as no trips are received on any of the six flux monitor inputs, the 1000 pps signal passes through to the output of Stage A. This signal then passes through a signal-conditioning inverter and appears as an input to Stage B of the logic block.

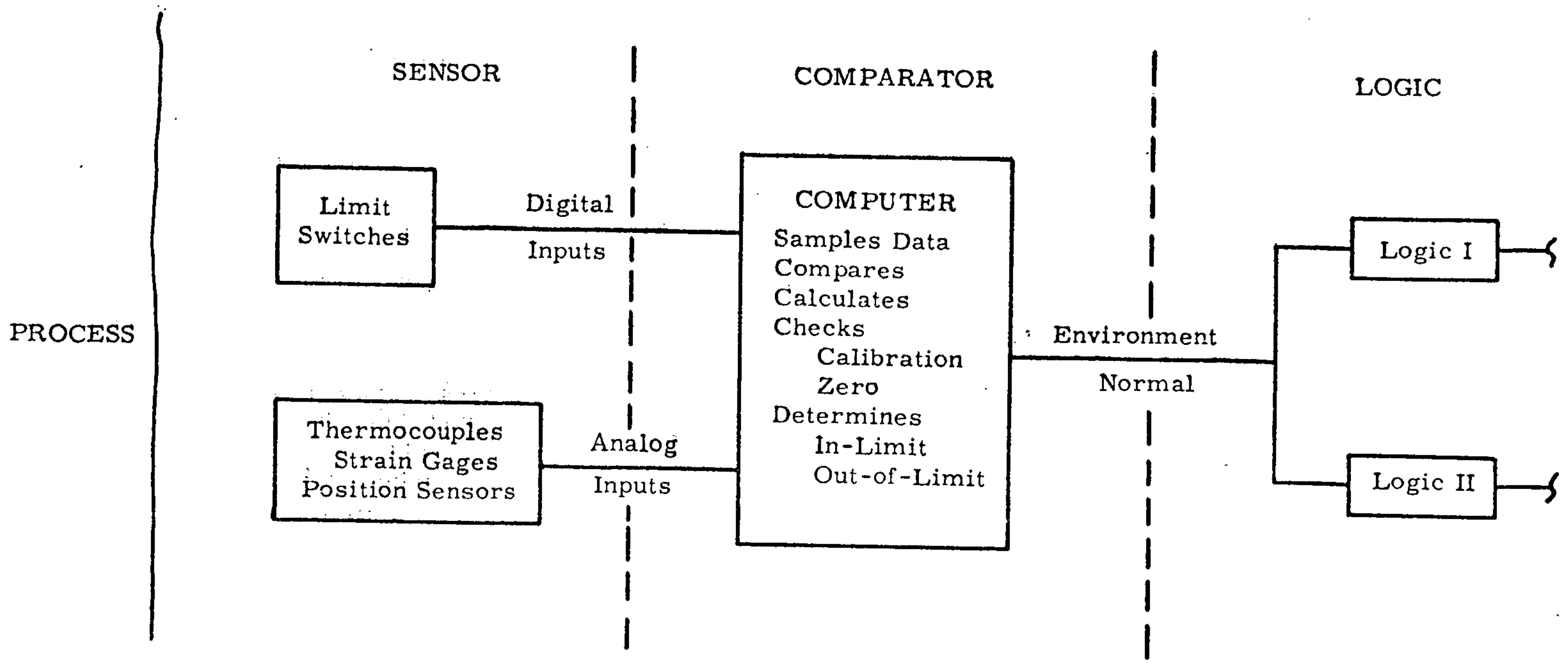


FIGURE F-6
Process Environment Channel

The remaining inputs to Stage B are the flux normal, nuclear check normal, and environment normal channels from PMACS, additional spare channels for signals presently unspecified but which might be needed for specific experiments, the manual scram input, and the lockout inputs. As long as no trips are received on any of these channels, the 1000 pps signal passes through to the output of Stage B of the logic block. It then passes through a second signal-conditioning inverter and amplifier to the logic block output. The 1000 pps signal is sensed by a pulse detector at the amplifier input, and if one pulse is missing from the pulse train, the output of the pulse detector ceases to furnish the normal signal to the lockout input of stage B of the logic block. This ensures that even a momentary trip will result in a reactor shutdown.

Though the two logic blocks are functionally identical, it should be emphasized that they are not identical in design and construction because the normal signal to one is zero volts whereas to the other it is negative. They are constructed on separate chassis, receive separate signals from the comparator block, and provide signals to different parts of the power supply block. Thus only independent, simultaneous, multiple failures in both logic blocks could cause an unsafe failure of the system.

Logic Stage IA, IIB

The logic blocks are composed of semiconductor diodes and resistors. In Logic Stage IA, shown in Figure 8, the normal inputs are zero volts and the trip inputs are negative, as shown in Figure 4. In Figure 8, the voltage at point 1 changes from zero to negative and back again as the 1000 pps signal is applied. As long as no trips appear on the other inputs, the anode of the 1000 pps

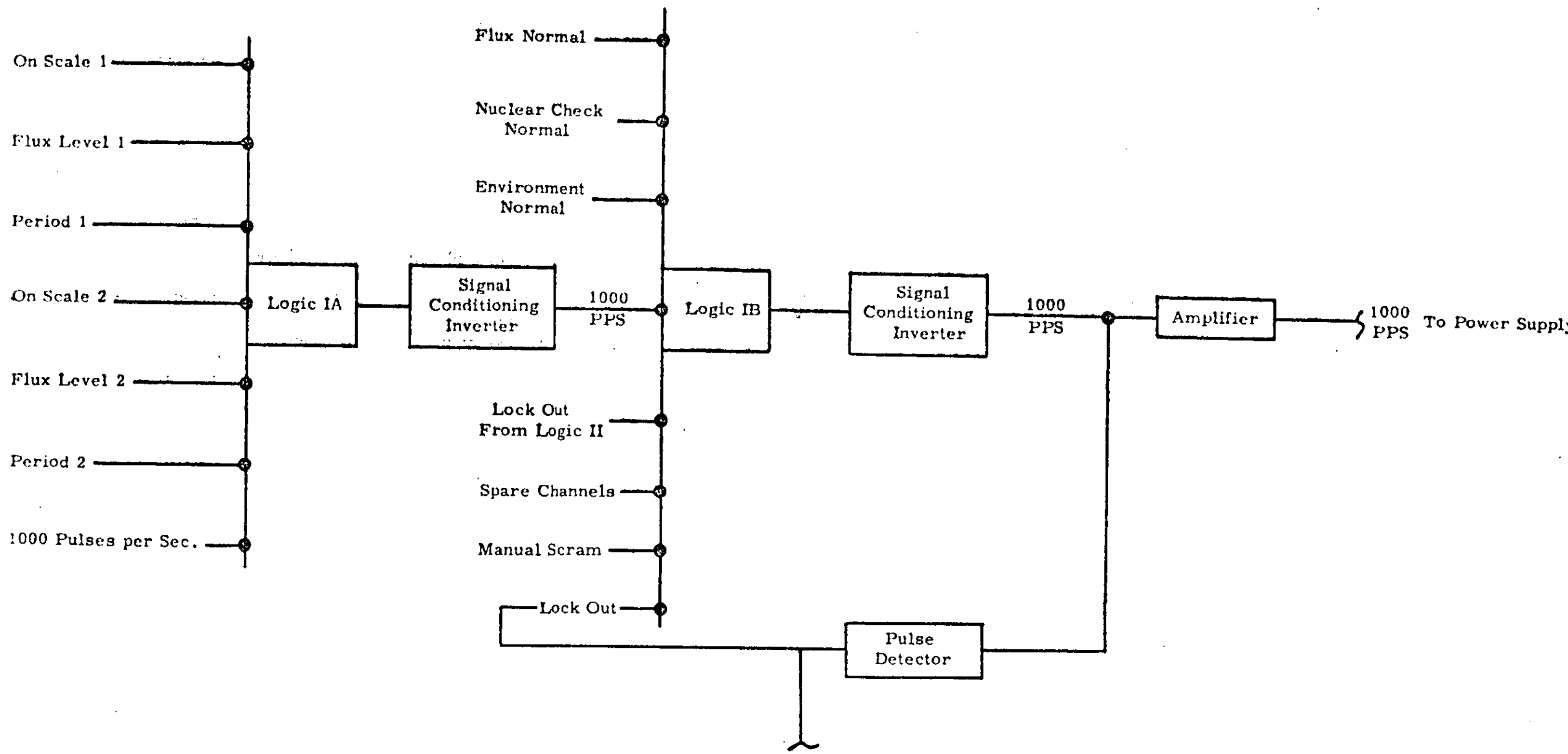


FIGURE F-7

Safety System Logic I Functional Diagram
(Logic II identical)

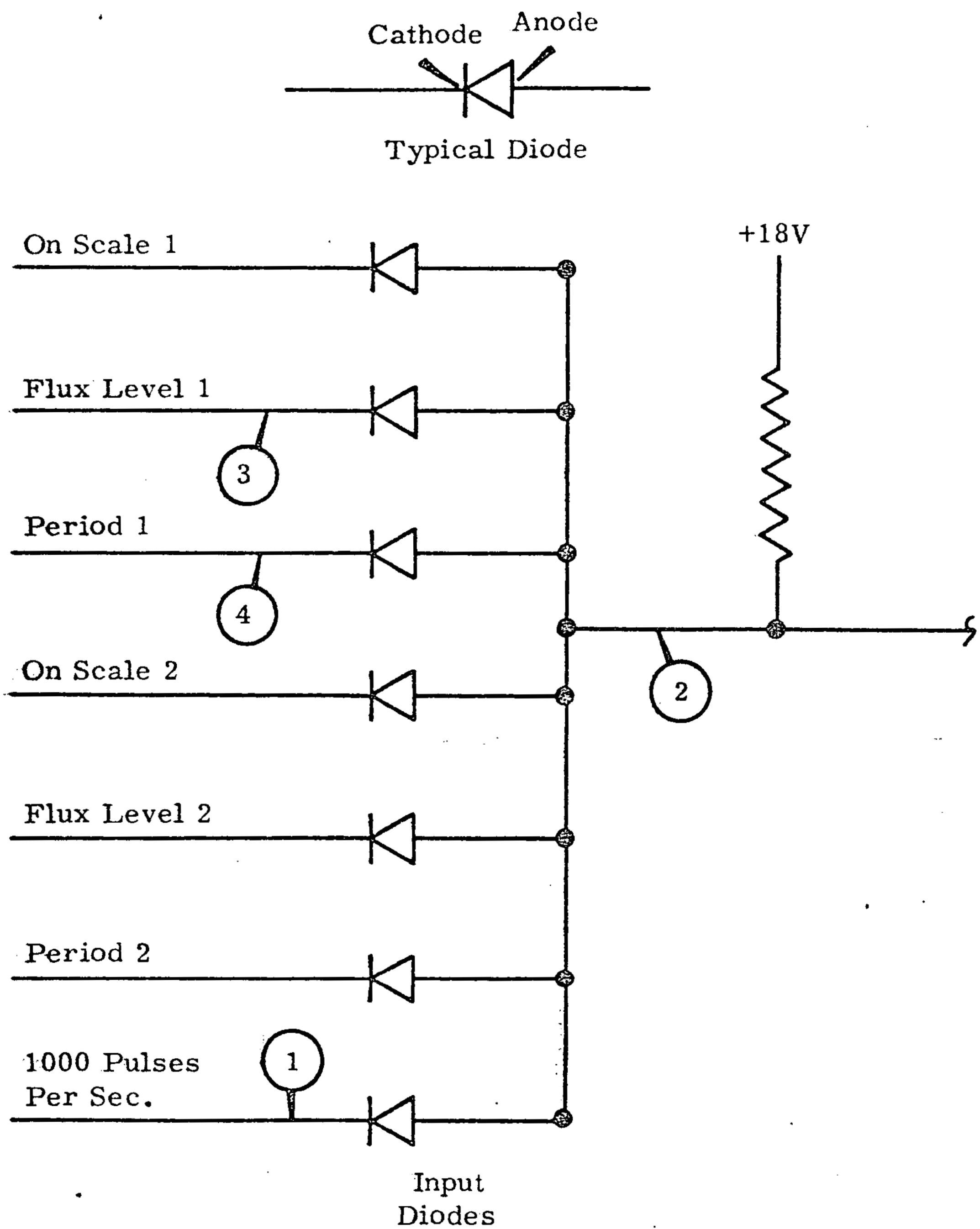


FIGURE F-8

Logic IA Schematic

(Logic IIB Similar)

input diode is always more positive than the cathode; the diode serves as a closed switch; and the 1000 pps signal passes, the values of the circuit components being so chosen that point 2 is always negative when point 1 is negative.

A trip of one of the relays of Figure 4 results in a negative voltage being applied to its respective diode in Figure 8, point 3 for example. The voltage at point 3 is the same as that at point 2 when point 1 is receiving a negative pulse. When point 1 is at zero volts, point 2 remains negative because the diode at point 3 serves as a closed switch. Therefore the anode of the 1000 pps diode is negative while the cathode is at zero volts, and the diode serves as an open switch. Therefore point 2 remains negative at all times, and the pulse signal does not appear at the output of Stage IA. Any single or multiple trip signal at the Stage IA inputs will thus interrupt the 1000 pps signal.

The functioning of Logic Stage IIB is the same as that of Stage IA, with the Stage B inputs, of course, in place of the Stage A inputs.

Logic Stage IIA, IB

In Logic Stage IIA, shown in Figure 9, the normal inputs are negative and the trip inputs are zero volts, as shown in Figure 4. With no trips at the other inputs, the 1000 pps signal applies a voltage at point 1 varying from negative to zero. When point 1 is negative it is at the same voltage level as point 2; when point 1 is at zero volts, the diode serves as a closed switch (anode positive with respect to cathode) and point 2 then switches to zero volts. The 1000 pps signal therefore passes the logic block.

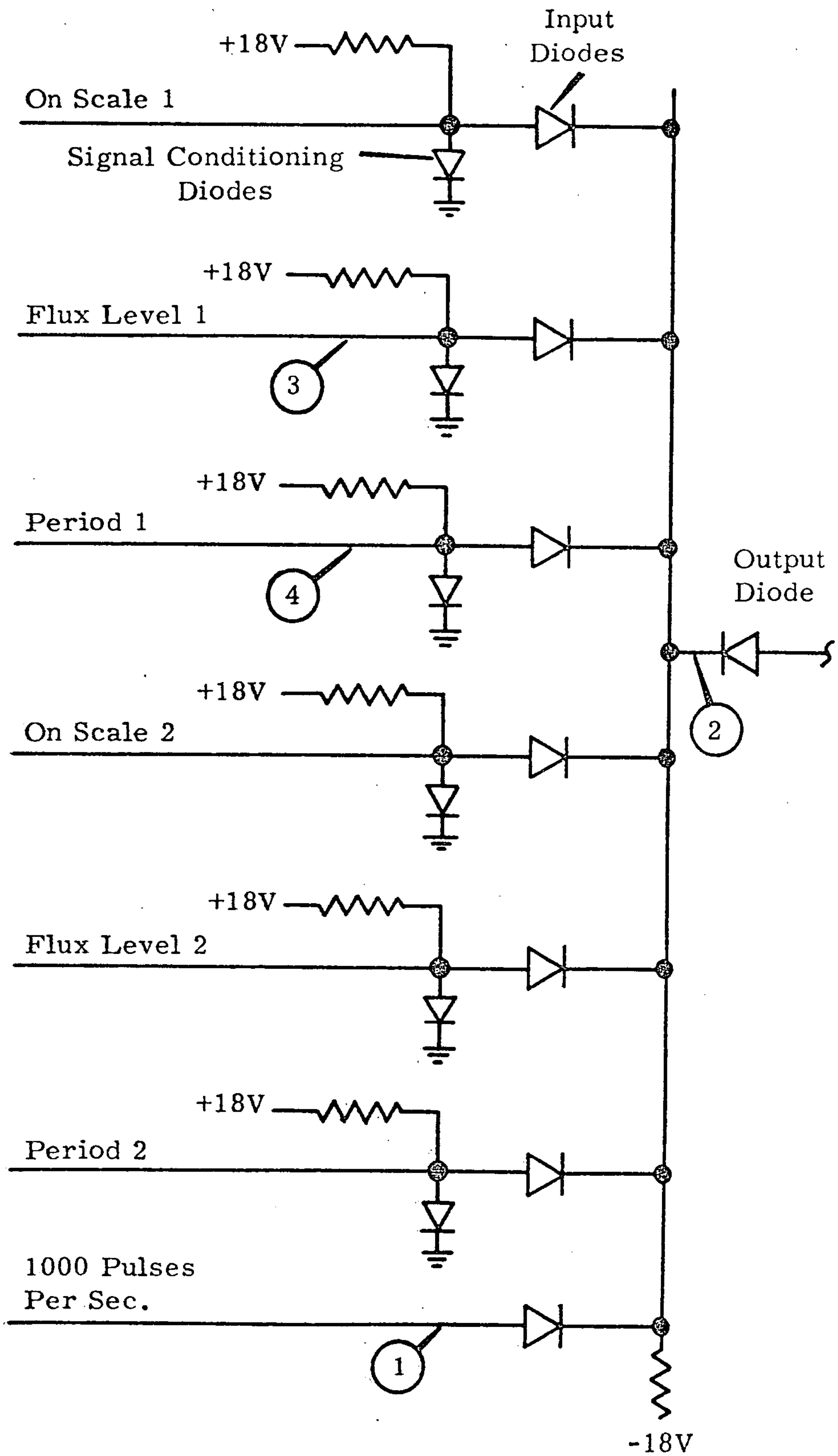


FIGURE F-9

Logic IIA Schematic

A trip of one of the relays of Figure 4 results in the normal negative voltage being removed from its respective diode in Figure 9, point 3 for example. The anode of the signal-conditioning diode at point 3 thereupon becomes more positive than the cathode and the diode serves as a closed switch. Point 3 therefore goes to zero volts (ground). The anode of the input diode is then positive with respect to its cathode, it serves as a closed switch, and point 2 drops to zero volts. Thus when the 1000 pps signal is at zero volts, points 1, 2 and 3 are all at the same voltage, namely zero. When the 1000 pps signal is negative, its diode serves as an open switch because the anode is more negative than the cathode, and point 2 remains at zero volts. The 1000 pps signal therefore does not pass the logic block.

The functioning of Logic Stage IB is the same as that of IIA, with the Stage B inputs in place of the Stage A inputs.

Signal-Conditioning Inverter (Figure 10)

The signal-conditioning inverter is a transistor which operates as shown in Figure 10. The normal output of the logic block is a 1000 pps signal varying from zero to about 6 volts negative. When the input to the transistor is at one of these levels, the output is at the opposite level. The phase of the signal is thus shifted 180°. Since power can be applied in the transistor, the signal is strengthened and the pulses are shaped for coupling to a later stage of the system. The inverter also prevents coupling of signals or events in later stages back into the stage which furnishes the input to the inverter.

The inverter cannot function by itself to give a 1000 pps output. Therefore if the 1000 pps input from the logic stage is interrupted, the output signal also will be interrupted. There is no unsafe mode of failure of the inverter, therefore.

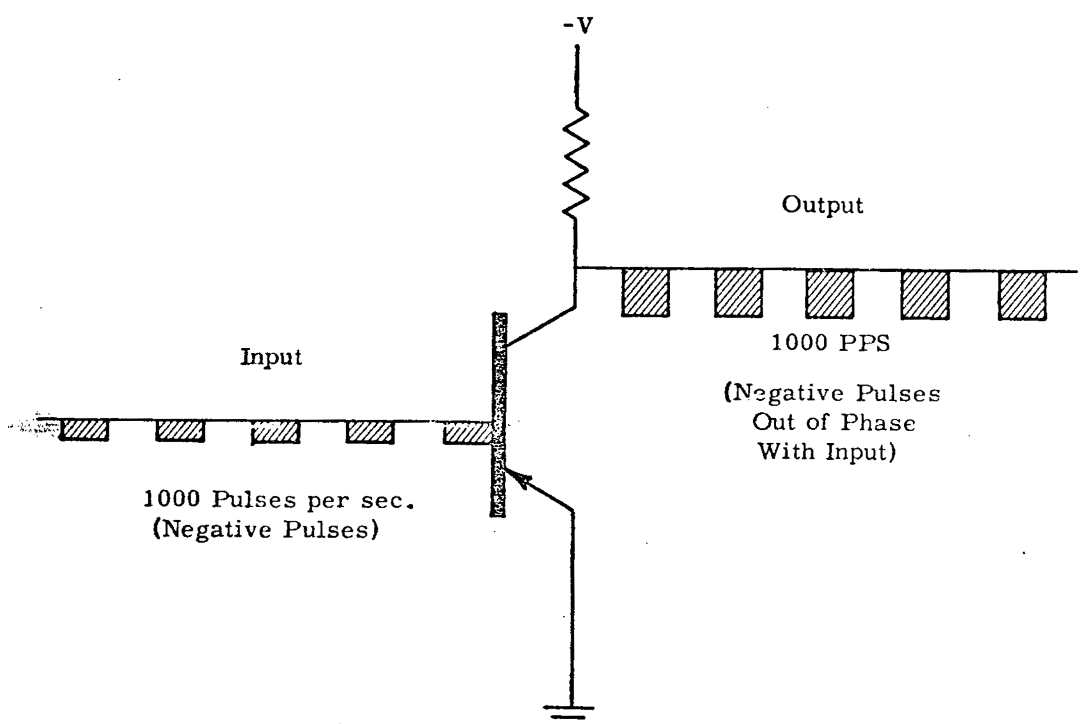


FIGURE F-10

Signal Conditioning Inverter

Amplifier; Pulse Detector

The amplifier shown in Figure 7 conditions the 1000 pps signal for input to the power supply block. The amplifier is incapable by itself of generating a 1000 pps signal, so any failures of the amplifier would be fail-safe.

The pulse detector also cannot generate a 1000 pps signal. It must be continuously supplied with a 1000 pps signal to maintain its steady output which is the lockout signal input to both Logic Stages IB and IIB. Any failures of the pulse detector would therefore be fail-safe.

Power Supply Block

Both 1000 pps outputs from the logic block are required to energize the power supply block, a schematic diagram of which is shown in Figure 11. The 1000 pps signals cause the power from the 28 volt dc source to appear at the primary of the output transformer as an alternating current; the corresponding ac from the secondary of the output transformer is rectified to form the dc power supply to the electromagnetic clutches. This design avoids the necessity of having to break the inductive electromagnetic clutch circuit by switching of any kind.

The normal operation of the power supply is as follows: The 1000 pps signal from Logic-I is applied to the primary of transformer T1, causing points A, B and C to alternate between a negative voltage and zero. The 1000 pps signal from Logic-II is 180° out of phase with that from Logic-I, it is applied to transformer T2 with the results that points D , E and F alternate between zero and a negative voltage, being always opposite to A, B and C. Noting the typical transistor shown in the insert, when negative voltage is applied at point 1,

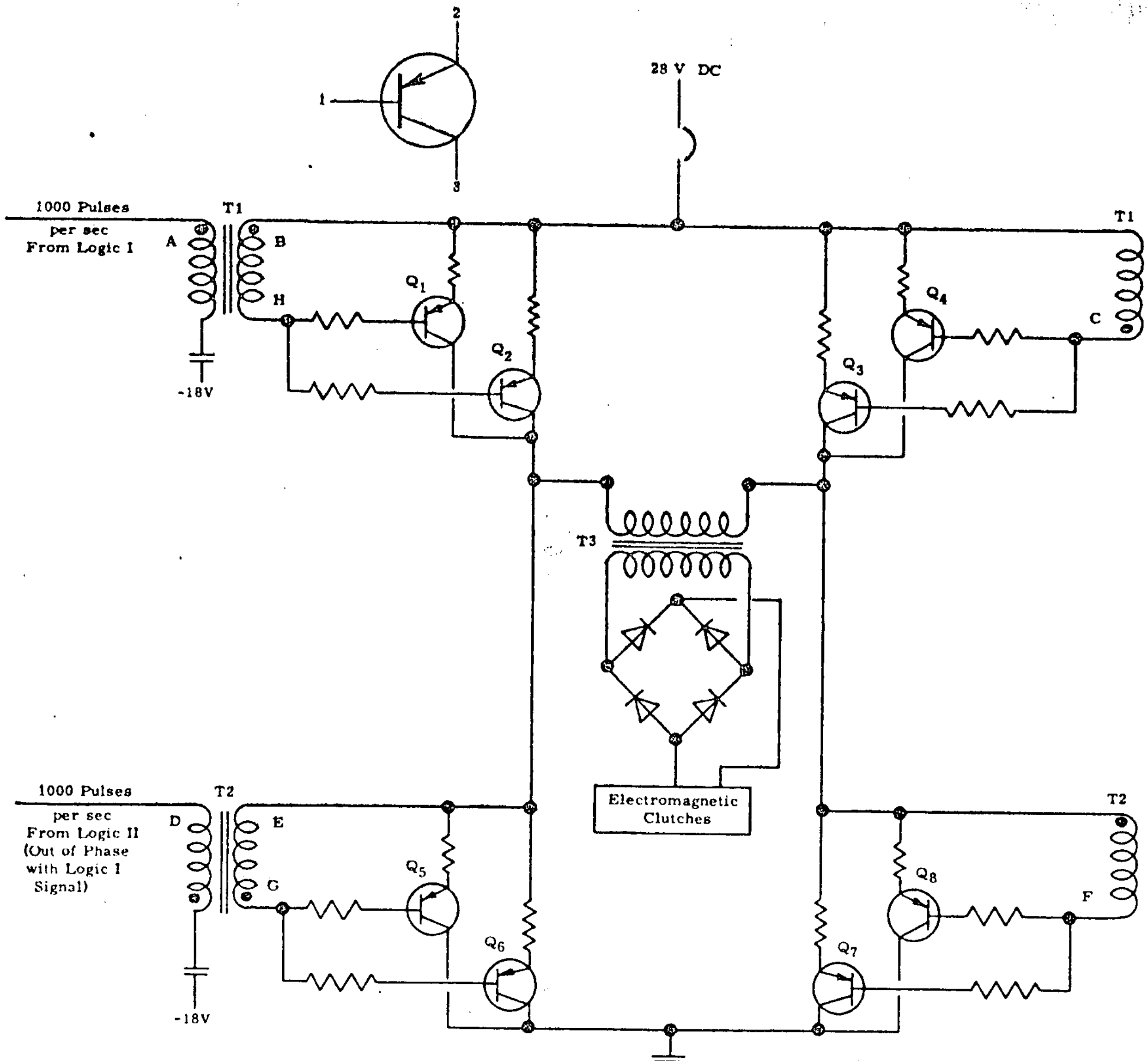


FIGURE F-11

Safety System Power Supply Schematic

current flows through the transistor from point 2 to point 3. A negative voltage at point C causes current flow in this manner through both Q_3 and Q_4. At the same time, point E is at zero volts and point G is negative. Current therefore flows through both Q_5 and Q_6. The circuit is thus complete from the 28 V dc supply through Q_3 and Q_4, through the primary of T3, through Q_5 and Q_6 to ground. On the opposite phase of the 1000 pps input signal, a zero voltage at B implies negative voltage at H, permitting Q_1 and Q_2 to conduct, and when B is zero F is negative, and Q_7 and Q_8 conduct. This closes the circuit from 28 V dc through the primary of T3 to ground in the opposite direction. Thus 28 V dc power is alternated in direction 1000 cycles per sec through the primary of transformer T3. This power is inductively coupled to the secondary of T3, and is rectified to form the dc holding current for the electromagnetic clutches.

The coupling of the two logic blocks through the lockout circuits ensures that both will shut down on any trip to either. Failure of this interlock is of as low probability as that of an unsafe failure of any input diode. However, should this interlock fail, a pulsating, rather than alternating, current would appear in the primary of T3 from the single logic block still passing the 1000 pps signal; the power at the secondary of the transformer in this case is insufficient to hold in the electromagnetic clutches.

Failure Modes and Consequences

Safety system failures have been discussed by J. E. Binns*. He placed system failures in two categories: failures of the first kind, which render the system incapable of taking necessary action, and failures of the second kind, which cause the system to take action although none is necessary. A third category may

*Ref. 7, page 12.1

also be added: component failures which do not cause the system to take action, yet do not render the system incapable of action on a trip signal. The goal of the design, construction, operation, and maintenance of a safety system is, of course, avoidance of failures of the first kind. The following analysis is intended to show how this goal has been attained for the HTLTR safety system.

In the detailed analysis it is helpful to classify failures as interconnection failures or component failures. The former are open circuits, short circuits, and single or multiple grounds that may result from loose or improper connections, damaged insulation, extraneous matter in the equipment, or poor maintenance techniques. Component failures may result from excessive operational and environmental stresses or from aging or wear.

Detailed analyses are presented only for the logic block. General analyses are sufficient for the other blocks of the safety system. This statement is justified in the discussion of each block.

Sensor-Comparator Block

The principles of redundancy, diversity, and frequent testing are relied upon for the proper functioning of this block. Considering the neutron flux instrumentation as the most essential for reactor safety, there are two channels of log count rate and period meters with relay trip outputs which are quite conventional and have proved reliable in many reactor-years of operation; there are also two channels of precise flux measuring instrumentation providing digital inputs to a stored program digital computer, which checks, compares, and calibrates the two channels continuously and provides a trip output to the logic block. One trip in any one of three modes from any one of the four channels provides an off-limit signal to the logic block. The two conventional channels are completely independent, but are identical in design and construction, so identical simultaneous failures in the two channels are not impossible, as experience has shown. The two digital channels are not completely independent, since both utilize the computer; they are, however, completely independent of the conventional channels and are completely unlike them in every respect, from the process sensor onward. Failures of the first kind in the digital channels are regarded as exceedingly unlikely, as failures of the computer occur either as absence of a signal or an egregiously erroneous signal, either of which would result in a trip. Furthermore, failure to process the flux information would be noted at once by failure to emit the nuclear check complete pulse. Multiple failures in all four channels resulting in a failure of the first kind of the entire system have thus been precluded by design.

Logic Block

This block has as input a 1000 pulses per sec signal. When this same signal appears at the output of the block as two amplified out-of-phase 1000 pps signals, the power supply to the electromagnetic clutches is energized. There is nothing in the internal circuitry of the logic block which could possibly generate the 1000 pps output signal. It is therefore sufficient to analyze the logic block in detail to determine whether multiple failures can give a path whereby the input signal can continue to pass through the block even though a trip signal has appeared at one of the inputs.

The contacts of the output relays of the comparator block of the neutron flux monitor channels are the inputs to Logic Stage A. Interconnection failures of these contacts will be analyzed in the discussion which follows.

(1) Grounding upper contact of a typical relay (Figure 4). (Includes grounds of any point of the Logic-I input circuits). Point 3 of Figure 8 would be at zero volts. The input diode would serve as a closed switch. Point 2 also would be at zero volts. Point 2 would be unable to follow the 1000 pps signal because of being grounded through the input diode. Therefore the 1000 pps signal would not pass.

(2) Open circuit from upper contact of a typical relay. (Includes open circuits at any point of the Logic-I input circuits).

This would result in an unsafe failure of Logic-I, since the negative trip signal would not go through and the short to ground as described in (1) would not occur. However, the negative normal signal would be removed from Logic-II and it would respond.

(3) Fused contacts of a typical relay.

This would result in an unsafe failure of the logic block, since the contacts would be unable to move from the normal position on de-energizing the relay coil. Reliance would be placed in this case on the proper functioning of flux monitor channel 2 or the proper functioning of the trip signals from the PMACS. The possibility of fused contacts of this relay is remote, since the current carried is small (about 6mA), the circuit contains no elements which could create surges on making or breaking the circuits, and there are no discoverable modes of failure of components which could load the relay heavily.

4

(4) Grounds and open circuits at the lower contact of a typical relay.

Similar analyses to those of (1) and (2) are obvious.

(5) Short circuit across the contacts of a typical relay.

In this case negative voltage would be applied to Logic-I in the same manner as a normal trip, resulting in a safe failure (reactor shutdown).

Interconnection failures elsewhere in the logic block are next considered.

(6) Grounds on the path of normal flow of the 1000 pps signal (points 1 and 2, Figure 8).

This would result in flow of the signal to ground and it would fail to pass, a safe failure.

(7) Open circuits on the path of normal flow of the 1000 pps signal.

This would obviously be a safe failure.

(8) Short circuits across two or more logic inputs (points 3 and 4, Figure 8).

A trip on either input would be recognized by both input diodes and the 1000 pps signal would not pass.

(9) Short circuit of the 1000 pps signal to one or more of the trip inputs (points 1 and 3, Figure 8).

This would result in a failure of the first kind of Logic-I. The 1000 pps signal at its input diode would alternate between a voltage more negative than usual and zero volts, and the diode would pass this signal. The signal conditioning inverter would restore the signal to normal and the Logic-I block would continue to provide an output. Logic-II, however, would cause a shutdown in the normal manner, and through the lockout coupling would shut down Logic-I as well.

Such a failure has been rendered improbable by design. The cable carrying the input 1000 pps signal does not lie side by side with any of the trip input cables. A short circuit could occur only at the logic block itself. At this point, component and input terminal separation minimizes the possibility of a short circuit. Multiple grounds would not function as a short circuit in this case since as was shown in (6) a ground on the path of the 1000 pps signal results in a safe failure.

Interconnection failures at other points in the logic block may be analyzed and shown to be equivalent in every case to one of the 9 discussed above.

Component failures are next considered. The diodes of which the logic circuits are composed can fail in only two ways — permanently open-circuited and permanently short-circuited. This results in 6 modes of failure of the logic block:

- open or short of the 1000 pps input diode
- open or short of an input diode on which a trip is received
- open or short of an input diode and receipt of a trip on another input.

These modes of failure and their consequences for Logic IA are shown in Table I:

TABLE I

MODES OF FAILURE OF LOGIC IA AND THEIR CONSEQUENCES

Mode of Failure	Tripped Diode	Normal Diode	1000 pps Diode	Failure of 2nd Kind	Failure of 1st Kind (1)
1	O*			No	Yes
2	S			No	No
3		O		No	No
4		S		No	No
5			O	Yes	-
6			S	No	Yes

*O - Open circuit failure

S - Short circuit failure

(1) Failure of first kind of Logic-I only. Logic-II would function to shut down the reactor.

Each mode of failure is discussed below, the entries in the last two columns of the table are justified, and when failure of the first kind is indicated, the alternative on which reliance is placed for reactor shutdown is stated. Where neither failure of the first or second kinds is indicated, the mode of failure is of the additional class mentioned in the opening discussion.

Mode 1: Consider as an example flux level channel 1 diode (point 3, Figure 8). With this diode open-circuited, a trip cannot get through on the channel and a failure of the first kind results. The system does not detect the fault and no failure of the second kind results. Reliance is placed on the proper functioning of Logic-II.

Mode 2: In the same location, with the diode short-circuited, the system does not detect the fault and no failure of the second kind results. The negative trip signal is directly connected to point 2. The 1000 pps diode functions as an open switch when the signal is at zero volts at point 1, and as a closed switch when it is negative. Point 2 therefore remains negative throughout the cycle and the 1000 pps signal does not pass. No failure of the first kind results.

Mode 3: Consider as an example an open circuit on period channel 1 diode and a trip on flux level channel 1. The open circuit cannot affect the functioning of the tripped channel, so no failure of either kind occurs.

Mode 4: A short on period channel 1 and a trip on flux level channel 1 would produce the same result, since the shorted channel ends in an open contact at the relay.

Mode 5: An open circuit of the 1000 pps diode would give the same result as any other open circuit in the 1000 pps signal--a failure of the second kind.

Mode 6: A short circuit of the 1000 pps diode would not be detected by the system, as the normal signal would appear at point 2. A trip input at point 3 causes the diode to function as a closed switch and applies a negative voltage at point 2. However, the 1000 pps diode being shorted, it could no longer act as an open switch on the zero volts part of the cycle. The circuit providing the 1000 pps signal has a low impedance compared to the negative trip input signal circuit. Therefore the voltage at point 2 would not remain negative during the zero voltage portion of the pulse, and the normal signal would pass to the inverter, resulting in a failure of the first kind. Reliance is placed on the proper functioning of Logic-II.

The failure mode analysis of Logic IA applies also to Logic IIB, with the exception that in Logic IA all trip signals result from contact closures, whereas in Logic IIB only the manual scram trip results from a contact closure. All other trip signals result from voltage level changes produced by the computer. These voltage level changes are low impedance connections to a negative bus or a ground bus, respectively, and thus shorting of an input diode cannot have the effect of a connection to an open relay contact. Failure modes and their consequences for Logic IIB are as shown in Table II.

TABLE II

MODES OF FAILURE OF LOGIC IIB AND THEIR CONSEQUENCES

Mode of Failure	Manual Scram Diode		Other Input Diodes		1000 pps Diode	Failure of 2nd Kind	Failure of 1st Kind(1)
	Tripped	Not Tripped	Tripped	Not Tripped			
1	O					No	Yes
2	S					No	No
3		O				No	No
4		S				No	No
5			O			No	Yes
6			S			Yes	-
7				O		No	No
8				S		Yes	-
9					O	Yes	-
10					S	No	Yes

(1) Failure of first kind of Logic-II only. Logic-I would function to shut down the reactor.

In failure modes 1 and 2, the manual scram diode is considered to have failed in the manner indicated, and then a scram signal from the manual scram circuit is received. In failure modes 3 and 4, the manual scram diode is considered to have failed, and then a trip of one of the other input diodes is received. In failure modes 5 and 6, an input diode is considered to have failed, followed by a trip on that channel. In failure modes 7 and 8, an input diode is considered to have failed, followed by a trip on another channel.

In failure mode 1, reliance is placed on both the proper functioning of Logic-I and the additional contact in the manual scram button which breaks the 120 V ac supply to the 28 V dc power supply circuit. In failure modes 5 and 10, reliance is placed on the proper functioning of Logic-I.

The failure mode analysis of Logic IA applies also to Logic IIA, with the exception that in Logic IIA failure of the signal-conditioning diodes could also occur. A short circuit of one of these diodes would result in a low impedance ground of the input signal. This would result in a scram, a failure of the second kind. An open circuit of a signal-conditioning diode results in the positive bias voltage being transmitted through the input diode in the event of a trip, where because of circuit characteristics it produces a low positive voltage at point 2. This results in the cathode of the 1000 pps input diode always being more positive than the anode, the diode thus functioning as an open switch and stopping passage of the 1000 pps signal. Thus the system functions normally. The analysis of the remaining modes of failure is similar to those above. The modes of failure of Logic IB are also similar to those of Logic IIA, again with the exception noted for IIB. The modes and consequences of failure for Logic IIA are shown in Table III. The extension to Logic IB is obvious.

TABLE III

MODES OF FAILURE OF LOGIC IIA AND THEIR CONSEQUENCES

Mode of Failure	Conditioning Diode	Tripped Diode	Normal Diode	1000 pps Diode	Failure of 2nd Kind	Failure of 1st Kind (1)
1	O				No	No
2	S				Yes	--
3		O			No	Yes
4		S			No	No
5			O		No	No
6			S		No	No
7				O	Yes	--
8				S	No	No

(1) Failure of first kind of Logic II only. Logic I would function to shut down the reactor.

Power Supply Block

Since the power supply block depends for its output on the active functioning of the switching transistors (Q, Figure 11), which are driven by the inductively-coupled out-of-phase signals from both logic blocks, and since the signals by themselves could not provide sufficient power to the electromagnetic clutches, it is apparent that open or short circuits or grounds in the transformers to which the signals are coupled or in the switching transistors could not cause a failure of the first kind. The only condition which could cause a failure of the first kind in this block would be a multiple interconnection failure which would directly connect the 28 V dc power supply to the electromagnetic clutches. This is precluded by the design of the system. The 28 V input terminals and the output terminals are well separated, and the system is protected against interconnections from multiple grounds by a ground detector in the electromagnetic clutch circuit.

RELIABILITY COMPUTATION

Failures in a single logic block do not cause a system failure. Since certain multiple failures could cause a system failure, the probability of such an occurrence has been calculated. We consider here the simultaneous failures of components in both flux monitor channels which prevent or preclude a safety system trip when the high flux level trip point is exceeded .

When electronic components are first installed, failures occur at a relatively high rate from production, test, or assembly faults. The failure rate quickly decreases and remains fairly constant for the useful life of the system. At the end of this period, the failure rate begins to increase as wear-out failures occur. The random and fairly constant occurrence of failures during the useful life period makes possible the use of statistical models to represent reliability. Reliability, the probability that a system will perform satisfactorily for at least a given period of time when used under stated conditions, is related to the component failure rate λ and time t by[9]

$$R(t) = e^{-\lambda t} \tag{1}$$

If an event occurs in s out of a total of n trials and does not occur in the remaining (n-s) trials, the probability of occurrence is the ratio s/n as the number of trials becomes large. Let the probability of the occurrence of an event E be P(E) and of the non-occurrence of the event $P(\overline{E})$. The sum of these probabilities is

$$P(E) + P(\overline{E}) = 1. \tag{2}$$

Two events A and B are independent, if the occurrence or non-occurrence of A has no dependence upon the occurrence or non-occurrence of B, and vice versa. The probability P(AB) that both events occur is

$$P(AB) = P(A) \cdot P(B). \tag{3}$$

Generally, then, for $A_1, A_2, A_3, \ldots A_i$ independent events the probability that all i events occur is given by

$$P(A_1 A_2 A_3 \cdots A_i) = P(A_1) \cdot P(A_2) \cdot P(A_3) \cdots P(A_i) \tag{4}$$

The probability that either event A or event B or both events occurs, for events which are not mutually exclusive, may be found from any of the following:

$$\begin{aligned} P(A \text{ or } B) &= P(A) \cdot P(\overline{B}) + P(B) \cdot P(\overline{A}) + P(A) \cdot P(B) \\ &= P(A) + P(B) - P(A) \cdot P(B) \qquad (5) \\ &= 1 - P(\overline{A}\,\overline{B}) \end{aligned}$$

Since the probability of satisfactory performance is R(t), the probability P(t) of failure is

$$P(t) = 1 - e^{-\lambda t} \tag{6}$$

for $\lambda t \ll 1$, the approximation

$$P(t) \cong \lambda t \tag{7}$$

is valid.

As an example consider a simple series circuit comprising a battery, a switch, and a lamp[9]. For the light to turn on, all three components must be in operating condition. If a failure is represented by 0 and the operating condition by 1, a functional truth table may be prepared showing the eight possible combinations of failure and operating condition:

TABLE IV

FUNCTIONAL TRUTH TABLE FOR SIMPLE CIRCUIT

Combination	Battery	Switch	Lamp	Circuit Failure
1	0	0	0	Yes
2	0	0	1	Yes
3	0	1	0	Yes
4	0	1	1	Yes
5	1	0	0	Yes
6	1	0	1	Yes
7	1	1	0	Yes
8	1	1	1	No

Assume that large samples of the batteries, switches and lamps have been tested and their useful-life failure rates determined. The probability that a circuit element will perform satisfactorily for at least a time t, its reliability, may be computed using Equation 1.

The reliabilities of the several components may then be combined to give the reliability of the entire circuit as in Equation 4, remembering that combination 8 is the only one giving successful operation of the entire circuit. The component reliabilities are given in Table V.

TABLE V

COMPONENT FAILURE RATES AND RELIABILITIES

Functional Block	Measured Failure Rate	Reliability
A (battery)	λ_A	$e^{-\lambda_A t}$
B (switch)	λ_B	$e^{-\lambda_B t}$
C (lamp)	λ_C	$e^{-\lambda_C t}$

The reliability of the circuit is then given by

$$R^*(t) = \left(e^{-\lambda_A t}\right)\left(e^{-\lambda_B t}\right)\left(e^{-\lambda_C t}\right)$$

$$= e^{-(\lambda_A + \lambda_B + \lambda_C)t}$$

The reliability of more complex systems may be similarly calculated when the failure rates of the components and the modes of failure of the system are known.

These methods will now be used to compute the reliability of the logic block in responding to high flux level signals from both flux monitor channels. These input signals are applied to Logic IA and Logic IIA as described above. The truth table for Logic IA failure modes developed from Table I is shown in Table VI.

TABLE VI

FUNCTIONAL TRUTH TABLE FOR LOGIC IA

Combination	Channel 1 Diode*	Channel 2 Diode*	1000 pps Diode Open	1000 pps Diode Short	Failure of 2nd Kind	Failure of 1st Kind
1	0	0	0		Yes	
2	0	0	1			Yes
3	0	1	0		Yes	
4	0	1	1			
5	1	0	0		Yes	
6	1	0	1			
7	1	1	0		Yes	
8	1	1	1			
9	0	0		0		Yes
10	0	0		1		Yes
11	0	1		0		Yes
12	0	1		1		
13	1	0		0		Yes
14	1	0		1		
15	1	1		0		Yes
16	1	1		1		

* 0 represents an open circuit failure, 1 represents all other situations (normal operation or short-circuit failure) since these do not cause failure of the first kind.

The probability expressions that describe the combinations yielding failures of the first kind are tabulated below in a notation similar to that used for the simple example:

TABLE VII

PROBABILITY EXPRESSIONS FOR LOGIC IA FAILURES OF THE FIRST KIND

Combination	Probability Expression
2	$P(t,0,C_1)\cdot P(t,0,C_2)\cdot[1-P(t,0,SN)]$
9	$P(t,0,C_1)\cdot P(t,0,C_2)\cdot P(t,S,SN)$
10	$P(t,0,C_1)\cdot P(t,0,C_2)\cdot[1-P(t,S,SN)]$
11	$P(t,0,C_1)\cdot[1-P(t,0,C_2)]\cdot P(t,S,SN)$
13	$[1-P(t,0,C_1)]\cdot P(t,0,C_2)\cdot P(t,S,SN)$
15	$[1-P(t,0,C_1)]\cdot[1-P(t,0,C_2)]\cdot P(t,S,SN)$

As an example of interpretation of the notation consider combination 9; this is the probability of coexistence of an open circuit of both channel 1 and channel 2 input diodes and a short circuit of the system-normal (1000 pps) input diode.

Combining the expressions in Table VII, the probability of a failure of the first kind of Logic IA is

$$P(t,1st,IA) = P(t,0,C_1)\cdot P(t,0,C_2)\cdot[1-P(t,0,SN)]\cdot[1-P(t,S,SN)] + P(t,S,SN) \qquad (8)$$

Using the information from Table III, we find the probability of a failure of the first kind of Logic IIA is

$$P(t,1st,IIA) = P(t,0,C_1)\cdot P(t,0,C_2)\cdot[1-P(t,S,SC_1)]\cdot[1-P(t,S,SC_2)]\cdot \left\{1-P(t,S,SC_i)\cdot[1-P(t,0,N_i)]\right\}^4\cdot[1-P(t,0,SN)] \qquad (9)$$

In this expression, SC_1 and SC_2 refer to the signal-conditioning diodes associated with the input diodes for channel 1 and channel 2, respectively. SC_i and N_i refer to the remaining signal conditioning and input diode combinations.

The probability of a failure of the first kind of the entire logic block with both high flux level scram signals present is the combined probability for coexistent failure of both Logic I and Logic II:

$$P(t,1st,Logic) = P(t,1st,IA) \cdot P(t,1st,IIA) \tag{10}$$

Failure Rate Data

To compute the numerical value of P(t,1st, Logic), data for the failure rate, λ, and for the time under consideration must be supplied. Failure rate experience for semiconductor diodes used in circuits similar to the logic circuits is given in Table VIII.

TABLE VIII

SEMICONDUCTOR DIODE FAILURE EXPERIENCE

Source Of Data	Diode Population	Operating Time, Hr	Diode Failures	Failure Rate, %/1000 Hr
1.1	13,080	1,400	0	0.0164*
1.2	13,393	2,000	0	0.0112*
1.3	12,700	1,373	0	0.0172*
1.4	12,700	1,121	0	0.0210*
2	3,000	13,140	3 (+)	0.0197*
3	8,000	7,500	1 (+)	0.0079*
4	167,958	1,020	244	0.0142
5				0.003
6				0.01

* 95% confidence level [1].
(+) Includes failures of other components as well as diodes. Assumed all diode failures in computing the failure rate.

Source 1: Four systems manufactured by the vendor of PMACS and of the safety system logic and power supply. Reference 2.

Source 2: A digital computer in the NRU computer control experiment being conducted by AECL. Reference 3.

Source 3: A chemical process control digital computer installation. Reference 4.

Source 4: A data-processing system. Reference 5.

Source 5: A military handbook value. Reference 8.

Source 6: An industrial handbook value. Reference 6.

The distribution[6] of diode failures is about 10% open circuit failure, 30% short circuit failure, and 60% drift failure. Since drift failure does not affect the performance of a diode as a switching element, the drift failures could be neglected. For computation convenience, they were lumped with the open and short failures and these were set equal at 50% each.

The failure rate chosen was that of source 1.2, the system of the PMACS vendor having the most test experience. Distributing this rate equally to opens and shorts and rounding up gives the failure rate λ = 0.006% per 1000 hr to be used in Equation (10).

The time t under consideration is the time interval between system tests[7]. Since the system is tested before experimentation is begun each day, or once during each 8 hour shift if experimentation is proceeding continuously, the longest time between system tests is 16 hours.

Probability Computation

The exponent in the exponential terms, λt, is

$$\left(\frac{0.006\%}{1000\text{ hr}}\right)(16\text{ hr}) = 9.6 \times 10^{-7} \text{ or approximately } 10^{-6}$$

and the approximation of Equation (7) is valid. Substituting in Equation (8) we find

$$\begin{aligned} P(t,1st,IA) &\cong (\lambda t)^2 (1-\lambda t)^2 + \lambda t \\ &\cong \lambda t \\ &\cong 10^{-6} \end{aligned}$$

From Equation (9), we find

$$P(t,1st,IIA) \cong (\lambda t)^2(1-\lambda t)^3[1-\lambda t + (\lambda t)^2]^4$$
$$\cong 10^{-12}$$

From Equation (10) the probability of failure of both Logic I and Logic II is

$$P(t,1st,Logic) \cong (10^{-6})(10^{-12})$$
$$\cong 10^{-18}$$

The complete failure of the safety system is, of course, of even lower probability because of the backup provided by the flux measurement system.

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