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MASTER

HAZARDS SUMMARY
FOR THE
L 77 LABORATORY REACTOR
FOR THE UNIVERSITY OF NEVADA
RENO

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UNIVERSITY OF NEVADA

Reno, Nevada

College of Engineering
Office of the Dean

September 14, 1962

Division of Licensing and Regulation
Atomic Energy Commission
Washington 25, D.C.

Gentlemen:

We are privileged to attach hereto our Hazards Summary for the L-77 Laboratory Reactor as prepared in support of our request for a Construction Permit and related licenses, grants, and documents pertaining to an installation proposed by the University of Nevada.

We will welcome the opportunity to amplify, clarify, or extend the representations made within the Hazards Summary, or any related documents as your office may require.

Respectfully,

H. B. Blodgett
Dean of Engineering

Approved:

Charles J. Armstrong
President

HAZARDS SUMMARY
FOR THE
L-77 LABORATORY REACTOR
FOR THE UNIVERSITY OF NEVADA
RENO

PART I

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ABSTRACT

This is the Hazards Summary Report for the planned installation and operation of an L-77 Laboratory Reactor in the Nuclear Engineering Building on the Reno campus of the University of Nevada. The report consists of two parts.

Part I presents site data, including information on the geography, geology, seismology, climatology, and hydrology of the area in which the reactor will be installed. This section also describes in detail, the reactor site and administration of the reactor facility.

Part II describes the reactor, its uses, and its performance characteristics, and presents analyses of the nuclear, radiation, and operational hazards.

I. GENERAL AREA

A. INTRODUCTION

The L-77 Laboratory Reactor to be installed at the University of Nevada, will be located on the Reno Campus at Reno, Nevada. The city of Reno had a population of 51,470 in 1960, and the adjacent city of Sparks had a population of 16,618. The remaining population of 16,655 in Washoe County is spread over an area of 6,281 square miles.

Water, electricity, and propane gas are furnished to the Reno area by the Sierra Pacific Power Company. Natural gas is currently being piped into the area and will soon be used in the cities of Reno and Sparks. The maximum capacity of the Sierra Pacific system is 150,000 kilowatts, and the current peak load is 90,000 kilowatts.

B. GEOGRAPHY

Reno is located at an elevation of approximately 4,500 feet above sea level, at the west edge of Truckee Meadows which is in a semi-arid plateau lying in the lee of the Sierra Nevada Mountain Range. To the west, the Sierras rise to elevations of 9,000 to 11,000 feet. The Truckee River flows from the Sierras eastward through Reno and out of the valley through Wadsworth Canyon, to drain into Pyramid Lake to the northeast. The Pyramid Lake Indian Reservation lies about 25 miles north of Reno, and Stead Air Force Base is about seven miles north of the University campus.

The reactor's proposed location with respect to the campus is shown in Figure 1. The area immediately to the north of the University is primarily open country and very hilly. Some light farming and ranching is done in this area. Residential areas surround the campus to the east, south, and west.

Figure 1. (Opposite Page)

Topographical Map, showing
the Location of the Univer-
sity of Nevada at Reno and
the proposed Reactor Site

4. Hallelujah Junction

Hallelujah Junction fault zone is located 24 miles north of Reno. Deep-seated slippages in this area have caused tremors in Reno at least once during the last ten years.

A map showing the faults, springs, and epicentral areas for the State of Nevada is shown in Figure 2. The nearest active fault is the Fallon-Stillwater area. A report of the U.S. Coast and Geodetic Survey⁽¹⁾ states that three quakes have taken place in the metropolitan area of Reno of intensity of V-VI and two of intensity VII-VIII on the Modified Mercalli Intensity Scale of 1931 (abridged). These quakes occurred in 1915, 1932, and in 1954.

E. CLIMATOLOGY AND METEOROLOGY

While temperatures on the whole are mild, the daily range between maximum and minimum is considerable, often exceeding 45°. Nights with minimum temperature over 60° are rare. Afternoon temperatures in mid-winter are moderate, and on an average, only six days a year fail to reach a temperature above freezing.

More than half the city's precipitation, falling largely as mixed rain and snow, occurs from December to March. Although there is an average of nearly 24 inches of snow a year, it is seldom that snow lies on the ground for more than three to four days at a time. Summer rain comes mainly as brief thunder-showers in the middle and late afternoons. While precipitation is scarce, abundant water is available from the high altitude reservoirs in the Sierra Nevada, where precipitation is heavy.

Humidity is very low during the summer months, and moderately low during the winter. Fogs are rare, and are usually confined to the early morning hours of mid-winter. Sunshine is abundant throughout the year.

General weather observations obtained by the Reno Weather Bureau are shown in Table 1. The mean hourly wind velocity is 6.4 mph - WNW. (See Table III.) According to the Weather Bureau at Reno, nocturnal temperature

inversions are frequent in the Reno area. They start in the evening and are most severe by sunrise. By mid-morning, they are dissipated. They are classified as light to moderate in severity.

TABLE I
LOCAL CLIMATOLOGICAL DATA

1961, Reno, Nevada,
U.S. Department of Commerce Weather Bureau

Mean Temperature	49.5°F
Record High Temperature	104°F
Record Low Temperature	-16°F
Mean Annual Precipitation	6.96 inches
Maximum Precipitation in 24 hours	2.37 inches
Mean wind - Speed and Direction	6.6 mph - WNW
Maximum wind - Speed and Direction	72 mph - S
Mean Annual Snow and Sleet	23.4 inches
Maximum Monthly	29.0 inches
Maximum in 24 hours	16.9 inches
Mean Annual Number of Days	
Clear	171
Partly Cloudy	91
Cloudy	103
Precipitation of 0.01 inch or more	47
Snow, sleet of 1.0 inch or more	8

More detailed weather information on the Reno area is given in Table II.

Data on mean hourly wind velocity and its prevailing direction are shown in Table III.

Figure 2. Faults, Springs, and Epicentral Areas of Nevada

TABLE II
TABULATION OF AVERAGE METEOROLOGICAL DATA
FOR 40-YEAR PERIOD
(1921-1960)

<u>Temperature</u> (of)	Annual	Jan.	Feb.	Mar.	Apr.	May	June	July	Aug.	Sept	Oct.	Nov.	Dec.
Mean Temp.	49.5	31.2	36.3	40.6	47.7	55.3	61.5	69.6	67.4	60.5	50.7	40.2	33.2
Daily Max.	67.6	45.7	50.9	56.5	65.6	73.2	80.8	91.8	90.1	82.3	70.2	57.1	47.2
Daily Min.	31.4	16.6	21.6	24.6	30.3	37.3	42.1	47.3	44.7	38.7	31.2	23.3	19.1
Record High	104	68	70	77	87	94	100	104	102	101	88	77	69
Record Low	-16	-16	-9	-2	13	20	25	35	32	21	13	7	-5
<u>Precipitation</u> (inches)													
Mean	6.96	1.04	1.05	0.70	0.46	0.48	0.42	0.23	0.23	0.22	0.55	0.64	0.94
Max. Monthly	5.25	4.05	1.88	2.02	1.63	1.64	0.97	0.84	0.23	1.02	2.14	2.04	5.25
Min. Monthly	0.00	0.01	0.22	0.05	0.03	T	0.00	0.00	0.00	0.00	T	T	0.01
Max. in 24 hrs.	2.37	2.37	1.18	1.21	1.60	0.80	0.66	0.80	0.17	0.80	1.27	1.23	2.16
<u>Wind</u>													
Mean Hourly (Speed and Direction)	6.6 WNW	6.6 S	6.4 S	7.9 WNW	8.2 WNW	7.8 WNW	7.6 WNW	6.8 WNW	6.5 WNW	5.5 WNW	5.7 WNW	5.1 S	5.2 S
Fastest (Mile and Direction)	72 S	66 SW	66 SW	68 S	66 SW	63 SW	66 S	54 S	57 S	48 S	66 S	59 SW	72 S

Data taken from period: 1921 - 1960

TABLE III
ANNUAL WIND VELOCITY & DIRECTION DATA

	Mean Hourly Velocity, MPH	Prevailing Direction
January	6.1	S
February	6.4	S
March	7.8	WNW
April	8.0	WNW
May	7.7	WNW
June	7.3	WNW
July	6.5	WNW
August	6.2	WNW
September	5.5	WNW
October	5.5	WNW
November	5.1	S
December	4.9	WNW
Year	6.4	WNW

F. HYDROLOGY

Due to the interbedded and lenticular nature of the fluvial deposits, the ground water movements are difficult to trace. In general, there are highly permeable and relatively impermeable stratas interbedded vertically, grading into one another horizontally. There are no known fractures in which underground water would tend to concentrate and flow.

Natural drainage from the University Campus is to the south toward the Truckee River which is approximately one mile distant. At flood stage, the surface of the river is still 35 to 40 feet below the proposed reactor site.

Cloudbursts or very rapid melting of the snows in the mountains to the west of Reno can produce flooding in downtown Reno. Two recent floods (1950 and 1955) were serious but did not get as far north from the river as the Southern Pacific Railroad tracks (see Figure 1). Due to the topography of Truckee Meadows (see Figure 1), it would be impossible for the Truckee River to flood the reactor site without first producing a large lake. A landslide in

Wadsworth Canyon - U.S. Highway 40, follows this canyon east from Sparks - could make the formation of a lake possible.

Block-N Creek flows along the east side of the campus. Its actual location is along the embankment on the east side of the area in which the Engineering Quadrangle is located (see Figure 3). Figure 4 shows the drainage area for Block-N Creek. The only probable way of getting flood water into the reactor site area would be for a cloudburst to take place over this area, which probably does not cover more than two square miles. According to the Reno Weather Bureau, the maximum precipitation on record that has taken place in 24 hours, is 2.37 inches, which would cause 253 acre-feet of water to flow down this creek. This creek has been enclosed in an underground conduit for its entire transit across the University Campus. For the reactor site area to be flooded, the cloudburst would have to be centered over the gulch containing Block-N Creek, and the underground conduit be blocked by debris. Furthermore, the position of the new Engineering-Mines Building (Bldg. E-M, Figure 3) provides protection for the Nuclear Engineering Building. Due to the grade to the south of the Nuclear Engineering Building, the water would continue to flow south, leaving the campus, and flowing down Evans Avenue toward the Truckee River. Most of the year, there is very little water in Block-N Creek.

Figure 3 - Engineering Quadrangle Building Index

On the Map: (Attached)

- Buildings:
- A. Houses the Nuclear Engineering Department. The nuclear reactor is to be placed in the south end of this building.
 - B. Houses two research laboratories belonging to the Desert Research Institute.
 - C. Will be removed
 - D. Will be removed
 - E. Houses the Civil and Electrical Engineering Departments, plus part of the Mackay School of Mines (M).
 - M. Houses the Mackay School of Mines.
 - G. Houses the University's greenhouses.
 - H-I. Houses the Biology Department (H) and the College of Agriculture (I).
 - J. Houses the Physics and Chemistry Departments.
 - F. Houses the Mechanical Engineering Department.

Engineering Quadrangle, Reno Campus
UNIVERSITY OF NEVADA

Figure 3. Engineering Quadrangle (Lettered Buildings)

WATER SHED BLOCK-N
CREEK

Figure 4
Topographic Map of the Drainage Basin of Block-N Creek
Reno Quadrangle Map - U. S. Geological Survey

1 0 1 2 3 4 MILES

3000 0 3000 6000 9000 12000 15000 18000 21000 FEET

1 5 0 1 2 3 4 5 KILOMETERS

CONTOUR INTERVAL 40 FEET
DOTTED LINES REPRESENT HALF INTERVAL CONTOURS
DATUM IS MEAN SEA LEVEL

18°
TRUE NORTH
MAGNETIC NORTH

APPROXIMATE MEAN
DECLINATION, 1950

48

18

II. REACTOR SITE

A. UNIVERSITY OF NEVADA

The proposed reactor site at the University is 4,528 feet above sea level, located essentially in a bowl, the rim of which is 1,500 to 2,500 feet high in the east, and 4,500 to 6,500 feet high on the west. The reactor's location in relation to the Reno-Sparks area is shown in Figure 1. The area immediately north of the University is primarily open country and very hilly. Some light ranching and farming is done in this area. Residential areas surround the campus to the east, south, and west.

The location of the reactor with respect to the campus is shown in Figure 5. The reactor will be in the Engineering Quadrangle (see Figure 3) which consists of approximately 400 people who are primarily engineering students and faculty. The University's total enrollment on the Reno Campus, including employees, is about 3,000.

B. REACTOR BUILDING

The reactor will be located in the Nuclear Engineering Building (A) near the east side of the campus (see Figure 3). There are two research laboratories in the north end of Building (B), which is immediately across from Building (A). Both these buildings are of the Butler type with walls of sheet steel.

C. REACTOR ROOM

The layout of the reactor room is shown in Figure 6. The room is about 20 feet wide and 35 feet long. An additional area of 7 by 13 feet gives the reactor room an overall length of 42 feet. The position of sinks, floor drains, cabinets, the reactor and console, etc., are all indicated on the drawing. The floor of the reactor room is of reinforced concrete designed to withstand a load of 800 pounds per square foot.

Reno

1. Morrill Hall
2. Stewart Hall
3. Library
4. Manzanita Hall
5. Infirmary
6. Humanities
7. Journalism
8. Ross Hall
9. Education
10. Travis Student Union
11. Artemisia Hall
12. Lincoln Hall
13. Getchell Library
14. Gymnasium
15. Mackay Stadium
16. Married Student Housing
17. Bldgs & Grnds Shop
18. Hartman Hall
20. U S Bureau of Mines
21. Observatory
22. Meteorology Station
23. Training Quarters
24. U S Naval Reserve
25. U S Marine Corps Reserve
26. Old Gymnasium
27. Nevada Bureau of Mines
28. Mackay School of Mines
29. Temporaries
30. Engineering Bldg
31. Electrical Engineering Bldg
32. Temporaries
33. Mechanical Bldg
34. Veterinary Bldg
35. Greenhouses
36. Mackay Science Bldg
37. Greenhouses
38. Fleischmann Home Economics
39. Fleischmann Agriculture
40. Dining Hall
41. Men's Dormitory

Figure 5. View of the Reno Campus of the University of Nevada Reno, Nevada

Scale $\frac{1}{8}" = 1'-0"$

ALL Rev. 1 8-27-62

Figure 6. L-77 Nuclear Reactor Room Layout.

D. HEAT & VENTILATION

The Nuclear Engineering Building (A) in Figure 3 is serviced by steam heat from a separate heating plant which also serves three other buildings. In addition, an auxiliary heating source in the controlled reactor area provides emergency heat in the event of failure of the main plant so that the minimum ambient air temperature in this area will be 65°F. An air-conditioning system maintains the air temperature in this area at a maximum of 80°F. Air is circulated from the air inlet louvres through the reactor room at the rate of 10 air changes per hour, and is exhausted through an absolute filter and an overhead vent. Ventilation requirements of the non-controlled areas of the nuclear building are served by a separate system.

Utilities and services are as follows:

- a. 110-220 three phase circuits
- b. City water outlets with sinks and city sewer contracts
- c. Steam heat from a separate heating plant serving this building and three other buildings in the immediate area.
- d. Regular janitorial and custodial services through the University Department of Buildings and Grounds.

III. ADMINISTRATION OF REACTOR FACILITY

A. ADMINISTRATIVE ORGANIZATION

No one will be given responsibility for either startup or operation of the reactor unless he has been properly licensed by the Atomic Energy Commission. All operators will be indoctrinated with the design and operation of the reactor, emergency procedures, and the official line of authority within the nuclear facility and the University of Nevada.

The line of authority within the University of Nevada and relative to the nuclear reactor facility is shown in Figure 7.

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.

The keys which permit operation of the reactor facility will be under the control of the licensed reactor operator. The responsibility for dispersing these keys will be that of the Responsible Engineer. (Lock and Key facilities are discussed in detail in Part II of this document).

As indicated in Figure 6, the reactor area is designated as a control area. Access is possible only through a single door except in case of an emergency when the locked doors on the west wall of the facility can be opened by breaking the emergency seal. During operation of the reactor, the operating light on the north wall of the controlled area will be "on" and readily visible by persons entering the controlled area.

On the same wall, immediately adjacent to the entrance door, will be located radiation film badges and a log-in sheet for visitors entering the controlled area. The door to the controlled area will be locked at all times when a licensed operator is not in the area. Entrance to the controlled area during reactor operation will be subject to the permission of the licensed operator.

**LINE OF AUTHORITY
UNIVERSITY OF NEVADA**

Figure 7

Senior and graduate students will perform various experiments on the reactor which are within the scope of experiments permitted by the license. These experiments will be carried out under the direction of a licensed operator. Under no conditions will a non-licensed student be permitted to perform operations at the reactor console unless (1) the operation is the result of a carefully planned experiment which has been reviewed by a licensed operator, and (2) a licensed operator is supervising said operation from a position accessible to view of the instruments on the console. Removal of exposure tube plugs and insertion or withdrawal of samples from the tubes will also be carried out under the direct supervision of a licensed operator.

C. HEALTH PHYSICS

Radiation and Radioactive Material Control

Responsibility for protection of personnel from radiological hazards, and responsibility for radioactive material control, resides with the health physicist shown on the Line of Authority Chart. Continuous monitoring of personnel exposures, maintenance of personnel exposure records, and review of operating procedures and restrictions will be included in these responsibilities. In addition, the health physicist will establish emergency evacuation and control procedures.

Operating procedures and restrictions shall include the following:

- a. Controlled radiation areas will be established.
- b. All personnel within controlled radiation areas will wear appropriate radiation-monitors on their persons.
- c. All licensed personnel and persons involved in reactor experiments will wear neutron-gamma dosimeters and, in addition, will wear neutron-gamma film badges during reactor operation and residence in the controlled area. During routine operation of the reactor, with all beam and exposure plugs in place, observers will be permitted into the controlled area with either neutron-gamma badges or neutron-gamma dosimeters on their persons.

- d. Control of exposure of all persons will be as specified in Title 10, Code of Federal Register, Part 20.
- e. Beam ports or exposure tubes will not be removed with the reactor in operation unless the portable beam catchers are in place in front of the tubes. The catchers will reduce total dose to 2.5 mrem/hr or less.
- f. Neutron-gamma film badges will be located at beam height at the points of maximum dose rate on the outside face of the building (in locked weatherproof enclosures) at locations A, B, C, D, and E shown in Figure 6.
- g. All film badges will be collected on a monthly basis or more often if required, for development and review. Dosimeters will be collected, read, and recharged on a weekly basis.
- h. Dry and wet radioactive wastes will be collected in suitably shielded and marked containers and stored in a marked area in the controlled area. Waste will be accumulated on a regular basis and handled as radioactive wastes. Large accumulations of waste will not be permitted.
- i. Typical alpha, beta, gamma, and air sampling detectors will be maintained at the facility.

REFERENCES

- (1) U. S. Coast and Geodetic Survey, Earthquake History of the United States, Part II, Revised (1960) edition, No. 41-1, P. 1, U. S. Department of Commerce.

PART II

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I. INTRODUCTION

A. GENERAL

The Model L-77 Laboratory Reactor is essentially an improved and perfected version of an earlier prototype Model L-47 Laboratory Reactor.¹ The summary hazards report for the initial Model L-77 Reactor,² which has been in operation since May 1958 at Atomics International, has been reviewed and approved by the Reactor Hazards Evaluation Branch of the United States Atomic Energy Commission.

Three sources of potential hazard exist in the operation of a reactor of this type: nuclear hazards, radiation hazards, and operational hazards. The chief hazard to the general public, which could arise from any of these causes, would be the release and dispersal of radioactive material from the reactor core. The inherent safety characteristics, as established in the Kinetic Experiments on Water Boilers (KEWB) Program described in section VIA, plus the design simplicity and the low power level (10 watts), combine to minimize the possibility of a nuclear excursion and further minimize the gravity of the effects should such an incident occur. Simplicity of structure, of instrumentation, and of operating procedures renders any type of accident highly improbable. In the event of negligence or malfunction, it would be virtually impossible for any serious damage or danger to result.

B. PURPOSES AND USES OF THE REACTOR

The Model L-77 Laboratory Reactor has been designed as an economic, easily operated reactor for use in educational institutions, industrial laboratories, research foundations, and in private reactor study groups. The simplicity of the reactor makes it an effective and versatile adjunct to teaching, and a convenient scientific research tool, as well as a demonstrative training aid.

The reactor may be used for the determination of data on nuclear reaction cross-sections and radiation effects, and in applied nuclear engineering. Laboratory experiments of educational value may be derived from its use, particularly in such fields as nuclear physics, nuclear engineering, chemistry, metallurgy, biology, and medicine. It may also be used for producing low activity isotope samples.

C. PERFORMANCE CHARACTERISTICS

Performance characteristics for the Model L-77 Laboratory Reactor are summarized in Table I. The values cited, although approximate, are considered to be representative. Radiation levels during full power operation are given in section IV.

TABLE I
PERFORMANCE CHARACTERISTICS

<u>Thermal Power Level</u>	10 watts
<u>Maximum Thermal Neutron Fluxes</u>	
In central exposure facility	3×10^8 n/cm ² -sec
In auxiliary exposure facilities	2×10^8 n/cm ² -sec
In beam tubes adjacent to core	1×10^8 n/cm ² -sec
<u>Gamma Dose Rates in Reactor Core</u>	3×10^3 r/hr
<u>Maximum Excess Reactivity from Fuel Loading</u>	0.5 %
<u>Control-Rod Reactivities</u>	
Regulating-rod worth	1.3 %
Safety-rod worth	1.3 %
Maximum single-rod differential worth (60-cycle current)	0.015 %/sec
<u>Moderating Material Reactivities in Exposure Tubes</u>	
Lucite filling all five tubes	0.5 %
Average lucite mass coefficient	5.4×10^{-4} %/gm
<u>Nuclear Reactivity Coefficients</u>	
Temperature coefficient	-0.03 %/°C
Mass coefficient of U ²³⁵	~0.022 %/gm
Water coefficient	~0.28 %/liter

Fig. 1. Model L-77 Reactor and Console

II. REACTOR DESCRIPTION

A. GENERAL DESCRIPTION

The Model L-77 Laboratory Reactor is a solution-type reactor using light water as moderator and enriched uranyl sulfate as fuel. The reactor and console are illustrated in Figure 1, and a cutaway of the core-shield assembly is shown in Figure 2. The design thermal-power level of this reactor is 10 watts. Because of the low power level, no cooling system is required.

A spherical stainless steel core contains the fuel solution. In the event of a major power excursion, the large throat area above the core permits rapid expulsion of expanding fuel solution into the overflow assembly, which is an annular holdup volume within the canopy. A gas recombiner located in the canopy serves to catalytically recombine the hydrogen and oxygen generated by decomposition of the moderator in the fission process. The core and canopy assembly is completely surrounded by three solid shield regions, a water shield, and the reactor shield tank.

The reactor is equipped with two control-safety rods. The rods pass through thimbles which penetrate the core and canopy assembly vertically. Steel tapes attached to the upper end of each rod are used to lift the rods. The rod drive mechanisms, located on the top of the shield tank, utilize electromagnetic clutches to engage the rod drive.

Operation of the reactor is observed and controlled from a desk-type console. The console displays pertinent operating information, such as control-rod positions and neutron-flux levels.

Reactor instrumentation consists, essentially, of two neutron channels, a pressure-monitoring system, and the electrical interlocking circuitry.

The experimental facilities include four small-diameter exposure tubes and a larger-diameter exposure tube, all of which completely penetrate the core and terminate at opposite sides of the water-filled shield tank; two beam tubes which extend from the shield-tank wall to the external periphery of the core; and two of the four instrumentation thimbles (the other two contain ion chambers). In

Fig. 2. Cutaway of Core-Shield Assembly

all of the experimental facilities, adequate shield plugs, which are stepped as necessary, are utilized to prevent radiation streaming when the facilities are not in use.

B. PRINCIPAL STRUCTURAL AND SHIELDING COMPONENTS

The core and canopy, the three-region solid shield assembly, the water-filled shield tank, the experimental exposure facilities, and the various conduits and piping, comprise the major structural and shielding components of the reactor.

1. Core and Canopy

The core and canopy, shown in Figure 3, consists of the 15-3/4-inch-OD spherical-core vessel, the canopy, the recombiner, the deflector plate, the overflow assembly, and the control-rod and exposure-tube thimbles. The canopy is welded to the top of the core and consists of a 9-inch-diameter, 5-inch-long cylinder with a 9-inch-diameter hemispherical head. The core and canopy assembly is fabricated from stainless steel with a wall thickness which has a maximum of 0.090 inch and a minimum of 0.068 inch. It is designed to withstand 300 psi in accordance with the ASME Unfired Pressure Vessel Code, and is pressure tested and helium leak tested to assure complete integrity. Extensive corrosion experiments and observations indicate that the corrosion rate of the stainless steel vessel will be less than 0.001 inch per year.

The gas recombiner unit, located within the hemispherical cap of the canopy, contains 1/8-inch-square cylindrical pellets of platinized alumina. Stoichiometric hydrogen and oxygen liberated by the decomposition of the moderator during the fission process are catalytically recombined and returned to the core proper.

In the event a major power excursion should occur, any fuel expelled from the core would impinge upon the deflection plate, from which it would be deflected into the overflow assembly. The deflection plate is a thin, stainless steel, member located above the throat of the overflow assembly and welded to the canopy.

The overflow assembly has a 3-liter annular retention volume formed of cylindrical stainless steel sections welded to the top of the core. Two small drain holes at the base of the overflow assembly permit any expelled fuel solution to return slowly to the core.

Fig. 3. Core and Canopy

2. Three-Region Solid Shield

Surrounding the core and canopy are three separate solid-shield regions. The first (innermost) solid region, which also serves as the reflector, is composed of a mixture of pelletized lead and diphenyl (60 percent lead, 40 percent diphenyl). The second solid-shield region contains a mixture of approximately 10 grams per liter of soluble boron in paraffin. The third (outer) solid shield region is a mixture of 60 percent pelletized lead and 40 percent paraffin, with the paraffin borated to approximately 15 grams per liter. These solid-shield regions are shown in Figures 2 and 4.

The inner two solid-shield regions completely surround the reactor core, while the third shield region encloses all but the bottom of the solid-shield assembly. The first region is separated from the second region by a 28-1/4-inch OD, 3/16-inch thick, 41-3/4-inch high, closed aluminum cylinder. The second and third regions are separated by a 34-3/4-inch OD, 3/16-inch thick, 54-inch high aluminum cylinder which is welded to the lower support gussetts. The entire three-region solid-shield assembly is enclosed by a 43-inch OD, 0.109-inch thick, 57-inch high, stainless steel cylinder, which is welded to the lower support plate. Four aluminum gusset plates, each 6-inches high, support the reflector cylinder. The gusset plates are supported by the lower-shield support plate, which is a 48-inch diameter, 3/8-inch thick, mild steel plate.

3. Reactor Shield Tank

The reactor shield tank is fabricated from mild steel. It is 8 feet in diameter by 7 feet 2 inches high, with a wall thickness of 1/4 inch. The tank is filled with water, which completely surrounds the solid-shield assembly and serves as an additional radiation shield. The center of the core vessel is on the vertical axis of the shield tank, 2 feet 8-3/4 inches above the base of the tank. Approximately 2 feet 2 inches of water surround the sides of the solid-shield assembly; a minimum of 1 foot 9 inches of water is above it; and 6 inches of water is below it.

On the shield tank are mounted the control-rod drive mechanisms, valve manipulators, water drain valve, vacuum gauge, float level gauge, source conduit access, instrumentation thimble terminal, instrumentation terminal connector, and the access ports to the experimental facilities. All access ports are locked and secured when not in use.

Fig. 4. Shielding and Piping

The top of the tank is bolted (not water tight) to a steel angle welded to the tank wall. The shield tank is structually designed so that it is safe for personnel to stand on top of it. Both the inside and outside surfaces of the tank are treated to prevent corrosion.

4. Experimental Exposure Facilities and Auxiliary Components

The experimental exposure facilities and their shield plugs are described in Table II.

TABLE II
EXPOSURE FACILITIES

Type of Facility	Number Of Tubes	Tube Dimensions		Shield Plugs	
		Inside diameter (inches)	Wall thickness (inches)	Approximate overall length of shielding (inches)	Shielding materials
Main exposure tube	1	1.37	0.065	37	Lead and borated paraffin
Auxiliary exposure tubes	4	0.75	0.065	39	Lead and borated paraffin
Gamma and neutron beam tubes	2	2.00	0.125	32	Lead and borated paraffin
Spare instrumentation thimbles	2	2.75	0.125	37	Borated paraffin

The core is completely penetrated by five horizontal exposure tubes, each of which is accessible from openings on both sides of the reactor tank. The main exposure tube passes through the core horizontally on a major diameter, and is welded at entry to the core on each side. The four auxiliary exposure tubes also pass through the core horizontally, but are located five inches on either side of the vertical core centerline and 2-1/4 inches above and below the horizontal centerline. The auxiliary tubes are also welded to each side of the core.

The two, stainless steel, horizontal beam tubes are on extensions of horizontal major core diameters, penetrating the solid-shield regions to within 1/8 inch of the core vessel.

Two instrumentation thimbles and two additional spare instrumentation thimbles extend horizontally from the shield-tank wall into the reflector region below the core. They pass 6-3/8 inches beyond the vertical centerline of the core. The centerlines of the thimbles are in a horizontal plane 15-1/2 inches below the horizontal core centerline. Two of these thimbles are tubes with a 3-inch OD, having centerlines passing 3 inches on either side of the vertical core centerline. The larger tubes with a 4-1/2-inch OD are located 9-3/4 inches from the core vertical centerline. The section of the instrumentation thimble within the solid-shield region is fabricated from aluminum.

Two control-rod thimbles completely penetrate the core vertically. They are stainless steel tubes each with a 1-1/4-inch OD and a wall thickness of 0.065 inches; they are symmetrically located 3 inches to either side of the core centerline and welded at entry to the core and canopy.

In addition to the above components which penetrate the solid-shield assembly, there are conduits and piping to conduct the recombiner heater electrical leads, to transfer fuel solution, to vent the core, and to contain the neutron source. These are shown in Figure 2. The recombiner heater conduit, which is a stainless steel tube with a 1/2-inch OD, 0.058-inch thick, terminates outside the solid-shield assembly in a tee, and is used to conduct the power and instrumentation leads from the recombiner unit. The fill and drain line and the vacuum line are stainless steel tubes with a 5/16-inch OD and a wall thickness of 0.065 inch. These tubes serve the purposes of gas handling, fuel transfer, core evaluation and related functions. The vacuum line and the fill and drain line each have one valve for controlling fluid flow. The source conduit is a stainless steel tube with a 1-1/2-inch OD and a wall thickness of 0.065 inch which extends vertically from the top of the shield to within approximately 1 inch above the horizontal midplane of the core.

C. PRINCIPAL MECHANICAL AND NUCLEAR COMPONENTS

1. Control Rod System

Control of the reactor is maintained by two vertical rods, illustrated with their drive mechanisms in Figure 5. Both rods also function as safety rods since they are both released under "scram" conditions. For convenience and operating procedural purposes, one rod is termed a "safety rod" since it must be fully withdrawn before the second rod, termed a "regulating rod," can be energized. Both rods are aluminum, 36 inches long and 1 inch in diameter, wrapped with a cadmium sheet for the lower 14 inches which serves as the poison (neutron-absorbing) section. Shielding for each rod consists of a 14-inch length of lead and borated paraffin.

Each rod controls 1.3 percent in reactivity. In the position of maximum differential worth, either rod can change reactivity at the rate of 0.015 percent per second. Maximum rod traverse is 13-3/4 inches at a constant speed of 4.3 inches per minute, based on utilizing 60-cycle electric current. When the rod drive is operated on 50-cycle instead of 60-cycle current, the rod speed and maximum differential worth are reduced to 3.7 inches per minute and 0.012 percent per second respectively.

The rod drive mechanism consists of a drive motor, a gear reducer, an electromagnetic clutch with a kickoff spring, a one-revolution drum, and a steel drive tape. These components appear in section view in Figure 5. Each drive mechanism is enclosed within a housing located on top of the reactor tank. Each housing is provided with a key-type locking mechanism.

The gear head drive motor is of small fractional horse power, acting through gear reducers with an effective ratio of 180 to 1. The electromagnetic clutch consists of two discs, between which is a kickoff spring, and a solenoid actuator which engages the discs and acts against the kickoff spring. The clutch serves to couple the drive motor to the one-revolution drum, over which the steel drive tape acts. Position indicators, mechanically geared to the rod-drives, transmit rod-position information to the console.

In order to raise the rods, the console must be turned on, the rod drives in the full down position, and all interlocks satisfied. When these conditions

Fig. 5. Control and Safety Rod System

have been met, the electromagnetic clutches are automatically energized. Whenever the electromagnetic clutches are deenergized, the same conditions as above must be reconstituted before the clutches can be reenergized. The kickoff spring assures positive disengagement of the clutch faces when the electromagnets are deenergized.

After all the conditions for rod withdrawal have been satisfied, the safety rod may then be withdrawn. The regulating rod can not be raised until the safety rod has reached its upper limit, indicated by the operation of the ROD OUT light. After the regulating rod has left its lower limit, as indicated by the extinction of the ROD IN light, either rod may be moved in either direction. A two-position selector switch makes it impossible, however, to move both rods simultaneously. Rod motion is controlled by a switch with a spring return from the upward rod movement setting.

2. Fuel and Fuel Handling System

Approximately 29 liters of fuel solution are used, the level of which is 6 inches above the core midplane. The fuel is uranyl sulfate, enriched in the range between approximately 20 and 90 percent in the uranium-235 isotope. The critical mass for this range of enrichments is between 1150 and 1400 grams. The actual quantity of fuel loaded into the reactor is limited to that amount which results in no more than 0.5 percent excess reactivity, with both control rods fully withdrawn and all exposure tubes empty except for the normal shielding plugs.

The fill and drain line, as shown in Figure 4, extends from the bottom of the core, through the shielding, to the top of the shield tank, where it passes through a remotely operated valve. The line continues down the side of the tank to the fuel drain outlet, where it is capped off and sealed. The outlet is located above the fuel solution level of the core.

3. Gas Handling System

A vacuum line extends from the top of the core to a point a few inches above the core-shield assembly, where it connects with a tee. The main line extends from the tee to the top of the shield tank, where it passes through a remotely operated valve, and continues from this point to the side of the tank

where it is capped off and sealed. The other line from the tee has a restriction to prevent overpressure, a diaphragm for containment, and beyond the diaphragm an extension into a filled vacuum gauge assembly. The vacuum gauge, mounted on the side of the reactor tank, operates a snap-action switch which scrams the reactor when core pressure is too high.

The core is normally maintained at approximately 26 to 28 inches of mercury vacuum. Core pressure is transmitted through a stainless steel bellows-type protector unit, through a liquid medium (either water or silicon fluid), to a bourdon-tube-type vacuum gauge. Before installation and filling of the pressure transmitting unit, it is completely helium leak checked. The bellows assembly is designed to withstand overloads up to 300 psia. Should the bellows develop a leak, the core atmosphere will still be contained by the gauge assembly, since the bourdon tube and connecting piping constitute an all-welded leak-tight assembly. Should the piping, bourdon tube, or any other part of the secondary side develop a leak, then the gauge will register atmospheric pressure and cause a reactor scram.

Gases evolved during the fission process, including fission products and water decomposition products, are accommodated by the internal gas-handling system. Stoichiometric hydrogen and oxygen are recombined by the action of a platinized alumina catalyst. No special provisions are made for the fission product gases because of their extremely low formation rates. A more detailed discussion concerning recombiner operation appears in the Appendix (section VIB).

4. Neutron Source

A neutron source, with an emission strength of approximately 10^6 neutrons per second, is inserted in a tube which extends from the top of the shield tank to a point immediately above and about one inch away from the horizontal midplane of the core. The source conduit is shown in Figure 2; it is capped and locked to prevent unauthorized removal of the source.

With the reactor shut down and the control rods fully inserted, the source creates a sufficient number of fissions in the core to cause observable readings on the micromicroammeters, which are used in conjunction with ionization chambers to give reactor power information. The low-level scram circuitry prevents reactor startup if the source is not at an effective location near the core.

D. INSTRUMENTATION

Reactor instrumentation includes the control console, two neutron-detection channels, interlocking circuitry, an annunciator circuit, and the process instruments.

1. Console

A desk-type control console, illustrated in Figure 6, displays pertinent operating information, houses all of the electrical and electronic components, and functions as the reactor control center.

2. Nuclear Instruments

The instrumentation block diagram is shown in Figure 7. Two channels of neutron instrumentation are provided, both of which must be operative in order to achieve or maintain reactor operability.

Channel I, a linear micromicroammeter circuit, includes the following components and characteristics:

- a) An uncompensated ionization chamber
- b) A battery power supply
- c) A range switching linear micromicroammeter with a range of approximately 3×10^{-13} to 10^{-3} amperes.
- d) Scram contacts at either end of the scale.

Channel II is a seven decade logarithmic micromicroammeter and period meter combination circuit and includes the following components and characteristics:

- a) An uncompensated ionization chamber
- b) A battery power supply
- c) A logarithmic micromicroammeter with a range of approximately 10^{-13} to 10^{-6} amperes, and with an adjustable high-power-level scram setting.
- d) A period meter with a range of three seconds to infinity, either plus or minus, and with an adjustable short period scram setting.

Fig. 6. Console

Fig. 7. Instrumentation Block Diagram

A selector switch on the console allows the recording of power readings on either channel as desired. The recorder has two speeds: one-half and two inches per minute.

3. Interlocking Circuitry and Scrams

Interlocks and trips are instrumented throughout the various circuits to prevent reactor startup or continued operation (scram the reactor), as appropriate, whenever any of the following conditions exist:

- a) Alternating current power not on.
- b) Key switch not on.
- c) Manual scram button depressed.
- d) Electric current not supplied to either the linear or the logarithmic micromicroammeter.
- e) High power level on either channel.
- f) Low power level on Channel I. (This interlock ensures the presence of the neutron source in its proper location prior to startup and protects against operation with loose cable connections, failures in the power supply and amplifier circuits, and with a desensitized neutron detector.)
- g) Short period on Channel II.
- h) High reactor core pressure (approximately 25 inches of mercury vacuum).
- i) Low water level in the reactor shield tank.

Additional interlocks ensure proper operation of the control and safety rod systems, as described (section IIC1).

Scrams are indicated visually on a panel mounted on the console by means of a drop-type signal-arm annunciator. Original scrams are also indicated audibly by a buzzer. The buzzer continues to sound until manually silenced.

4. Instrument Adjustments

The instruments in the scram circuitry are adjustable over the following ranges:

- a) Low-power-level linear micromicroammeter: This is a range-switching instrument covering 0.3 decades of power on each range. The low-level scram contact can be adjusted over the full range of the instrument from source level to full power.
- b) High-power-level linear micromicroammeter: As above, this scram contact is adjustable over the entire range of the instrument. The scram set point is normally at 150 percent of full power.
- c) Period, logarithmic micromicroammeter: The scram set point is adjustable over the range from infinity to 3 seconds. It is normally set at 5 seconds.
- d) High-power-level logarithmic micromicroammeter: The scram set point is adjustable over the entire range. It is normally set at 150 percent of full power.
- e) Low shield-tank water level: The scram set point is adjustable over the top 4 inches of the shield tank.
- f) Reactor-core pressure: The scram set point is pre-set by the supplier at 5 inches of mercury absolute; the setting is checked before installation. It is possible, however, with some difficulty, to change the set point over the range from 0 to 30 inches of mercury absolute.
- g) Other scrams (e.g., manual and loss of power): These do not involve any adjustments.

5. Nonnuclear Instruments

Because of the inherent safety and simplicity of this reactor concept, a minimum of process instrumentation is employed. A vacuum gauge monitors and displays the reactor-core pressure. A microswitch contact on the pressure relay initiates a scram whenever the core pressure becomes greater than approximately 25 inches of mercury vacuum. A level indicator, attached to the top of the reactor shield tank, causes the reactor to scram when the water level in the tank falls below a predetermined level.

A thermocouple attached to the recombiner monitors the recombiner temperature. The information is transmitted to a console indicator. Positions of the control rods are read on the console to the nearest hundredth of an inch.

6. Manual Circuit Breaker

At the bottom rear of the console there is located a main, push-pull type, circuit breaker which may be used to shut off all electric power except the battery power supplied for the nuclear instruments.

III. NUCLEAR HAZARDS ANALYSIS

A. PERFORMANCE DURING NORMAL OPERATION

From information on the performance of the L-77 reactor and of similar reactor types, and from the particular safety features of the L-77 reactor design, quite definite safety characteristics may be concluded.

Because of the low power level (10 watts), there are no reactivity control problems resulting from fission-product buildup nor from transients during normal operational power changes. The buildup of radioactive materials is not dangerous from the radiological standpoint (see section IV and section VID). Transmutation of elements by neutron capture is negligible. Neither high temperatures nor radical temperature fluctuations exist. In addition to the various safety circuits which prevent excessive temperature rise, the reactor has a negative temperature coefficient of reactivity (approximately -0.03 percent per degree Centigrade). A low core pressure results in boiling at approximately 50° C, with accompanying voids to reduce reactivity.

The approximate neutron-flux distributions during 10-watt operation are shown in Figure 8. The majority of the water shield (extending to 122 centimeters from the center of the core) is not shown, the thermal and fast neutron fluxes decreasing essentially exponentially in this region. At the reactor shield-tank surface the thermal and fast neutron fluxes are approximately 40 and 10 n/cm²-sec, respectively. Reactivity characteristics are summarized in Table I.

B. NUCLEAR EXCURSION

The L-77 reactor is designed to minimize the possibility of a nuclear excursion, and further to minimize the seriousness of effects if an excursion should occur. Scram circuitry, based on both high and low power levels, makes it impossible for an operator to desensitize or oversensitize the neutron detectors to the extent that they become ineffective in giving proper readings. Period and high-power-level scram circuitry also serve to prevent excessive power rise. The low excess reactivity in the reactor prevents obtaining a stable period of less than 3.3 seconds unless material which adds reactivity should be inserted near the reactor core. A graph of reactivity vs stable period is presented in Figure 9.

Fig. 8. L-77 Neutron Flux Distributions for 10-watt Operation

Fig. 9. L-77 Reactivity vs Stable Period

If the control rods should not function properly, so that the excess reactivity would not be controlled by the rods, several factors would still act to control the reactor sufficiently to prevent a serious excursion. Assuming a starting core temperature of 20°C and the maximum value of excess reactivity (0.5 percent), the negative temperature coefficient of reactivity (-0.03 percent per °C) would allow a temperature increase to only 37°C, assuming good mixing. A higher localized temperature could possibly result from a sudden power surge. However, the low core pressure (26 to 28 inches of mercury below atmospheric pressure) results in boiling at approximately 50°C. This boiling, and the evolution of radiolytic gases (H₂ and O₂), would further act to decrease the magnitude of any nuclear excursion. In addition, the low initial core pressure and the presence of a gas recombiner (see section IIC3 and section VIB) minimize the magnitude of a potential explosion of stoichiometric hydrogen and oxygen during an excursion. The low core pressure also serves to reduce integrated power during an excursion (see section VIA).

An overflow assembly, having a retention volume of approximately three liters, is provided above the reactor core to prevent repetitive excursions. The 3-1/16-inch-ID throat opening at the top of the core sphere minimizes any expansion pressure accompanying an excursion. The overflow assembly has two small drain holes for slow return of fuel solution to the core, preventing any immediate reinsertion of reactivity after solution is ejected from the core sphere into the overflow assembly. Because of the low excess reactivity (0.5 percent maximum from fuel loading and 1.0 percent maximum when also adding moderating material in all experimental facilities), it is not considered possible that any excursion could be violent enough to expel fuel solution from the reactor core. The provision for overflow from the core is intended as an added safety factor.

In the unlikely event of an extreme power excursion or hydrogen-oxygen explosion, the strength of the core vessel would serve to contain the explosion. The 300-psi allowable steady pressure within the core is undoubtedly very conservative with regard to resistance to suddenly-applied loads. The three-region solid shield surrounding the core vessel would further serve to absorb energy from an explosion.

The inherent stability of the solution-type reactor under conditions of rapid reactivity insertion has been demonstrated in the KEWB Program. A description of experimental results from the KEWB Program is presented in the Appendix (section VIA).

C. CONTROL ROD SYSTEMS

The safety rod and the regulating rod are designed to be failsafe. The rods are sustained in a raised position by means of a geared electric motor acting through an electromagnetic clutch. Deenergizing the electromagnets results in a decoupling of the motor drive from the rod drive pulley, causing the rods to fall back into the core. The electromagnets are deenergized by loss of power, a manual scram or any scram signal from the instrumentation system.

Rapid reactivity insertion by rod movement is prevented by mechanically limiting the speed of each rod to 4.3 inches per minute (when operated on 60-cycle current), corresponding to an average reactivity change of 0.007 percent per second. Maximum differential worth of each rod is 0.015 percent per second. Only one of the two rods may be raised at a time (see section IIC1).

Each rod controls 1.3 percent reactivity, which is more than twice the maximum excess reactivity loaded into the reactor. The only ways in which the rods could fail to insert during an intended scram would be through either the failure of the circuits to deenergize the electromagnetic coupling, or the jamming of the rods in their thimbles. The operator could manually scram the reactor or turn off the power if the circuits were not properly performing their function. In the unlikely event of rod jamming or combined circuit and operator failure, the reactor would be shut down by the inherent shutdown mechanisms (described in section IIIB).

A key switch must be used to turn on the console and energize the magnets before the control rods may be raised. Use of the key is controlled by the operator.

The entire length of both control rods lies within the outer tank. In order for a rod to be lifted out of this position, the housing over the drive mechanism must be removed. To prevent removal of both rods, operating procedures call for removing only one cover plate at a time and then replacing the first rod and cover plate before removing the second cover plate. Before fastening each cover

plate, the operator is required to check that the tape running through the cover plate is properly attached to the rod below and to the one-revolution pulley above, and that the rod is manually seated in its fully-lowered position.

Having the rods completely within the outer tank minimizes the chance of their being jammed by any accident or explosion originating outside of the tank, or from an earthquake. Fire would not affect them, nor would extreme weather conditions. Clogging or distortion of the thimbles or rods, or snagging of the rod support tape, would be necessary to prevent the rods from falling. Such clogging, distortion or snagging would not be likely to occur to both rods simultaneously except during a violent explosion or shock. Under reasonable circumstances at least one rod would be expected to fall into the core to shut down the reactor.

No accident or malfunction related to the electromagnets, limit switches, gear reducers, motors or instrumentation could in any way prevent the rods from being dropped into the core upon deenergizing the magnets. The failure of an electromagnetic clutch to couple the drive motor with the drive pulley would result in the rod being located in its lowermost position, fully inserted in the reactor core. Because of the small excess reactivity normally present in the reactor, criticality cannot be attained with either rod fully inserted.

D. FUEL LOADING

The fuel fill and drain line and the vacuum line are the only openings into the reactor core. Each of these lines has a valve and a sealed end cap.

Because of the valves and seals it is improbable that there could be any accidental insertion of fuel solution, other than at the time the lines have been opened intentionally to load fuel into the reactor core. Procedures require that the fuel loading be done in discreet increments, and that instruments be carefully checked during the process. The total uranium-235 loaded into the core is between 1150 and 1400 grams, for enrichments in the range from 19 percent to fully enriched uranium.

Figure 10 shows the equipment setup for the addition of fuel to the L-77.

Fig. 10. Equipment for L-77 Fuel Addition

The fuel is received in aqueous sulfate form (at a concentration of approximately 500 grams of uranium per liter when at 20 percent enrichment) in one-half-liter polyethylene bottles. The fuel loading line is attached to a polyethylene tube inserted into a bottle, and the fuel is transferred to the core by the action of the pressure differential between the atmosphere and the partially evacuated core. All fuel loadings are prepared in individual aliquots and transferred to the core by the same method.

Each new fuel addition is mixed with the existing core loading by permitting air to bubble through the core from the fuel line to the vacuum ballast tanks. These tanks are evacuated into auxiliary ballast tanks. The activity of each ballast tank is monitored, and the method of safe disposition of the gas determined by the activity level.

The amount of each fuel addition is determined by the graphic extrapolation of reciprocal multiplication data. Loadings for excess multiplication are done in a manner which permits determination of the mass coefficient of reactivity and the reactivity worth of the control rods.

The reactor is shut down during fuel loadings by fully inserting both control rods. The worth of these control rods is 2.6 percent. Since approximately 50 grams of uranium-235 are required to increase reactivity one percent, a loading error of approximately 130 grams of uranium-235 would be required to override the reactivity worth of the control rods. Normal fuel loading is limited to not more than 0.5 percent excess reactivity, with all experimental facilities empty and shield plugs in place.

During fuel loading the fuel solution is mixed with water to near the optimum moderation (a hydrogen to uranium ratio of between approximately 450 and 520), so that little reactivity increase could result from any later addition or subtraction of water in the solution.

E. EXPERIMENTAL LOADINGS

The maximum reactivity effect of inserting a moderating material such as lucite into all five experimental facilities is approximately a 0.5 percent increase. Withdrawal of a neutron absorbing material from an experimental tube can increase reactivity no more than the excess reactivity due to the fuel loading in the core plus the

moderating materials in the experimental facilities, which total to a maximum of 1.0 percent.

Operational procedures prohibit the insertion or removal of experimental samples from the central exposure tube while the reactor is in operation, unless this insertion or removal is part of a carefully planned and approved reactor test (such as a pile oscillator experiment).

F. MAXIMUM CREDIBLE ACCIDENT

The maximum credible accident for the L-77 reactor is assumed to be the instantaneous release of all the available excess reactivity, which is 1.0 percent when all exposure facilities in the core are filled with moderating material in the form of lucite. It is further assumed that the control rods fail to insert to stop the transient. The details of the transient are presented in the Appendix (section VIC), including the analysis of KEWB data, graphs, assumptions, and numerical results.

The transient may be initiated by either of two methods. One method would be through withdrawal of both control rods, such as by pulling on the drive tapes or rotating the one-revolution drums. This would first require removal of the housings over both drive mechanisms, which might be done during maintenance of control mechanisms. With either of the two control rods inserted the reactor is subcritical.

The second possible method of initiating the transient would be by the rapid withdrawal of neutron absorbing material from an experimental facility. In order to obtain the maximum 1.0 percent reactivity step the neutron absorber would have to be so strong that, for the reactor to be critical with the absorber in the experimental facility, both control rods would have been completely withdrawn and all other experimental facilities filled with lucite or similar moderating material. Maximum reactivity insertion could be obtained by rapid replacement of the neutron absorber with a moderating material (lucite) which might have been used as a reflector adjacent to the absorber.

Normally, a reactor scram would automatically result from short period or high power level occurring during the transient. The operator could also scram the reactor by pushing the manual scram button or turning off the key

switch. However, for this analysis it is assumed that there is no operator action for five minutes, and that the instrumentation does not function or the control rods fail to drop for some other reason (such as from being jammed in their thimbles).

The resulting transient, caused by either the withdrawal of both control rods or of a strong neutron absorber, could be terminated by reinserting one of the control rods or a neutron absorber. The pressure increase within the reactor core is calculated to be insufficient to cause distortion of the control rod thimbles or exposure tubes; therefore, there should be no interference to prevent reinsertion. However, it is pessimistically assumed that the transient continues uncorrected for five minutes. A discussion of further continuation of the transient (for no action taken to terminate the transient within five minutes) is also presented in the Appendix (section VIC).

Based on experimental results of the KEWB reactor at low core pressures (see section VIC including Figure 16), the 1.0 percent reactivity step would be expected to cause the reactor power to rise on approximately a 50-millisecond stable period, terminating within 20 seconds at a peak power of approximately 10 megawatts. Following the peak, the power would drop off rapidly. The integrated power during the initial five minutes of the transient would be approximately 2 Mw-sec, and the reactor power level at the end of this period would be approximately 10 kilowatts.

For the first five minutes of the transient the total integrated radiation dose at the outer surface of the reactor shield tank (not at an open beam port) would be approximately 330 mrem. This is slightly more than one percent of the 25 rem permissible accidental or emergency whole-body dose occurring once in a lifetime.³

As a result of such an excursion, the rate of production of hydrogen and oxygen may temporarily exceed the capacity of the recombiner. The internal core pressure could reach a maximum of approximately 48 centimeters of mercury. The worst hazard would occur if the recombiner initiated an explosive recombination when the pressure was at its maximum. If this happened, a calculated pressure peak of approximately 180 psia would momentarily occur

within the core. Since the core vessel is designed and tested to withstand a sustained pressure differential of 300 psi, a substantially greater peak pressure could be tolerated without failure.

IV. RADIATION HAZARDS ANALYSIS

A. OPERATIONAL RADIATION HAZARDS

The Model L-77 Laboratory Reactor has been designed to present a minimum amount of radiological hazard during routine operation. This inherent high degree of radiation safety has resulted from three basic design principles:

- 1) Radiation shielding of a type, quantity, and sequence to permit maximum experimental exposures with a minimum of environmental radiation
- 2) Complete containment of fuel and fission products within the core, canopy, and associated gas and fuel piping adjacent to the core
- 3) Maximum simplicity of design and operation resulting in high system reliability

Table III presents some typical activity and radiation levels which are possible during operation of the reactor. These values are based on continuous 10-watt operation and therefore overestimate the levels which would generally be present during routine reactor operation.

TABLE III
MAXIMUM OPERATIONAL ACTIVITY LEVELS

Location	Dose Rate of Activity Levels
Reactor shield-tank outer surface	7 mrem/hr
Five feet from shield-tank surface	1 mrem/hr
Induced activity (saturated) in shield water (based on Los Angeles city water)	$4 \times 10^{-6} \mu\text{c/ml}$
Outer surface of solid shielding 20 minutes after shutdown	< 1 mr/hr
Induced activity in stainless steel core (10-month run at 10 watts, 60-day decay)	2.6 mr/hr-lb at 10 cm
Total fission product activity (long 10-watt run, 1-minute decay)	60 curies
Saturated activity of volatile fission products (Xe, Kr, I, and Br)	15 curies

Tolerance levels (300 mrem or less per 40-hour week⁴) are maintained outside of the reactor shield tank when exposure facilities are closed. Fast neutrons contribute from 20 to 25 percent of the total dose rate at the reactor shield-tank surface, the balance resulting from gamma rays.

B. EXPERIMENTAL RADIATION HAZARDS

Experimental full-power (10-watt) reactor operation results in radiation and radioactive isotopes of levels as indicated in Table IV.

TABLE IV
EXPERIMENTAL RADIATION DOSE RATES

Location	Dose Rate
Beam-tube port of open exposure facility at outer shield surface (both shield plugs removed)	3 rem/hr
Central-tube port of open exposure facility at outer shield surface (both shield plugs removed)	1 rem/hr
100-gm sample of cobalt-59 irradiated 1 week (1 meter from source)	1.1 mr/hr
1-gm sample of sodium-23 irradiated to saturation (1 meter from source)	24 mr/hr
1 meter from unshielded container filled with 1 liter of fuel solution (1-year, 10-watt operation, 5-day cooling)	580 mr/hr
Surface of 4-inch, lead-shielded, 1-liter container filled with fuel which had been drained 1 day after 1000 hours of 10-watt operation	16 mr/hr

C. CONSEQUENCES OF REACTOR MALFUNCTION

1. Containment

Fuel and fission products are contained within the core and canopy, fuel fill and drain line, and the vacuum line. These components are manufactured from stainless steel of adequate thickness and integrity to essentially eliminate the possibility of leakage during the expected lifetime of the reactor. Control-

rod thimbles and exposure facility tubes which penetrate the core and canopy are also made of stainless steel, and are welded to the core at the points of intersection. Nevertheless, the consequences of the severe accidents required to breach the containment, or of equivalent errors of operating personnel, may be seen to result in maximum radiation exposures far short of lethal values.

2. Loss of Shield Water

If a major break in the reactor shield tank, resulting in total loss of the shield water, should occur simultaneously with failure of the low-water-level scram interlock, the operating reactor would produce a maximum estimated radiation level at the tank surface of 2 rem/hr. Twenty minutes after shutdown the radiation level at the tank surface would be reduced to approximately 0.5 mr/hr. The accumulated exposure from such an event should be small, since the large quantity of water released would attract the immediate attention of the operator.

3. Fuel Leak to Shield Water

A break in the fuel containment components would result in increased pressure of the hydrogen and oxygen gas in the core, resulting in actuation of the core high-pressure scram interlock. If the fuel and fission products were somehow assumed to be entirely displaced to the outer water shield after a long 10-watt power run, the maximum dose rates outside the shield would be less than 1 mr/hr.

In the event of a leak of all fission gases to the air space above the shield water, the maximum dose rate on top of the shield tank could momentarily approach 5 r/hr, but would decay rapidly. The fission gases would leak from the air space in the tank to the outside, since the tank is not gas tight.

4. Simultaneous Fuel Leak to Shield Water and Shield Water Leak to Reactor Room

In the event that following a long 10-watt power run the fuel solution and fission products were all released to the shield water in some manner and subsequently all shield water released to the reactor room floor, the dose rate in a 20-foot by 20-foot room would be below 540 mr/hr after one minute of decay. Although release of this material would require major decontamination, the levels involved would be far from catastrophic.

5. Fuel Release Directly to Reactor Room

Because of the secure containment of fuel solution, its release to the reactor exterior is extremely unlikely. However, as a result of such events as earthquake, sabotage, or operating errors the fuel could conceivably be released to the reactor room by fracture of an exposure tube, or by pressurization through the fuel drain system.

Following a long 10-watt run, the fuel-solution activity will reach approximately 2.1 mc/ml (60 curies total activity) one minute after shutdown. Release of this activity would require immediate room evacuation due to the likelihood of high airborne activity concentrations and external dose rates. If 1 cm³ of the solution should become airborne in a 20-foot-square by 15-foot-high room, the resulting air activity concentration would be approximately $1.4 \times 10^{-5} \mu\text{c}/\text{cm}^3$. Assuming 10 years of reactor operation at 40 hours per week, the maximum possible combined air-activity concentration of the 26-year-half-life strontium-90 isotope and its daughter yttrium-90 from 1 cm³ of the solution would be approximately $1.2 \times 10^{-8} \mu\text{c}/\text{cm}^3$. This is approximately 20 times the maximum permissible concentration³ for strontium-90 plus yttrium-90. All other isotopes would have less hazardous activities in air. Gamma radiation from the spilled solution may be characterized by the dose rate per unit volume of approximately 3.0 r/hr per liter at a distance of one meter.

6. Fission-Gas Release to Reactor Room

The events postulated (paragraphs C4 and C5) could, depending on the nature of the core leakage, release the fission gases accumulated in the core void space to the room atmosphere. Measurements of fission-gas accumulation for the WBNS, KEWB, and SUPO reactors have all indicated that only the inert-gas isotopes of xenon and krypton and their daughters are normally present in this space.

If it were assumed that the reactor would be enclosed in a room having a volume of 6,000 ft³ held at slightly lower air pressure than the adjoining areas and ventilated by a flow of 600 cfm, that the void space gases were totally expelled to the reactor room from the operating reactor after several weeks of 10-watt operation, that the event remained undetected except by reactor

circuitry causing a scram due to high core pressure, and that the room was not vacated, then the total accumulated internal and external dose to a room occupant would not be more than approximately 4 rem.

This total maximum 4 rem dose, based on whole-body exposure to the inert-gas isotopes, is 16 percent of the 25 rem permissible accidental or emergency whole-body dose occurring once in a lifetime.³

7. Fission-Gas Release to Environs

The hypothetical accident of most serious consequences to the environs beyond the reactor room is assumed to be one in which 100 percent of the volatile fission products are released from the reactor after a long 10-watt power run. The Appendix (section VID) presents calculations which show that, even under the most pessimistic assumptions, the maximum dose that a person could possibly receive at any location outside the building would be approximately a 77-mr whole-body gamma dose plus a 6-rad thyroid dose from inhalation.

The 6-rad thyroid dose is approximately biologically equivalent to the 77-mr whole-body gamma dose, and though the effects are not necessarily additive, it is noted that this total dose is equivalent to 0.6 percent of the 25-rem permissible accidental or emergency whole-body dose occurring once in a lifetime.³

D. RADIOACTIVE WASTE DISPOSAL

No gaseous discharges are necessary from the L-77 reactor, and hence no stack is used. If, however, it should become necessary to withdraw radioactive gases from the interior of the system, it may be done with the use of evacuated cylinders. This technique has been employed for some time with the WBNS, KEWB, L-47, and L-77 reactors at Atomics International. Because of the design of the L-77 reactor, little if any liquid or solid waste is anticipated.

V. OPERATIONAL HAZARDS ANALYSIS

A. GENERAL SAFETY OF ROUTINE AND NONROUTINE OPERATIONS

1. Operator Surveillance

A single operator is in control of the reactor and the reactor room at all times. The small space requirements of the reactor permit the operator to have audio communication with all persons in the reactor room and to visually observe all activities except on the opposite side of the reactor.

2. Physical Hazards

There are no obstructions on the reactor tank to cause accidents which might result in personal injury or loss of control of the reactor. Simplicity of structure, of instrumentation, and of operating procedures makes it very unlikely for any sort of accident to occur. If any error should occur, such as through negligence or accident, the reactor design is such that it would be virtually impossible for any serious damage or danger to result.

3. Operating Procedures and Log

Operating procedures are prepared as regulations and instructions for the operator. They provide checklists to cover reactor startup, shutdown, and other routine and nonroutine operations. The startup checklist includes overall preoperational checking of the reactor components, verification of proper functioning of interlock and scram circuitry, and steps in bringing the reactor to operating power. The shutdown checklist renders the reactor secure against unauthorized operation and tampering.

A complete log of all pertinent information relative to reactor operation and performance is required by operating procedures. These include records of times of operation, power levels, difficulties encountered, incidents of any sort, experiments performed, samples exposed (including description, location, exposure time and level, and activity at discharge), instrumentation performance, tapping of fuel and gas lines, reactivity changes, and maintenance performed. Procedures for nonroutine operations require that radiation levels be checked and recorded as a routine matter before and during operation, and that water samples be analyzed periodically.

B. LOCK AND KEY FACILITIES

All exposure ports, beam tubes, and instrumentation thimbles terminate either in steel boxes recessed into the shield-tank wall, or are flush mounted at the shield-tank wall. An ignition-type key lock is used at each termination to secure the ports against insertion of unscheduled or unauthorized experiments or samples into the reactor. Each lock requires a positive effort in key manipulation to either open or close a port.

The key locks are opened by a master key which is kept by the operator. Personnel must consult the operator to obtain the master key before making use of any experimental facility. This system tends to minimize personnel over-exposures, and allows the operator to maintain control over reactor experimentation as well as operational activities. Operating procedures require that the operator check the security of experimental facilities before and after reactor operation.

To prevent unscheduled or unauthorized operation of the reactor, the console switch is also of the ignition type, requiring a key for operation. The main power switch is located within the console and ordinarily remains closed. The key switch on the console is a two-position switch, which, when turned to ON, energizes the interlock circuitry, the clutch-magnet power supply, and the rod-drive motors. With the key turned to OFF the only circuits energized are the regulated power supply, and the channel I and II neutron detectors and amplifiers.

There are three keys used in the operation of the reactor. One is for the console key switch, one for the source-conduit lock and rod-drive locks, and one for all experimental-facility locks. Operating procedures require that the console and experimental-facility keys be kept in the possession of the operator during periods of reactor use, and in the possession of responsible supervision at all other times; the control rod and source key will be kept by responsible supervision at all times. Because of these conditions for control of the keys, there is little probability of accidentally breaching the security of the reactor during operation or shutdown.

VI. APPENDIX

A. KEWB REACTIVITY TRANSIENTS

1. General

The Kinetic Experiments on Water Boilers (KEWB) Program has been conducted by Atomics International for the Atomic Energy Commission to study transient characteristics of aqueous-solution-type reactors. The KEWB Reactor core is basically similar to the L-77 reactor core, though smaller (the KEWB core having an ID of 12.3 inches compared to 15.6 inches for the L-77 core), of greater wall thickness and in general of more complex design than the L-77.

The transient characteristics of the L-77 reactor may be derived from the KEWB experimental data. Much greater excess reactivity is permitted in the KEWB reactor than the 0.5 percent provided by fuel loading in the L-77 reactor, so that short periods and severe transients may be obtained in KEWB.

Three series of experiments (A, B, and C) have been run in the KEWB reactor. The A series was conducted for transients caused by low reactivity steps (of the order of the excess reactivity in the L-77 reactor). The B series determined characteristics of higher reactivity steps. The C series included studies of initial core pressure effects. The A and B series results are summarized briefly below. The recent C-series results have not been summarized, but indicate that low initial core pressure has a strong tendency to reduce the severity of transients. The L-77 reactor core pressure is substantially below the pressures used in the KEWB A and B series, so L-77 transients would be of less magnitude. The L-77 maximum credible accident analysis as described (paragraph C), utilizes data from KEWB C series transients.

2. KEWB A Series Transients⁵

The A series of power excursions were induced in the reactor by rapid withdrawal of control rods to a number of different positions. In addition to furnishing data on the dynamic behavior of the reactor, the transients were designed to determine if the radiation levels accompanying normal reactor operation were sufficiently intense to effect recombination of the hydrogen and oxygen gases evolved from the core solution before dilution of these gases was accomplished by the sweep-gas stream.

Most of the transient tests which have been made were restricted to examinations of peak power as a function of reactivity, and included only that reactivity region in which gas production occurs at rates low enough to prohibit expansion pressure from occurring. The gas recombination system was excluded from the system, and a ballast tank was utilized to accommodate the gas produced during the excursions. This allowed examination of the core vessel alone with respect to the gas-ignition question. No coolant was circulated through the core, and in most of the excursions the reactor was shut down manually immediately after the peak power had been attained.

The reactivity range investigated included values up to very near prompt criticality, though all measured reactor periods were greater than one-half second. The power rise in every case was sufficiently slow to be followed accurately with a conventional chart recorder with fast pen-travel provisions (one-half second for full-scale travel). The signal to the recorder is furnished by a boron-lined ion chamber after amplification in a logarithmic amplifier. Period measurements taken from this information channel provided an accurate check on the direct-reading period channels which furnish information to the recording oscillograph.

A typical power trace is shown in Fig. 11. This was obtained with a reactivity release of 0.45 percent, which gives a reactor period of 4.8 seconds. As shown in the trace, the power reached a peak of 40 kilowatts, oscillated briefly with decreasing amplitude and then decayed with a negative period of about 80 seconds. The peak power is limited by the core temperature rise with significant assistance from the gas formation. The oscillations which follow the first power periodic behavior are damped and then finally disappear as the sustained high-power operation produces a sufficiently large temperature rise to compensate for the added reactivity. The final power-decay rate is determined by the temperature rise and the delayed neutron lifetimes.

A plot of peak power vs reactivity release and stable reactor period is shown in Figure 12. Earlier data from reactors at Livermore, California,⁶ and at the Los Alamos Scientific Laboratory⁷ are included.

Fig. 11. KEWB Power Trace for a Reactivity Release of 0.45 Percent

Fig. 12. KEWB A Series Peak Power vs Reactivity Release and Stable Reactor Period

3. KEWB B Series Transients⁸

The B series transient program was a study of the dependence of transient peak power, energy release, and core pressures upon input reactivity step. The test core was loaded with an excess reactivity of 4 percent. Step reactivity inputs up to approximately 2 percent have been investigated.

The transients described here were initiated by an automatic poison (neutron absorbing material)-rod withdrawal device. The poison rod is a boron carbide element in an aluminum sheath, and is withdrawn from the horizontal glory-hole by a weight and pulley system upon a signal from the automatic sequence timer. Time required for total removal of the poison rod from the core ranges from 0.5 to 1.0 second. The reactivity controlled by the poison rod is sufficient to keep the reactor far subcritical with the normal control rods fully withdrawn.

In preparation for a transient, the reactor is made critical at a low power, 4 watts, with the poison rod fully withdrawn. (Low-power critical rod position varies with temperature and core-water inventory.) After noting the critical control-rod position, the poison rod is fully inserted and the control rods are further withdrawn by an increment computed to give the desired positive period. After allowing the reactor power to decay to essentially zero, control is turned over to the automatic sequence timer. During a transient, the normal instrumentation scrams are bypassed and the reactor is limited only by its inherent physical design features.

Figure 13 is a plot of normalized peak power vs stable reactor period. Normalization was to an initial temperature of 25° C.

It may be noted that the trend in peak power dependence upon initial pressure is the same here as was observed in the early B series,⁹ i. e., higher initial core pressures result in higher transient peak powers, with the excursions initiated at near-atmospheric pressure giving about 15 percent higher peaks than those initiated at 15 centimeters of mercury.

Energy of the "first burst" is defined as the energy released at the time of the power minimum which occurs after the power peak. The magnitude of this minimum is usually on the order of 500 to 50,000 times smaller than the power peak. Figure 14 shows the relationship between energy release of the

Fig. 13. KEWB B Series Peak Power vs Stable Reactor Period

first burst and stable reactor period. There was a significant difference in energy release for various initial pressures. Differences of this order were not noted in the energy release up to peak power. This indicates that initial pressure has a marked effect on the trailing edge of the transient.

On very slow transients, (periods longer than 100 milliseconds), slow smooth pressure rises corresponding to the gradual system-pressure increase brought about by the production of hydrogen and oxygen, are indicated by all pressure transducers. With shorter periods, the pressure at the bottom of the core begins to develop a maximum shortly after the time of peak power. As period is decreased further, the maximum becomes more pronounced and moves to within a few milliseconds after time of peak power. For periods of less than 30 milliseconds, a distinct pulse is seen. Figure 15 shows the increase in the pressure pulse as seen by the transducer at the bottom of the core, indicating an inertial phenomenon in the fuel solution.

Approximately 10 milliseconds after this inertial pulse, a very narrow pulse is seen by the transducer located just above the normal fuel solution level. The amplitude of the latter pulse as a function of reactor period is also shown in Figure 15. The pulse apparently is caused by the fuel solution impinging upon the transducer as it is ejected from the sphere.

B. TESTS ON THE CATALYTIC RECOMBINATION OF HYDROGEN AND OXYGEN¹⁰

A mockup of the catalytic recombiner for the L-77 reactor has been tested successfully at powers up to 400 watts. The unit was operated for approximately 32 hours at an equivalent reactor power of 11 watts. The initial system pressure of 1.5 inches of mercury absolute increased to only 1.7 inches of mercury absolute during the run, a change of only 0.2 inches of mercury. The temperatures in both the recombiner and mockup core were essentially ambient at powers in the vicinity of 10 watts. No external heat was supplied to the catalyst during operation.

In another series of tests, a number of test capsules were prepared in order to obtain information on the reaction of $2\text{H}_2 + \text{O}_2$ in the presence of possible catalyzing agents. The capsules were evacuated and filled with a stoichiometric

Fig. 15. Pressure Pulse Amplitude vs Stable Period in KEWB

mixture of hydrogen and oxygen to an initial pressure of approximately one atmosphere. The pressure and temperature of each capsule were recorded as a function of time in order to measure the rate of combination of the gases.

In summarizing the results of these tests, the following points should be noted:

- 1) In the absence of a radiation field, it appeared that only an active dry catalyst such as platinum could cause ignition of a stoichiometric hydrogen and oxygen mixture.
- 2) A dry platinum catalyst (at up to 100° C initial temperature) did not cause ignition when the system pressure was below 16 inches of mercury absolute.
- 3) A high gamma field (about 10^5 r/hr) not only did not ignite the gas mixture, but did not produce an increase in the rate of combination.

Apparently a threshold pressure exists for the combination of a $2\text{H}_2 + \text{O}_2$ mixture. A capsule containing a glowing filament was filled with different initial quantities of $2\text{H}_2 + \text{O}_2$. It was found that below approximately two to three inches of mercury absolute there was no detectable ignition or explosion.

C. MAXIMUM CREDIBLE ACCIDENT

With the normal loading of uranium in the reactor and with the experimental facilities void, there is available a maximum of 0.5 percent excess reactivity which conceivably could contribute to an excursion. In addition, the core experimental through holes, if completely filled with lucite, may contribute up to an additional 0.5 percent reactivity. The maximum credible accident, therefore, is taken to be the instantaneous release of all the possibly available excess reactivity, which is 1.0 percent, assuming all control rods are inoperative. For comparison purposes, a 0.5 percent instantaneous reactivity release is included. The possible methods of initiating the maximum credible accident are described in section IIIF.

The KEWB transients of the C series allow a reasonable estimate to be made of the L-77 transient behavior following 0.5 percent and 1.0 percent reactivity step inputs. The KEWB power curves for step transients of these

magnitudes are shown in Figure 16. The initial temperature and pressure at which the transients were initiated duplicate the normal operating point of the L-77. The marked difference in the form of the power traces of the 0.5-percent and 1.0-percent reactivity step transients is a result of the latter transient being initiated with a completely filled core so that expulsion of the fuel occurred. After fuel expulsion, a second power peak occurred as the fuel drained back into the core.

From Figure 16 the power and integrated energy curves indicate that the total energy releases, at the end of five minutes following initiation of the transient, are approximately 1450 and 1750 kw-sec for the 0.5-percent and 1.0-percent reactivity inputs, respectively. Under these conditions, based on the normal 7-mrem/hr radiation level at the point of highest intensity on the reactor tank surface, the maximum total integrated radiation dose in five minutes would be approximately 280 mrem and 330 mrem, respectively. At the reactor console the dose would be considerably less. The reactor operator would have time to take emergency action and then leave the immediate vicinity without receiving a radiation dose in excess of the 300 mrem maximum permissible weekly radiation dose under normal occupational conditions.⁴

Of somewhat more concern is the possibility of a hydrogen-oxygen pressure buildup within the canopy region and a subsequent explosion triggered by the recombiner. From the KEWB data, the pressure buildup in the L-77 at the end of five minutes, not considering the effect of the recombiner, would be approximately 70 centimeters of mercury after adjusting for the difference in void volume and gas-evolution rates. However, the L-77 gas recombiner, located within the canopy area, will continuously recombine the hydrogen and oxygen evolved during the transient. The recombiner will not, however, be able to recombine the gases as fast as the formation rates. The effect of the recombiner on the equilibrium pressure may be estimated by the equation

$$\delta p(\theta) = \frac{K}{V} \int_0^\theta ([P(\theta) - C_1] d\theta ,$$

Fig. 16 KEWB C Series Transients

where $\delta p(\theta)$ = pressure incremental change above initial value
(atmospheres)

K = constant (liters of gas/kw-sec)

V = void volume (liters)

$P(\theta)$ = reactor power at time θ (kw)

C_1 = equivalent recombiner capacity (kw)

The constant K has a somewhat different value for each fuel enrichment; here, K is assumed to be 0.00353 liters/kw-sec for 90-percent enrichment.

From laboratory tests, minimum C_1 is one kilowatt; at higher temperature C_1 is greater. Based on the first five minutes of power trace A (Figure 16) and a recombiner capacity of one kilowatt, the maximum gas pressure increase above the fuel solution equals approximately 38 centimeters of mercury; therefore, the total gas pressure equals 43 to 48 centimeters of mercury absolute.

Under these conditions, laboratory tests with the vapor-pressure recombiner have shown that as the pressure reaches approximately 4 psia, a series of small explosions would occur in which very low overpressures would be experienced, tending to maintain the maximum pressure in the core vessel at or under 4 psia.

Even if the recombiner should trigger an explosive recombination, a resultant 20-fold increase in instantaneous pressure¹¹ would give a maximum overpressure of approximately 170 psi. This is well within the containment capability of the reactor core vessel and shield complex.

Continuation of the transient for a one percent reactivity step, beyond the initial five-minute period discussed above, would be as follows if no action were taken to reduce reactivity. For an additional period of approximately five minutes the power level of approximately 10 kilowatts would be maintained, raising the temperature of the fuel solution to slightly above 53° C and making the reactor subcritical because of the negative temperature coefficient of reactivity. Reactor power would gradually be reduced to a lower value at which heat generation would balance heat losses to the outside of the reactor shield tank, with a temperature difference of approximately 33° C being maintained

between the fuel solution and the ambient outside air. This low power level could be maintained for an extended period of time unless a control rod or neutron absorber should be inserted into the reactor core through the experimental facilities or control rod thimbles. The recombiner would be expected to prevent any serious pressure buildup.

It is concluded then that no serious danger exists either to the operating personnel or the reactor as the result of a 1.0 percent (or less) step reactivity insertion.

D. CALCULATION OF MAXIMUM ENVIRONS RADIATION FROM L-77 FISSION GAS RELEASE

To determine the maximum possible exposure which could be received by a person outside of the building which houses the reactor, an analysis is made of the possible radiation release during a hypothetical accident in which the fission gases are released. The following assumptions and bases of calculations are used:

- 1) There are 15 curies of volatile fission products (Xe, Kr, Br, and I) present in the core (based on saturated activity from a continuous 10-watt power run).
- 2) Release from core to environs equals 100 percent of the volatile fission products.
- 3) All volatile fission products appear simultaneously at the reactor room exhaust vent.
- 4) The exhaust vent release height is five meters (16.4 feet).
- 5) The average wind speed is 1 m/sec (2.2 mph).
- 6) Radioactive decay after reactor shutdown is not considered.

These assumptions are pessimistic, tending to overestimate the true hazard. The assumption of no radioactive decay has the most pronounced effect in maximizing possible radiation levels. The maximum total integrated gamma dose may be calculated by the following formula:¹²

$$TID_{(max)} = \frac{2Q}{\pi e u h^2} \frac{\text{curie-sec}}{m^3} ,$$

where:

Q = activity level of material released (curies)

$\bar{u}$ = average wind speed (m/sec)

h = release height (meters)

It should be noted that this value of total integrated dose is independent of all meteorological parameters other than wind speed. The formula is not valid for wind velocities approaching zero, because of the assumption of no radioactive decay. Inserting the assumed numerical values, the following result is obtained:

$$TID_{(max)} = 0.140 \frac{\text{curie-sec}}{m^3} ,$$

Since $1 \text{ curie-sec}/m^3 = 0.547 \text{ E roentgens}$, and assuming an average gamma energy of 1 Mev, we may express the total integrated external gamma dose as:

$$TID_{(max)} = 77mr$$

This represents only 0.3 percent of the 25-roentgen permissible accidental or emergency whole-body dose occurring once in a lifetime.³

It has been calculated¹³ that of the biological effect due to exposure to a volatile fission product cloud, approximately 50 percent is due to the gamma radiation and the other 50 percent originates with the thyroid dose delivered by the iodine isotopes. A thyroid dose of 2,000 rad is suggested¹³ as the equivalent in biological significance to a 25-rem whole-body dose. On this basis the estimated thyroid dose, biologically equivalent to the whole-body dose, would be 0.3 percent of 2,000 rad, or 6 rad.

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