BNWL-CC-225 REV1

~~Rev 1~~

February, 1967

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FINAL SAFEGUARDS ANALYSIS

HIGH TEMPERATURE LATTICE TEST REACTOR

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January, 1966

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MAY 31 1968			

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- 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 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 have 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.

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 (west) face, for complete access to the reactor core for alterations, and a smaller door on the rear (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. A drawing of the water-cooled joint is included in Appendix G, Figure G-6.

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 R.1.

The pressure in the gas circulation system is controlled at the blower inlet, the low pressure point in the system. At this point it is maintained slightly above atmospheric pressure, about 14.9 psia, by the purge and makeup flows. A purge flow is taken direct from the reactor outlet at about 15.3 psia, via the purge cooler and monitors, through a flow controller receiving signals from PMACS, a 3-way diversion valve, and filters to the stack. The purge flow control is set to maintain a preset rate. The 3-way valve then diverts the purge flow to recycle unless the gas composition is out of limit, thus conserving makeup nitrogen consumption.

The flow of makeup nitrogen is controlled by PMACS to maintain the system pressure accurately at the control point. Precise pressure control is required for experimental precision; it is easily attained in this system because the pressure can change only very slowly by the action of the controls. The volume of the system is large, and the input and output flows are small, limited by pipe sizes and pressure differentials. Control to the precision of the measuring device is thus readily achieved by restricting the rate of change of the control position to low values.

The design pressure of the reactor gas envelope and all parts of the circulating gas system is 5 psig and full vacuum. The operating pressure varies from about 15.0 psia (0.3 psig) at the blower inlet to about 18.0 psia (3.3 psig) at the blower discharge. The reactor gas envelope operating pressure will be controlled accurately at or near 15.3 psia (0.6 psig). The two rupture disks are designed to rupture at 5 psig.

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 through a small universal joint which eases stresses

on the graphite caused by differential thermal expansion of the shell and graphite stack. 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. (The rod is shown fully open, i.e. in its most reactive position, in the figure.) Graphite cylinders about the same size

FIGURE V-1

5.7a R.1

FIGURE V-2a

Control Rod: Drive Mechanism

Control Rod: In-Reactor Portion

FIGURE V-2b

they are connected by the latch. The pulses are counted and converted to exact rod position by the computer. The pulses are produced by PMACS at a maximum rate of 125/sec. The maximum rate of rod movement is therefore 0.03125 in/sec. With all rods ganged together, the maximum rate of reactivity insertion is 2 to 3¢/sec.

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. Dennergizing 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 shut-down 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 safety rods is provided. All four rods drop into the core together on a reactor scram. Single rods may be dropped on tests. The design of the rod and its operating mechanism is shown in Figure V-3; details of the design and operation follow:

The control element is a blade, about 7 inches wide by 1/2 inch thick overall, and about 138 inches long. In the core, it rides freely in a graphite-lined slot 8 inches by 1-1/2 inches in cross-section, in the fixed driver zone. Thus binding or interference with its free fall is negligible. The basic structure of the blade is a welded ladder-like frame of 1/2 inch TD-nickel rod. Into the frame are inserted neutron poison plates 6 inches square and 1/16 inch thick, covered and held in place with 0.030 inch TD-nickel sheet assembled to the ladder frame with TD-nickel rivets. A stainless steel support assembly, clamped to the top of the frame, couples the blade to the 1/8 inch stainless steel aircraft cable which raises the rod from its position of maximum negative reactivity. The rod structure is flexible, but amply strong in service, while suspended from its cable support.

The neutron poison material is a cermet of 4% gadolinia, 4% europia, and 92% nickel by weight. Its physical properties, both at ordinary and at operating temperature, are similar to those of nickel. Specimens of the material were exposed in contact with graphite in slightly moist nitrogen to temperatures of 1000 C for over 200 hr, with little corrosion

or effects on the cold physical properties. The adequacy of the material for the physical service conditions is thus demonstrated. The reactivity worth was determined to be equivalent to that of sheet cadmium and blocks of B_4C -graphite.

The control blade is lifted, lowered, and dropped if a scram occurs, by the mechanism shown in Fig. V-3. The rod lifting cable winds on a grooved drum to prevent overlap of the cable lays. The drum is mounted on the main drive shaft, which is driven through a magnetic clutch by a geared stepping motor in raising or lowering the rod under power. A disk brake is attached to the drum to decelerate the rod near the end of its travel in free fall in a scram.

The operation of the rod in a scram cycle is as follows: The cycle begins with all parts in the position shown in the figure, with the rod fully withdrawn. The scram signal interrupts the current to the magnetic clutch, releasing it. (As discussed in the description of the safety circuit, operator action is required to restore current to the clutch.) The rod drops of its own weight, accelerating at the same time the drum, the brake disk, and the gear train to the cylindrical cam. When the bottom end of the rod is 12 inches above the end of travel, the rotation of the cam releases the cam follower, permitting the brake arms to clamp together about the hinge pin under the force of the spring. The brake arms are attached to the frame only at the pivot pin; small rotations about this pin equalize the pressure applied to the brake shoes. The braking force is transmitted to the frame through the stiffness of the brake arms and the shear strength of the pivot pin.

To withdraw the rod, it is necessary (1) that the safety circuit be made up to provide current to the magnetic clutch, and also to apply nitrogen pressure to the cylinder to release the brake, (2) that PMACS is operating to provide current pulses to the stepping motor, and (3) that the operator has selected and signaled to PMACS a rod or rods to be withdrawn. The same conditions must be met to lower the rods under power.

This mechanism permits effective scram action at any point in the rod travel, not just at the fully-raised position. If a scram were to occur at any position above which the cam releases the brake, the rod will fall in a normal manner as just described, except that the kinetic energy would be less than normal, and the rod would stop above the end of travel. At 12 inches above the end of travel, the point of application of the brake, the lower end of the rod is about 1 ft 4 in. above the bottom of the reactor, and the rod is nearly 100% effective. If a scram were to occur below the brake closing position, with the brake held open by the nitrogen cylinder, the rod would fall freely about 12 inches with no damage, and would contact a solid stop with poison still left to enter the core at the top. If the brake were to jam in the open position, damage to the rod might occur in a scram, but the entire negative reactivity insertion would be made. Jamming of the brake in such a manner as to prevent the fall of the rod in any position above the release point of the cam is not possible, since the cam and follower present a solid mechanical interference against such an event.

A prototype assembly has been tested in over 100 drops at room temperature. A short assembly has been exposed to operating temperatures in the large mockup, and has survived several drops with no effect. Confidence in the performance of the VSR's has thus been established. The time required for full insertion of the rods including instrument delays, magnet release time, and brake action 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 58 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 ac electric power to the 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:

Neutron flux channel down scale (low level trip)

Neutron flux channel up scale (high level trip)

Short reactor period

PMACS self-audit signal failure (10 audits per second)

Movement of access doors to reactor enclosure

Each of the first three shutdown functions is provided by two duplicate channels, each consisting of a fission detector and an ion chamber, amplifiers and period meters, an overlap circuit, a high voltage power supply, and a recorder. If either channel produces an off-normal signal for any of the three functions, the reactor is scrammed automatically. The system is so designed 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 measurement signals for nuclear experimentation, is arranged in dual channel configuration to provide two signal channels to PMACS. In response to off-normal signal levels from either channel, PMACS will trip the safety circuit, at the levels of downscale (low level), upscale (high level), and

short period corresponding to those of the fission chambers. Thus two entirely separate systems of different design have been provided for monitoring and shut down control of the neutron flux level.

Certain non-reactor conditions and off-standard reactor process conditions may be programmed in PMACS to cause an automatic non-scrum reactor shutdown, set-back of the reactor heaters, and shutdown of the circulating gas blower. Some of these are the following:

Nuclear incident monitor

Loss of pressure in gas system

Very high impurity level in gas system

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

The effective operation of the safety circuit is regularly tested, to further ensure reliability. The operation and settings of the trip circuit in PMACS is checked as a part of the initialization program, by which PMACS is placed in a state of readiness at the start of the operating shift. This is done in three steps: (1) The program "Checkerboard" is called. This program is furnished by the Digital Equipment Corporation as an all-purpose checkout of the PDP-8 computer operation. It operates all parts of the basic computer and provides the operator with a series of statements of correct functioning via the Teletype. (2) The program "PNIT" is called. This is our initialization program for the Astrodata equipment and for the memory fields of the computer. This checks PMACS nuclear instruments for on-scale readings, verifies all interlocks, verifies the normal position of all safety and control rods,

and starts the tenth-second pulses to the safety circuit. It then makes up the safety circuit in the selected mode. (3) The program "FLUX" is called. This program compares the low level, high level and short period trips with artificial signals as a check on the trip settings. Interim high level and short period trips can be entered in "FLUX" if desired; these trips are at lower level and longer period than the main trips, which are known as "backup" trips. The settings of the backup trips cannot be changed except by rewriting the "FLUX" program, a difficult task. See Appendix F, page F.6 R.1, for further details of the PMACS self-checking system.

The relay outputs of the independent fission counter channels consist of two relays in parallel. That is, for each of the three outputs, low level, high level, and short period, two relays are installed. These are so connected that identical failure would result in a scram signal; the possibility of simultaneous non-identical failure, though remote, is still present. Therefore, with one channel bypassed, artificial signals can be fed into the amplifiers and correct response of the relays noted at the proper trip setting by signal lamps. Initially, these checks will be made daily; as experience with the equipment is accumulated, the frequency may be adjusted. The initial frequency was based upon the calculations of Appendix F.

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. The reactor will not be started up nuclearly unless a shutdown margin is assured with provision for the failure of the control or safety rod of greatest worth.

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 $(3.1/2)(240)/(26)(3600) = 0.0040\phi/\text{sec}$. For abnormal conditions the cooling rates are still slower, as shown in Section XI.

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 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. 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 heat control, and experiment recording and analysis. The reactor safety circuit receives signals from the two neutron flux channels of PMACS as described in detail in Appendix F.

The flux monitor subsystem provides a signal proportional to neutron flux level. Two banks of uncompensated boron-lined 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 performed 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 can be computed by PMACS by the inverse multiplication method on an off-line basis.

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.

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.

H. Moisture Monitor

The moisture monitor is a recent type. Moisture is absorbed from the gas stream on a film of electrolyte. The current flow through the electrolyte is a function of the amount of moisture absorbed. The electrolysis products, hydrogen and oxygen, are carried off in the gas stream.

The sample is introduced via a flowmeter and flow controller at 10 cc/min. The circuit is so arranged that the moisture meter reads directly in parts per million by volume. The meter has five ranges, with full scale at 100, 300, 1000, and 10,000 ppm respectively.

Backflow is prevented by check valves. The sample is filtered at the input. Flow is established by a vacuum pump.

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. The fueled parts of the control rods will, however be present; tests will confirm that no attainable configuration of these fueled segments would be critical.

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. The mechanical integrity of the rods will be verified and the scram times measured after installation. The VSR will already have been installed and tested. 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

D. Operating Procedures

1. Startup

Changes in the test core loadings will be effected by incremental loading procedures. 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

XI. ACCIDENT ANALYSISA. 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 (page 9.7)
- control of excess reactivity (pages 9.7, 10.4)
- maintenance of worth of shutdown mechanisms (page 9.8)
- routine functional testing of safety system components (page 8.1)
- safety analysis of experiments (page 10.4)

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 chambers and ion chambers and the other two channels with

ion chambers only, of a different design.

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 component failures
 - 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 maxi-

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 connectors would have to be replaced after the incident.

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 might be anticipated. An analysis has shown that no damage to the floor would occur for an interruption of 4 hours. Furthermore, 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 connected to a hose connection provided for drainage 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.

Construction materials for the reactor and its appurtenances have been tested in small assemblies and in a large mock-up of the reactor, containing graphite, insulation material, metal structural parts, a prototype control rod, and a vertical safety rod (VSR), thermocouples designed for high temperature service, and other parts prototypical of the reactor design.⁴ The mock-up was first heated with graphite heater rods of a flat design. These were found to be marginal from the standpoint of warpage, so a modified I-beam design was subsequently tested in the mock-up and has been adopted for the reactor. The mock-up was provided with a circulating nitrogen atmosphere, using components and materials similar to those of the reactor equipment. Experience with the mock-up has shown the importance, for good service life of reactor materials, of a moisture removal system in the circulating gas stream, and one has been provided for the reactor. Mock-up experience has also shown the unsuitability of boron carbide in graphite as a neutron poison material at these temperatures and atmosphere conditions, dictating the change to the cermet described on page 5.10. The final mock-up test of 1000 hr at 1000 C,⁴ comparable to the time at operating temperature (800 C and above) expected for the reactor during one to two years, has shown that all materials which must last for the 10 year life-time of the reactor can confidently be expected to do so.

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 such that the bars will not experience temperatures in excess of 860 C. The tests have shown that these important structural members will perform without failure. Only minor surface effects have been noted in specimens examined by metallography⁴ 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 was present as a contaminant in the nitrogen atmosphere, minor surface change was all that was noted in samples of Inconel and Hastelloy-B placed at locations in the brick insulation, only where the temperature was above 800 C.

The only possible high temperature material failure having safeguards importance is believed to be a creep rupture or corrosion failure of the hanger rods suspending the firebrick insulation of the top of the reactor from the steel shell. 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 might 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. Binns⁷ as falling into two categories, which he calls failures of the first kind and

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

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 boron-lined 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.

PMACS provides highly precise measures of flux level and period at intervals of one second. It also checks its own operability and accuracy every 10 seconds against an artificial signal input. Nuclear shutdown (not a scram trip) results if any comparison is not acceptable. This is discussed in greater detail in Appendix F, page F.6 R.1.

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 contacts 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 used 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 an active thermocouple would cause shutdown of the heaters by PMACS. Continuous temperature monitoring of all heater rods is assured while they are energized.

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 and compute the average from among those remaining, so long as 50% remain. If fewer than 50% are serviceable, PMACS will shut down the heaters.

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 check of each thermocouple system is also available to the operator by requesting a log-out on the typewriter.

Thermocouple failures would not cause a hazardous condition. The

APPENDIX BMain 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 100 C, the following is obtained, where T is the equilibrium absolute temperature of the mixture:

$$\begin{aligned} \text{Heat input} &= \int_{273}^{1273} (6.50 + 0.00100T) dT + (100.077)(18.016)n \\ \text{Heat output} &= 9,720.53 n + \int_{273}^T (6.50 + 0.00100T) dT + n \int_{373}^T (8.22 + \\ &\quad 1.5 \times 10^{-4} T + 1.34 \times 10^{-6} T^2) dT \end{aligned}$$

Equating these and performing the integrations,

$$\begin{aligned} 7273.0 + 1802.99 n &= 9,720.53 n + 6.5 T + 5 \times 10^{-4} T^2 + n (8.22 T + \\ &\quad 7.5 \times 10^{-5} T^2 + 0.4467 \times 10^{-6} T^3) - 1811.76 - 3099.68n \end{aligned}$$

Introducing values of T and solving for n:

<u>T, °K</u>	<u>n, moles water</u>	<u>P, fraction of initial pressure</u>
1273	0.000	1.00
1173	0.0505	0.97
1073	0.1075	0.93
973	0.1720	0.90
873	0.2453	0.85
773	0.3293	0.81
673	0.4262	0.75
573	0.5392	0.69
473	0.6726	0.62

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

APPENDIX ECOMPARISON 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
- ^{238}U in driver fuel
- Shutter-type control rods without poison
- Rare earth oxides instead of cadmium as the poison in the 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.

SENSOR AND COMPARATOR BLOCKSNeutron Flux Monitor Channels (Figure 4)

Neutron flux is monitored from startup to higher flux levels by two channels. Each channel includes both (1) a fission counter with a pre-amplifier driving a log count rate circuit (LCRM) plus a period circuit (PM), and (2) a boron-lined ion chamber whose current output drives a logarithmic amplifier (Log N) plus a period circuit. Dual inputs are required to cover the extended range of flux levels, about 8 decades. Internal circuitry provides automatic overlap at the decade from 10^5 to 10^6 nv.

Reactor shutdown occurs if either channel provides an out-of-limit signal for flux level (on-scale and high flux) or for short period. 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 allowing a power excursion by the flux measurement channels discussed below.

A comparison logic in each flux level and period circuit in both LCRM and Log N units establishes an internal logic level. From logic equations derived from a truth table, the internal logic levels are matrixed to drive one on scale, one high level or one period trip relay combination per channel over an input range from about 10 nv to 2×10^8 nv. As long as the input signals stay 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

* Two single-pole double-throw (SPDT) relays in series-parallel.

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 boron-lined 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 analog signals are amplified and converted to pulses whose frequency is proportional to input current. A counter receives and registers each pulse, for processing by the computer and associated logic circuitry under program control. As the PMACS flux program is presently written,* when enabled by an on-scale response from the ion chambers and AD converter, the program reads in raw flux data via storage registers to the computer ten times per second and:

1. calculates period and power ten times a second,
2. accumulates and calculates a high accuracy period for experimental purposes once every ten seconds,
3. completes all calibration and a zero offset check on both channels periodically,
4. performs a range check every second for each channel (each of which has 12 ranges).

From the above, the program causes a scram if:

1. the calibration or zero check is poor,
2. ranges for the two flux channels differ by more than four,
3. fewer than 200 counts appear on the most sensitive range (on-scale trip),
4. the set-point period level is exceeded (period set point trip),

* The programs are subject to revision as final tests are made.

5. the period is less than 8 seconds regardless of the period set point (period back up),
6. the power level exceeds the set-point (high level set point trip),
7. the count is increasing on the top range (high level trip back up).

Alarms, in addition, are provided whenever the calibrate and zero checks are not normal but not yet at the trip point and whenever the ratio of the periods calculated for the two channels differs from 1.00 by 1% or more.

Any calculated values and experimental data may be stored on micro-tape as requested by the operator.

As remarked in the preceding section, 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.

APPENDIX GKEY WORKING DRAWINGS

To permit review in greater detail than that given in the display drawings and descriptions in the main text, there are appended 25 key working drawings of the building, the reactor, and its controls. The drawings have been retitled for the purpose of this report, but still bear the original drawing number for further reference if needed.

The figure numbers and titles in this report and the original numbers are listed below:

G-1	Building Plans	H-3-26605
G-2	Building Sections	H-3-25012
G-3	Reactor Cross-Section	H-3-25176 Sh. 2
G-4	Reactor Cross-Section at Heater Bus Bars	H-3-25089 Sh. 2
G-5	Reactor Longitudinal Section	H-3-25176 Sh. 1
G-6	Copper-to- Graphite Electrical Connection	H-3-25089 Sh. 3
G-7	Reactor Heaters Electrical Diagram	H-3-25061
G-8	Control Rod Drive Assembly	H-3-25229 Sh. 1
G-9	Control Rod Drive Bill of Materials	H-3-25229 Sh. 2
G-10	Control Rod Drive Details	H-3-25229 Sh. 3
G-11	Control Rod Drive Details	H-3-25229 Sh. 4
G-12	Control Rod Drive Details	H-3-25229 Sh. 5
G-13	Control Rod Drive Details	H-3-25229 Sh. 6
G-14	Control Rod Drive Details	H-3-25229 Sh. 7
G-15	Control Rod Drive Details	H-3-25229 Sh. 8
G-16	Control Rod Assembly	H-3-25228 Sh. 1
G-17	Control Rod Details	H-3-25228 Sh. 2
G-18	Safety Rod Drive Assembly	H-3-25230 Sh. 1
G-19	Safety Rod Drive Details	H-3-25230 Sh. 2
G-20	Safety Rod Drive Details	H-3-25230 Sh. 3
G-21	Safety Rod Drive Details	H-3-25230 Sh. 4
G-22	Safety Rod Drive Details	H-3-25230 Sh. 5
G-23	Safety Rod Assembly	H-3-25231 Sh. 1
G-24	Safety Rod Details	H-3-25231 Sh. 2
G-25	Safety Rod Details	H-3-25231 Sh. 3

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S E C O N D F L O O R P L A N
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1	DWG NO WAS SK-3-11954	NDL S-F-14		

REF. NO.	DWG. NO.	DRAWING TITLE
4	H-3-25007	ARC BSMT. FLOOR PLAN - 19-6
3	H-3-25009	ARC. FLOOR PLAN AT EL. +14'-6"
2	H-3-25008	ARC. FLOOR PLAN AT EL. 0'-0"
1	H-3-25000	DRAWING INDEX

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SCALE AS SHOWN		THE PACIFIC NORTHWEST LABORATORIES A DIVISION OF BATTTELLE MEMORIAL INSTITUTE PACIFIC NORTHWEST LABORATORY RICHLAND, WASH.
DRAWN BY RL BROWN	CHECKED BY WRF	
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CLASSIFIED BY		BNWL-CC-225
CLASSIFICATION NONE		
BUILDING PLANS		HIGH TEMP LATTICE TEST REACTOR
DWG. NO. H-3-26605

GENERAL ELECTRIC

SPECIAL SPECIFICATION

APPROVED FOR CONFORMANCE WITH DESIGN CRITERIA

NOTE: EL. 0-0 IS EQUIVALENT TO EL. 398-0

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BUILDING SECTIONS			
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DATE WJ. ING.		HIGH TEMP. LANTHES TEST REACTION	
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DATE	2/1/64	DATE	
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4 DETECTOR MOUNTING
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REACTOR CROSS-SECTION		HTLTR	
PROJ. NO. CAH-100	ISSUE NO. 2	ISSUE NO. 1	ISSUE NO. 1
DATE 10-9-66	CHECK PRINT	CHECK PRINT	CHECK PRINT
LTC: 12-28-66 - ISSUED FOR APPROPRIAL		LTC: 12-28-66 - ISSUED FOR APPROPRIAL	
COMMENTS INCORPORATED PER IRC		COMMENTS INCORPORATED PER IRC	
REFERENCE DRAWINGS		REFERENCE DRAWINGS

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DIMENSIONS ARE IN INCHES
TOLERANCES ARE:
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+ .0000 - .0000
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SURFACE ROUGHNESS
WITH ASA B

7 OR 8

1 1/16 DIA HOLE THRU NEAR SIDE
TND 1-BUNC-28 FAR SIDE
C BORE 2 1/16 DIA < 3/4 DEEP NEAR SIDE
CHAMFER HOLE 1/16 x 45°
TYPICAL 8 PLACES

H-3-25090
PT. NO. 6 (REF)

5 SEE NOTE #7

H-3-25090
PT. NO. 2 (REF)

SECTION Z-Z (TYP 6 PLACES)

VIEW 13-13
EAST FACE

LEGEND

- HEATER ROD - FASTENED TO BUS
- HEATER ROD - FASTENED TO PILE
- ⊗ HORIZONTAL CONTROL ROD
- ⊙ ALTERNATE POSITION FOR HORIZONTAL CONTROL ROD
- 21HTR60 - HEATER ROD NO. SEE H-3-25059

NOTES

- UNLESS OTHERWISE SPECIFIED:
1. ALL GRAPHITE SHALL BE PER HW-7567.
 2. REAM HOLE THRU PT. #2 .002 OVERSIZE.
 3. MATING PARTS MUST MATCH FOR FULL SURFACE ELECTRICAL CONTACT
 4. ON ALL METAL PARTS - BREAK ALL CORNERS
 5. ALL CUT SURFACES SHALL BE IN ACCORDANCE WITH ASA B 46.1 1962 125
 6. ALL ITEMS TO BE AS SPECIFIED OR APPROVED EQUAL
 7. TORQUE ALL 1" GRAPHITE SCREWS 60"
 8. TORQUE ALL 2" GRAPHITE SCREWS 100"
 9. ALL GRAPHITE PARTS SHALL HAVE 1/8 TO 1/4 CORNER, EDGE & FILLET RADII.
 10. GRAIN DIRECTION OF GRAPHITE TO BE WITHIN 5° OF DIRECTION NOTED ON DWG.

H-3-25089 2/8 2

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BATTELLE-NORTHWEST	
BY	DATE
1-4-60	3/4/60

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RICHLAND OPERATION OFFICE

SCALE NONE	VITRO ENGINEERING CO. & A DIVISION OF VITRO CORP. OF AMERICA
DATE 1-4-60	G-4 BNWL-CC-225
REACTOR CROSS-SECTION AT HEATER BUS BARS	
PROJ. NO. CAH 100	PROJ. TITLE HTLTR
PLAN NO. 318	SHEET NO. 7308
H-3-25089 2/8 2	

SPECIFIED
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IN ACCORDANCE

REV	BY	DATE	DESCRIPTION
2	H-3	6-9-59	REVISED CENTER BUS BARS TO 6" LENGTH MAX
1	GRW	5/5/56	REVISED TO 6'-0" MAX. GRAPHITE LENGTH
1			LETTER-DATED 12-3-62 15260 FOR COMMENTS INCORPORATED PER A.E.C.
1			CHECK PRINT

DRAWING STATUS

APPROVAL AUTHORITY

RECORD PLACES

NORTH - ELEVATION

SHEET 2

A

INSTRUMENT DUCT

H-3-25126

H-3-25151

H-3-25151

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H-3-25176 SHEET 1/2 REV. 0

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RICHLAND WASHINGTON					
APPROVED FOR CONFORMANCE WITH DESIGN CRITERIA					
(PAGE 1 OF 14) APPENDIX A, II & III					
BY <i>[Signature]</i> FOR GENERAL ELECTRIC DATE <i>[Date]</i>					
DWG REV NO.	APPR ST	DATE	DWG REV NO.	APPR ST	DATE

REACTOR HEATER BUS ARRGT.
H-3-25089

STRUCTURAL CONCRETE
H-3-25031, H-3-25027 & H-3-25032

U. S. ATOMIC ENERGY COMMISSION RICHLAND OPERATION OFFICE			
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DESIGNED BY <i>[Signature]</i>	DATE 10-17-64	G-5 BNWL-CC-225	
CHECKED BY <i>[Signature]</i>	DATE 10-17-64	REACTOR LONGITUDINAL SECTION	
APPROVED BY <i>[Signature]</i>	DATE 10-17-64	CLASSIFICATION UNCLASSIFIED	
PROJECT NO. CAH 100		PROJECT TITLE H3LTR	
DWG. NO. 318		SHEET NO. 1500	
DWG. H-3-25176		SHEET NO. 1500	

DRAWING INDEX		L.T.R. 12-28-64 - ISSUED FOR APPROVAL	
H3-25000	DRAWING INDEX	H3-25000	COMMENTS INCORPORATED PER A.E.C.
H3-25001	DRAWING INDEX	H3-25001	COMMENT PRINT. A
H3-25002	DRAWING INDEX	H3-25002	CHECK PRINT
H3-25003	DRAWING INDEX	H3-25003	CHECK PRINT
H3-25004	DRAWING INDEX	H3-25004	CHECK PRINT
H3-25005	DRAWING INDEX	H3-25005	CHECK PRINT
H3-25006	DRAWING INDEX	H3-25006	CHECK PRINT
H3-25007	DRAWING INDEX	H3-25007	CHECK PRINT
H3-25008	DRAWING INDEX	H3-25008	CHECK PRINT
H3-25009	DRAWING INDEX	H3-25009	CHECK PRINT
H3-25010	DRAWING INDEX	H3-25010	CHECK PRINT
H3-25011	DRAWING INDEX	H3-25011	CHECK PRINT
H3-25012	DRAWING INDEX	H3-25012	CHECK PRINT
H3-25013	DRAWING INDEX	H3-25013	CHECK PRINT
H3-25014	DRAWING INDEX	H3-25014	CHECK PRINT
H3-25015	DRAWING INDEX	H3-25015	CHECK PRINT
H3-25016	DRAWING INDEX	H3-25016	CHECK PRINT
H3-25017	DRAWING INDEX	H3-25017	CHECK PRINT
H3-25018	DRAWING INDEX	H3-25018	CHECK PRINT
H3-25019	DRAWING INDEX	H3-25019	CHECK PRINT
H3-25020	DRAWING INDEX	H3-25020	CHECK PRINT
H3-25021	DRAWING INDEX	H3-25021	CHECK PRINT
H3-25022	DRAWING INDEX	H3-25022	CHECK PRINT
H3-25023	DRAWING INDEX	H3-25023	CHECK PRINT
H3-25024	DRAWING INDEX	H3-25024	CHECK PRINT
H3-25025	DRAWING INDEX	H3-25025	CHECK PRINT
H3-25026	DRAWING INDEX	H3-25026	CHECK PRINT
H3-25027	DRAWING INDEX	H3-25027	CHECK PRINT
H3-25028	DRAWING INDEX	H3-25028	CHECK PRINT
H3-25029	DRAWING INDEX	H3-25029	CHECK PRINT
H3-25030	DRAWING INDEX	H3-25030	CHECK PRINT
H3-25031	DRAWING INDEX	H3-25031	CHECK PRINT
H3-25032	DRAWING INDEX	H3-25032	CHECK PRINT
H3-25033	DRAWING INDEX	H3-25033	CHECK PRINT
H3-25034	DRAWING INDEX	H3-25034	CHECK PRINT
H3-25035	DRAWING INDEX	H3-25035	CHECK PRINT
H3-25036	DRAWING INDEX	H3-25036	CHECK PRINT
H3-25037	DRAWING INDEX	H3-25037	CHECK PRINT
H3-25038	DRAWING INDEX	H3-25038	CHECK PRINT
H3-25039	DRAWING INDEX	H3-25039	CHECK PRINT
H3-25040	DRAWING INDEX	H3-25040	CHECK PRINT
H3-25041	DRAWING INDEX	H3-25041	CHECK PRINT
H3-25042	DRAWING INDEX	H3-25042	CHECK PRINT
H3-25043	DRAWING INDEX	H3-25043	CHECK PRINT
H3-25044	DRAWING INDEX	H3-25044	CHECK PRINT
H3-25045	DRAWING INDEX	H3-25045	CHECK PRINT
H3-25046	DRAWING INDEX	H3-25046	CHECK PRINT
H3-25047	DRAWING INDEX	H3-25047	CHECK PRINT
H3-25048	DRAWING INDEX	H3-25048	CHECK PRINT
H3-25049	DRAWING INDEX	H3-25049	CHECK PRINT
H3-25050	DRAWING INDEX	H3-25050	CHECK PRINT

2 MATL. GRAPHITE

50 PLATE 1' THK. MATL. 304

23 FLANGE MATL. SYNTHANE - C - CR GRADE SYNTHANE CORP OAKS, PENN.

UNLESS OTHERWISE SPECIFIED, DIMENSIONS ARE IN INCHES. FRACTIONS DECIMALS. SURFACE ROUGHNESS WITH ASA B.

H-3-250-89	SHEET 3 8	REV 1
------------	-----------------	----------

APPROVED
FOR CONFORMANCE
WITH DESIGN CRITERIA
BATTELLE-NORTHWEST
DATE: 3/14/85

U. S. ATOMIC ENERGY COMMISSION RICHLAND OPERATION OFFICE			
SCALE NONE		VITRO ENGINEERING CO. A DIVISION OF VITRO CORP. OF AMERICA	
DRAWN BY <i>[Signature]</i>	DATE <i>7/6/54</i>		
CHECKED BY <i>E. B. [Signature]</i>	G-6 BNWL-CC-225		
COPPER-TO-GRAPHITE ELECTRICAL CONNECTION			
APPROVED <i>[Signature]</i>	DESIGNED BY <i>[Signature]</i>	PROJ. NO. CAH-100	PROJ. TITLE HTLTR
APPROVED <i>[Signature]</i>	DESIGNED BY <i>[Signature]</i>	BLK. NO. 318	BLK. NO. 7308
CLASSIFICATION NONE		H-3-25089

LEGEND:

HC- CORE HEATER TRANSFORMER
 HS- SIDE HEATER TRANSFORMER
 HT- TOP HEATER TRANSFORMER
 HB- BOTTOM HEATER TRANSFORMER

NOTES:

1. CHANGE WIRE NO: PREFIX TO GO-D, GO-C, GO-D, etc. FOR CIRCUIT BREAKERS B, C, & D
2. TRANSFORMERS HB, HC, HS & HT HAVE TAPS TO PROVIDE THE FOLLOWING SECONDARY VOLTAGES 22.8, 25.2, 27.8, 30.0, 32.5, 34.9, 37.6 & 40.0
3. PURCHASE & INSTALL 180-120V, 10, 50VA TRANSFORMERS. LOCATE IN RESPECTIVE UNIT SUBSTATION ENCLOSURE.

△ NIC

HEATER BUS CONNECTION DIAGRAM (REAR VIEW)

ELECTRICAL SCHEMATIC TRANSFORMERS HC, HS, HT & HB

H-3-25061 1/3

U. S. ATOMIC ENERGY COMMISSION RICHLAND OPERATION OFFICE		VITRO ENGINEERING CO. A DIVISION OF VITRO CORP. OF AMERICA	
3	COORDINATION CHECK INCORPORATED VENDOR DATA & RE-ISSUED FOR APPROVAL REVISED SAT. REACTOR CONTROL.	SCALE NONE	G-7
2	RE-ISSUED FOR APPROVAL DEVICE IN EACH FEEDER PMACS ZONE FI, ADDED HALL EFFECT CHANGED TEMP. CONTROL TO	BNWL-CC-225	REACTOR HEATERS ELECTRICAL DIAGRAM
1	A.E.C. LETTER 10-22-64 COMMENTS INCORPORATED PER ISSUED FOR APPROVAL - PHASE II.	CAH-100	HIGH TEMPERATURE CALTICE TEST REACTOR
0		318	7302
REFERENCE DRAWINGS		H-3-25061	
H-3-25089 REACTOR HEATER ARRGT.			
H-3-25087 CND PLAN & ELEVATIONS - HTS			
H-3-25057 ELEM DIAG REACTOR AUX			
H-3-25051 ELECTRICAL ONE LINE DIAG			
H-3-25000 DRAWING INDEX			
DRAWING STATUS			
NONE

TAP #10-24UNC-2B
THRU TO 2 AT ASSY
& PLACES

COVER, REF
DWG H-3-25227

9/32 DIA, 2 HOLES
AT ASSY

1 ASSEMBLY

SECTION "B-B"

SECTION "A-A"

H-3-25229 1/3 3

NOTE: FOR GEN NOTES & BILL OF MATL SEE SH "2"

REV 3 W/ 2/1/66
APPROVED FOR COMPATIBILITY
WITH REACTOR DESIGN
VITRO ENGINEERING COMPANY

NO.	DESCRIPTION	DATE	BY	CHKD
1	WAS DWG H-3-25228	1-1-66	WJL	
2	DELETE PARTS 36 & 37	6-1-66	WJL	
3	(6) ADDED SECTION CC	6-17-65	WJL	
4	(A) ADDED PARTS 95, 96 & 97	6-17-65	WJL	

NO.	DESCRIPTION	DATE	BY	CHKD
1	H-3-25227 CONTROL ROD COVER			
2	H-3-25228 CONTROL ROD			
3	HWS-8504 CONTROL ROD ASSEMBLY			

CHECK PRINT	DATE	REVISION	DATE	APPROVED FOR CONST.
SCALE	NONE	APPROVED FOR PURCHASE		
DRAWN	HK ZORNES	APPROVED FOR		
CHECKED	W. J. L.	DATE		
APPROVED	W. J. L.	DATE		
CLASSIFICATION	NONE	PLATE NO.	318	YEAR
			1901	

G-8 BNWL-CC-225
CONTROL ROD DRIVE ASSEMBLY

H-3-25229 1/3 3

4	92	SCR, SET #10-24 UNC-2A x 1/2 LG
2	93	SCR, CAP #10-24 UNC-2A x 1/2 LG SOC
4	94	DOWEL 1/2 DIA x 1 3/4 LG (AMER S
1	95	TUBE :
1	96	ASSEMBLY
1	97	TUBE
1	98	SPACER
2	99	YOKE
1	100	SPIDER
4	101	SWIVEL PIN
1	102	ROD EXTENSION
2	103	SPACER

SST
SST
SST
SHT 6
SHT 8
SHT 8
H-3-25228-PART 4

6	47	SCR $\frac{1}{8}$ -16UNC-2A X $1\frac{1}{2}$ LG HEX HD	SST
6	48	NUT #10-32UNC-2B HEX	SST
4	49	CAM FOLLOWER MCGILL BRG	*CF $\frac{1}{16}$
2	50	CAM FOLLOWER MCGILL BRG	*CF $\frac{3}{16}$
32	51	SCR $\frac{1}{4}$ -20UNC-2A X $\frac{3}{4}$ LG FLAT HD	SST
1	52	SOLENOID AUTOMATIC SWITCH	*106576 3270C
1	53	SCR #6-32UNC-2A X $\frac{3}{4}$ LG SOCHD CAP	SST
1	54	NUT #6-32UNC-2B HEX	SST
1	55	SCR $\frac{1}{4}$ -20UNC-2A X $1\frac{1}{2}$ LG HEX HD	SST
1	56	SPRING $\frac{1}{16}$ DIA WIRE, CLOSE WOUND	TENSION SST
2	57	BALL BEARING NEW DEPARTURE	*77R-6
1	58	GEAR BOSTON GEAR	*NA-24
1	59	GEAR BOSTON GEAR	*NA-50
2	60	SNAP RING TPUARC	*N5000-87
2	61	GEAR BOSTON GEAR	*NA25B
1	62	STEPMOTOR DIGITORK	*M135BY
1	63	CONTROL UNIT DIGITORK	SRIBNR
1	64	MATCHING UNIT DIGITORK	*RCSBRIM
1	65	POWER SUPPLY DIGITORK	*PS1JHT
4	66	BALL 250 DIA	BEARING STL
1	67	SNAP RING TPUARC	*S100-131
1	68	SEAL CHICAGO RAWHIDE	*14361
1	69	O-RING LINEAR	*11-214
1	70	BUMPER $\frac{3}{16}$ OD X $1\frac{1}{2}$ ID X $\frac{1}{4}$ THK	NEOPRENE
8	71	SCR $\frac{3}{8}$ -16UNC-2A X $1\frac{1}{2}$ LG HEX HD	SST
1	72	DIAPHRAGM BELLFRAM CORP	*1-100-400
2	73	BALL BEARING KAYDON	*KA-42-CP
2	74	ROLLPIN ESNA CORP	*79-040-187-0625
4	75	SCR $\frac{1}{4}$ -20UNC-2A X $\frac{3}{4}$ LG HEX HD	SST
2	76	ROLLPIN ESNA CORP	*79-048-250-2000
6	77	SCR $\frac{1}{2}$ -13UNC-2A X $1\frac{1}{4}$ LG HEX HD	SST
1	78	SPRING, WIRE SIZE .177 DIA, SPRING OD .812 $\pm$.010, TURNS 61, FREE LENGTH 15", PITCH .245, ENDS CLOSED & GROUND	SST
16	79	NUT $\frac{1}{4}$ -20UNC-2B HEX	SST
1	80	O-RING $\frac{1}{2}$ ID X $7\frac{1}{4}$ OD X $\frac{1}{4}$ THK	SILICONE RUBBER
1	81	GEAR-20 PITCH; $1\frac{1}{2}$ " PA; $\frac{1}{2}$ " FACE; 75 TEETH, $1\frac{1}{2}$ " DIA HUB; $\frac{1}{2}$ " HUB PROJ, $\frac{1}{2}$ " BORE, 3.750 PD BOSTON GEAR	STL *GA75
1	82	LIMIT SWITCH GE *CR9441-E	1A
1	83	BUSHING BOSTON GEAR	*3810-4
1	84	SLEEVE $\frac{1.000}{.998}$ OD X $\frac{.876}{.873}$ ID X .375 LG	SST
1	85	THRUST BEARING BOSTON GEAR	*A016
1	86	ROLLPIN ESNA CORP	*79-018-078-1000
1	87	ELBOW SWAGelok	*3600-8-6
10	88	SCR $\frac{1}{4}$ -20UNC-2A X $\frac{3}{4}$ LG FL HD CAP	SST
1	89	ROLLPIN $\frac{1}{8}$ DIA X $\frac{13}{16}$ LG ESNA	*79-028-125-2812
1	90	SCALE	SH-3
2	91	SCR #6-32UNC-2A X $\frac{1}{4}$ LG	SST

GENERAL NOTES

- UNLESS OTHERWISE SPECIFIED
- 1 REMOVE ALL BURRS & BREAK SHARP EDGES TO $\frac{1}{64}$ MIN R
- 2 TOLERANCES: FRACTIONAL $\pm \frac{1}{32}$, DECIMAL $\pm .005$, ANGULAR $\pm \frac{1}{2}^\circ$
3. ALL MACHINED SURFACES (ASA B46.1 1962)
4. PAINT ALL CSTL PARTS (EXCEPT MACHINED SURFACES) WITH AMERCOAT GRAY ENAMEL IN ACCORDANCE WITH MFR INSTRUCTIONS
5. SHIM OR ADJUST MOTOR MTG AS REQUIRED TO ELIMINATE ALL BACKLASH IN GEARS
- 6 PART #2 MUST BE FREE TO BOTTOM OUT WHEN PART #24 IS RELEASED WITH PART #32 IN ANY POSITION ON ITS PATH OF TRAVEL. TESTS SHOULD BE CONDUCTED IN POSITION SHOWN IN ELEVATION & IN POSITIONS ROTATED 90°, 180° & 270° ABOUT Q.

7. ALL MATL TO BE AS SPECIFIED OR APPROVED EQUAL QUALITY.

QTY.	PT. NO.	DESCRIPTION	
1	1	ASSEMBLY	SH 1
1	2	PISTON ROD	SH 3
1	3	CUP RETAINER	
1	4	CUP RETAINER	
1	5	BUSHING	
1	6	ROLLER	
1	7	NUT	
1	8	NUT	
1	9	ROD	
1	10	BRACKET	
1	11	SCREW	
1	12	FRAME	SH 4
2	13	SPACER	SH 4
1	14	CYLINDER (FLANGE)	
1	15	CYLINDER (CAP)	
1	16	NIPPLE	
1	17	OUTBOARD SUPPORT	
1	18	CROSS BAR	
1	19	SUPPORT	
2	20	U-BOLT	
1	21	GEAR	SH-5
1	22	GUIDE	
1	23	LIMIT SWITCH BRACKET	
1	24	LOCK RING	
1	25	SHIM PLATE	
1	26	ELBOW	
2	27	CAM FOLLOWER BRACKET	
1	28	CLAMP	
1	29	BRACKET	
1	30	STOP BLOCK	
1	31	BALL NUT WITH SCR & FLANGE	
1	32	HOUSING	
1	33	MOTOR MOUNT	
1	34	CLEVIS	
1	35	MOUNTING PLATE	SH 6
1	36	CLAMP	
1	37	COVER PLATE	
1	38	MOUNTING PLATE	
1	39	COVER PLATE	
1	40	SHAFT $\frac{.375}{.374}$ DIA X $\frac{3}{8}$ LG	COLD ROLL STL
2	41	EYE BOLT $\frac{1}{4}$ -20UNC-2A X $1\frac{1}{2}$ LG SHANK, $\frac{1}{2}$ EYE ID	STL
1	42	DIFFERENTIAL XMFR SANBORN	ADD 24-DCOI-1000
1	43	SWITCH (DPDT) MICRO SWITCH	*DTE-2RN
8	44	SCR $\frac{1}{4}$ -20UNC-2A X $1\frac{1}{2}$ LG SOCHD CAP	SST
2	45	SPRING $1\frac{1}{2}$ MAX OD, CLOSE WOUND EXTENSION TYPE, HOOK ENDS, 80" LOAD WHEN EXTENDED TO 20 (HOOKS INCLUDED), 40" LOAD WHEN EXTENDED TO 14 (HOOKS INCLUDED)	SPRING WIRE
2	46	EYE BOLT $\frac{1}{4}$ -20UNC-2A X $3\frac{1}{2}$ LG SHANK $\frac{1}{2}$ EYE ID	STL

H-3-25229

SHEET 28 REV. 4

APPROVED FOR COMPATIBILITY
WITH REACTOR DESIGN
VITRO ENGINEERING COMPANY
REV 4 WSE 2/86

U. S. ATOMIC ENERGY COMMISSION RICHLAND OPERATION OFFICE

SCALE NONE		THE PACIFIC NORTHWEST LABORATORIES A DIVISION OF BATTELLE MEMORIAL INSTITUTE PACIFIC NORTHWEST LABORATORY RICHLAND, WASH.	
DRAWN BY HKZ	CHECKED BY J. E. J. 1/1/65	APPROVED RASMUSSE	
RESPONSIBLE ENGINEER RASMUSSE		G-9 BNWL-CC-225	
APPROVED W. W. Brown		CONTROL ROD DRIVE BILL OF MATERIALS	
CLASSIFIED BY G. W. S. 1/1/65		CLASSIFICATION NONE	
FILE NO. 318		INDEX NO. 1901	
H-3-25229		2 8 4	

4	WAS 2 REQD	REV. 1/1/65	REV. 1/1/65	REV. 1/1/65
3	ADD PARTS 96 THRU 103	REV. 1/1/65	REV. 1/1/65	REV. 1/1/65
2	PARTS 96 & 97 & 26	REV. 1/1/65	REV. 1/1/65	REV. 1/1/65
1	ADDED PARTS 95, 96, 101	REV. 1/1/65	REV. 1/1/65	REV. 1/1/65

NO.	DESCRIPTION	REV BY DATE	APPRO BY	FOR DATE
1	DESCRIPTION			
2	REVISIONS

SCR. MATL TO BE 17-4 PH
1100° HEAT TREAT
SCR TOL'S: .0003 THD TO THD
.0005 TOTAL
.0005 IN ANY INCH

32 1060 STL

HELI-COIL INSERT
#3585-BCN-0750
10 PLACES

35 6061-T6 AL

36 SST
#16 (0625) 5A

38 6061-T6 AL

PIN NO	WIRE FUNCTION
A	Φ A
B	Φ B
C	Φ C
D	COM
E	INTERLOCK
F	FB-A
G	FB-B
H	-12 V
J	MOTOR RETURN
K	IN LTS, COM
L	↓, NO
M	↓, NC
N	OUT LTS, COM
P	↓, NO
Q	↓, NC
R	LVDT
S	↓
T	↓
U	↓
V	SCRAM EV
W	" "
X	SCRAM LTS COM
Y	" " , NO
Z	" " , NC

CONTROL ROD PLUG CONNECTION

PIN NO	WIRE FUNCTION
A	Φ A
B	Φ B
C	Φ C
D	COM
E	INTERLOCK
F	FB-A
G	FB-B
H	-12 V
I	MOTOR RETURN

STEP MOTOR PLUG CONNECTION

QTY.	PT. NO.	DESCRIPTION	
X	1	HTLTR CONTROL ROD ASSY	
AR	2	CORD, PWR, 2 #16, 1 1/8" OD, 300V, 10A/COND TYPE SJO, COIL TYPE	WHITNEY BLAKE 4-8022-00-91
	3	CORD, PORTABLE, 2 #16, 300V, TYPE SJO	
	4	, 4 #16,	
	5	, 6 #16,	
	6	, 9 #16,	
10'	7	, 24 #16,	
AP	8	CABLE CLAMP, NYLON	
I	9	TERMINAL BOARD, 2 PIN	
	10

EL. "A-A"

NOTES

1. MOUNT ALL PORTABLE CORDS TO LOWER COVER WITH CABLE CLAMP, PART 8.
2. EVERY PRECAUTION MUST BE MADE TO PROVIDE FREE MOVEMENT OF MECHANICAL PARTS WITH NO DAMAGE TO PORTABLE CORDS.
3. EXPAND PWR CORD, PART 2, BETWEEN TERM BD & MAX. TRAVEL OF SCRAM EV, BEFORE SECURING WITH CABLE CLAMP. DO NOT ALLOW ANY SLACK IN PWR CORD WHEN SCRAM EV IS AT ITS MIN. TRAVEL.

H-3-25229 7/8 0

APPROVED FOR COMPATIBILITY
WITH REACTOR DESIGN
VITRO ENGINEERING COMPANY
M. S. Gorman

CHECK PRINT	COMMENT PRINT	APPROVED FOR CONST.	U. S. ATOMIC ENERGY COMMISSION PACIFIC NORTHWEST LABORATORY BATTELLE MEMORIAL INSTITUTE
DATE 7/29/64	ISSUE NO. 1	DATE 7/29/64	
SCALE		APPROVED FOR PURCHASE	
DESIGN <i>J. M. Gorman</i>	DATE 7/29/64	DATE 7/29/64	
CHECKED <i>J. M. Gorman</i>	DATE 7/29/64	DATE 7/29/64	
REVISED <i>J. M. Gorman</i>	DATE 7/29/64	DATE 7/29/64	
PROJECT COW-100	DWG. NO. 668983	CLASSIFICATION G-14	BNWL-CC-225
DATE 2-24-65			
REFERENCE DRAWINGS		H-3-25229	
NEXT USED ON		7/8	

CONTROL ROD DRIVE
REF DWG H-3-25229 SHT 1

99 101 100 99 102 98 97 103

10

TRIM OR SHIM HERE
AS REQD TO OBTAIN
"0" POSITION AS
SHOWN ON DWG
H-3-25228 SHT 1

96 ASSEMBLY

Technical drawing of a shaft with two views: front and side.

Front View (Left):

- Overall length: $15 \frac{7}{8} \pm \frac{1}{16}$
- Distance from left end to center of hole: $2 \frac{5}{8} \pm \frac{1}{16}$
- Shaft diameter: 3 DIA

Side View (Right):

- Distance from left end to center of hole: $2 \frac{5}{8} \pm \frac{1}{16}$
- Hole diameter: $1 \frac{3}{4}$ DIA
- Thread specification: $2 \frac{1}{4} - 4 \frac{1}{2}$ UNC-2B (BOTH ENDS)
- Minimum radius: $\frac{1}{32}$ MIN R (TYP)
- Minimum thread relief: MIN THD RELIEF (TYP)

[illegible]

32/ ALL OVER
 (98) NUCLEAR GRAPHITE

100 — STL

CONTROL ROD
REF DWG H-3-25228 SHT 1

103

17

Technical drawing of a circular part. The drawing shows a cross-section of a ring with a central hole. The outer diameter is dimensioned as 2.625 and 2.623. The inner diameter is dimensioned as 1.313. A note indicates a tolerance of $2 \frac{3}{4}$ DIA. The drawing is a top view with a center line and a section line.

Technical drawing of a shaft with a keyway. The shaft has a diameter of .250 DIA. The keyway is .1/16 WIDE x 1/16 DEEP. The key is .1/4 - 20 UNC-2A. The shaft has a length of .3750 DIA and a tolerance of .3743 DIA. The key is labeled MIN THD RELIEF. The keyway is labeled 1/16 WIDE x 1/16 DEEP SLOT. The shaft is labeled .250 DIA. The key is labeled .1/4 - 20 UNC-2A. The shaft has a length of .3750 DIA and a tolerance of .3743 DIA.

(101) HARDEN R/C 35-40
STL

Technical drawing of a shaft assembly with dimensions and tolerances:

- Overall length: $18 \frac{3}{4}$
- Distance from left end to start of second step: $5 \frac{5}{8}$
- Distance from start of second step to end: $1 \frac{3}{4}$
- Left end thread: $\frac{7}{16}$ -14 UNC-2A
- Second step diameter: 1.438 DIA (nominal), 1.435 (minimum)
- Third step diameter: 1.5 DIA (nominal), 1.4964 (minimum)
- Right end thread: $\frac{7}{16}$ -14 UNC-2B

102 304-L SST

FOR GENERAL NOTES & BILL OF MATL SEE SHEET #1

APPROVED FOR COMPATIBILITY
WITH REACTOR DESIGN
VITRO ENGINEERING COMPANY
W. A. Green 2/3/66 M

U. S. ATOMIC ENERGY COMMISSION
RICHLAND OPERATION OFFICE

THE PACIFIC NORTHWEST LABORATORIES
A DIVISION OF BATTELLE MEMORIAL INSTITUTE
PACIFIC NORTHWEST LABORATORY
RICHLAND, WASH.

BNWL-CC-225

CONTROL ROD DRIVE
DETAILS

SCALE NONE		
DRAWN BY RTO	CHECKED BY R.H.F.	DATE 1/6/61
RESPONSIBLE CHG. RASMUSSEN		
APPROVED S.C. [Signature]		G-
APPROVED [Signature]		C
APPROVED W.W. Brown		
CLASSIFIED BY J. [Signature]		1/6/61

310	W 4-193029	DATE: 1907
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CLASSIFICATION
NONA

310	W 4-193029	DATE: 1907
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11225279	APR 1961	100	100
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[illegible]

FOR CONTINUATION
SEE H-3-25229 SHT B

1 ASSEMBLY

QTY	PT NO	DESCRIPTION	SH #
X	1	ASSEMBLY	
1	2	ROD	SH #2
1	3	SPACER	
11	4	SPACER	1a, 1b, 1c
1	5	TUBE	
8	6	BUSHING	1a
8	7	INNER FUEL CYLINDER	1b
7	8	BUSHING	1c
7	9	POSITION CYLINDER	1d
7	10	OUTER FUEL CYLINDER	1e
1	11	TUBE	
7	12	SPACER	1f
1	13	SPRING	
1	14	PLUG	
1	15	NUT	
1	16	WASHER	
1	17	TUBE	
1	18	TUBE	

[illegible]

QTY.	PT. NO.	DESCRIPTION
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H-3-25228

NOTE
FOR GENERAL NOTES & BILL OF MATL SEE SHEET 1

REV 2 WSG 2/4/66
APPROVED FOR COMPATIBILITY
WITH REACTOR DESIGN
VITRO ENGINEERING COMPANY

NO.	DESCRIPTION	REV	APPRO	REFERENCE DRAWINGS
2	DELETE PART 3 REDESIGN PART 13	M.H.S.	A.C.C.	
1	(1) WAS 1: 3.83 CH WAS 3.83 VIEW OF PART 7 (1) WAS 500 PROJECTION (1) ADDED DETAIL (2) ADDED C-BORE (1) ADDED 507 (3) LG WAS 3.933 (1) DIA WAS 3.933 (4) LG WAS 4.0 (1) LG WAS 4.0 (5) LG WAS 4.0 (1) LG WAS 4.0	17 King 6 12/65		

CHECK PRINT	COMMENT PRINT	APPROVED FOR CONST.	U. S. ATOMIC ENERGY COMMISSION HANFORD ATOMIC PRODUCTS OPERATION GENERAL ELECTRIC
DATE 2-1-67	DATE 1/1/67	APPROVED FOR PURCHASE	
SCALE NONE	DATE 1/1/67	APPROVED FOR DE	
PROJECT CAN-100	CLASSIFICATION C-48012	DATE 1/1/67	
DRAWN BY G. W. S. S.	CHECKED BY	DATE 1/1/67	
DRAWING TITLE		DATE 1/1/67	
REFERENCE DRAWINGS		DATE 1/1/67	
G-17		DATE 1/1/67	
BNWL-CC-225		DATE 1/1/67	
CONTROL ROD DETAILS		DATE 1/1/67	
H-3-25228		DATE 1/1/67

1 ASSEMBLY

4	50	SCR $\frac{1}{4}$ -16UNC-2B X $\frac{1}{2}$ LG	SOC HD CAP	SST
1	51	LOCKWASHER $\frac{1}{8}$ SPLIT (AMER STD)		SST
3	52	BALL BRG #77R6	NEW DEPARTURE	
5	53	SNAP RING #5100-37	TRUARC	
1	54	AIR CYLINDER #A65-1 $\frac{1}{2}$ (1" STROKE)	MILLER	
1	55	COTTER PIN $\frac{3}{32}$ DIA X 1" LG	SST	
1	56	CASTLE NUT $\frac{7}{16}$ -20UNC-2B	SST	
1	57	WASHER $\frac{7}{16}$ FLAT	SST	
1	58	PIVOT PIN #NB16B	BOSTON GEAR	
1	59	ROLLPIN #79-032-156-0813	ESNA CORP	
2	60	BALL BRG #77-R12	NEW DEPARTURE	
3	61	BALL BRG #77-R8	NEW DEPARTURE	
1	62	SPACER $\frac{3}{8}$ OD X $\frac{1}{2}$ ID X $\frac{1}{2}$ LG	STL	
1	63	SPACER $\frac{3}{8}$ OD X $\frac{1}{2}$ ID X $\frac{1}{2}$ LG	STL	
1	64	SPACER $\frac{3}{8}$ OD X $\frac{1}{2}$ ID X $\frac{1}{2}$ LG	STL	
2	65	SNAP RING #5100-25	TRUARC	
4	66	NUT 1"-BUNC-2B, HEX JAM	SST	
1	67	ROLLPIN #79-032-156-2000	ESNA CORP	
1	68	MAGNETIC CLUTCH	SH-6	
1	69	LIMIT SWITCH CR 9441-E 1A	GENEPALELEC	
2	70	ROLLPIN #79-040-187-0750	ESNA CORP	
8	71	SCR $\frac{1}{16}$ -18UNC-2A X 1" LG	SOC HD CAP	SST
1	72	REDUCER #VLW113 (150:1 RATIO)	BOSTON GEAR	
4	73	SCR $\frac{3}{8}$ -16UNC-2A X 1" LG	SOC HD CAP	SST
16	74	SCR $\frac{1}{4}$ -20UNC-2A X $\frac{3}{8}$ LG	SOC HD CAP	SST
2	75	SNAP RING 5100-62	TRUARC	
1	76	RECEPTACLE KPT-07H-20-16PN	CANNON ELECT	
1	77	STEP MOTOR #M135BY (115V)	DIGITORK	
1	78	WINDOW $\frac{1}{8}$ X $\frac{1}{8}$ X $\frac{1}{8}$	TEMP & GLASS OPTICAL CLEAR	
12	79	SCR $\frac{3}{8}$ -16UNC-2A X $\frac{1}{2}$ LG	SOC HD CAP	SST
1	80	KEY $\frac{1}{16}$ SQ X $\frac{1}{2}$ LG	SST	
2	81	ROLLPIN #79-040-187-0750	ESNA	
1	82	LIMIT SW #DTE-2RN DPDT	MICRO SWITCH	
1	83	CLUTCH HUB	SHT 2	

QTY.	PT. NO.	DESCRIPTION	
	X 1	ASSEMBLY	
	1 2	END PLATE	SH-5
	1 3	HOUSING	SH-2
	2 4	GASKET	SH-5
	1 5	COVER	SH-5
	1 6	LIMIT SWITCH BRACKET	
	1 7	FRAME	SH-4
	1 8	SUPPORT BAR	
	4 9	BEARING HOLDER	
	1 10	PIVOT PIN	SH-3
	1 11	BRAKE ARM (PIVOT)	
	1 12	DRUM	SH-4
	1 13	BRAKE DISK	SH-3
	1 14	RING GEAR	SH-4
	2 15	EYE BOLT	SH-3
	1 16	WINDOW CLAMP	SH-5
	1 17	GASKET	
	1 18	CAM FOLLOWER SHAFT	SH-3
	1 19	GASKET	SH-5
	1 20	ADAPTER	
	1 21	SHAFT	SH-5
	1 22	IDLER PIN	
	1 23	IDLER GEAR	SH-4
	1 24	CAM	
	2 25	SHOE PIN	SH-3
	1 26	SHOE STUD (LONG)	
	2 27	BRAKE SHOE	
	1 28	BRAKE ARM (HINGE)	
	1 29	HINGE PIN	
	1 30	SHOE STUD (SHORT)	
	1 31	DRUM COVER	SH-5
	1 32	GEAR	
	1 33	GEAR	
	1 34	COUPLING #FCBB-12 ONE END $\frac{3}{8}$ OTHER END $\frac{1}{2}$	BOSTON-GEAR
	86 35	SCR $\frac{1}{4}$ -20UNC-2A X $\frac{1}{2}$ LG	SOC HD SST
	10 36	SCR $\frac{1}{4}$ -20UNC-2A X $\frac{3}{8}$ LG	SOC HD SST
	112 37	LOCKWASHER $\frac{1}{8}$ SPLIT (AMER STD)	SST
	4 38	SCR #10-24UNC-2A X $\frac{1}{2}$ LG	SST
	52 39	LOCKWASHER $\frac{3}{16}$ SPLIT (AMER STD)	SST
	1 40	AIRCRAFT CABLE $\frac{1}{8}$ DIA. 7X19 20:0" LG	SST
	1 41	FORK END AIRCRAFT CABLE TERMINAL #AN-667-4	AMERICAN CHAIN & CABLE
	5 42	SCR #10-24UNC-2A X 1" LG	SOC HD SST
	9 43	LOCKWASHER #10 SPLIT (AMER STD)	SST
	1 44	HEX NUT #10-24UNC-2B	SST
	4 45	SCR $\frac{3}{8}$ -16UNC-2A X $\frac{3}{8}$ LG	SOC HD CAP SST
	1 46	SCR, SET $\frac{3}{16}$ -18UNC-2A X $\frac{1}{8}$ LG, HEX SOC CAP	SST
	10 47	NUT $\frac{1}{4}$ -20UNC-2B HEX	SST
	1 48	SPRING, TENSION, 13 COILS, 12 GA (.105) DIA., .700 O.D.	SPRING STL
	1 49	KEY $\frac{1}{16}$ SQ X $\frac{1}{2}$ LG	SST

GENERAL NOTES

1. ALL MOVING PARTS ARE TO BE ASSEMBLED FREE OF BIND
2. PART 76, RECEPTACLE, HAS MATCHING PLUG NO. KPT06P-20-16S CANNON ELECTRIC CO.

3. AT ASSEMBLY DRILL $\frac{1}{16}$ DIA HOLE THRU PART 33 & 69 FOR PART 67

4. INSERT PART 49 IN PART 12 & PLUG WELD TO PART 13 AFTER BOLTING

5. REMOVE ALL BURRS & BREAK SHARP EDGES TO $\frac{1}{64}$ MIN R

6. TOLERANCES: (UNLESS SPECIFIED) FRACTIONAL $\pm \frac{1}{64}$, DECIMAL $\pm .005$, ANGULAR $0^{\circ}30'$

7. ALL MATL TO BE AS SPECIFIED OR APPROVED EQ QUALITY

8. ALL MACHINED SURFACES $\sqrt{32}$ (ASA B46.1 1962)

H-3-25230

SHEET 17 REV. 2

M 2	ADDED PART 83	REV. 8/15/65	REV. 1/2/66	REV. 1/2/66
M 1	(1) CHANGED NOTE (2) QTY WAS 2 AND LG $\frac{1}{2}$ WAS 116 (3) QTY WAS 48 (4) WAS PART 37 (5) QTY WAS 90 (6) QTY WAS 4 (7) WAS PART 35 (8) QTY PARTS 6, 8, 32, 33 & 68 TO SHOW CHANGES OF (9) CHANGED ASSY DWG	REV. 8/15/65	REV. 1/2/66	REV. 1/2/66
NO	DESCRIPTION	REV. BY	APP'D BY	FOR DATE
	REVISIONS			

H-3-25232	SAFETY ROD COVER
H-3-25231	SAFETY ROD
HWS-8505	SAFETY ROD ASSEMBLY

REP. NO.	DWG. NO.	DRAWING TITLE
		REFERENCE DRAWINGS
		NEXT USED ON

U. S. ATOMIC ENERGY COMMISSION RICHLAND OPERATION OFFICE	
SCALE NONE	THE PACIFIC NORTHWEST LABORATORIES A DIVISION OF BATTTELLE MEMORIAL INSTITUTE PACIFIC NORTHWEST LABORATORY RICHLAND, WASH.
DRAWN BY HKZ	CHECKED BY DATE 1/2/65
RESPONSIBLE ENGINEER RASHUSSELI	APPROVED BY W. Brown
APPROVED BY W. Brown	CLASSIFIED BY W. Soder
G-18	BNWL-CC-225
SAFETY ROD DRIVE ASSEMBLY	
PLATE NO. 318	INSTR. NO. 1801
NO. H-3-25230	SHEET NO. 17 REV. 2

3 304 SST
 $\frac{1}{8}$ THK UNLESS
 OTHERWISE SPECIFIED

20 UNC-2A

C STL
(EXCEPT LINING)

13 C STL

FOR GENERAL NOTES AND BILL OF MATL SEE SH #1

28 C STL
USE STD CHANNEL
IF DESIRABLE

REV 1, NSG 2/7/66
APPROVED FOR COMPATIBILITY
WITH REACTOR DESIGN
VITRO ENGINEERING COMPANY
H. S. Gamm

CHECK PRINT		COMMENT PRINT		APPROVED FOR CONST.		U. S. ATOMIC ENERGY COMMISSION	
DATE	ISSUE NO.	DATE	ISSUE NO.	DATE	ISSUE NO.	DATE	ISSUE NO.
1/1/67	1	1/1/67	1	1/1/67	1	1/1/67	1
SCALE NONE				APPROVED FOR PURCHASE			
DRAWN BY: [Signature]				APPROVED FOR DES			
CHECKED BY: [Signature]				APPROVED FOR DES			
DESIGNED BY: [Signature]				APPROVED FOR DES			
DEL. DATA: [Signature]				APPROVED FOR DES			
FOLDER: CHN-100				CLASSIFICATION: NONE			
DRAWING NO: 132123				PLATE NO: 318			
DATE: 9-15-65				PLATE NO: 1901			
DATE: 10-24-65				PLATE NO: H-3-25230			
DATE: 10-24-65				PLATE NO: 3 7			

NO.	DESCRIPTION	REV BY	DATE	APP'D BY	FOR	DATE	REFERENCE DRAWINGS
1	ADD "USE STD CHANNEL"	PHLS	8-18-65	PHLS			

G-20 BNWL-CC-225
SAFETY ROD DRIVE
DETAILS

OD DRIVE
H-3-25230

QTY.	PT. NO.	DESCRIPTION	MATL OR SH NO.
X	1	ASSEMBLY	
X	2	SUB-ASSEMBLY	SH-3
1	3	GASKET	SH-2
1	4	BUMPER PLATE	
1	5	HOUSING	
1	6	BRACKET	SH-3
1	7	CLAMP	
2	8	SIDE RAIL	
1	9	TIP	
19	10	FILLER	
1	11	FILLER	
2	12	SKIN	
20	13	SPACING	
180	14	RIVET	
2	15	RIVET	
4	16	SPRING $\frac{3}{4} \times 1\frac{1}{2}$ MD DIE	RAYMOND MFG
8	17	JAM NUT $\frac{3}{8}$ -16UNC-2B HEX	SST
4	18	SCR $\frac{3}{8}$ -16UNC-2A $\times \frac{1}{2}$ LG, SOC HD CAP	SST

H-3-25231 1/3 0

GENERAL NOTES

UNLESS OTHERWISE SPECIFIED

1. REMOVE ALL BURRS & BREAK SHARP EDGES TO $\frac{1}{64}$ MIN RADIUS.
2. TOLERANCES: FRACTIONAL $\pm \frac{1}{32}$, DECIMAL $\pm .005$, ANGULAR $\pm \frac{1}{2}$.
3. ALL MACHINED SURFACES $\sqrt{32}$ (ASA B46.1 1962)
4. ALL MATERIAL TO BE AS SPECIFIED OR APPROVED EQ QUALITY

APPROVED FOR COMPATIBILITY
WITH REACTOR DESIGN
VITRO ENGINEERING COMPANY
W. S. Gann

U. S. ATOMIC ENERGY COMMISSION RICHLAND OPERATION OFFICE			
SCALE NONE		BNWL-CC-225	
DRAWN BY HKZ DATE 12/15/55		THE PACIFIC NORTHWEST LABORATORIES A DIVISION OF BATTELLE MEMORIAL INSTITUTE PACIFIC NORTHWEST LABORATORY, RICHLAND, WASH.	
RESPONSIBLE ENGINEER D. RASMUSSEN		G-23	
APPROVED <i>[Signature]</i>		SAFETY ROD ASSEMBLY	
APPROVED <i>[Signature]</i>		CLASSIFIED BY G. W. Sellen 1/3 0	
APPROVED <i>[Signature]</i>		BLOB. NO. 318 PROJ NO CC-100 INDEX NO. 1301	
REFERENCE DRAWINGS H-3-2532 SAFETY ROD COVER H-3-25230 SAFETY ROD DRIVE HWS-3508 SAFETY ROD ASSEMBLY		Dwg. NO. H-3-25231 SHEET NO. 1 OF 3	
SHEET USED ON		CLASSIFICATION NONE	

NO.	DESCRIPTION	REV. BY	APPROVED BY	FOR	DATE	SHEET USED ON
1	H-3-2532 SAFETY ROD COVER					
1	H-3-25230 SAFETY ROD DRIVE					
1	HWS-3508 SAFETY ROD ASSEMBLY

NOTE FOR
SEE SHEET

4 HOLES
SP ON 2 1/2 DIA BC

1/2 DIA HOLE
IN LINE
WITH 5/32 HOLE
SHOWN IN
ENLARGED VIEW

QTY.	PT. NO.	DESCRIPTION
------	---------	-------------

DOWEL: 4 EQ SP ON 5 DIA
BC SEE "X-X"

3/8-16 UNC-2B
EQ SP ON 5 1/2 DIA BC
4 HOLES

ST

3/16 DIA DOWEL PIN 1 LG
PRESS FIT IN PLACE
(2 PLACES)
AT ASSY
WITH PART 7

NY 300 SST

SECTION G-G

F-F

14 ANY 300 SST

12 6061-T6 AL

DOWEL 1.000 DIA X 3/8 LG
PRESS FIT IN PLACE

SECTION "X-X"

SECTION C-C

24 6061-T6 AL

23 MODIFY
BOSTON #NB40

H-3-25230

NOTE
FOR GENERAL NOTES & BILL OF MATL SEE SHEET #1

REV 1, W36 2/3/66
APPROVED FOR COMPATIBILITY
WITH REACTOR DESIGN
VITRO ENGINEERING COMPANY
W.S. Chapman

CHECK PRINT	COMMENT PRINT	APPROVED FOR CONST.	U. S. ATOMIC ENERGY COMMISSION HANFORD ATOMIC PRODUCTS OPERATION GENERAL ELECTRIC
DATE	ISSUE NO.	DATE	
SCALE NONE	DATE	APPROVED FOR PURCHASE	
DRAWN	DATE	APPROVED FOR DES	
CHECKED	DATE	DATE	
DESIGNED	DATE	DATE	
PROJECT	DATE	DATE	
CLASSIFIED	DATE	DATE	
UNCLASSIFIED	DATE	DATE	
CLASSIFIED	DATE	DATE	
UNCLASSIFIED	DATE	DATE	
CLASSIFIED	DATE	DATE	
UNCLASSIFIED	DATE	DATE	

G-21 BNWL-CC-225
SAFETY ROD DRIVE
DETAILS

(1) DIM WAS 15 1/2					
(2) ADDED 4 THD HOLES					
(3) RELOCATED G HOLES					
(4) REDREW PART NO 8

(32) MODIFY AS SHOWN
OHIO GEAR # ST1696- $\frac{3}{4}$ BORE

33 MODIFY AS SHOWN
OHIO GEAR #ST1696 ⁵/₈ BORE

20 ANY 300 SST

(16) 304 SST

$\frac{1}{4}$ -20 UNC-2B
 $\frac{1}{2}$ DEEP
 4 HOLES
 $\frac{1}{4}$
 $\frac{1}{4}$
 $12 \frac{13}{16}$
 $1 \frac{1}{4}$
 $\frac{3}{8}$
 $2 \frac{1}{2}$
 $\frac{3}{8}$
 (2) ANY 300 SST

FOR GENERAL NOTES AND MATL LIST
SEE SH #1

APPROVED FOR COMPATIBILITY
WITH REACTOR DESIGN
VITRO ENGINEERING COMPANY
J. J. Quinn

REV 2, WSF 2/3/66

CHECK PRINT	☒	COMMENT PRINT	☒	APPROVED FOR CONST. BY [] FOR EXPIRY DAY []	U. S. ATOMIC ENERGY COMMISSION HANFORD ATOMIC REACTORS OPERATION GENERAL ELECTRIC &	
ISSUED NO.	DATE	ISSUE NO.	DATE	APPROVED FOR PURCHASE BY [] FOR []	G-22	BNWL-CC-225
SCALE	NONE			APPROVED FOR DES. BY [] FOR []	SAFETY ROD DRIVE DETAILS	
DRAWN	AHL, 10-2-68					
CHECKED	[Signature]	DATE	10-2-68			
MAN. BY	M K Jills	DATE	9/7/68			
DES. DIV.	RASMUSSEN	DATE				
PROJECT	CRN-100	SHEET NO.	32123			
					PLATE NO. 318	WELD NO. 1901

FIGURE V-1

5.6 R.1

LEGEND

P - PRESSURE	- BLOCK VALVE
Pd - DIFFERENTIAL PRESSURE	- CHECK VALVE
T - TEMPERATURE	- REMOTE OPERATED VALVE
F - FLOW	- REMOTE OPERATED 2 WAY 3 PORT VALVE
I - INDICATOR	
● - PMACS	
NC - NORMALLY CLOSED	
NO - NORMALLY OPEN	
FC - FAILS CLOSED	
FO - FAILS OPEN	
FN - FAILS NORMAL	