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HAZARDS SUMMARY REPORT
FOR THE
WALTER REED ARMY MEDICAL CENTER
NUCLEAR RESEARCH REACTOR

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FOR THE
WALTER REED ARMY MEDICAL CENTER
NUCLEAR RESEARCH REACTOR

By

HEINZ H. CAPPEL

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Principal Contributors

ROBERT S. HART
JAMES O. HENRIE

Other Contributing Agencies

U. S. ARMY CORPS OF ENGINEERS
WALTER REED MEDICAL CENTER

ATOMICS INTERNATIONAL

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I. INTRODUCTION AND SUMMARY

The proposed nuclear research reactor and its associated irradiation facilities at the Walter Reed Army Medical Center, are designated for medical and biological research and radioisotope production. The reactor, designed to operate at 50 kilowatts, will be located in the addition to the Army Service Graduate School at Walter Reed Army Medical Center in Washington, D.C.

The reactor is of the homogeneous solution type. A series of two enclosures provide containment barriers against dispersal of the radioactive elements produced in the core. The primary enclosure is of stainless steel and includes the core and gas handling system. The secondary enclosure, which consists of an aluminum envelope and steel sub-pile room liner, completely surrounds the primary system.

In this report, the reactor, its operation, location and the using agency are described in detail. The potential hazards and controlling safeguards are objectively evaluated and assessed. The maximum credible accident was postulated as the failure of the primary and secondary enclosure in the core region, resulting in the release of the core contents to the reflector. Since the reflector region is sealed to prevent argon-41 leakage the maximum credible accident does not present a serious hazard to the occupants of the reactor room.

In the evaluation of the maximum credible accident for the reactor, it was postulated that a step increase in reactivity of 3 percent was introduced. Before attaining the maximum credible accident, however, it had to be assumed that a H_2-O_2 reaction would follow the events described above. The pressure pulse of the reaction following the excursion was calculated to be large enough to damage the reactor core vessel. Simultaneous damage to the secondary enclosure was then postulated without regard to its possibility. The probability of occurrence of this succession of accidents is extremely small, especially since the step reactivity input investigated assumes the release of all reactivity in the reactor. Experience with the KEWB reactor, which is designed to study large power excursions, has shown that large step reactivity increases are exceedingly difficult to obtain.

As additional information, it was hypothesized that all volatile fission products (including iodine) were released instantaneously to the reactor room and also to the outside. The results indicate that even under these hypothetical conditions the associated radiological hazards allow sufficient time for emergency measures to be enacted.

II. GENERAL AREA

A. GEOGRAPHY

The reactor is located at the Walter Reed Army Medical Center in Washington, D.C. The city, coextensive with the District of Columbia, is situated on the Potomac River and its eastern branch, the Anacostia.^{1, 2, 3} The population in 1954 was estimated at 840,000 in the city and 600,000 in the environs.

B. GEOLOGY

The proposed site is situated near the western extremities of a massive to schistose granite-like mass (Laurel gneiss) of uncertain geologic age that dips steeply to the northwest. Recent studies suggest the bedrock may be highly metamorphosed sediments which were subsequently granitized. Except for the ravine of nearby Rock Creek, outcrops of rock are uncommon. Depth of decayed rock is irregular and often deep (30 feet) and soils derived thereof are micaceous silts. Locally thin patches of agglomerated sand, gravel, and clay of cretaceous age unconformably overlie the bedrock.

C. SEISMOLOGY

The area surrounding the proposed site is not likely to experience severe or even moderate earthquake damage for three reasons:

- 1) Its position is a region of deep-focus earthquakes (a region in which tremors are detected over large areas but with little energy evident in the central region).
- 2) The five shocks (years 1828, 1852, 1861) listed for the District of Columbia indicate it is not a seismic area. However, it has felt a number of shocks originating in Maryland and especially in Virginia; these shocks undoubtedly were detectable in the eastern two-thirds of the District where the alluvial character of the ground would aid the propagation of earth tremors.
- 3) The unweathered, fault-free crystalline bedrock should offer adequate resistance to vibrations and lateral displacements resulting from any future earthquake activity.

D. CLIMATOLOGY

The highest temperature for any one day occurs about the middle of the afternoon at Washington, D.C. The lowest temperature for the day usually occurs about sunrise. Figure 1 shows by months the percent chance of exceeding a given maximum or minimum temperature based upon 46 years of record. The average of maximum and minimum temperature during the winter months varies between 30° and 60°. During May through September the average temperature varies between 60° and 80°. At Washington the lowest recorded temperature was -15° and the highest was 106°

The local area, on the average, receives a uniform and abundant supply of moisture because of its geographical location with respect to the usual storm paths. The principal rain producing storms for the area can be divided into three types - the occluding cyclones, the dissipating tropical storms, and thunder storms. The average annual precipitation is approximately 40 inches. Figure 2 shows expectancy diagrams of precipitation amounts by months. Although the maximum monthly total precipitation has exceeded 17 inches, the most likely range is between 2 inches and 5 inches. The maximum 24-hour precipitation exceeds 7 inches but the most likely maximum 24-hour precipitation ranges between 0.50 and 1.50 inches. Figure 2 also shows the percent of days in any given month with precipitation equal to or exceeding a specified value.

High velocity winds are infrequent in Washington. The occurrence of strong winds are usually caused by hurricanes, tornadoes, or thunder storms. Although approximately 10 hurricane centers have passed directly over the Washington, D.C. metropolitan area, only four tornadoes have been observed in the District of Columbia. Most thunder storms recorded in Washington are not accompanied by extremely high winds; the maximum recorded velocity of 100 mph was caused by thunder storm activity during 9 June 1928. Other thunder storms have recorded 50-80 mph winds. The following tabulation indicates the variation in high wind speeds (mph) from Weather Bureau Records:

<u>Average Annual</u>	<u>Maximum Average Monthly</u>	<u>Maximum Velocity (5 min.)</u>	<u>Peak Gust</u>
7.0	10.7	53	77

Figure 1. Daily Maximum and Minimum Temperatures

Figure 3 shows monthly surface wind roses for Washington, D.C.

E. HYDROLOGY

The installation site topography is hilly with elevations varying between 250 and 350 feet above mean sea level. The average slope is approximately 5 percent. The ground cover can be considered 50 percent impervious, 50 percent turf.

The site area is serviced by storm sewers designed to carry an estimated 35 cubic feet per second. No serious incidence of local flooding is recorded although some ponding is experienced at curb inlets. The site area is approximately 100 feet above the normal water level of Rock Creek and is not susceptible to fluvial flooding.

Storm drainage from the site is discharged by sewers into Rock Creek and subsequently flows into the Potomac River. Rock Creek, at present, carries some combined sewage overflows into the Potomac River. The Rock Creek basin under proposed plans would have minor upstream flood control and a sewerage diversion conduit to reduce the silt and pollution load of the creek. The proposed diversion conduit would have a capacity of 1962 cubic feet per second below the mouth of Rock Creek.

Figure 3. Wind Direction

III. REACTOR SITE

A. WALTER REED ARMY MEDICAL CENTER

The Walter Reed Army Medical Center comprises Walter Reed Army Hospital, Walter Reed Army Institute of Research and Walter Reed Central Dental Laboratory. There also is the Armed Forces Institute of Pathology.

The hospital has an average of 1300 bed patients daily, exclusive of outpatients. The research institute spearheads medical, dental and veterinary educational and research programs. The pathology institute is the central laboratory of pathology for military and other government agencies. Figure 4 is a plan-view of the hospital grounds.

B. REACTOR BUILDING

The reactor will be located in an addition to the Army Medical Service Graduate School, Figure 5.

The major activity to be conducted in this addition will be medical and biological research. This work will be conducted on a continuous basis by approximately 200 permanently assigned personnel consisting of military and civilian medical doctors and scientists. The basement floor will be devoted entirely to radiological laboratories. The proposed 50 kilowatt medical research nuclear reactor and the first, second, and third floors will have a total of eighty laboratory modules. The partitions for these modules will be movable, making the laboratories flexible in size. It is anticipated that one to four permanently assigned personnel can use a laboratory concurrently. Students will be assigned to the school on an individual basis to work on a specific item of research which may run from a week to a year or more. The fourth floor will be devoted entirely to rooms housing small animals.

C. REACTOR ROOM

The reactor rooms are in an annex to the graduate school addition, Figure 4. The room housing the reactor is about 25 feet high, 35 feet wide and 65 feet long.

There are two sets of gas-tight doors forming airlocks, which give access to the reactor, one on the reactor room floor and one on the ground floor. The

Figure 4. Plan View and Population Density

Figure 5. Walter Reed Army Medical Center,
Addition to Army Medical Service Graduate School

double gas-tight doors are interlocked so that either door being opened will de-energize a holding device on the other door, preventing it from opening. The equipment door at the reactor room floor level is interlocked with the reactor so that, when it is opened, the reactor is shut down. The door also contains a key lock to prevent shutdown of the reactor by indiscriminate opening of the door.

D. AIR-SEALS

To keep the reactor room relatively air-tight and prevent excessive gas leakage, the walls, floors and structural ceilings in the reactor room and sub-pile areas are coated with an epon resin. The gas permeability of the coating has been determined experimentally⁴ and it is estimated for the specified pressure difference of one centimeter of mercury, that the leakage will be much less than 5 percent of the room volume per day. The leakage will be tested by the time-pressure decay method and shown to meet this requirement.

E. VENTILATION

Negative air pressures of about 1/8 inch H₂O with respect to outdoor atmosphere will be maintained in the reactor room, so that air will leak into, rather than out of, the building. Pressures and temperatures are automatically regulated.

Both supply and exhaust air valves in the reactor room are of the butterfly type and are capable of maintaining the gas-tight integrity of the room. The valves are remotely operated and can be closed from the supervisor's office, located next to the control room.

The air will be changed a minimum of six times an hour, with the exhaust being discharged into a vent on the penthouse roof. This exhaust will be adequately monitored and any measurable increase in activity will cause an annunciator at the control console to give visual and audible alarm.

The sub-pile area will be provided with a 3 inch diameter line which runs from the sub-pile room to the equipment room. Also, a 2 inch line will be run from the reflector area to the equipment room. These lines will be capped. Space has been provided for installing valves and a blower on these lines connected

to the exhaust duct. The valves are normally closed. This will make ventilation of the sub-pile room possible if airborne radioactivity levels in the sub-pile room prove to be too high for easy access. Any discharge from this area will be authorized by the Chief of Reactor Operations and the Health Physics supervisor and under no condition will the exhaust leaving the building be above tolerance levels.⁵

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IV. DESCRIPTION OF THE L-54 REACTOR

A. GENERAL DESCRIPTION

The L-54 Reactor is a homogeneous solution-type reactor, designed to operate at 50 kilowatts of thermal power and to produce a maximum thermal neutron flux of about 10^{12} neutrons/cm²-sec at the center of the reactor core. Exposure facilities are provided through which the neutron flux is available in varying intensities for experimental purposes.

An aqueous solution of enriched uranyl sulfate dissolved in distilled normal water is used as the fuel solution. This is contained in a spherical core tank. A sealed aluminum envelope surrounds the core assembly and part of the graphite reflector; this envelope, together with the steel-lined sub-pile room, constitutes a secure secondary enclosure. The rectangular graphite reflector and the thermal column is 5 feet square by 8 feet long, and is enclosed on all but one side by an aluminum tank. Dense concrete surrounding the reflector serves as a biological shield.

The reactor is provided with an overflow tank, a fuel handling system for filling and draining the core tank, a cooling system for maintaining the proper fuel temperature, and a gas handling system for confining and controlling the reactor atmosphere and recombining radiolytically decomposed water. Power level in the reactor can be controlled either manually or automatically by means of a vertical control rod system. The control instrumentation is such that in the event of a power surge or a malfunction the control-rod system is automatically actuated to shut down ("scram") the reactor. Numerous circuits provide for automatic scrams or prevention of startup for other safety reasons, and fail safe arrangements are incorporated.

Below the reactor is a sub-floor level. Here are located a sub-pile room, valve room, recombiner room, and exposure or set-up room. The exposure and valve rooms are separated from the recombiner and sub-pile rooms by thick dense concrete walls and a gas-tight door.

Sectional views of the reactor are shown in Figures 6 and 7, and of the sub-floor plan in Figure 8.

Figure 6. L-54 Reactor (horizontal section)

Figure 7. L-54 Reactor (vertical section)

Figure 8. L-54 Reactor (sub-floor plan)

B. OPERATING PARAMETERS

Table I summarizes various characteristics of the reactor. The numerical values given in the table were extrapolated from measured data obtained from similar 50 kilowatt solution type reactors.

TABLE I
SUMMARY OF CHARACTERISTICS OF THE L-54 REACTOR

Reactor type	Thermal
Power	
Thermal power level, maximum	50 kilowatts
Power density, maximum	2.5 kilowatts/liter
Power density, average	1.8 kilowatts/liter
Flux and radiation levels (during 50 kilowatt operation)	
Thermal flux at center of core	1.5×10^{12} neutrons/cm ² -sec
Fast flux at center of core	2×10^{12} neutrons/cm ² -sec (0.5 Mev)
Thermal flux at outside of bismuth window	5×10^{11} neutrons/cm ² -sec
Thermal flux at exterior face of horizontal thermal column	5×10^9 neutrons/cm ² -sec
Radiation level at outer surface of biological shield	Less than 2.5 millirem/hr
Gamma dose in void space between horizontal thermal column and shield	Less than 300 rep/hr
Reactivities	
Reactivity at 20°C, zero power	3 percent
Reactivity at equilibrium power	
Beam holes open	0.2 percent
Beam holes filled with graphite	1.1 percent
Reactivity controlled by the four control rods fully inserted	8 percent
Reactivity controlled by only one control rod fully inserted	2 percent

Mass coefficient of reactivity	0.025 percent/gram U^{235}
Overall temperature coefficient of reactivity	-0.016 percent/ $^{\circ}F$
Fuel	
Composition	Enriched uranyl sulfate (UO_2SO_4) dissolved in distilled normal water
Enrichment in U^{235} , approx.	90 percent
Solution concentration	50 grams U per liter
Zero power critical mass	1000 grams U^{235}
Operating mass	1200 grams U^{235}
H = U^{235} atomic ratio	580
Fuel solution volume in core	26.5 liters
Solution temperature at 50 kilowatt operation	175 $^{\circ}F$
Coolant (numerical values refer to 50 kilowatt operation)	
Primary	
Composition	Distilled normal water
Temperature at inlet to core	Below 90 $^{\circ}F$
Temperature rise through core	Approximately 25 $^{\circ}F$
Flow rate to reactor core	Up to 15 gpm
Flow rate to gas recombiner	5 gpm
Secondary	
Composition	Normal water
Flow rate	Up to 50 gpm
Reflector	Graphite
Control and safety rods	
Number of rods	Four
Absorbing material	Boron

C. REACTOR CORE

The core consists of a 15-3/4 inch diameter spherical tank, made of stainless steel varying in thickness from 0.072 inch to 0.093 inch. It is shown

pictorially in Figure 9. Stainless steel cooling tubes are coiled within the core in ten parallel sections. A horizontal central exposure tube gives access to the core interior for experimental purposes. Four control rod thimbles extend vertically through the core tank for control purposes.

The exposure tube and each rod thimble consist of two concentric sections, the outer of which is seal-welded to the core-tank wall and the inner of which is seal-welded to the aluminum envelope forming part of the secondary enclosure. This construction provides the necessary passages into the core and yet maintains both primary and secondary enclosure seals.

A vertical overflow tank at the top of the core provides space for any fuel solution which may be forced out of the core in the event of a power surge. The fuel solution is forced up through a stand-pipe into the overflow tank and drains back slowly into the core through a 1/8 inch opening at the bottom of the overflow tank.

The overflow tank has a capacity of approximately 2 liters, and will hold that amount of fuel solution an average of 16 inches from the center of the core tank. Expanding fuel solution is carried up a 2 inch (I. D.) pipe, is deflected by a curved plate above the top of the pipe, and collects in the annular space surrounding the pipe. An entrainment eliminator at the top of the overflow tank, above the curved plate, reduces the number of droplets of fuel solution being carried into the gas handling system. Cooling of the overflow chamber walls results in condensation of water vapor which escapes with the sweepgas. The condensate washes down the inside surfaces of the chamber; this prevents the buildup of uranium salts on the wall surface.

D. PRIMARY AND SECONDARY ENCLOSURES

Radioactive materials within the reactor are completely contained in a sealed double enclosure.

The primary enclosure consists of all parts which come in direct contact with the fuel solution or with the radioactive gas atmosphere. These parts include the core assembly and the components and piping of the fuel handling and gas handling systems, all of which are fabricated from corrosion-resistant steel. The entire

Figure 9. Core Assembly

system is welded except for connections to the recombiner pump and pressure transducers. These mechanical connections have been carefully designed and tested and are located in accessible areas. All welds are performed to exact specifications and are die checked, X-rayed and mass spectrometer helium leak checked. The system is also pressure tested at 300 psig.

A secondary enclosure completely surrounds the primary enclosure, to prevent the escape of radioactive gases or liquids in the event of a leak anywhere in the primary enclosure. The secondary enclosure consists of an aluminum envelope enclosing the core assembly and some of the reflector graphite, plus the steel-lined sub-pile room. The aluminum envelope is sealed to the sub-pile room lining by lead o-rings. The sub-pile room has a steel gas-tight door separating it from the valve room.

The aluminum secondary enclosure assembly is helium leak checked. The steel sub-pile room liner welds are die checked. After installation the entire secondary enclosure system will be leak checked by the time-pressure decay method.

The atmosphere within the secondary enclosure is circulated by a small pump to an ionization chamber located in the set-up room. The chamber provides continuous monitoring for radioactivity. Should the primary enclosure fail, escaping radioactive gases would be contained by the secondary enclosure while automatically sounding an alarm because of the increased radioactivity level of the enclosed atmosphere. Since the volume within the secondary system is some 250 times that of the primary system, the release of gases from the primary enclosure could cause no appreciable pressure change in the secondary system.

E. COOLING SYSTEM

The reactor cooling system is shown schematically with the other piping systems, in Figure 10. It is constructed entirely of corrosion-resistant steel and is designed to maintain the average core-solution temperature at 175°F for a maximum power level of 50 kilowatts.

The primary coolant is distilled water, which is circulated at up to 15 gpm through the core cooling coils. An additional amount of the primary coolant

1

bypasses the core and is routed at about 5 gpm through the recombiner heat exchanger, which maintains the desired temperature of the water in the recombiner tank.

The core cooling coils consist of ten parallel sections of 1/4 inch tubing having a wall thickness of 0.028 inch. The total length of the coil tubing is approximately 90 feet. The primary cooling water is recycled in a closed system. It transfers heat to the secondary coolant in the main heat exchanger.

All components of the cooling system except the core cooling coils, the recombiner heat exchanger, and piping leading to them are located in one assembly in the sub-pile room. This assembly is shown pictorially in Figure 11. It includes the main heat exchanger, the pump, the surge tank, the throttle valves, temperature indicators, low level and high pressure alarms, and conductivity cell.

The primary coolant pump is a motor-driven, constant-speed, centrifugal pump. A 5 gallon surge tank, connected into the primary system on the inlet side of the pump, serves as the coolant reservoir and provides expansion space for the system. The primary coolant circuit may be vented by opening a valve from the surge tank.

The secondary coolant is water, furnished directly from water mains and circulated in a once-through process to cool the primary coolant. The flow rate will be varied according to the temperature of the incoming water and the power level of the reactor; flow rates from 5 to 50 gpm are anticipated.

The secondary coolant is continuously monitored for radioactivity before being discharged. A pressure monitor and a liquid level indicator in the primary surge tank, as well as a coolant conductivity cell, monitor the primary cooling system continuously for leaks. It should be noted that as long as either the primary or secondary system remains intact, the secondary coolant cannot become radioactive.

F. FUEL HANDLING SYSTEM

The fuel handling system, shown schematically in Figure 10, consists of a safe-geometry fuel drain tank and the necessary piping and valving connecting the tanks to the reactor core and to the external servicing connections.

Figure 11. Cooling System Components

The drain tank is located within the secondary enclosure, below the sub-pile room floor level. Lead bricks are used to shield the sub-pile room from the tank. This arrangement is shown in Figure 8. The tank has sufficient capacity to contain all the reactor fuel solution.

External connections to the primary system are located in the exposure room. Service valves in the lines leading to these connections are located in the sub-pile room and can be remotely operated from the valve room. Fuel solution can be transferred from the drain tank to the core tank by evacuating the core tank atmosphere. The core tank can be drained by gravity to the drain tank.

G. GAS HANDLING SYSTEM

The gaseous atmosphere above the fuel solution, comprising water vapor, gaseous fission products, hydrogen and oxygen, is maintained in a closed system referred to as the gas handling system. The system is shown schematically in Figure 10.

The principal functions of the gas handling system are to:

- 1) Recombine hydrogen and oxygen and return the resulting water to the core tank.
- 2) Minimize the possibility of hydrogen-oxygen explosions, and if one should occur, to confine it safely.
- 3) Confine the fission product gases and provide a means for their disposal, if necessary.
- 4) Provide, within reasonable limits, surge tank capacity to minimize pressure fluctuations within the reactor atmosphere.

The gas handling system, as part of the primary enclosure, is a closed system that is pressure-tested at 300 psig. During normal operation the pressure is initially set just below atmospheric pressure. The principal components of the system are a recombiner, a pump and exhauster used for circulating the sweep gases, and piping connecting the recombiner assembly with the core tank. The primary coolant maintains a constant low temperature in the recombiner

tank. External connections for servicing the system are located in the exposure room. Service valves located in the sub-pile room are remotely operated from the valve room.

Pressure, temperature, and flow measurements are made at various points throughout the system. A dangerous condition or malfunction, such as a major leak in the system, a blocked line, or an inoperative component, will be reflected in these measurements and will result in automatic reactor shutdown.

The system is initially charged with oxygen gas, which serves as the carrier for radiolyzed water products and fission product gases. The large excess of oxygen which is thus always present aids in effecting recombination of hydrogen with oxygen when circulated through the catalyst chamber.

To maintain circulation within the system, a pressure differential is created by pumping water through an ejector-type nozzle. The circulating water passes from the recombiner tank through a heat exchanger, a water pump, the nozzle, and back into the recombiner tank. The heat exchanger is cooled by the reactor primary coolant. It removes the heat of recombination of hydrogen and oxygen, and maintains the recombiner tank water temperature at less than 115°F. The water pump is of the canned-rotor type and is provided with aluminum-oxide, water-lubricated bearings. This pump creates a pressure differential of 40 psi at a flow of 25 gpm which causes a gas pressure drop across the water jet of approximately 20 inches of water.

Gas formed in the fuel solution in the core tank passes through the entrainment eliminator at the top of the overflow tank and then is drawn downward to the recombiner. The gas enters the recombiner near the water nozzle outlet where it is entrained in the water stream, carried upward, and emptied into the recombiner tank.

From the recombiner tank, the gas passes through another entrainment eliminator into the catalyst chamber. This entrainment eliminator reduces the amount of entrained water entering the catalyst chamber. The recombining catalyst is platinum deposited on cylindrical pellets of aluminum oxide. The pellets are 1/8 inch in diameter by 1/8 inch long, with the platinum content amounting to 0.3 percent by weight. The catalyst chamber is equipped with a

heater which is used during reactor startup to augment the catalytic action. When the bed temperature reaches approximately 250°F, the heater automatically shuts off.

Temperature is maintained by the recombining reaction. The recombined water vapor is swept from the catalyst chamber to the core by the carrier gas moving at about 50 ft/sec. There is a continuous transfer of water from the core tank to the recombiner tank. To maintain equilibrium in the system, an overflow weir, which maintains a constant water level, is installed in the recombiner tank. Water overflowing the weir is drawn into the circulating gas stream and carried back to the core tank.

The total gas volume in the gas handling system is approximately 30 liters. About 3 gallons of water are required as working fluid in the recombiner tank. The relatively large volume in the tank above the water level provides ballast space for smoothing out pressure variations.

H. CONTROL ROD SYSTEM

Four identical vertical control rods are provided for power regulation and reactor shutdown. Normally one of these four rods functions as a regulating rod, and three as shim rods. Each rod controls approximately 2 percent reactivity. The regulating rod may be automatically regulated at a power level above approximately one watt. Applying manual control to raise any other rod stops upward motion of the automatically-controlled rod, but does not prevent its downward motion.

Each rod is fabricated from stainless-steel tubing and contains, as the neutron-absorbing material, four boron-carbide cylinders approximately 3/4 inch in diameter and 4 inches long. As shown in Figure 12, the rods move within vertical guide tubes. The lower section of each guide tube consists of an aluminum thimble which extends through a stainless-steel sleeve into the core tank. The sleeve is seal-welded at each end to the core tank walls and the thimble is seal-welded to the upper surface of the secondary enclosure envelope. This construction preserves both the primary and the secondary enclosure seals.

The rod drive mechanism is of the rack and pinion type, actuated through a gear-reduction unit by a reversible electric motor. This mechanism is located

Figure 12. Reactor Control Rod Mechanism

above the reactor shield. The drive assemblies are readily accessible for adjustment, calibration, or repair. The rod travel is 15 inches and the rod withdrawal rate is 0.1 inch per second.

The drive mechanism and the control rod itself are separate parts coupled by a direct-current magnet located at the bottom of the drive mechanism. When the magnet is de-energized, the control rod falls freely by gravity for about 13 inches and then is decelerated for the last 2 inches by the air-cushioning effect of the control-rod piston riding in the cylinder formed by the walls of the rod housing. When deceleration begins the control rod is approximately 85 percent effective. The magnets can be de-energized automatically by any of the reactor scram circuits or they can be de-energized manually at the control console. Each rod is provided with microswitches that operate position signal lights on the control console indicating "ROD IN", "ROD OUT", "MAGNET IN", "MAGNET OUT". After all rods have been fully inserted and any one rod is then raised, a warning bell sounds for 30 seconds. Five "Reactor On" warning lights appropriately located in the reactor building are on when any of the four rods are not fully inserted.

Only one rod may be withdrawn at a time, whether by automatic or manual control. Any number of rods may be inserted simultaneously, even when a rod is being withdrawn. Toggle springs on the manual control switches stop the outward motion of any rod after the operator's hand is removed from the switch.

I. REACTOR INSTRUMENTATION

The principal functions of the reactor instrumentation system are to:

- 1) Regulate, indicate, and record reactor power level.
- 2) Measure and indicate pressure, temperature, and flow in the gas handling system and in the cooling system.
- 3) Provide interlocks to insure that certain safety requirements have been satisfied.
- 4) Provide for radioactivity monitoring.
- 5) Provide automatic scram or alarm circuits associated with each of the preceding functions to safeguard against any condition that could result in a dangerous incident.

The instrumentation Block Diagram is shown in Figure 13. The manner in which the instrumentation fits in with the piping systems is shown in the Piping and Instrumentation Diagram, Figure 10. All operating information is indicated or recorded at the control console.

Power Level is measured by three types of sensing elements. In the range from about 0.001 watt to 1 watt, two fission chambers are used. In the range from about 0.1 watt to well above full power (50 kilowatts), two gamma-compensated ion chambers are used. An uncompensated ion chamber serves in the range from about 10 to over 50 kilowatts. The instrument ranges overlap to provide continuous measurements over the entire span of power from source level at shutdown to full power.

The output pulses from each fission chamber pass through a preamplifier and a linear pulse amplifier. In Channel I the pulse amplifier output signal is transmitted to a logarithmic count rate meter, which also provides a source interlock that permits reactor startup only when the combination of source strength and shutdown multiplication provide an adequate count rate. In Channel II the pulse amplifier output signal is fed to a linear count rate meter whose output represents counts per unit time and is proportional to reactor power level.

In the power level range from 0.1 watt to 50 kilowatts, one ion chamber (in Channel III) feeds a logarithmic amplifier which produces two output voltages. One is proportional to the logarithm of the neutron flux ($\log N$ flux) and is fed to the $\log N$ recorder. The other output voltage is proportional to the time derivative of the $\log N$ flux, and therefore, is a measure of the inverse period of the reactor. This voltage is fed, through the period amplifier, to the period indicator.

The output current from the ion chamber of Channel IV, operating the power level range from 0.1 watt to 50 kilowatts, is fed to a linear micro micro ammeter whose output voltage is proportional to reactor power level; its output voltage is transmitted to a recorder-controller, which is connected to the regulating rod for automatic control.

Since the first four channels do not offer extremely fast action in initiating a scram in the event of a rapidly rising period when near full power, the uncompensated ion chamber of Channel V is introduced for control only in the range

Figure 13. Instrumentation Block Diagram

from about 10 to over 50 kilowatts. This ion chamber is connected to a safety chassis, and initiates rod release for a scram within 50 milliseconds after the reactor power exceeds its pre-designated level.

J. REACTOR FUEL AND NEUTRON SOURCE

The reactor fuel is an aqueous solution of Uranyl Sulphate. The Uranium is to be approximately 90 percent enriched in U-235. The fuel requirements for the reactor are further described in Table I.

Detailed specifications have been written to control the purity of the fuel solution and additives. The halides are limited as a corrosion prevention measure. Maximums of 25 ppm of fluorene and 5 ppm of all other halides, based on uranium metal, are specified and controlled. Based on the final fuel solution, rather than uranium metal, these maximums are decreased by a factor of about 25.

The neutron source is specified to produce a minimum of 10^6 neutrons/second. The source is contained in a doubly enclosed steel container and is inserted in a graphite stringer adjacent to the reactor core.

K. REFLECTOR

The reactor core is surrounded by a graphite reflector consisting of machined reactor-grade graphite blocks. The blocks are 4-1/16 inches square and up to 50 inches long. The pieces immediately surrounding the core assembly, are accurately machined to appropriate shapes and sizes.

The reflector forms a right prism approximately 8 feet long, 5 feet wide, and 5 feet high. The bottom, top, and three sides of the reflector are enclosed in an aluminum tank. The thermal-column end of the reflector is not enclosed by the tank. Other openings in the tank are provided for the experimental facilities.

L. SHIELDING

The major portion of the biological shielding consists of dense concrete as shown in Figures 6, 7 and 8. This concrete has a density of approximately 225 lb/ft³. It is composed primarily of magnetite (Fe_3O_4) ore, colemanite ($\text{Ca}_2\text{B}_6\text{O}_{11}$) ore, portland cement and water. Shield plugs are provided for the

thermal column, exposure tubes, and other experimental facilities. The thermal-column end of the reflector is shielded by a movable, motor-driven steel-lined concrete-filled door; and space provided at the other end for a second horizontal thermal column is shielded by stacked concrete blocks.

The movable concrete door which shields the thermal column is motor-driven. The drive unit is located between the outer two wheels of the concrete door, and is easily accessible for maintenance. The door is operated by OPEN, CLOSE, and STOP pushbuttons mounted on the door's outer face. "Open" and "close" limit switches are also provided to automatically stop the door when it reaches either extreme of travel. An alarm sounds when the door is being operated. A key switch locks the door when not in use.

A bismuth gamma-ray shield adjacent to the secondary enclosure and a lead gamma-ray shield above and below the bismuth minimize the gamma-flux in the horizontal thermal column. A borated paraffin shield, called the beam catcher, reduces the thermal neutron flux entering the sub-pile room.

M. SUB-FLOOR AREA

The sub-floor plan, shown in Figure 8 includes the sub-pile room, recombiner room, valve room and exposure room. The sub-pile room is separated from the valve and exposure rooms by thick dense concrete walls. The other walls of the sub-floor are of ordinary concrete.

The sub-pile room has a steel plate liner on the walls, floor and ceiling, as part of the secondary enclosure. A gas-tight door provides access to the sub-pile room from the valve room when the reactor is not operating. Because it contains components of the gas handling system in which radioactive gases are circulating, the sub-pile room is not accessible while the reactor is operating. An interlock requires that the gas-tight door be closed before the reactor can be operated. Opening of the door while the reactor is in operation will automatically initiate a scram. An interlock furthermore allows the opening of the gas-tight door only when the recombiner room shield door is closed.

To protect personnel working in the sub-floor area, key operated Lock-Safe switches are installed in the valve room and the sub-pile room which, when

locked in the SAFE position, will prevent reactor startup. The keys are removable only in the SAFE position. Finally, an interlock button (reset on the outside of the gas-tight door) will have to be operated before the reactor can be started; this prevents the experimenter from being trapped in the sub-pile room.

Within the sub-pile room, an extension of the concrete wall shields the cooling system components from direct radiation from the gas handling system, so that the room can be entered and the cooling system approached without direct exposure to the gas handling system.

A wheel-mounted irradiation cart, 16 inches wide by 36 inches long is situated in the sub-pile room. A guide channel and an electrically powered chain drive lead into the recombiner area allowing the cart to be moved into regions of varying gamma fluxes. The cart and shield door are interlocked to prevent contact and remotely controlled from the reactor control room. Manual overrides are provided for both the cart and shield door.

All valves in the fuel and gas handling systems can be remotely operated by means of valve handles which extend through the wall into the valve room. All of these valves are normally closed. Lines extend from points in the fuel handling and gas handling systems through the wall of the sub-pile room, terminating at capped fittings in the exposure room.

N. EXPERIMENTAL FACILITIES

The experimental facilities of the reactor are shown in Figures 6, 7 and 8. These include:

- 1) Two horizontal beam tubes, 4 inch diameter
- 2) Four vertical beam tubes, 4 inch diameter
- 3) Two horizontal beam tubes, 3 inch diameter
- 4) One straight pneumatic tube, 2 inch diameter
- 5) One curved pneumatic tube, 1.5 inch diameter
- 6) One central exposure tube, 1.5 inch diameter
- 7) Two 6 inch square and two 6 inch round thermal column access ports

- 8) One access port opposite the thermal column, 24 inches high x 28 inches wide x 12 inches deep
- 9) One horizontal thermal column, approximately 5 feet x 5 feet x 5 feet
- 10) One thermal column access port, 1 foot square x 5 feet deep
- 11) One thermal column access port, 2.8 inches high x 36 inches wide x 20 inches deep
- 12) One thermal column access port, 60 inches square x 20 inches deep
- 13) One vertical thermal column, 54 inch diameter at its upper surface, and 24 inches thick.

The tube facilities consist of steel sleeves extending through the concrete shield, and aluminum thimbles or liners which extend into the immediate vicinity of the core. Each tube facility is equipped with a graphite reflector plug and one or more dense concrete and steel shielding plugs.

The horizontal thermal column (refer to Figure 6) consists of a 5 foot square column of graphite extending from the core to the thermal column shield door. In the center are removable graphite stringers, providing an access port about 28 inches high, 36 inches wide and 20 inches deep.

The shield on the side opposite the thermal column, consists of removable concrete blocks. A removable section of graphite about 24 inches square and 12 inches deep is provided in this area.

The vertical graphite thermal column (Figure 6), 24 inches thick and 54 inches in diameter, is in a container lined on its walls and partially on its base with a thin annulus of boral for neutron shielding. The container is structurally reinforced for lifting as a unit. Holes in flanges at its top permit securing an experimental exponential tank to the graphite pedestal.

Exposure facilities extend from the exposure room into the recombiner room near the gas recombiner (Figure 8) in order to utilize the gamma activity produced by the fission-product gases in the gas-recombiner tank. These facilities consist of three 4 inch steel sleeves and aluminum thimbles which extend through the dense-concrete walls of the recombiner room and one 1-1/2 inch stainless

steel thimble inside a stainless steel sleeve which passes through the gas recombiner. As with the beam tubes, each port is equipped with a plug to be installed for shielding purposes when the port is not in use.

V. ADMINISTRATION OF THE REACTOR FACILITY

A. ADMINISTRATIVE ORGANIZATION

The Director of the Division of Nuclear Medicine and Chemistry will be responsible to the Director of Walter Reed Army Institute of Research for policies and practices established for the use of this facility.

To execute this responsibility it is proposed that the Director of Walter Reed Army Institute of Research appoint a Reactor Advisory Committee which will consist of the Division Director, the Technical Director for the Reactor, the Research Advisor, the Chief of Reactor Operations and the Health Physics Supervisor. This committee may be enlarged if experience warrants it. The Advisory Committee will review reactor procedures and formulate all rules and regulations concerning the use of the reactor, its maintenance, safeguard measures, emergency procedures and security. Responsibility for specific supervision and enforcement rests with the Division Director.

Only persons who have been properly licensed by the Atomic Energy Commission will be given responsibility for reactor startup and operation. Training of specific individuals both at the facilities of the manufacturer of the reactor, Atomics International, and other reactor sites is contemplated. The operators will be indoctrinated thoroughly relative to the design and operation of the reactor, emergency procedures, and the official line of reporting within the organization.

The basic operations organization and the line of responsibility is indicated below.

B. REACTOR SURVEILLANCE

To prevent access to the reactor and its facilities by persons other than authorized individuals, and to implement the radiation health protection of personnel and visitors, the following surveillance procedures will be established.

During the working day there will be a receptionist in the lobby leading to the reactor room and sub-floor level to control visitors and personnel. Anyone wishing to enter the reactor area must pass a locked door buzzer system controlled by the receptionist. Radiation dosimeters such as film badges will be issued at this point. Visitors will register properly and be provided with an escort for the duration of their stay. After hours the doors will be key-locked. The door to the reactor area is provided with an alarm system to detect tampering.

Entrance to the building after normal working hours is controlled by a guard, who will check the entrance to the reactor rooms at regular intervals. In addition, the Center Military Police will patrol the entire medical center area on an all-night basis.

C. HEALTH PHYSICS

The Health Physics and Radiation Engineering functions are part of the Division of Nuclear Medicine and Chemistry. The responsibility for formulating radiological safety procedures in the reactor building resides with the Health Physics Supervisor, who is also designated the Radiological Safety Officer for Walter Reed Army Medical Center.

It will be the explicit responsibility of the Health Physicist to supervise the radiological safety aspects of any experiment, to direct waste disposal and decontamination procedures, utilizing where necessary other members of the staff, and to render reports of any noteworthy incidents to the reactor advisory committee.

The Health Physicist will be charged with making periodic inspections of the following physical facilities:

- 1) Air handling facilities
- 2) Laboratory surfaces

- 3) Health physics equipment
- 4) Alarm systems.

It will be the further responsibility of the Health Physicist to act as accountability officer for the fuel and all radioisotopes within the medical center. To this end, records will be maintained on all storage facilities, on disposal procedures and similar operations.

The health physics laboratory will be centralized in a specific area within the reactor building structure. The laboratory will house instruments which include ionization chambers as well as Geiger counter portable rate meters, at least one high dosage rate gamma ray probe, and adequate alpha and neutron detectors. Furthermore, portable air monitoring equipment for use in areas where air-borne contamination is likely will be at hand. The Health Physicist will be responsible for checking records on automatic air monitoring and surface monitoring equipment which will be installed in the reactor area, and in the appropriate radio-chemical laboratories. These procedures will assure adequate instrumentation for the laboratory, with the advantage of permanent recording of the conditions prevailing in the laboratory at any time. Located in the health physics laboratory will be gas masks, decontamination clothing, and similar safety devices.

Before individuals are permitted to work within the reactor building, they will be given detailed instruction by the staff on radiological safety procedures. This instruction will include the rules of radiological safety, the use of survey instruments and the specific details of the action to be taken in an emergency.

Personnel exposure will be monitored by film badges supplied and read by the U.S. Army Signal Corps, Lexington Signal Depot, Lexington, Ky. These badges will be worn by all personnel in the reactor area from the time they enter until they leave. Special dosimeters will be issued to operating and research personnel as needed.

D. RADIOACTIVE WASTE DISPOSAL

The proposed radioactive waste disposal procedures will be in accordance with existing Army procedures. All radioactive waste created by the reactor

located at Walter Reed Army Institute of Research will be accumulated in one central location. This waste will be sealed in appropriate cartons and transported to the U.S. Army Chemical Corps facility at Edgewood, Maryland, in accordance with Army Regulation 755-380. It will then be taken by the above mentioned organization and placed in concrete filled drums and dumped at sea by the U.S. Navy according to AEC regulations. The withdrawal of radioactive gases from the interior of the system would be accomplished by the use of evacuated cylinders and transported to the above disposal group should it become necessary.

VI. REACTOR OPERATION

A. REACTOR STARTUP

Prior to reactor startup, instruments and auxiliary equipment are placed in operation, and the entire reactor is checked for indication of any condition which might lead to unsafe operation. The pre-operational procedures for reactor startup are compiled in a check list similar to Figure 14, and a scram checklist as shown in Figure 15. Each check list is filled in by the reactor operator and examined by the shift supervisor. Only if all conditions on the check list are satisfied will the reactor be started up.

The reactor is brought up to operating power by withdrawing the control rods, one at a time, until the reactor period is positive. As the desired power level is approached, one control rod is moved inward until the period becomes infinite and the power level remains constant.

During startup, the reactor power level will be increased slowly to maintain maximum sensitivity in measuring power level. The linear count rate meter and the electrometer which feeds the power level recorder will be kept on the lowest ranges possible as the power increases, being switched to the next higher range only as the power reading approaches full scale.

The step-by-step procedures for pre-operational inspection and reactor startup are outlined in an operating and maintenance manual published by Atomics International.⁶

B. NORMAL OPERATION

During normal operation the reactor may be controlled automatically, though the power level and period are under the surveillance of the operator. Whenever the power level is to be changed, all instruments will be checked for normal readings similar to the procedure before startup.

At regular intervals (once each hour) the temperature and pressure readings will be checked and recorded. Since certain reactor parameters (core temperature, coolant temperature difference, catalyst-bed temperature and sweepgas flow) vary with reactor power, these reactor parameters will be checked for consistency.

OPERATIONAL CHECKLIST

[illegible]

Figure 14. Typical Preoperational Checklist

SCRAM CHECKLIST

[illegible]

Figure 15. Typical Safety Circuit Checklist

C. REACTOR SHUTDOWN

At the end of a normal operating period, the reactor will be shut down by the operator by pushing the scram button. The following equipment and instruments will then be shut off in the order indicated:

- 1) catalyst-bed heater
- 2) recombiner pump
- 3) primary coolant pump
- 4) chart drives on all recording instrument
- 5) console power
- 6) water supply to secondary cooling system

D. REACTOR SCRAM

The instrumentation and control system components are divided into two categories with reference to alarms and scrams. In the first category are those instruments whose abnormal indications, while not serious enough to scram the reactor, sound an alarm at the console and activate the annunciator. Example of such indications are high surge-tank pressure and low surge-tank level. In the second category are those instruments whose abnormal indications will scram the reactor. Scram is caused during startup or normal operation by any of the following conditions:

- 1) core tank temperature excessively high
- 2) catalyst-bed temperature excessively high
- 3) linear power level excessively high
- 4) log N level excessively high
- 5) reactor period excessively short
- 6) recombiner gas pressure excessively high
- 7) sub-pile room door not closed
- 8) Lock-Safe switch to valve room in SAFE position
- 9) Lock-Safe switch to sub-pile room in SAFE position

- 10) power failure
- 11) failure to any ion chamber power supply
- 12) safety chassis relay not energized
- 13) sweepgas flow rate below minimum value
- 14) earthquake, acceleration above set point
- 15) radioactive gas in secondary enclosure above set point
- 16) Manual scram switch open

A scram causes the current to the control-rod magnets to be cut off, thus allowing the rods to drop into the core. Also the main alarm sounds, and the condition causing the scram is indicated on the annunciator panel. For a scram resulting from a reactor power surge the overall response time between the ionization-chamber signal and the full insertion of the four rods is less than one-half second.

E. INSPECTION AND SERVICING

The reactor and the auxiliary equipment is designed for service free operation under normal operating conditions: inspections other than those specified under reactor startup and normal operations are not required. If, however, it is desired to inspect or service any part of the reactor or auxiliary equipment after operations at appreciable power, it is essential that procedures are followed which will minimize radiation hazards to personnel. All radiation hazards associated with reactor servicing procedures depend on the particular conditions under which the servicing is performed. For this reason, protective measures must be determined each time an operation is performed.

The procedures to be followed for servicing of the various components are outlined in the operating and maintenance manual for the L-54 published by Atomics International.⁶

VII. REACTOR HAZARD ANALYSIS

The ultimate potential hazard is the widespread dispersal of the fuel and its inventory of fission products. Spills of radioactive liquids; small scale releases of radioactive gas; excessive radiation intensities resulting from localized shielding failures, from open beam tubes, or from improper design of experimental equipment shielding --- all these represent rather localized incidents. While hazardous, such incidents do not affect more than a few members of the reactor staff, who are fitted by their professional training and experience to apply the necessary counter-measures. Such incidents do not affect the general population near the reactor site.

In order for widespread dispersal to occur, it would be necessary to breach the primary enclosure and the secondary enclosure.

The energy required to initiate the sequence of container breaks from the core and primary enclosure outwards can come from two sources: a nuclear excursion and/or a chemical reaction in the core.

A. NUCLEAR EXCURSIONS

The energy released during a nuclear excursion can cause rupture of the primary container only by raising the pressure, or temperature of the core tank contents to a sufficiently high value. Because of the large negative temperature and void coefficient of reactivity of the water boiler type reactor, the reactor will shut itself down a fraction of a second after the initiation of any kind of excursion. Only small temperature increases would result, much too small to cause any damage to the structure. Transient pressures, however, would occur during some types of excursions and the cause of these pulses and their characteristics will be discussed in section H, below.

B. POTENTIAL CAUSES OF EXCURSIONS

Nuclear excursions could be initiated by withdrawing absorbing material from, inserting reactive material into, or flooding of the reactor core or reflector region. Withdrawal of absorbing material is accomplished by removal of such material from the experimental facilities in the core or reflector, or by withdrawing

control rods from the core. Addition of reactive material could result from insertion of such material into either the core, reflector or experimental facilities or by changing the U/H ratio. In the following paragraphs each condition of this section is analyzed and evaluated.

C. CONTROL ROD WITHDRAWAL

The reactivity controlled by each control rod is approximately 2 percent; an uncontrolled withdrawal of a rod therefore, either by the operator or due to a malfunction of the automatic rod regulating system with simultaneous failure of the automatic scram circuit, would introduce a ramp reactivity with a total of about 2 percent. Removal of more than one control rod under these conditions does not seem credible, since only one rod is manipulated by the automatic control and interlocks prevent the removal of more than one rod at a time. Even if all rods were withdrawn simultaneously and at maximum speed, the resulting excursion would be less severe than the step input excursion discussed in the next section.

D. REMOVAL OF NEUTRON ABSORBER

The majority of experiments conducted with the L-54 reactor will consist of placing different materials into the experimental facilities in the core or reflector region. Since all materials are neutron absorbers to a greater or lesser extent, the removal of these materials while the reactor is still in operation might introduce a step function of reactivity. This occurrence, however, is only possible as a result of gross negligence by the reactor personnel or deliberate disregard of established operating procedures. In such case there would be no effect greater than that caused by the release of the excess reactivity loaded into the reactor at the time of the excursion. Since the excess reactivity at zero power (70°F) is about 3 percent for the design power of 50 kilowatts, it is possible to insert a step reactivity of this amount into the reactor. However, since this would infer that all control rods had been withdrawn before removal of the absorber without making the reactor critical, an occurrence of this kind would also require that the control and safety system be inoperative.

An incident such as the one described above is highly unlikely, in fact it seems that this kind of accident would require definite planning. In section

VII-G and VII-H, the effect of such a step input of reactivity on power, temperature and pressure will be discussed in detail.

E. ADDITION OF REACTIVE MATERIAL

A supercriticality incident would require the insertion of reactive material into either the core or reflector, through the central exposure facility, removable graphite stringers, fuel lines, gas lines, control or safety rod openings, or cooling coils. The most probable ways of obtaining such an insertion of reactivity would be through overloading the core with fuel solution or inserting material of unanticipated reactive character into the experimental facilities. The amount of reactivity introduced as a result of the incident discussed above, would depend upon the particular circumstance and cannot be evaluated in advance. However, the probability of such an occurrence with simultaneous control and safety system failure is considered to be very small. The inherent safety features of the reactor (discussed in section G, below) would minimize any resulting nuclear excursions.

F. U/H RATIO

A change in the U/H ratio can occur most likely as a result of loss of the primary coolant or because of a break in the core region of the coolant system. Loss of cooling water from the core would result in a loss of reactivity. This loss is estimated to be less than 1 percent.⁷

A leak in the primary cooling system, with a consequent introduction of cooling water in the fuel solution would result in an increase in reactivity. This is largely due to the improved geometry of the system (approaching spherical shape) and the cooling of the fuel solution. Since the cooling water circulation time is relatively large compared to other reactor constants, even for a total break, with maximum water influx, this can only be considered as a slow ramp increase in reactivity with an estimated total of about 1 percent.⁸

G. INHERENT SAFETY

All experimental evidence on the solution type reactor shows that this kind of reactor is inherently stable under virtually all conditions of reactivity insertion.

The factors responsible for the self-regulating behavior of this type reactor are the large negative void coefficient, the large negative temperature coefficient, and the function of the overflow system.

A temperature coefficient of reactivity of about -0.025 percent per degree centigrade has been established for this reactor. This is consistent with experimental data on the Livermore (L-3), Armour (L-8) and KEWB reactors, all of which are solution-type reactors similar to the L-54. The measured temperature coefficients for those reactors are all negative, and have ranged from -0.015 to -0.03 percent reactivity per degree centigrade.

In the Kinetic Experiment Water Boiler (KEWB) program, conducted by Atomics International for the United States Atomic Energy Commission, the dynamic behavior of the solution-type reactor has been investigated extensively. The experimental data⁹ show that the radiolytic gas coefficient of reactivity is significant in effecting reactor shutdown both below and above prompt critical.

Although the KEWB reactor differs in some details from the L-54, the kinetic behavior of the L-54 is not expected to differ essentially, and conclusions as to the behavior of this reactor under different conditions of reactivity insertion are therefore drawn from the KEWB experiments. Figure 16 shows a typical plot of power, energy and overall reactivity as a function of time for an initial step increase of 1.03 percent reactivity. A similar distribution of power and overall reactivity can be expected in the L-54 for the possible step function discussed in section VII-D. A plot of peak power as a function of period in the range from 2 milliseconds to 30 seconds, corresponding to step reactivities of about 0.1 to 3.7 percent is shown in Figure 17. The temperature as well as the final pressure increase in the core, is very small since the energy release is strongly limited by the shutdown mechanisms.

Table II shows the latest data for step and ramp reactivity inputs for the KEWB. It can be seen that the integrated power released in a 3 percent step input is only about 3.3 Mw-sec. This is barely enough to raise the fuel temperature, without cooling, by about 40°F. The reactor will stabilize at a relative low power level after the initial power surge. This level for the KEWB is between 2 and 150 kilowatts, depending on the primary coolant flow rate.

Figure 16. Power, Energy and Reactivity During a Transient with a Reactor Period of 32 Milliseconds in KEWB

Figure 17. Peak Power versus Reactor Period

TABLE II
KEWB DATA

Reactivity Input	$\frac{\Delta k}{k}$	Stable Period	Peak Power	Integrated Power
Step	3%	2.8 msec	300 Mw	3.3 Mw-sec
Ramp	0.13%/sec	60 msec/min.	2.1 Mw	

Even though quick release of energies of the above magnitude will not result in large fuel temperatures, there will be a tendency for the solution to be expelled from the core, at least during the initial stage in which gas bubbles form but remain in the solution.

The overflow system is designed to hold two liters of solution at an average distance of 16 inches from the center of the core tank. The expanding fuel solution is carried up to the 2 inch (I.D.) pipe, is deflected by the curved plate, and collects in the annular space surrounding the pipe (Figure 10). This fuel drains back to the core through a 1/8 inch hole in the inside wall. About 2 minutes are required to return 2 liters. This time interval should permit effective shutdown measures to be taken even if the rod system is inoperative.

Table III lists some of the characteristics of solution type reactors which have been built by Atomics International. All the reactors, with the exception of the L-54 Milan which is in the construction stage, have been operated successfully and without nuclear incidents from the startup date indicated.

H. TRANSIENT PRESSURES IN THE CORE

Transient pressures inside the core are observable with reactor periods of less than 30 milliseconds. These pressures occur when the expanding solution hits the constriction at the entrance of the 2 inch pipe. In all experiments performed with the KEWB, the pressures developed were largest at the top of the sphere. They occur in the form of a pulse having increasing magnitudes for

TABLE III
CHARACTERISTICS OF SOLUTION TYPE REACTORS

Reactor	Location	Startup Date	Maximum Power (kw)	Maximum Thermal Flux (n/cm ² -sec)	Enrichment (%)	Critical Mass (gr)
WBNS	Downey, Cal.	1952	.002	2×10^8	90	634
L-3	Livermore, Cal.	1953	.5	2×10^{10}	90	694
L-6	Santa Susana, Cal.	1956	1.5	7×10^{10}	93	756
KEWB-1	Santa Susana, Cal.	1956			90	1680
L-8 Armour	Chicago, Ill.	1957	50	2×10^{12}	90	1200
NCSR II	Raleigh, N.C.	1957		8.6×10^{11}	90	800
L-54 Japan	Tokai Mura, Japan	1957	62	10^{12}	19	1170
L-55 Denmark	Riso, Denmark	1957	2		19.9	984
L-47	Canoga Park, Cal.	1957	.005		89	1196
L-54 Frankfurt	Frankfurt, Germany	1957	50	10^{12}	19	1200
L-54 Berlin	Berlin, Germany	1958	50	10^{12}	19	1200
L-54 Milan	Milan, Italy	1959	50	10^{12}	19	1200
L-77	Canoga Park, Cal. (Univ. of Wyoming)	1958	.01		90	1150
L-77	Geneva, Switzerland (Puerto Rico)	1959	.01		20	1380

decreasing reactor periods. The pressure pulse observed for a step reactivity input of 3 percent has a magnitude of about 220 psi and a duration time of approximately 2 milliseconds. In section X-B, the significance of these pressure pulses concerning potential hazards is discussed in detail.

VIII. CHEMICALLY INDUCED REACTIONS

The most probable reaction among the chemical constituents of the primary enclosure is between the hydrogen and oxygen that are produced by radiolysis of the core solution during reactor operation. The uranyl sulfate solution present in the core produces approximately 18 liters at STP of stoichiometric hydrogen-oxygen mixture per kilowatt-hour of operation.

The reactor atmosphere at startup is initially pure oxygen and some water vapor, somewhat but not appreciably below 1 atmosphere total pressure. The core solution temperature at 50 kilowatts is about 175°F. At this temperature, the equilibrium partial pressure of water vapor from the uranyl sulfate solution is about 0.46 atm. This represents a maximum value. The minimum equilibrium partial pressure of water vapor in the system is about 0.03 atm, determined by the temperature of the water working fluid in the jet-pump, which is about 75°F. During operation at 50 kilowatts, the ratio of hydrogen to oxygen varies from a maximum near 2, just at the fuel solution surface, to a minimum near zero, downstream from the catalyst bed.

The thermodynamically favored hydrogen-oxygen reaction can proceed at rates characterized by the designations, and within the composition limits, discussed below.

A. INFLAMMATION

This is the smooth burning of hydrogen in oxygen. The flammability range of composition is defined as that in which a combustible mixture at atmospheric pressure and room temperature will propagate in a tube 2 inches or more in diameter. The lower limit of flammability for hydrogen-oxygen mixtures is ¹⁰4.65 percent hydrogen by volume. The upper limit is 93.9 percent hydrogen. Mixtures outside these limits will not propagate a flame, or any of the more rapid reactions discussed below.

B. EXPLOSION

In a normal explosion, as in smooth burning, the process is controlled by chemical reaction rate and mixing processes, and is characterized by a relatively low velocity of flame propagation.

The explosion limits for mixtures of hydrogen and excess oxygen are not known to have been measured. However, for stoichiometric mixtures of hydrogen and oxygen diluted by water vapor the lower limit for explosion is about 17 percent of stoichiometric mixture by volume.¹¹ Assuming that excess oxygen is equivalent to excess water vapor in its effect on the explosive properties of the mixture, this value translated to the present reactor atmosphere indicates that the lower limit for explosion is about 11 percent hydrogen by volume.

C. DETONATION

In a detonation, the reaction proceeds at a very high rate as in an explosion, but the process is largely a hydrodynamic phenomenon and is characterized by a shock wave which travels through the unburned gas at supersonic velocity.

Hydrogen-oxygen mixtures initially at one atmosphere pressure give detonability limits of 15 to 90 percent of hydrogen by volume. Although these data do not eliminate the possibility of existence of a detonation wave in mixtures outside the range, there is clear experimental evidence that such a wave rapidly degenerates into a compression wave.¹²

The composition limits are summarized in Table IV. These limits are relatively insensitive to pressure and, to a lesser extent, to temperature. For example, the explosion limit quoted is nearly constant from 1 to 68 atmospheres.

TABLE IV

FLAMMABILITY, EXPLOSION AND DETONATION LIMITS IN H_2 - O_2 MIXTURES

(Initial Conditions: Pressure = 1 atm. Temperature 65°F)

	Lower Limit (volume % H_2)	Upper Limit (volume % H_2)
L-54 Design Point	<4	
Inflammation	4.65	93.9
Explosion	11	--
Detonation	15	90

D. EXPERIMENTAL STUDIES OF H_2-O_2 REACTIONS

Theoretical analysis¹³ and recent experiments performed by Atomics International on stoichiometric H_2-O_2 reactions have shown that the maximum resulting pressures at the reflecting surfaces are about 40 times ambient. The pressure pulses decay in an exponential fashion and have a time constant of about 0.2 msec. Experiments performed on prototypes of the core and gas handling system have yielded the following results.¹⁴

- 1) In normal operation at 50 kilowatts, there was no detectable ignition or explosion.
- 2) In simulated 3 megawatt-second nuclear runaways, in which 15 liters of stoichiometric H_2-O_2 were introduced very quickly into the core tank, no explosion occurred in the core. Flashbacks occurred at the catalyst, but did not propagate back into the core. No detonation occurred.

These results have a high reliability. They indicate that a detonation is unlikely to occur in the system.

E. CORROSION

Extensive long term corrosion tests have been conducted at Atomics International on various stainless steels. These tests, coupled with information developed at Oak Ridge National Laboratory and other installations, demonstrate that corrosion of stainless steels by clean, almost chloride free uranyl sulfate solutions at temperatures under 212°F is not a problem.

Coupons of commercial stainless steels of the following types: 304 (19% Cr, 10% Ni), 309 (23% Cr, 14% Ni), 310 (25% Cr, 21% Ni), 316 (18% Cr, 12% Ni), 321 (19% Cr, 10% Ni, Ti), and 347 (19% Cr, 11% Ni, Cb), have been tested in baths of uranyl sulfate solutions up to 2.5 molar with added sulfuric acid. The averaged results are summarized in Table V.

Before placing in the test baths, all coupons were pickled in a 1:1 HNO_3 solution containing 1.5 percent hydrofluoric acid, followed by passivation in 1:1 HNO_3 and a distilled water rinse. As can be seen by inspection of the table,

all corrosion rates are extremely low, never exceeding 0.2 mils/yr. Metallographic examination of the test coupons indicates no intergranular or pitting type of corrosion attack.

TABLE V
CORROSION TEST OF STAINLESS STEEL COUPONS

Stainless Steel Type	Uranium Concentration gms/liter	Molarity H_2SO_4	Corrosion Rate mils/yr
304	270	0.5	0.102
309	270	0.5	0.043
310	270	0.5	0.043
316	270	0.3	0.102
321	250	0.5	0.165
347	270	0.3	0.086
304	600	1.0	0.178
310	600	1.0	0.074
316	600	1.0	0.130
316	450	0.5	0.097
321	450	0.5	0.113
347	450	0.5	0.080

Coupon surfaces were found to be free of discolorations or other unaccounted for blemishes. Because the fuel used in these reactors originates as uranium hexafluoride, there has been some concern about excessive amounts of fluorides carrying through the chemical processing into the finished sulfate. Corrosion checking of coupons exposed in solutions with up to 50 ppm fluorine has indicated no accelerated corrosive action. Coupons containing sample welds have similarly been checked, and weight losses are consistent with the results reported for unwelded samples.

After operation for three years, the primary system of the solution type reactor at the Radiation Laboratory of the University of California was opened

up for inspection of the inside surfaces. Micro-techniques of chemical analysis were also utilized to analyze the fuel solution for corrosion products, iron, chromium, and nickel. The ratio of these metals was found to correspond roughly to the composition of the 18-8 stainless steel (type 316L). It was determined that not more than 2 grams of stainless steel total had been removed from the core vessel. This corresponds to an average corrosion rate of approximately 0.02 mils/yr, a factor of two less than the lowest laboratory results. Metallographic examination of the core vessel again showed no sign of intergranular or pitting type of attack.

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IX. RADIATION HAZARD ANALYSIS

A. RADIOACTIVITY INVENTORY

1. Fission Products

Using the method outlined by Blomeke and Todd,¹⁵ the fission product inventory after long operation (10^8 sec) at 50 kilowatts is 1.54×10^5 curies. The more volatile isotopes (Xe, Kr, Br and I) comprise a theoretical source of 2.7×10^4 curies at one minute decay, with Br and I initially making up about 50 percent of the total.¹⁶

2. Other Sources

There are no other sources of radioactivity of primary importance associated directly with the reactor during normal operation. The activation of the stainless steel core and its internal components could approach a source strength of the order of ten curies after long operation at 50 kilowatts. Some argon-41 (A^{41}) production in void spaces inside the biological shielding will occur. The maximum concentration possible (with normal air as the source) is $\sim 3 \mu\text{c/cc}$ in the central exposure tube. The maximum amount possible in a given location is $\sim 40 \text{ mc}$ (at $\sim 3 \times 10^{-2} \mu\text{c/cc}$), which can be obtained only by taking out all of the removable graphite blocks in the thermal column and irradiating the entire static air space remaining. The primary coolant water is highly purified and its activation is inconsequential in view of the small volume and the short half-lives of the products formed.

B. RADIATION HAZARDS DURING NORMAL REACTOR OPERATION

1. Radiation Levels

Radiation levels in the immediate vicinity of the reactor are, under normal conditions and full power, expected to be below 7.5 mr/hr. Some experiments, however, will necessitate removal of some of the graphite stringers or cement blocks in the experimental facilities; this might change the radiation levels adjacent to the reactor. The safety precautions, however, which have been adopted by the Division of Nuclear Medicine and Chemistry, will insure radiological safety for all personnel working at the reactor. The radiation level in the rest of the reactor room is expected to be less than 2.5 mr/hr.

4. Radioactive Waste

During the critical assembly and possibly as the result of a servicing operation which requires access to the primary system (fuel sampling), small quantities (1 - 10 gallons) of miscellaneous contaminated materials (such as rubber gloves, absorbent paper, rinse solutions and glasswater) may be accumulated.

Some quantities of A^{41} will be produced during normal reactor operation, but these will be ordinarily dilute enough to permit direct and continuous disposal by the existing ventilation system. During the critical assembly and possibly at infrequent intervals thereafter, limited quantities (1 - 10 cubic feet) of fission gases may be accumulated in the waste tanks provided.

The only significant exception to the estimates given above would occur in the event that extensive decontamination of the primary fuel system became necessary. In this case, the following volumes of active waste might result:

Liquid	100 gallons
Solid	20 cubic feet (uncompressed)
Gaseous	20 cubic feet (STP)

About 80 percent of these volumes would be of relatively low level (no shielding required).

From the foregoing summary, it can be seen that the radiological hazard associated with the accumulation and disposal of the active waste that results directly from the operation of the reactor under normal conditions will be of minor significance.

C. DIRECT RADIATION DOSE RESULTING FROM A POWER EXCURSION

The integrated dose from a power excursion can be related to the dose rate at any point during normal full power operation of the reactor in the following manner,

$$D = 5.6 \times 10^{-6} M K \quad \dots (3)$$

where,

D = integrated dose from power excursion (mr)

M = dose rate at 50 kilowatts at point of interest (mr/hr)

K = integrated power of excursion (kw - sec)

With the available excess reactivity, the maximum possible power excursion would be of the order of 3.3 Mw-sec. Substituting in equation (3) above,

$$D = 1.85 \times 10^{-2} \text{ M (mr)}$$

Thus the maximum integrated direct dose would not be greater than that received in the same location in about 1.3 minutes of 50 kilowatt operation.

D. RADIATION HAZARDS RESULTING FROM THE RELEASE OF FISSION PRODUCTS FROM THE PRIMARY AND SECONDARY CONTAINMENT ENCLOSURES

It is most probable that any release of fission product activity from normal containment would preferentially involve the more volatile isotopes. These fission products, composed largely of Xe and Kr isotopes, are constantly being swept from the fuel solution by the formation of H_2 and O_2 in the water disassociation process when the reactor is operating. Subsequently, they are entrained in the sweepgas stream which circulates through the recombiner-core loop. The detailed isotopic makeup on this source, beyond the verification of the presence of the more prominent Xe and Kr isotopes, has not been rigorously determined experimentally. It is generally felt, on the basis of some limited experiments with various water boiler reactors (KEWB, WBNS) and with the North Carolina State reactor, that the iodine and bromine isotopes are almost completely inhibited, relative to the Xe and Kr isotopes, from leaving the fuel solution. The most probable explanation of this is based on their (I and Br) chemical activity and solubility. Nevertheless, since such an assumption cannot be quantitatively specified at this time, the calculations which follow are based on a 100 percent release of the I and Br fission product inventory as well as that of the Xe and Kr. At one minute decay, following long operation at 50 kilowatts, the makeup of the fission products has been calculated as shown in Table VI.

TABLE VI
VOLATILE FISSION PRODUCT INVENTORY

Isotope	Curies
Xe + Kr	1.0×10^4
I	1.3×10^4
Br	0.4×10^4
Total	2.7×10^4

This activity is assumed to be instantaneously and uniformly dispersed throughout the reactor room. In computing the radiological hazard resulting from such a hypothetical release, it can be shown that two areas of potential radiation dose are dominant; the direct whole body gamma dose and the thyroid dose from the inhalation of the iodine isotopes.¹⁷ Therefore, the calculations are limited to these two considerations.

1. Radiological Hazards - Interior of Reactor Room

Obviously, an instantaneous dispersion of fission product activity of the assumed magnitude (2.7×10^4 curies) in the reactor room is of grave concern to the occupants and immediate evacuation is mandatory. The dose received by anyone in the room while evacuating is highly dependent on the particular location and movements of the individual relative to those same characteristics of the fission product release. However, a reasonable limit can be set by placing the individual in the center of the reactor room with all the activity instantaneously dispersed through its volume. It is further assumed that the ventilation system has been shut down. The gamma dose rate at one minute decay has been calculated as ~ 3 r/min. The infinite thyroid dose due to the inhalation of iodine isotope (chiefly I^{131} and I^{133}), under the same conditions, would be $\sim 7 \times 10^3$ rad per minute of exposure.

2. Radiological Hazards - Exterior to Reactor Room

Three sources of potential radiation exposure are considered for the environmental areas outside of the reactor building, gamma exposure from the fission products contained within the reactor room, gamma exposure from the leakage plume, and the thyroid dose from the iodine fission product leakage.

a. Direct Gamma Dose from Contained Fission Products

The accumulated gamma doses at several selected distances due to the contained fission products are shown in Table VII. The shorter distances have some particular significance as noted below:

300 feet	closest occupied building
500 feet	closest hospital patients
700 feet	closest boundary of Walter Reed Medical Center (Alaska Ave.)

Depletion of the source by leakage is neglected in this calculation.

b. Direct Gamma Dose from Leakage Plume of Fission Products

Leakage from the reactor room is assumed to be 5 percent of the room volume per 24 hours. The downwind doses were calculated using the nomogram developed by Holland¹⁸ for two types of weather conditions (see Table VIII). The leakage is assumed to occur as a point source at ground level, and an average gamma energy of 0.7 Mev was used. The integrated doses at several distances directly downwind are shown in Table VII, for exposure periods of one, two, and four hours under the most unfavorable weather condition (inversion).

c. Thyroid Dose

To estimate downwind concentrations of the iodine isotopes leaking from the reactor room, the general air diffusion expression developed by Sutton¹⁸ and later modified for a fallout correction by Chamberlain¹⁹ was used. A continuous point-source release model at ground level was chosen. The downwind concentration is given by,

$$X(x, 0) = \frac{2Q}{\pi C^2 \bar{\mu} x^{2-n}} \cdot e^{-\frac{4v_g}{n\mu C\sqrt{\pi}}} \quad \dots (4)$$

where

- $X(x, 0)$ = ground concentration at point x , located directly downwind for $y = 0$ (curie/m³)
- Q = release rate (curie/sec)
- C = generalized diffusion coefficient for isotropic turbulence ($C = C_x = C_y = C_z$)
- $\bar{\mu}$ = average wind speed (meters/sec)
- x = distance downward (meters)
- n = non-dimensional parameter associated with stability
- v_g = velocity of particle deposition (for iodine vapor), (meters/sec)

The diffusion parameters associated with the meteorological conditions were chosen consistent with present convention^{18, 19} for two weather situations. These parameters are summarized in Table VIII.

TABLE VII

SUMMARY OF ENVIRONMENTAL DOSES FROM GROUND LEVEL LEAKAGE OF
RADIOACTIVITY UNDER INVERSION (STABLE) CONDITIONS

Distance from Reactor Building (ft)	Infinite Gamma Dose from Fission Products Contained in Reactor Room - (r)	Gamma Dose from Fission Product Leakage - (r)			Thyroid Dose from Iodine Leakage (rad)		
		Exposure Time (hrs)			Exposure Time (hrs)		
		1	2	4	1	2	4
300	0.080	0.020	0.038	0.072	22	35	65
500	0.030	0.014	0.026	0.048	8.0	13	24
700	0.015	0.011	0.019	0.036	4.2	6.5	13
2640 (0.5 mi)	0.001	0.004	0.007	0.013	0.25	0.40	0.75
5280 (1.0 mi)		0.002	0.004	0.007	0.070	0.11	0.20

TABLE VIII

METEOROLOGICAL AND DIFFUSION CONSTANTS USED

Stability	$\bar{\mu}$ (meters/second)	C	n	v_g (meters/second)
average (neutral)	3	0.20	0.25	1.3×10^{-2}
inversion (stable)	1	0.05	0.55	6.0×10^{-3}

The accumulated "infinite" thyroid dose was calculated at several distances downwind for exposure periods of one, two and four hours under the more adverse weather conditions (inversion). These values are shown in Table VII. No allowance was made for changing weather conditions.

d. Instantaneous External Release of Reactivity

If the release of radioactivity occurred instantaneously at an elevation above ground level, the resulting hazards would be somewhat different than those previously calculated. To examine this situation, it is assumed that all of the activity is released instantaneously as a point source at the elevation of the reactor room ventilation exhaust duct. In this case, the maximum downwind concentration was calculated by¹⁸

$$\chi_{\max} = \frac{2Q}{e\pi\bar{\mu}h^2} \quad \dots(5)$$

where,

- $\chi_{\max}$ = maximum integrated exposure at any point downwind
(curies-sec/meter³)
- Q = activity released (curies)
- $\bar{\mu}$ = average wind speed (meters/sec)
- h = release height (meters)

The release height was taken as 24 meters (~80 feet), with no additional lift due to velocity or temperature. Depletion of the source by fallout as it traveled downwind was neglected. The distance downwind at which this maximum exposure occurs is also of interest and may be calculated by

$$d_{\max} = \left(\frac{h^2}{C^2} \right)^{\frac{1}{2-n}} \quad \dots(6)$$

Table IX summarizes the integrated doses at the points of maximum exposure for the two types of weather conditions considered.

TABLE IX
SUMMARY OF MAXIMUM ENVIRONMENTAL DOSES FROM
INSTANTANEOUS ELEVATED RELEASE

Weather Type	Integrated Dose		Maximum Dose Distance
	Whole Body Gamma (r)	Thyroid (rad)	
average (neutral)	0.65	18	790 feet
inversion (stable)	1.5	54	3.1 miles

X. ACCIDENT EVALUATION

In the previous parts of this report, the reactor operation, potential causes of accidents and the hazards arising from a radioactivity release, were developed and evaluated. Since it is impossible to predict any particular accident and its consequences with any degree of accuracy, only one particular circumstance will be investigated: the accident which would yield the highest possible radiological hazard, but which is still considered credible.

A. DESCRIPTION OF MAXIMUM CREDIBLE ACCIDENT

The highest possible radiological hazards would be realized when the reactor contents (fuel solution, and fission products) are released to the surroundings. This, however, is only possible when both the primary and secondary enclosures are damaged, at least to the extent that both envelopes lose their gas tightness.

An accident of this type can originate from two sources, an internal pressure increase in the reactor or the gas handling system, or an external force to the reactor structure. Since externally applied forces can only be caused by earthquakes, floods or tornadoes which have a negligible probability of occurrence, or by willful attempts of an individual to destroy the reactor, they are impossible to predict and their evaluation will therefore be omitted. An internal pressure increase again, might originate from two sources, a pressure pulse as a result of a step or a fast ramp increase in reactivity (section VII-C and VII-D) or from an H_2-O_2 reaction (section VIII).

Since the pressure pulse (section VII-H) is largest in the top of the core, and the core vessel is the weakest part of the system, and since the H_2-O_2 mixture most conducive to a reaction (stoichiometric) occurs in the top of the core also, the two conditions of pressure increase will only be investigated for the reactor core vessel.

B. CAUSE OF MAXIMUM CREDIBLE ACCIDENT

The pressure pulses which have been observed in the KEWB during a large step-input of reactivity or during a H_2-O_2 reaction could occur in the L-54 under similar circumstances. The effect of these pressure increases on the structural

integrity of the core, depends not only on the magnitude of the pressure wave, but also on the time that this pressure acts on the vessel wall. For very short pulses of loading, the yield point and the ultimate stress of a material increases. Experimental results of H_2-O_2 detonation loading on 304 stainless steel specimen have been reported²⁰ and it was found that the ratio of the dynamic ultimate yield stress to the static stress is about 1.28. If it is assumed that this ratio is the same for SS 347, the maximum stress the vessel can experience without rupturing is:

$$\sigma_{\max} = 1.28 (80000) = 102400 \text{ psi}$$

1. Maximum Stresses in the Core Vessel During a 3.3 Mw-sec Nuclear Excursion

A stress analysis for fuel expansion impulse pressures for a 3 percent step-reactivity input has been made by use of equations developed for dynamic internal loading of spherical shells.²¹ The radial deflections are given in equations (7) and (8), assuming elastic behavior. (This assumption is justified because the elastic limit of a material increases for shock loading and in fact approaches the ultimate stress.)

$$u_r = K \left(1 - \frac{t}{T} - \cos w t + \frac{\sin w t}{w T} \right) \quad \dots (7)$$

for $0 < t < T$

$$u_{r \max} = K \sqrt{1 + \frac{2}{(w T)^2} - 2 \frac{\cos w T}{(w T)^2} - 2 \frac{\sin w T}{w T}} \quad \dots (8)$$

for $t > T$

where
$$K = \frac{p}{w^2 \rho d}$$

$$w = \sqrt{\frac{2 E}{\rho b^2 (1 - \nu)}}$$

u_r	=	radial deflection (in.)
p	=	internal pressure (psi)
ρ	=	mass density (lb sec ² /in. ⁴)
E	=	modulus of elasticity (psi)
d	=	shell thickness (in.)
b	=	shell outer radius (in.)
t	=	time (sec)
T	=	duration time of pressure pulse (sec)
ν	=	Poissons ratio

The maximum impulse pressure for a 3.3 Mw/sec excursion is about 220 psi (section VII-H). Using this value and the minimum L-54 shell thickness of 0.071 inch and radius of 7.88 inches, the deflection was found from equation (7) to be a maximum at $t \sim 10^{-5}$ sec and is equal to

$$u_r = 4.5 \times 10^{-3} \text{ in.}$$

the maximum stress resulting from this deflection is equal to

$$\sigma = 18,850 \text{ psi}$$

This is smaller than the yield strength of type 347 stainless steel (about 30,000 psi, not allowing for welds or areas of stress concentration) and since the yield strength of a material increases for shock loading, it can be concluded that the core vessel will not yield or fail under the conditions considered.

2. Maximum Stresses in the Core Vessel During a 3.3 Mw-sec Nuclear Excursion and H₂-O₂ Reaction

The fiber stresses resulting from a (stoichiometric) H₂-O₂ reaction, were calculated on the assumption that this reaction would follow the maximum possible nuclear excursion considered (3.3 Mw-sec).

The gas pressure before the reaction, will increase from 1 to 1.55 atm for a gas production rate of 18 liter/kw-hr and a total gas handling volume of 30 liters. Since the pressures at the reflecting surface, after the reaction, are about 40 times ambient (section VIII-D) the resultant pressure will be 62 atm or about 910 psia.

Equations (7) and (8) were solved for the resulting maximum deflection, using a pressure of 910 psia and a pressure pulse duration time of 2×10^{-4} sec (the approximate observed pressure time constant for H_2-O_2 reactions). The maximum deflection was found to be

$$u_r = 1.92 \times 10^{-2} \text{ in.}$$

and the resulting stress level

$$\sigma = 77,000 \text{ psi.}$$

This is smaller than the ultimate stress, the factor of safety, however, is only about 1.25, again neglecting stress concentrations or weld factors.

This margin of safety, in a conservative evaluation, must be considered too small to assume that the vessel will stay intact. However, it is considered improbable that the core tank would be fragmented, even if the core were unsupported. The core tank is in fact completely supported by close-fitting graphite. This would prevent any tendency of the vessel to fragment. It appears that the worst that could happen would be weld failure, or similar failure resulting in release of the core gas or solution through a relatively fine opening. This type of release would be entirely confined by the secondary enclosure.

The pressure pulses due to expansion and H_2-O_2 reaction cannot occur simultaneously (expansion occurs when gas bubbles are still in solution). A second excursion cannot occur if it is assumed that the core ruptures after the first, as the changing geometry will render the reactor subcritical. However, since the exact stress concentration and distribution over the core vessel is impossible to predict, and since the primary and secondary enclosures are adjacent in some places, (a minimum of 3 inches of graphite separates both enclosures, except in the central exposure tube where they are concentric) it will be assumed, at the risk of being overly conservative, that the secondary enclosure is damaged also in the accident considered. This would result in a release of fission products into the reflector region.

C. CONSEQUENCE OF MAXIMUM CREDIBLE ACCIDENT

Rupture of the reactor vessel and possible damage to the gas tight integrity of the secondary enclosure will therefore be considered the result of the maximum

credible accident. The reactor shield and room walls or ceiling will not be affected by the accident described above, since the energy released is much too small to cause damage beyond the reflector region.

The secondary enclosure, even if impaired extensively, as well as the shield and reflector, would retain essentially all of the fuel solution.

The secondary enclosure in the core and reflector region is further enclosed by the reflector tank. The reflector tank is constructed of 1/2 inch thick aluminum plate and the dense concrete biological shield is cast around it on four sides. The end opposite the thermal column is closed with 1/4 inch aluminum plate and bolted and sealed at its periphery. This is backed up by dense concrete blocks which are further covered at the outer shield surface with a 1/4 inch thick end plate.

The thermal column shield door is sealed to the reflector tank when in its closed position. The door has a lock switch and by procedure will always be closed during reactor startup and operation. Most of the experimental facilities entering into the reflector region are enclosed by aluminum thimbles. The four exposure facilities which do not contain thimbles, contain sealed shield plugs. The central exposure tube is protected by the stainless steel primary enclosure sleeve, the aluminum secondary enclosure sleeve, and an additional aluminum liner which is sealed to the steel sleeves passing through the biological shield. The control rods operate in a stainless steel primary sleeve, lined with secondary enclosure aluminum thimbles and each rod thimble is somewhat sealed at the upper surface of the biological shield by a close fitting (approximately 0.001 inch clearance) bushing.

The various tertiary seals described were designed to prevent diffusion of argon-41 from the reflector region into the reactor room. It is seen that this also provides a reasonably good tertiary barrier in the unlikely event that the primary and secondary system were ruptured in the vicinity of the reactor core.

Therefore, with the 3 containment enclosures in the core area, the credible accident postulated cannot be considered to allow emergency quantities of radioactive materials to escape from the primary system to the reactor room to such an extent that evacuation and emergency procedures cannot be effectively enacted. However, even if instantaneous release of all volatile fission products into the

reactor room is postulated, it is apparent from the analyses of section IX that eight or nine minutes would be available for evacuation before the whole body gamma dose exceeds the acceptable emergency dose of 25 r.

A similar acceptable emergency limit for a thyroid dose is not officially available, although values ranging up to 2000 rad have been proposed.¹⁷ Even with this upper limit for thyroid exposure, about one-half minute of evacuation time would be available. The assumption of a finite leakage rate to the reactor room would reduce the potential thyroid exposure to the occupants by a large factor. Outside of the reactor building, a similar situation exists. Moreover, the accumulative leakage does assume constant weather conditions, including wind direction. It does not seem reasonable for such meteorological conditions to persist for longer periods than eight hours. In any case, evacuation procedures would be instigated within this time interval if deemed advisable.

The operating procedures prescribe immediate shutdown of the ventilation system in case of a significant release of radioactivity to the reactor room. Nevertheless, the possibility exists that, through either mechanical or human failure, such action might not be taken in time to prevent a release of radioactivity through the room ventilation exhaust. In order to evaluate the resulting environmental hazards from this situation, the maximum downwind concentration was calculated by equation (5) and the results are given in Table IX.

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SUPPLEMENT NO. I
TO THE
HAZARDS SUMMARY REPORT
FOR THE
WALTER REED ARMY MEDICAL CENTER
NUCLEAR RESEARCH REACTOR
AI-3739

ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN AVIATION, INC.
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I. INTRODUCTION AND SUMMARY

This supplement No. I to Report AI 3739 further describes the reactor core for the Walter Reed Army Medical Center nuclear research reactor, including a cooling system which was recently added, incorporates a section outlining the reactor startup procedure, and corrects some sections of the original report. The changes indicated are either the result of minor errors in the original report or reflect design improvements which are being incorporated in the building and the reactor. None of the changes adversely affect the safety aspects of the facility or add in any way to the maximum credible accident.

II. REACTOR CORE*

The sweep gas returning from the recombiner enters the reactor core vessel in the neck area just above the spherical section (Figure 1). This causes immediate dilution of the radiolytic gases leaving the fuel solution during reactor operation. The gas then moves up into the overflow chamber, through the entrainment eliminator and then through the sweep-gas-out line to the recombiner.

The reactor overflow assembly and the upper sections of the core vessel are separately cooled. These cooling systems cause water vapor to condense on the inner surfaces of the upper core and overflow assembly in order to prevent the accumulation of fuel salts or other materials which otherwise might collect in these areas by evaporation of fuel droplets. The cooling system for the upper core vessel area is in parallel with the core cooling system and operates whenever the core cooling system is operating. However, the overflow assembly cooling system is separately valved and will be operated only if it becomes necessary to reflux that area.

*This section supplements Section IV-C of AI-3739.

Figure 1. Layout of Reactor Core Assembly

III. REACTOR STARTUP

After installation, the reactor primary system is again pressure-tested and helium leak-checked, and the recombiner water is added. Each system is operated, and all pressure, flow and temperature instrumentation is calibrated. All fluid mixing, loading and transfer procedures are checked with water. The source is installed and the nuclear instrumentation, including a minimum of three auxiliary startup channels, is operated for at least 24 hours preceding the initial loading. The nuclear instrumentation is calibrated insofar as it is practicable and all interlocks and scrams are checked for proper operability. During the loading operation, two of the four control-safety rods are fully withdrawn and two are completely inserted.

The fuel is shipped to the reactor site in over-safe geometry containers. The containers are unpacked and inspected one at a time, and their contents repacked and stored in a controlled area. The uranium is packaged in denominations sufficiently small to avoid undue handling of fuel during the critical assembly.

A typical fuel transfer is effected as follows. The core and fuel drain tank are initially evacuated to about 2 psia. The core fill-and-drain line is connected to a 1/4-inch polyethylene tube. Fuel is drawn from the fuel bottle through the polyethylene tubing into the fuel drain tank. Additional bottles of fuel are added until the planned fuel loading is in the fuel drain tank. All valves are then closed and the two valves between the fuel drain tank and the core are opened. The fuel drain-tank vent-line valve is carefully opened to allow the fuel drain-tank pressure to approach atmospheric pressure. As the fuel is transferred to the core, a reduced pressure is maintained in the core by intermittently venting the core atmosphere to evacuated gas storage tanks. After the fuel is transferred, air is allowed to flow from the fuel drain tank up through the fuel in the core to insure complete transfer from the fuel drain tank and adequate mixing of the fuel in the core. All valves are then closed, and nuclear data are obtained, recorded and analyzed.

The following basic fuel loading schedule has evolved from previous experience on more than 12 comparable reactors of the aqueous homogeneous type. The initial charge into the reactor consists of water, corrosion inhibitors and sulphuric acid, but no uranium. In subsequent additions, the ratio of uranium to water is adjusted so that a minimal variation of the atomic ratio of hydrogen to U^{235} results. The loading of fuel is made in sufficiently conservative additions of U^{235} and water, such that the experimental nuclear parameters measured before and after critical assembly are used to revise and adjust the remaining loading schedule. Five loadings of fuel before the reactor is made critical are planned. Tentatively, these are at 500, 250, 125, 60 and 30 grams of U^{235} . Proportionate water additions will be made to result in a core critical volume of 24.5 liters. The critical loading will then be made by adding enough fuel to make the reactor critical, plus fuel equivalent to one tenth the experimentally-determined fuel worth of a control rod. These rod worths, as well as the critical mass, are determined by graphical extrapolation of reciprocal multiplication data as fuel is added.

The first post-critical loadings are made in two steps with a total of 0.5 liter of water. The difference in critical position on the control rod is noted. From these observations a water coefficient of reactivity is computed. The magnitude of the negative temperature coefficient of reactivity is measured. The second set of post-critical loadings is made with sufficient fuel to result in a reactivity of 0.01. The remaining loadings consist of fuel and water additions, which will be adjusted to satisfy the operational nuclear reactivity of 0.03 and the core solution volume of 26.5 liters. The reactivity will be determined by period measurement techniques. For the above procedures, reactor power will be limited to a maximum of 1 kw. Temperature and power coefficients of reactivity will be measured from ambient to operating temperature and from zero power to 50 kw.

IV. CORRECTIONS TO AI-3739

Report AI-3739 is modified in part to read as follows:

Section III-C, REACTOR ROOM (Page 15):

The reactor rooms are in an annex to the graduate school addition (Figure 4). The room housing the reactor is about 25 feet high, 35 feet wide and 65 feet long.

There is one set of gas-tight doors forming an airlock, which provides access to the reactor room on the reactor room floor. These doors are interlocked so that either door being opened will de-energize a holding device on the other door, preventing it from opening. An equipment door at the reactor room floor level and a personnel door at the ground floor level are interlocked with the reactor so that when either door is opened the reactor is shut down. These doors also contain key locks to prevent shutdown of the reactor by the indiscriminate opening of either door.

Section II-E, VENTILATION, second paragraph (Page 18):

Both supply and exhaust air valves in the reactor room are of the butterfly type and are capable of maintaining the gas-tight integrity of the room. The valves are remotely operated and can be closed from the control room.

Section IV-C, REACTOR CORE, first line (Page 26):

The core consists of a 15-3/4-inch diameter spherical tank, made of stainless steel varying in thickness from 0.085 to 0.109 inch.

Section IX, D, Equation 4 (Page 77):

$$X(x, 0) = \frac{2Q}{\pi C^2 \bar{\mu} x^{2-n}} \exp \left[-\frac{4u_g x^{n/2}}{n \bar{\mu} C \sqrt{\pi}} \right]$$