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UNITED STATES
DEPARTMENT OF THE INTERIOR
GEOLOGICAL SURVEY
WASHINGTON, D.C. 20242

JAN 13 1967



Director, Division of Reactor Licensing
U.S. Atomic Energy Commission
Germantown, Maryland 20767

Attention: Mr. Roger S. Boyd,
Research and Power Reactor Safety Branch

Gentlemen:

In accordance with discussion with and instructions from your staff, I am pleased to transmit for your review and approval two copies of Hazards Summary Report for a 1000 KW TRIGA Mark-1 Reactor at the U.S. Geological Survey, Denver, Colorado. Under separate cover, an additional 35 copies are being sent to the office of Mr. Roger S. Boyd.

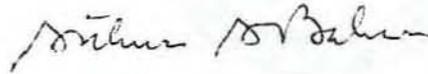
Close coordination has been maintained with the General Atomic Division of the General Dynamics Corporation through all phases of the report preparation, and guidelines were obtained from previous Hazards Reports acceptable to AEC.

The report includes a description of the site, reactor specifications, and an evaluation of potential hazards, an outline of the reactor administration and organization, and proposed procedures related to the reactor facility.

The Geological Survey requests that a construction permit be issued for a 1000 KW TRIGA Mark-I Reactor purchased from the General Atomics Division. There is an urgent need for a construction permit to make certain renovation to the newly constructed Nuclear Science Building (Building 15) at Denver, preparatory to installation of the reactor. Contractual schedule calls for completion of the reactor installation by September 1967.

Your approval of the Hazards Report and the early issuance of a construction permit to the Geological Survey will be appreciated.

Sincerely yours,

A handwritten signature in cursive script, appearing to read "Arthur B. Baker".

Acting Director

Enclosure

DOCKET NO. 50-274

UNITED STATES
DEPARTMENT OF THE INTERIOR
GEOLOGICAL SURVEY

DOCKET NO. 50-274

HAZARDS SUMMARY REPORT FOR A
1000 KW TRIGA MARK-I REACTOR
AT THE
U. S. GEOLOGICAL SURVEY
DENVER, COLORADO

This report has been compiled by the U. S. Geological Survey from data furnished by several sources, including the General Dynamics Corporation, General Services Administration, the U. S. Weather Bureau, and the U. S. Geological Survey. The report has not been edited for conformity with U. S. Geological Survey nomenclature and format.

December 1966



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1. INTRODUCTION

This report is part of an application by the U. S. Geological Survey, Department of the Interior, Denver, Colorado, for a Construction Permit, a special Nuclear Materials License for the fuel element inventory and a [REDACTED] Po-Be startup source, and a Class 104 License for a utilization facility to permit the installation and operation of a research, isotopes-production and training reactor (TRIGA-MARK I) to be purchased from the General Atomic Division of General Dynamics Corporation. Included are a description of the site, reactor specifications, and an evaluation of potential hazards, an outline of the reactor administration and organization, and proposed procedures related to the reactor facility.

The Nuclear Science Building in which the reactor will be installed is presently under construction. This building will be occupied by professional and technical personnel, most of whom have experience in handling, using, and measuring radioactive material, and are authorized to do so by existing AEC licenses (5-1399-8 and SNM-111). The building will be used for basic studies to improve methods and techniques to enhance scientific knowledge about water and earth materials. These studies will involve analytical techniques utilizing natural and induced radioactivity for determining the isotopic composition of earth materials related to geochronology and crustal evolution; areal distribution and geochemistry of minor elements; the genesis of ore deposits, the occurrence, distribution and geochemical behavior of trace elements in natural water; sorption and exchange studies; analysis of industrial pollutants, and pesticide residues;

measurements of transport of trace elements in stream sediments and their distribution between liquid and solid phases; and studies of the occurrence, distribution, and geochemistry of natural radioelements and fission products in environmental waters. The reactor is considered as an integral part of this research program.

The TRIGA* Mark I 1000 kw reactor is a light-water-cooled and -reflected reactor using advanced TRIGA U-ZrH fuel-moderator elements developed by the General Atomic Division of General Dynamics Corporation. This reactor is designed for continuous steady-state operation at 1000 kw and may also be pulsed repetitively to yield a burst having a prompt energy release of about 16 Mw-sec, a peak power of about 1,600,000 kw, and a pulse width at half maximum of about 11 msec. The technical specifications are summarized in Appendix 1 - I.

The safety of the TRIGA reactor lies in the large prompt negative temperature coefficient that is an inherent characteristic of the uranium-zirconium hydride fuel-moderator material. Thus, even when large sudden reactivity insertions are made and the reactor power rises on a short period, the excess reactivity is compensated automatically because the fuel temperature rises simultaneously so that the system returns quickly to a normal power level before any heat is transferred to the cooling water.

The inherent prompt shutdown mechanism of TRIGA reactors has been demonstrated extensively during more than 10,000 transient tests conducted at the two prototype TRIGA reactors in General Atomic's laboratories in San Diego. These tests involved step insertions of reactivity of up to 3.5 % $\delta k/k$. This demonstrated safety has permitted the location of TRIGA

* TRIGA trademark registered in U. S. Patent Office.

reactors in urban areas in buildings without the pressure-type containment usually required for other research reactors of similar power level and excess reactivity. At present, in addition to the 3 prototype reactors at General Atomic, 20 graphite-reflected TRIGA Mark I and Mark II reactors and 5 water-reflected Mark F and Mark III reactors are in operation; 15 of these TRIGA reactors are equipped for pulsing operation. The Mark III and Mark F reactors are licensed for routine pulsing with reactivity insertions of up to 2.1% $\delta k/k$.

The U. S. Geological Survey 1000 kw TRIGA Mark I is similar to the prototype Mark III except that the Mark I has a graphite and water reflector surrounding the core, rather than water only.

The U.S.G.S. TRIGA Mark I reactor will use fuel elements of an advanced type; the cladding is 0.02-inch-thick stainless steel, and the hydrogen-to-zirconium atom ratio in the fuel material is approximately 1.7 to 1. The improved performance and safety of the advanced fuel has been amply demonstrated by tests in the prototype reactor at General Atomic's laboratories in San Diego, and in five other TRIGA reactors. The prototype reactor has been in operation for five years with the advanced fuel.

In addition to being tested at steady state continuously for thousands of hours at powers up to 1500 kw, the core in the prototype reactor has been pulsed safely over 6600 times with reactivity insertions up to 3.5% $\delta k/k$. The result of a 3.5% $\delta k/k$ insertion is a power pulse with an initial period of about 1.5 msec and a peak power of about 8400 Mw. The prototype reactor is licensed by the United States Atomic Energy Commission to be pulsed routinely with insertions up to 3.2% $\delta k/k$.

The TRIGA family of reactors was developed for use by universities and research institutions as a general-purpose research and training facility. A prototype has been in operation since May 1958 at General Atomic in San Diego, California. The U. S. Geological Survey's reactor will be equipped with conventional safety rods and electronic scram equipment as additional safety measures.

Initial loading and start-up of the reactor will be supervised by General Atomic personnel in accordance with the procedures described in Appendix 1-II. At least one U. S. Geological Survey employee will be trained by General Atomic in the operation and maintenance of the reactor. The U. S. Geological Survey plans to hire additional reactor facility personnel, including senior operators and operators. Other U.S.G.S. personnel will be trained to operate the reactor after routine operation is established.

Table 1.1

GENERAL INFORMATION REQUIRED BY 10 CFR 50.33

- a. Applicant: U. S. Geological Survey, Department of the Interior
- b. Address: Federal Center, Denver, Colorado 80225
- c. Business: Research
- d. Applicant is not an individual
- e. License class, use, and duration:
 1. Construction permit for the installation of a 1 Mw pulsing Mark I TRIGA reactor. Permit is requested as soon as possible for a duration of 9 months.
 2. Class 104 license for operation of above reactor. Installation completion date estimated June-December 1967. License duration: 40 years. The reactor will be used primarily for research and

development, usually including isotope production related to geologic and hydrologic studies, and secondarily for instructing potential reactor operators.

3. Special Nuclear Materials license for the fuel element inventory and for a [REDACTED] Po-Be startup source. Duration and expiration date to be same as Class 104 license.
 4. By-product Materials License 5-1399-8 to be amended to provide for increase in material resulting from reactor operation.
- f. The applying organization is a part of the U. S. Government and is funded by Congress.
 - g. The scientific staff associated with the reactor use is composed of Appendix 1-III chemists, physicists, geologists, and geophysicists/who are qualified and authorized to use radioactive materials in research (Licenses 5-1399-8 and SNM-111). The reactor operating staff has not been employed, but Key members will be employed prior to reactor startup.
 - h. Earliest and latest dates for completing construction: February 1 and September 1, respectively. Depends greatly on effective date and duration of construction permit.
 - i. Reactor utilization does not include generating and distributing electric power.
 - j. This application does not contain classified defense information.

2. SUMMARY

This report shows that the 1000 kw TRIGA Mark I design makes safe and adequate provisions for the more routine considerations such as argon-41 generation, contamination from a fuel element cladding failure, radiation dose rates during full power operation, etc. Following is a brief summary of the detailed analyses contained herein on the two major credible accidents, loss of cooling accident and reactivity accident.

2.1. LOSS OF COOLING ACCIDENT

Should the pool be accidentally instantaneously emptied of water, radiation levels in the reactor room would be high from the decay gammas in the core. However, calculations indicate that the fuel element cladding would remain intact, and there would be no injuries to operating personnel or danger to the surrounding population from dispersal of fission products.

If the reactor had been operating at 1000 kw for infinite time just prior to the instantaneous loss of cooling water, with convective cooling of the core by air, the maximum fuel temperature would be less than 800°C , a temperature well below the 1400°C melting point of the stainless steel cladding. The equilibrium pressure resulting from fission gases, entrapped air, and hydrogen at 800°C is about 140 psi. This pressure produces a stress of 5100 psi, whereas the yield stress for the stainless steel cladding is $>12,000$ psi at 800°C . The main hazard would be from the high radiation levels from the unshielded core. After 1 day the direct radiation dose at the top of the pool would be about 1200 r/hr. However, the scattered radiation dose at the edge of the pool would be about 0.3 r/hr, thus making it possible to take corrective action after the loss of water.

2.2. REACTIVITY ACCIDENT

The limitation on the size of the reactivity insertion permitted for routine operation of the aluminum-clad fuel for TRIGA reactors was determined by the temperature at which the $\text{U-ZrH}_{1.0}$ fuel undergoes the phase transition from alpha to beta phase, which results in dimensional changes in the fuel material. This temperature limitation was determined to be 530°C . Since the stable gamma phase of the $\text{U-ZrH}_{1.7}$ fuel material utilized in the stainless-steel-clad TRIGA reactor fuel does not undergo such a phase transition, the pulsing limits for these fuel elements are set by the equilibrium hydrogen pressure within the fuel element. This pressure is a function of temperature and must not exceed the rupture stress of the fuel element cladding. For the 0.02-inch-thick stainless steel cladding, the rupture pressure has been measured to be 1800 psi at 100°C . The fuel temperature at which the equilibrium hydrogen pressure will be 1800 psi is about 1000°C . The peak fuel temperature occurring during 1-Mw steady-state operation (about 350°C) and resulting from the routine reactivity insertions of $2.1\% \delta k/k$ from zero power (about 450°C) are both well below the limit.

Although the reactor control system is interlocked to prevent such an occurrence, consideration was given to pulsing the reactor with a $2.1\% \delta k/k$ insertion (the total worth of the transient rod) when all of the other rods are completely withdrawn and the core is fully loaded. (To do this, conditions of the operating license must be deliberately violated and several interlocks and scrams must be bypassed.) Under such circumstances, the maximum total reactivity to be compensated by the steady-state temperature would be $2.8\% \delta k/k$ (4.9% , the maximum reactivity in the core, minus 2.1% , the total worth of the transient rod). The resulting steady-state power would be about 1.4 Mw, the average fuel temperature would be about 237°C , and the peak fuel temperature would be about 405°C at the steady-state condition before pulsing.

The average temperature rise resulting from the sudden insertion of 2.1% $\delta k/k$ is 233°C . The average fuel temperature at the conclusion of the pulse from 1.4 Mw would then be $237^{\circ}\text{C} + 233^{\circ}\text{C} = 470^{\circ}\text{C}$. The peak temperature rise would be 399°C for a total peak temperature of $405^{\circ}\text{C} + 399^{\circ}\text{C} = 804^{\circ}\text{C}$, well below the maximum temperature permitted. The resulting pressure would be about 60 psi, well below the rupture pressure of 1800 psi. Details of the analysis of this accident are found in Section 8.7.

Thus, it is concluded that these two major credible accidents will present no hazard to the health and safety of the public.

3. SITE

3.1 SITE LOCATION AND ADJACENT AREA

1. The reactor will be installed in the Nuclear Science Building [REDACTED] at the Denver Federal Center, Jefferson County, Colorado.

The Federal Center is a complex of U. S. Government offices and laboratories occupying 690 acres of land which is owned and maintained by the U. S. Government. It is 7 miles southwest of the central Denver business area, 3 miles west of the Denver city limits, 1-1/2 miles southwest of Lakewood, Colorado, and about 6 miles southeast of Golden, Colorado (fig. 3-1).

2. The Nuclear Science Building is near the eastern boundary of the Federal Center (fig. 3-2) and is under construction; completion is estimated for November, 1966. The building plan (fig. 3-3) provides space for about 40 U. S. Geological Survey employees engaged in radioactivity measurements and uses of materials ranging in intensity from natural levels to several curies.

3. [REDACTED]

The general area is open to the public from 6 A.M. to 6 P. M. on Monday through Friday, but access is limited to specified employees at all other times. The general area also includes areas which are restricted at all times to specified personnel. Some of the Nuclear Science Building will be unrestricted during normal working hours; restrictions in other areas of the building will range from no access to other than specified personnel to access when accompanied by user personnel.

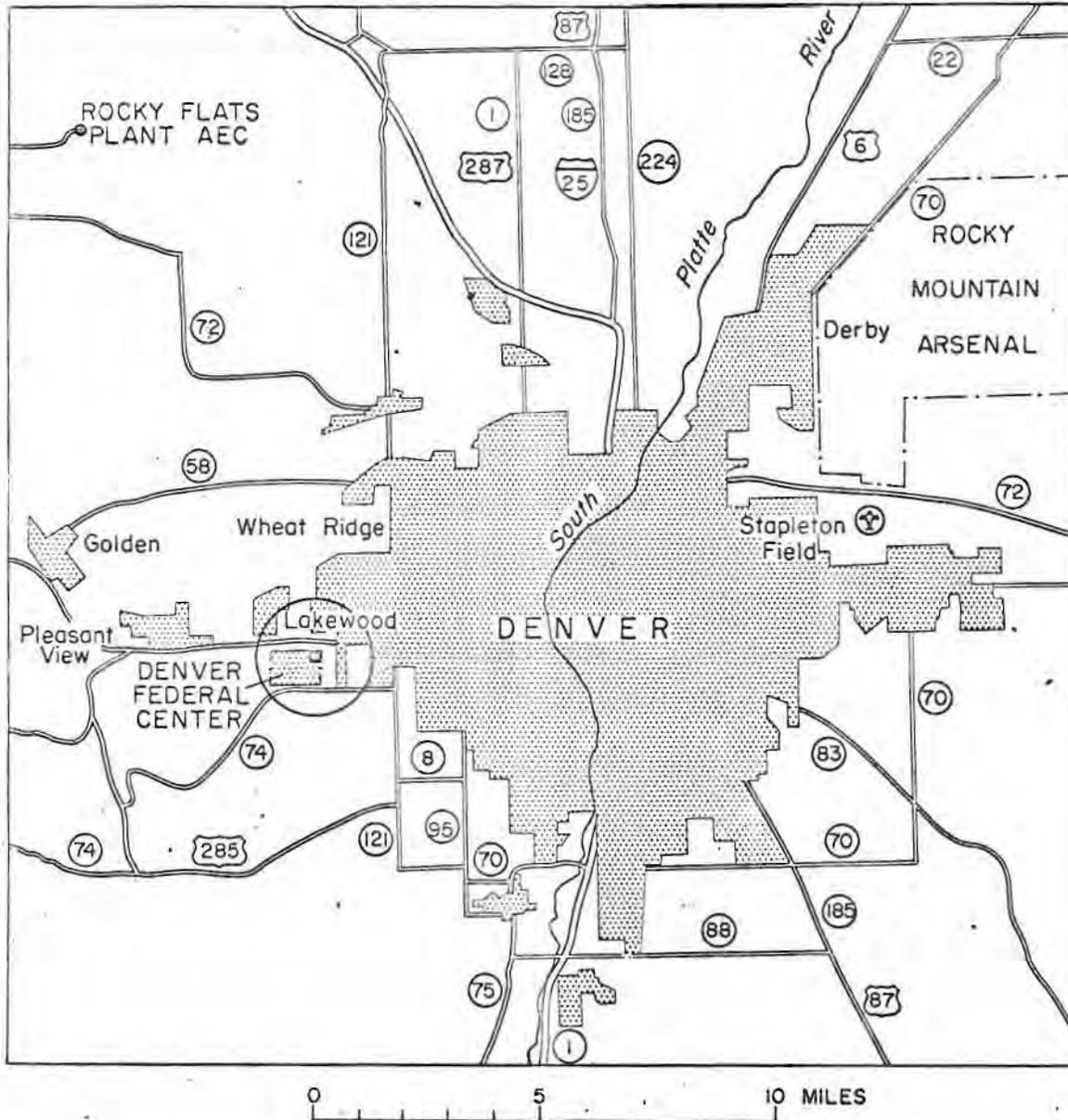


Fig. 3-1 -- Index map of Denver, Colorado and vicinity. Circle centered in Denver Federal Center has 7,000-ft. radius from reactor site.



Fig. 3-2. Index map, Denver Federal Center and Nuclear Science Building

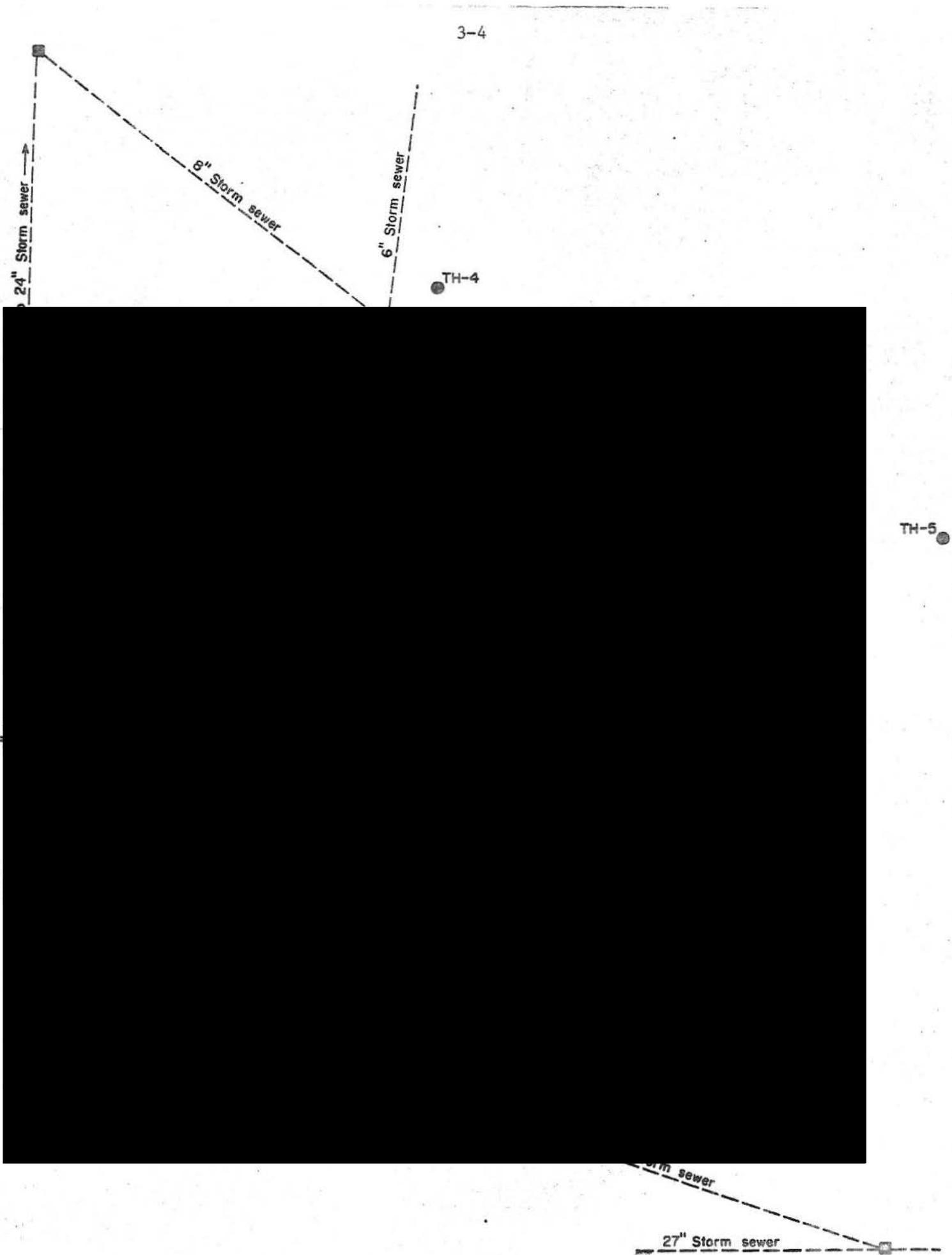


Fig. 3-3. Building plan, Nuclear Science Building

4. The daytime occupancy of personnel at the Federal Center is about 6,000; during the night less than 500 are on duty. No residences are within the Federal Center boundary. Occupants include the following U. S. Government Departments: Agriculture; Air Force; Army Reserves; Commerce; Health, Education and Welfare; Interior; Navy; and Treasury, plus the Civil Service Commission; Emergency Planning Office; General Services Administration; and Veterans Administration. Private enterprise is limited to daytime operation of cafeterias, a bank, and 2 airline ticket offices.

5. The population within the Denver metropolitan area which includes the vicinity of the Federal Center is about 1 million people (1964). The area around the Federal Center consists of open land, residential, and commercial property. The resident population in the vicinity of the Federal Center and within 7,000 ft. (approximately 1.33 miles) of the reactor site is shown in Figure 3-4. The population is based on a count of residences and an assumption of 2.8 occupants per residence (oral communication, U. S. Bureau of Census). The Federal Center occupies about 25 percent of the area within a 1.25-mile radius of the reactor site. Excluding the area covered by the Federal Center, about 75 percent of the remaining area is residential. The remaining area is zoned residential, but is presently open ground. The area includes two public schools (3,000 ft. southeast and 3,200 ft. northeast of the reactor site), a small stadium, and a county health clinic.

6. The general nature of activities and population within the Federal Center are expected to remain virtually constant for several years. Residential areas within a 1.25 mile radius of the reactor site have devel-

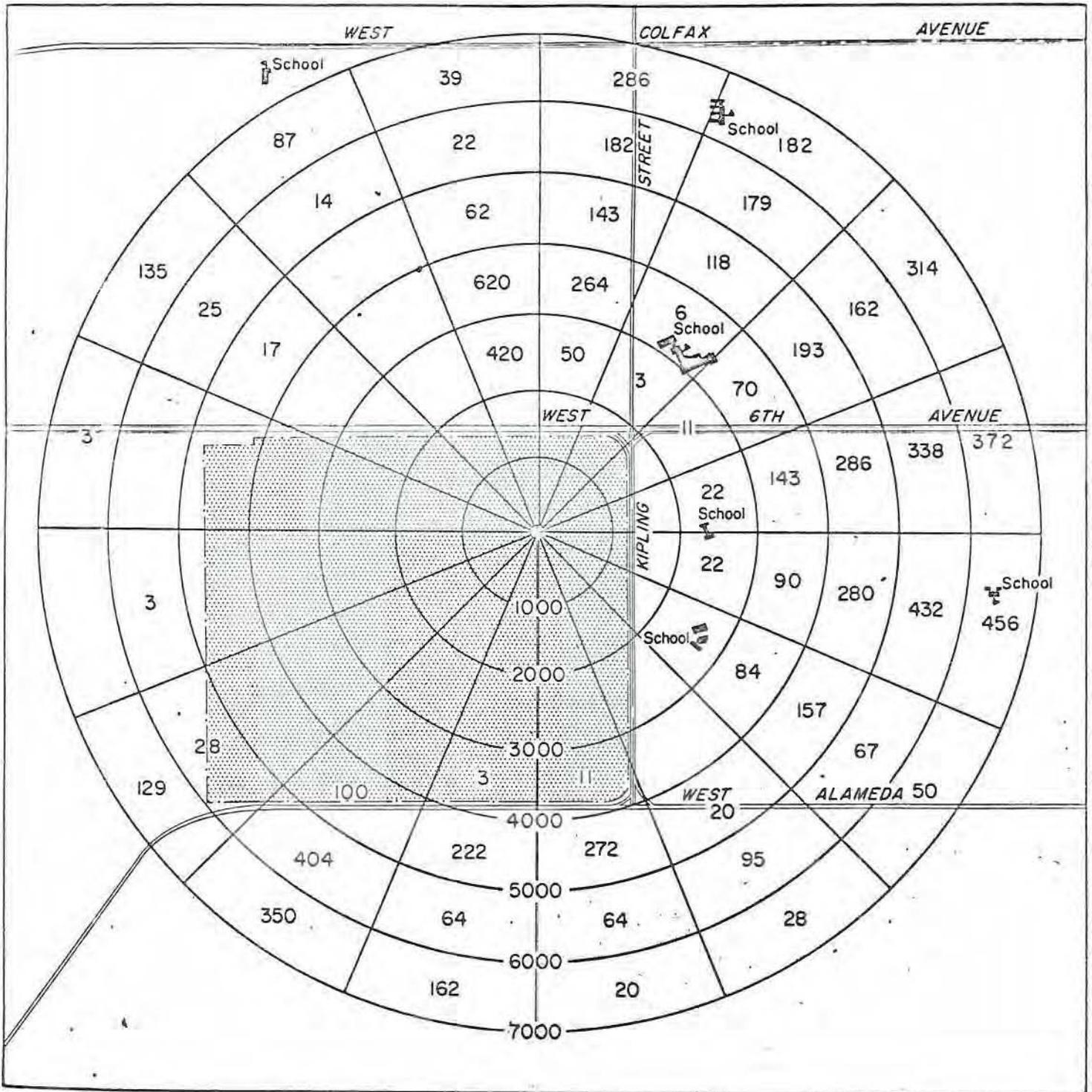


Fig. 3-4 -- Resident population distribution within 7,000 feet of reactor site. Numbers on circles represent distance from reactor. Numbers in radial segments show resident population; unnumbered segments contain no residences.

3.2 METEOROLOGY

1. The meteorological data were obtained from the U. W. Weather Bureau station at Stapleton International Airport located about 13 miles ENE of the Federal Center. General weather patterns at the two areas are related, but some aberrations cause local turbulence created by the foothills of the mountains near the Federal Center; records of these differences are not available.

2. Climatological summaries are presented in Table 3-1. Ten-year averages of wind velocities and direction for each month are shown in Figure 3-5.

3.3 HYDROLOGY

1. The site is 5,583 ft. above mean sea level. It is about 2,000 ft. from any definable drainage channel and about 5 miles from and 350 feet above the major river (the South Platte) which flows through the Denver area. Natural drainage from the site is to the northeast. Storm sewers with inlets near the site drain to the east and northeast, and discharge into Lakewood Gulch which discharges into the South Platte River. An irrigation canal lateral crosses the Federal Center from northwest to southeast and passes about 750 feet west and 10 feet higher than the reactor site. Possibility of overflow is remote but if it does occur the entire flow will be confined to streets and gutters and will not reach the reactor room. The sewage system (fig. 3-3) from the Nuclear Science Building drains into the Denver sewage system.

AVERAGE TEMPERATURE

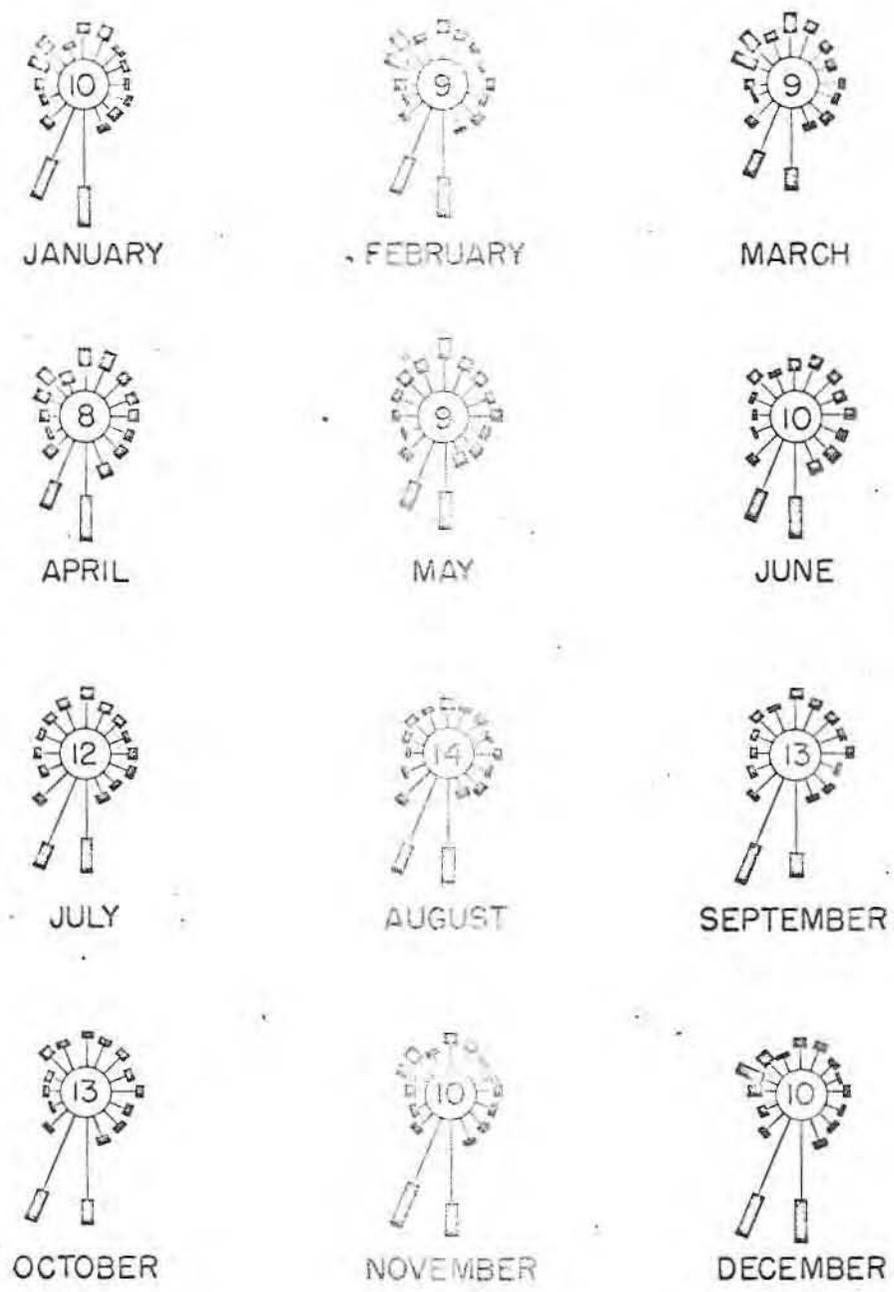
TOTAL PRECIPITATION

DENVER, COLORADO
STAPLETON INT AP

Year	Jan	Feb	Mar	Apr	May	June	July	Aug	Sept	Oct	Nov	Dec	Annual
1898	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1899	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1900	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1901	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1902	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1903	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1904	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1905	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1906	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1907	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1908	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1909	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1910	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1911	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1912	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1913	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1914	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1915	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1916	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1917	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1918	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1919	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1920	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1921	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1922	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1923	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1924	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1925	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1926	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1927	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1928	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1929	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1930	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1931	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1932	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1933	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1934	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1935	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1936	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1937	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1938	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1939	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1940	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1941	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1942	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1943	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1944	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1945	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1946	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1947	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1948	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1949	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1950	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1951	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1952	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1953	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1954	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1955	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1956	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1957	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1958	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1959	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1960	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1961	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1962	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1963	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1964	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2
1965	16.9	42.3	37.2	55.5	54.6	65.2	74.0	71.9	61.8	51.2	40.4	32.1	50.2

Year	Jan	Feb	Mar	Apr	May	June	July	Aug	Sept	Oct	Nov	Dec	Annual
1898	.46	.16	.44	.80	1.07	.17	1.78	2.02	.27	.63	.80	.62	9.12
1899	.02	1.11	1.00	2.23	3.43	.62	.90	2.36	.22	.63	.59	.63	13.16
1900	.53	.88	.83	1.39	1.85	1.91	.85	1.20	1.11	.99	.89	.49	10.12
1901	.91	2.21	3.32	4.08	2.34	1.60	.84	1.17	.91	.81	.18	.82	12.07
1902	.01	2.03	1.00	1.10	1.58	.22	.32	.91	.50	.7	.41	.67	8.53
1903	.68	.94	.19	3.74	4.92	.27	2.53	.87	2.18	.62	.29	.06	16.65
1904	.23	.18	1.36	.41	2.23	1.88	2.69	2.21	3.49	1.83	.13	.42	16.80
1905	.28	.49	.77	1.34	1.07	3.19	1.01	.95	.34	.64	.33	.93	11.34
1906	.43	.25	1.76	3.18	3.94	1.73	1.00	2.26	3.34	.44	1.46	.83	20.43
1907	.82	1.14	1.06	1.08	1.25	1.06	.17	.18	.87	.98	.37	.58	8.48
1908	1.01	.67	2.26	1.46	1.89	.10	1.24	.25	4.05	.35	.76	.38	14.50
1909	1.11	.28	1.33	3.28	3.71	2.93	1.29	1.84	2.48	2.44	.62	.86	22.05
1910	.66	.91	.68	4.17	1.12	3.08	1.02	.77	.81	2.94	.28	.15	16.59
1911	.23	.12	.43	1.04	2.96	1.22	.72	1.28	.07	.27	.41	.37	9.12
1912	1.08	.25	2.89	3.92	1.73	.82	3.34	.48	.7	.06	.52	.37	15.94
1913	.70	.49	.13	2.55	2.32	2.02	1.29	2.55	1.17	.78	.40	.09	15.34
1914	.84	.77	.52	2.09	1.95	.82	1.60	1.38	1.18	.88	.27	.04	14.29
1915	.37	.87	1.04	1.30	4.61	2.76	1.53	1.27	.91	3.41	.73	.27	19.06
1916	1.44	.44	1.71	2.32	1.84	1.04	.80	.41	.45	.16	.63	.26	12.62
1917	1.17	.03	2.29	1.46	3.21	4.27	1.35	.92	.28	1.36	.01	.33	18.78
1918	.47	.20	.31	2.98	2.80	3.32	.56	.27	1.58	1.12	1.00	.37	13.93
1919	.83	.78	1.47	2.01	1.78	2.27	.82	4.47	.97	2.16	1.17	.89	19.43
1920	.01	.68	2.12	2.75	3.06	.12	1.06	1.41	.54	.18	1.31	.19	13.43
1921	.39	1.39	1.15	1.29	2.68	1.46	1.98	1.25	.20	.44	1.00	1.02	14.23
1922	.23	.04	.49	.88	.60	.66	1.95	.51	.77	.06	.57	.71	7.51
1923	.23	.85	1.14	.48	2.47	1.39	2.99						

SURFACE WIND ROSES
DENVER, COLORADO



0 10 20 30 40 50 PERCENT
MEASURED FROM CIRCUMFERENCE OF CIRCLE
EXPLANATION

- 9 Percent of calms (0-3 M.P.H.)
- 4-12 M.P.H.
- ▬ Over 12 M.P.H.

Fig. 3-5 -- A TEN-YEAR SUMMARY OF WIND VELOCITY AND DIRECTION BY MONTHS FOR DENVER, COLORADO

2. The alluvium underlying the general area has low permeability and will locally yield small amounts of water to wells, possibly up to 20 gallons per minute. No pumped wells are within the Federal Center and any in the general vicinity are used only for lawn watering. The ground water level at the site is about 20 feet below the reactor room floor and slopes gently to the east.

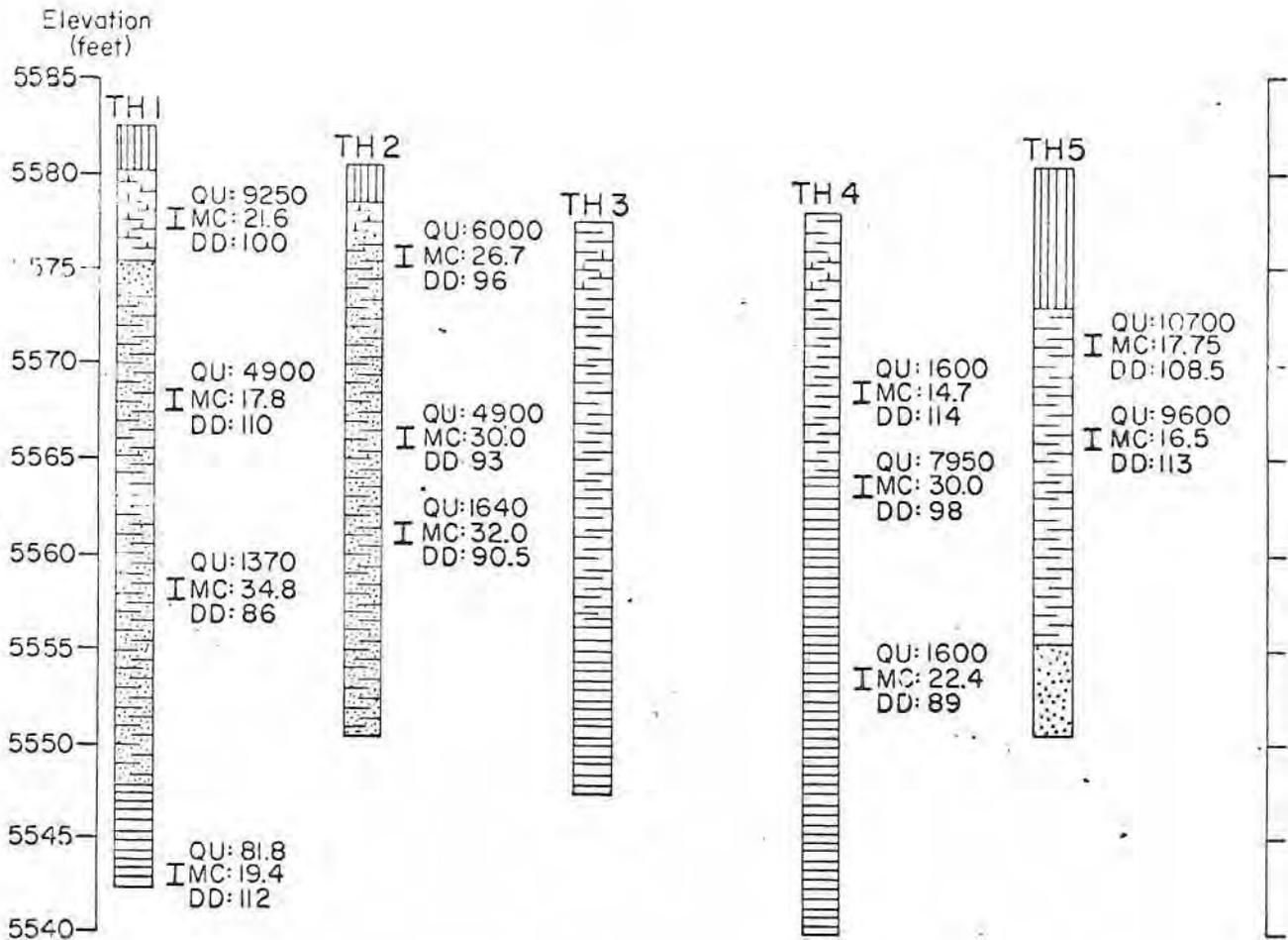
3.4 GEOLOGY

1. The Federal Center is situated on alluvium which extends from the foothills of the Rocky Mountains on the west to several miles east of Denver. Test borings near the reactor site (fig. 3-6) show that the underlying near-surface material is composed of gravel, sand, and shale.

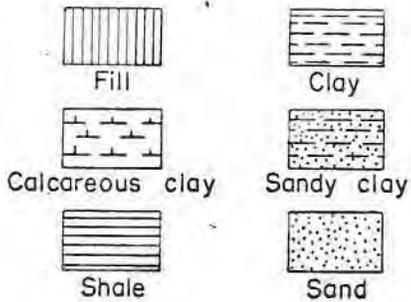
2. No geologic faults are indicated by surficial evidence within the area.

3.5 SEISMOLOGY

1. The site is in a seismically "quiet" area (Eppley, R. A., 1965, Earthquake history of the United States: U. S. Dept. of Commerce, No. 41-1). Occasional tremors centered near Derby, Colorado, about 15 miles northeast of the site have been recorded since 1962; these are measurable, but are not otherwise discernable at the Federal Center.



EXPLANATION



QU: Unconfined compression strength, in pounds per square foot
 MC: Natural moisture content, in percent
 DD: Natural dry density, in pounds per cubic foot

Fig. 3-6 -- Geologic and physical properties data from test holes

4. NUCLEAR CENTER FACILITIES

4.1 GENERAL

The reactor facility will be composed of two separate areas - the control area and the reactor room (presently designated as room [REDACTED] and an adjacent office [REDACTED]).

The reactor room will be located within a portion of the building which is constructed as follows:

Foundation: - - - Separate spread footings generally 6 feet below top of floor slab.

Floor: - - - 6-inch concrete slab on grade.

Frame: - - - 12-inch concrete bearing walls with concrete grade beams at wall openings. Concrete poured in place roof beams with integral one-way solid concrete roof slabs 5 inches thick. Top of roof slab is 17'-6" above floor slab.

The control area will be located within a portion of the building which is constructed as follows:

Foundation: - - - Separate spread concrete footings generally 7 feet below top of floor slab.

Floor: - - - 5-inch concrete slab on grade.

Frame: - - - Steel columns and beams without fire protection, open web steel roof joists and 1-1/2-inch steel roof decking, 9'-4" above floor slab.

The building was designed according to the Structural Engineering Handbook of the General Services Administration. Design loads are as follows:

Roof Live Load - - - 30 psf

Wind Load - - - Fastest wind for a 50-year period of recurrence

Seismic Load - - - Not considered

Allowable Soil Pressure - - - 3,000 psf minimum for dead load
6,000 psf maximum for dead load plus
live load.

4.2 REACTOR ROOM

The reactor room is 21 feet by 36-1/2 feet as shown on fig. 4-1. A new wall with door will be constructed between the reactor room and the control area. The reactor will be situated near the bottom of a water-filled pit in the center of the reactor room. The pit is to be 8 feet in diameter and 26 feet deep, and will be lined with aluminum 1/4 inch thick. All seams and joints in the liner will be welded and leak-tested; 20 percent of the welds will be radiographed as a check of internal weld integrity. The lower 10 feet of the liner will be surrounded by reinforced concrete 4 feet thick on the sides and 3 feet thick on the bottom to prevent unacceptable soil activation. The upper portion of the liner will be surrounded by concrete one foot thick. The outside of the aluminum liner will be coated for corrosion protection; the outside of the surrounding concrete will be waterproofed.

The reactor room is served by an existing 12-foot-wide 13-foot-high outside access door. A bridge crane having a capacity of 5 tons will be installed. The crane will be controlled from a pendant push-button control box independent of reactor control circuitry.

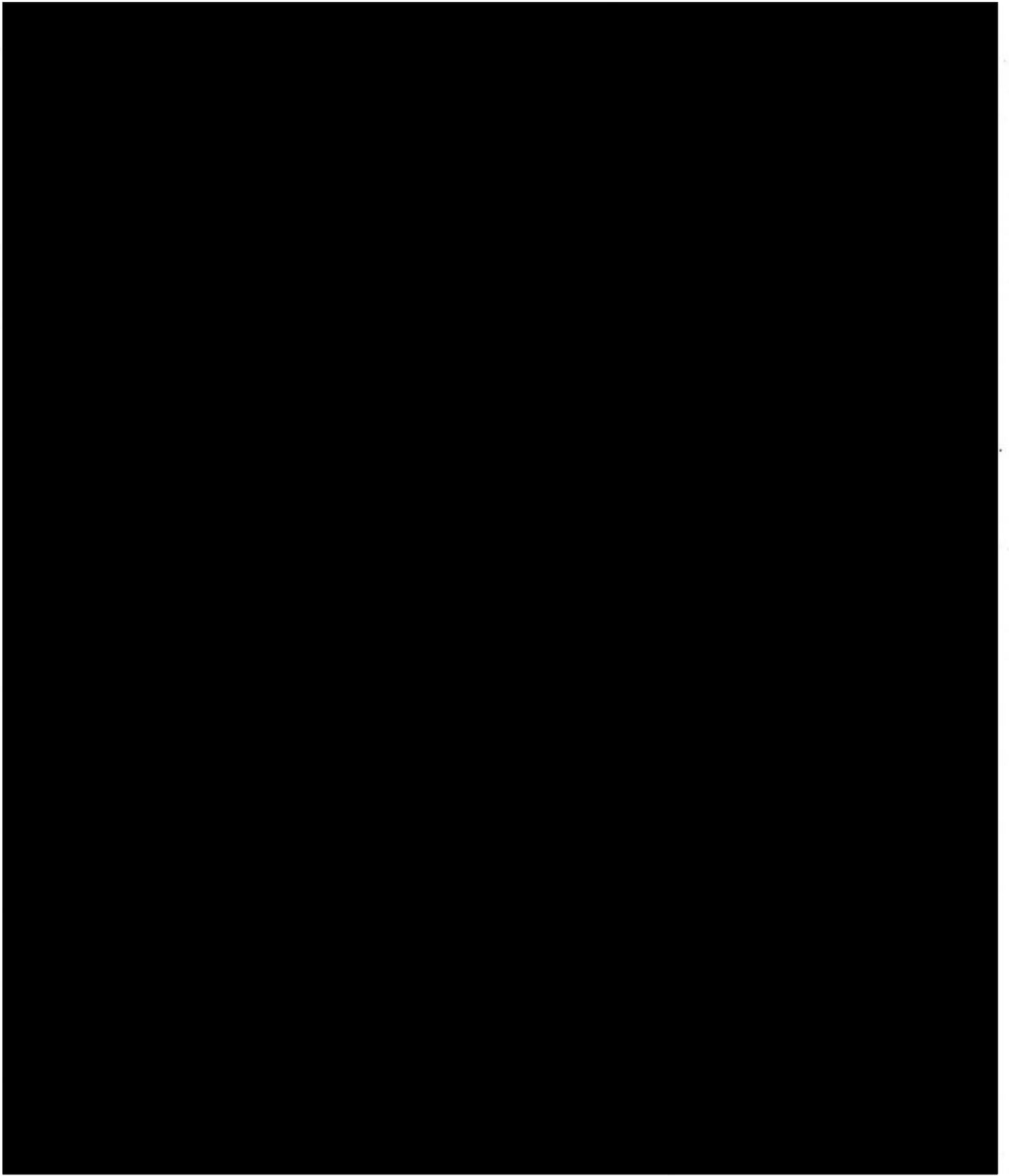


Fig. 4-1 -- Plan view, Reactor Facility Area

4.3 CONTROL ROOM

The control room will consist of a 9-foot by 19-foot control area and a 9-foot by 14-foot office area. In addition to the reactor control console, the control room will contain an experiment instrumentation area and a test set-up area. The control console will be located to provide direct visual communication between the console and the reactor room.

4.4 AIR CONFINEMENT

The reactor room will be designed to permit substantial confinement of contaminated air within the room. There will be no windows in the exterior walls. There will be special flashing and sealing materials at trench and piping penetrations. Openings for ventilation ducts will be sealed with epoxy mastic. Piping and wiring penetrations will be sealed with rubber grommets or with a suitable calking compound. All reactor room doors will have weather seals and door closers sufficient to allow maintenance of negative air pressure by means of the ventilation system.

4.5 VENTILATION AND FILTER SYSTEM

The existing supply air systems to the reactor room and the control room are to be retained. The control room is supplied by a dual duct high velocity system separate from the supply to the reactor room. The reactor room receives filtered 100 percent fresh air from Zone No. 3 of A/C Unit No. 3 located in the basement mechanical equipment room. Zone No. 3 serves the reactor room, radioactive materials room, and neutron generator room. An automatic positive-sealing damper is to be installed in the supply duct to the reactor room to permit confining the air within the room in the event the room air is accidentally contaminated.

The exhaust system for the reactor room, presently consisting of an exhaust fan mounted on the roof, is to be modified to provide an automatic positive-sealing damper to isolate the reactor room, a separate small-volume duct to exhaust air from the reactor room through an absolute filter mixing with a large volume of outside air and discharging through the roof to slowly purge the reactor room while maintaining a slight negative pressure within the room.

The reactor room ventilation system normal cycle is shown on Figure 4-2. The reactor room ventilation system with the reactor room isolated is shown on Figure 4-3. The reactor room ventilation system dilution cycle is shown on Figure 4-4.

A radiation monitoring system is to be installed in the exhaust air discharge to monitor the activity of the exhaust air. If the exhaust air activity becomes excessive, an automatic control system will cause the system to operate as described below.

Automatic operation is accomplished by a two-position selector switch labeled AUTO ISOLATE and DILUTE PURGE. With the switch in the AUTO ISOLATE position, a high radiation signal will cause the automatic dampers in the supply air and exhaust air ducts to close and cause the exhaust blowers to shut down. After evaluating the cause of the high radiation signal, the operator may turn the switch to the DILUTE PURGE position, causing the exhaust blower to start and slowly purge the room.

All automatic dampers will complete their operation within three seconds after receipt of signal.

The exhaust air will be discharged vertically upward through a stack extending 4 feet above the roof with a minimum velocity of 1200 feet per minute at all times.

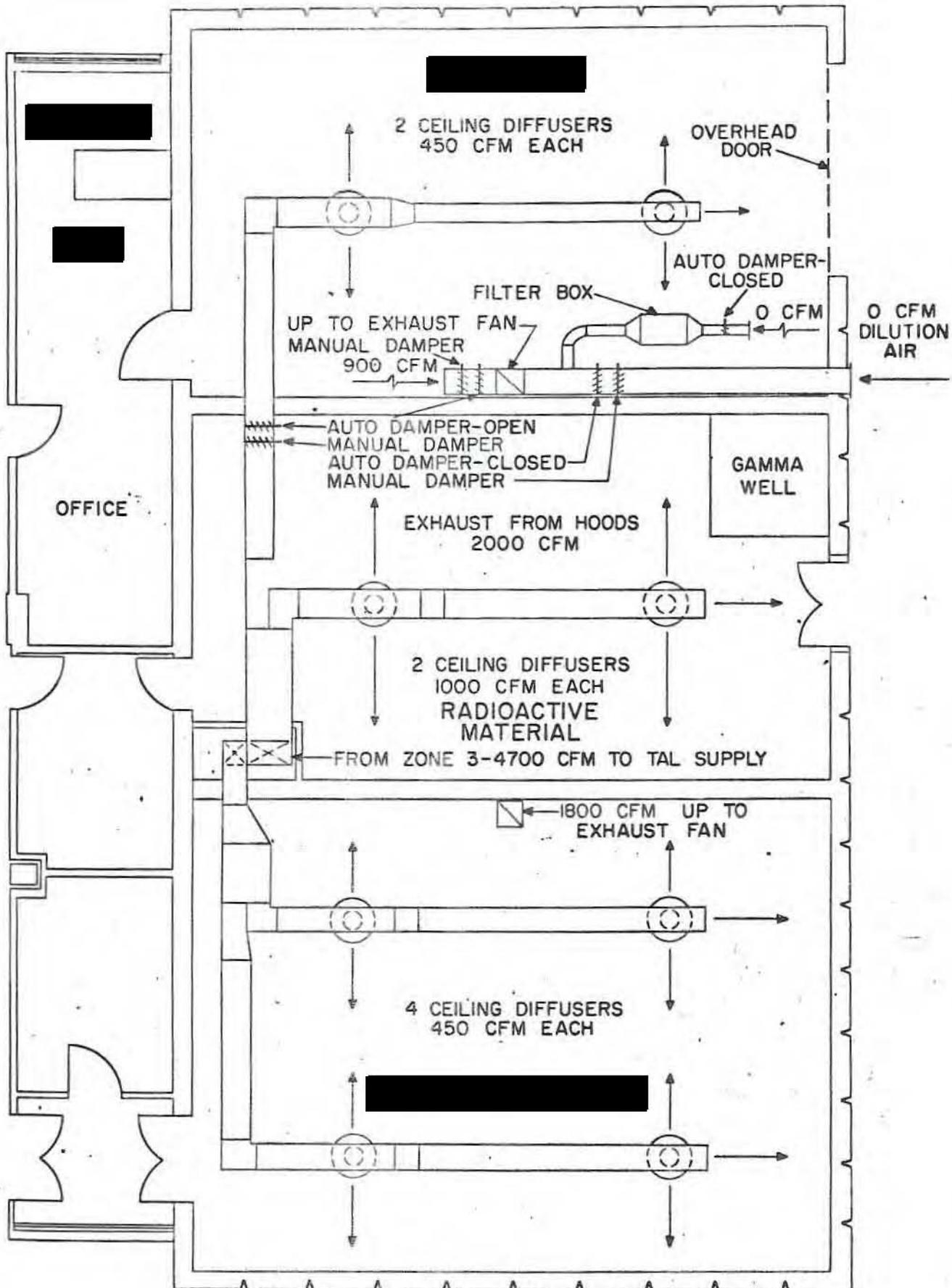


Fig. 4-2 -- Ventilation system, normal cycle

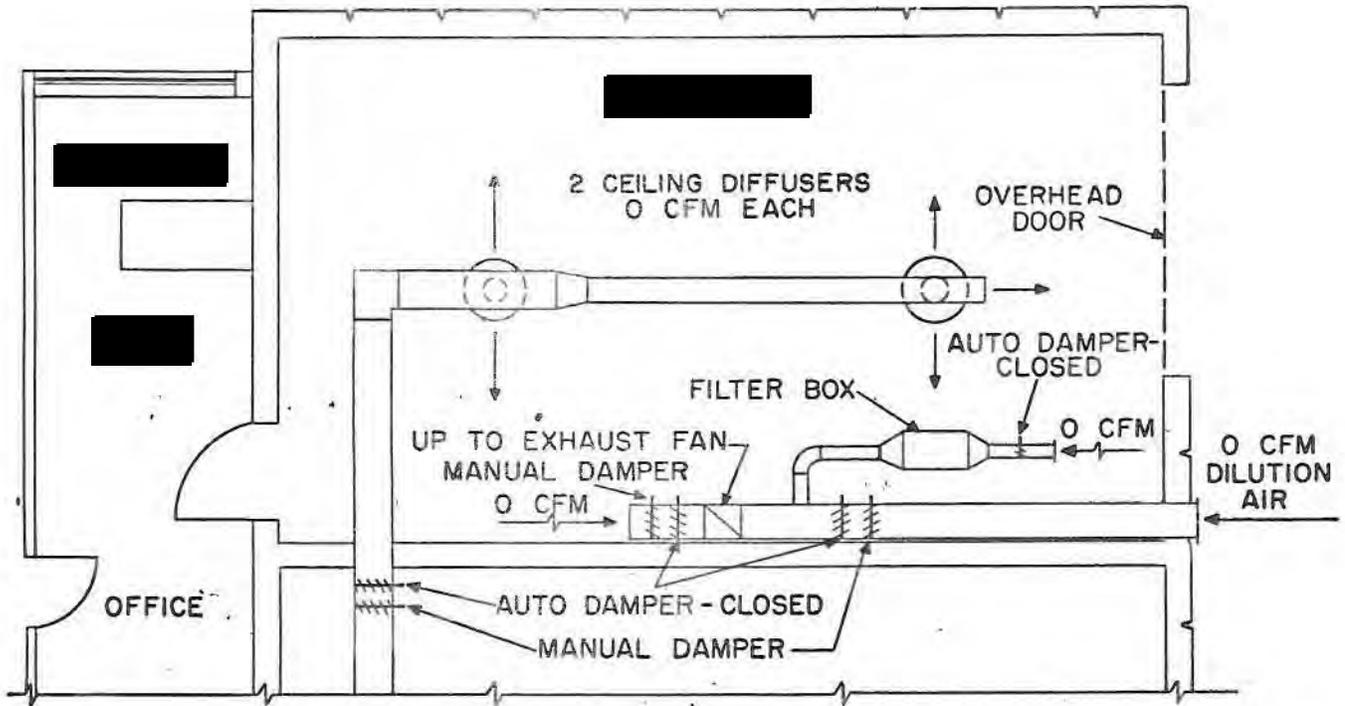


Fig. 4-3 -- Ventilation system, reactor room isolated

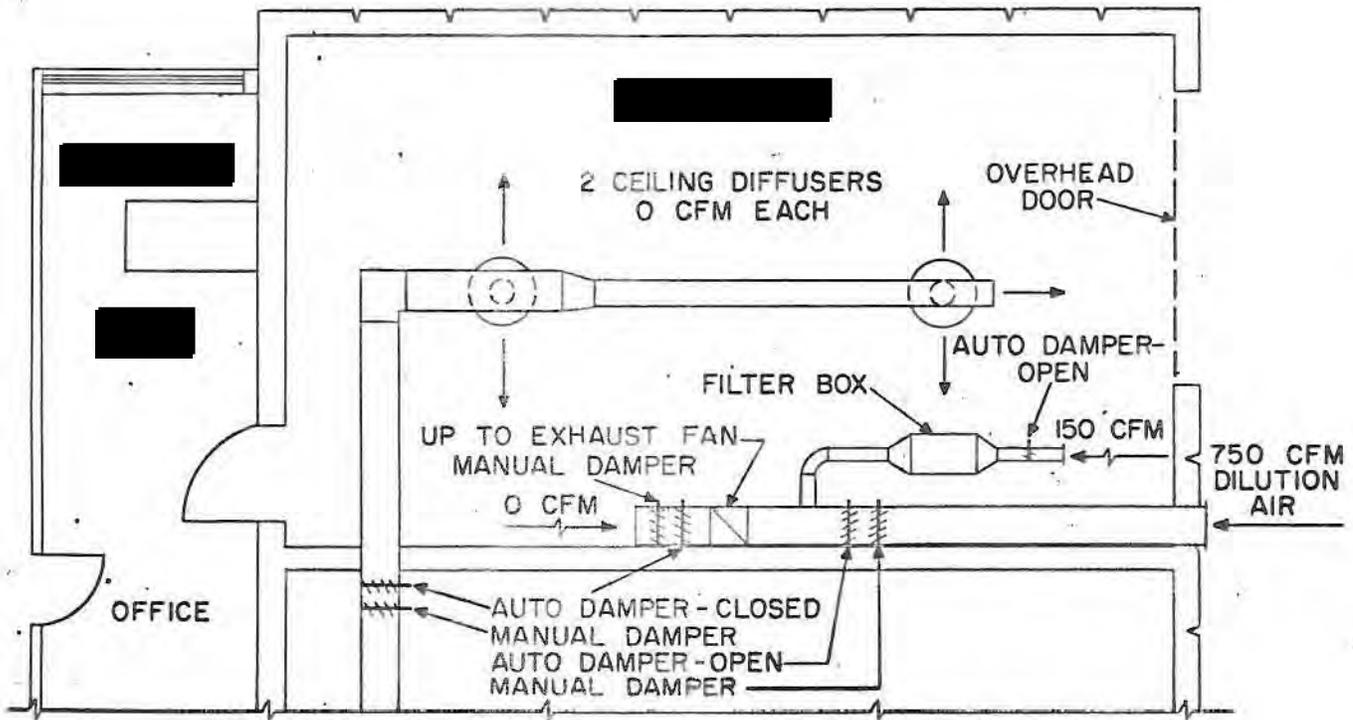


Fig. 4-4 -- Ventilation system, dilution cycle

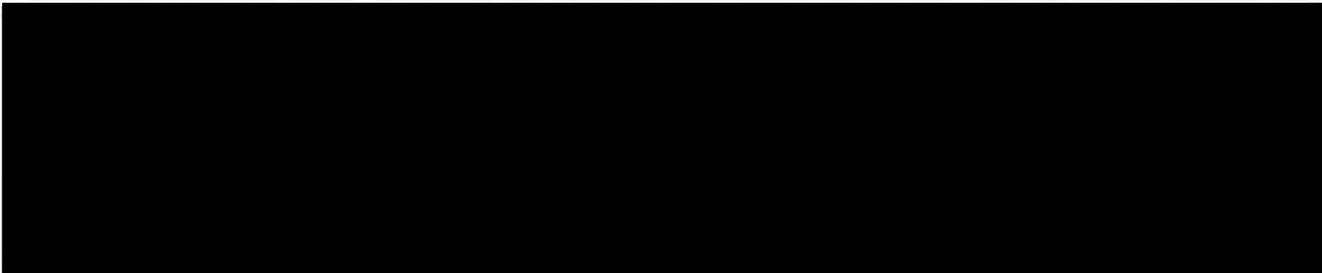
5. REACTOR DESCRIPTION

5.1. GENERAL DESIGN FEATURES

The TRIGA Mark I reactor developed by the General Atomic Division of General Dynamics Corporation, is an inherently safe reactor designed for advanced research in the various fields of nuclear research and education. The TRIGA Mark I reactor operates routinely at steady-state thermal power levels up to 1000 kw and has the capability of being pulsed repetitively to a peak power of approximately 1,600,000 kw.

Used in conjunction with the TRIGA Mark I reactor are facilities for high-energy neutron and gamma radiation studies, radiation effects testing, and sample activation. These facilities include a central thimble, an isotope production facility surrounding the core (rotary specimen rack), one or more pneumatic transfer systems, facilities for "in-core" irradiations and a large reactor pool for in-pool irradiations (see Fig. 5-1).

The reactor core is operated near the bottom of a large open pool. Approximately 20 feet of water above the core provides vertical shielding. The resulting design dose rate is approximately 10 mrem/hr at the water surface from gamma radiation direct from the core, neglecting N-16.



The fuel-moderator elements used in the 1 megawatt TRIGA Mark I reactor, which were developed by General Atomic, consist of a homogeneous mixture of uranium-zirconium hydride in which the H-to-Zr atom ratio is

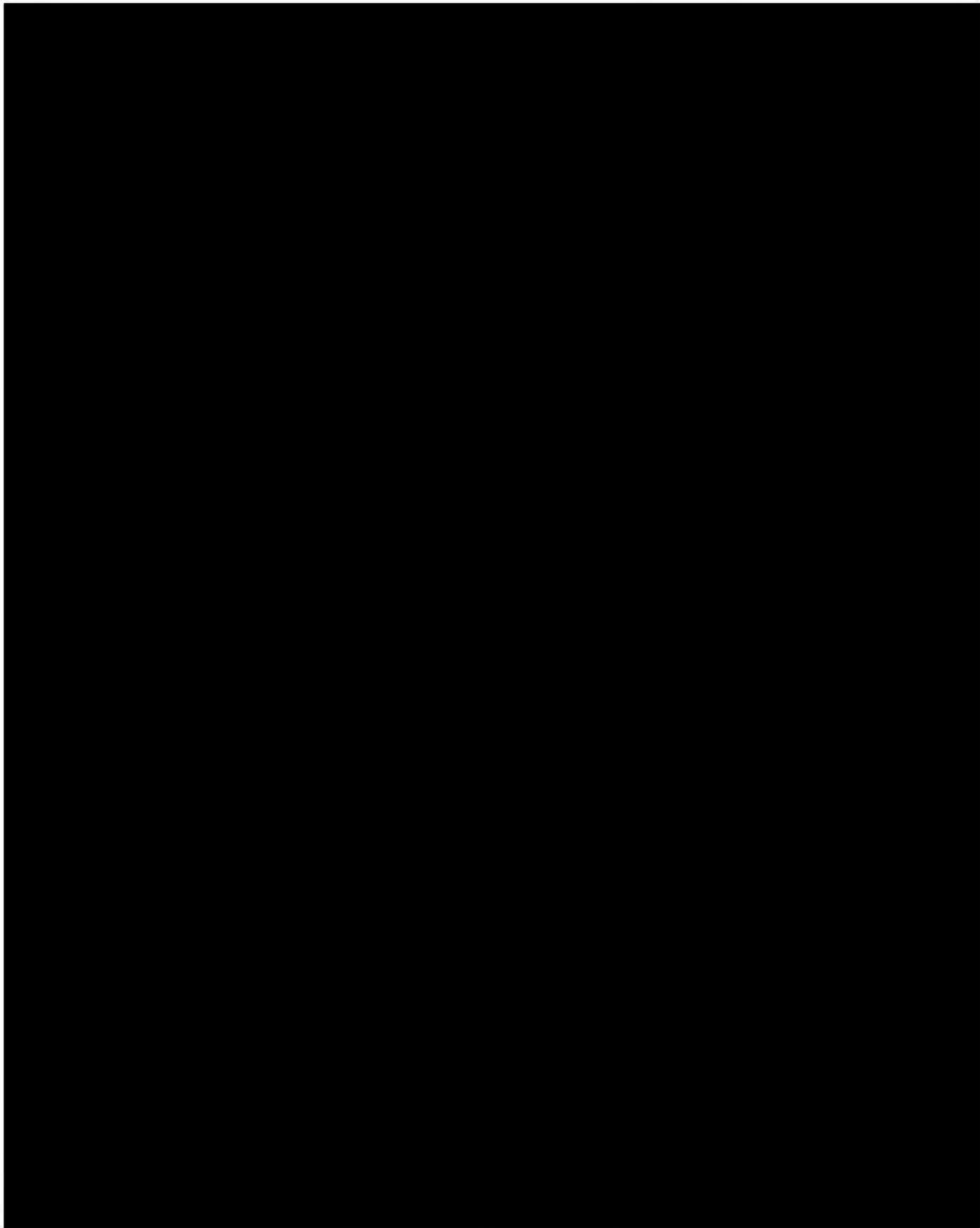


Fig. 5-1--Cross section of TRIGA Mark I reactor

approximately 1.7 to 1 and the uranium is enriched to 20% in U^{235} . The active portion of the fuel-moderator elements is 15 inches long and 1.43 inches in diameter. The elements are clad with stainless steel. The unique feature of these fuel-moderator elements is the prompt negative temperature coefficient of reactivity, which gives the TRIGA reactor its built-in safety by automatically returning the reactor power to a normal power level in the event of a power excursion.

The reactor core consists of a cylindrical lattice of fuel-moderator elements, control rods, and a polonium-beryllium neutron source. The water that surrounds these components occupies about one-third of the core volume. The core excess reactivity will be 4.9% $\delta k/k$. Graphite end sections of the fuel-moderator elements serve as the top and bottom reflectors. Neutron reflection in the radial direction is provided by approximately 1 foot of graphite enclosed in an aluminum housing. Initially, the outer regions of the core will contain graphite dummy elements.

The core components are contained between top and bottom aluminum grid plates which are supported by the reflector housing. The grid plates locate and support the core components. The top grid plate has 126 holes for fuel elements and control rods arranged in six concentric rings around a center hole.

The control rods pierce both grid plates and are guided by the grid plates. A restraining safety plate attached to the reflector housing beneath the bottom grid plate prevents the possibility of a control rod dropping out of the core. Three poison rods with fueled followers are connected to the electrically driven rack-and-pinion control rod drives located directly above the core on the bridge. The fourth poison rod (transient rod), located in an aluminum guide tube, has an air-filled follower. This transient poison rod is connected to a pneumatic-electromechanical drive, also mounted on the bridge. Physical access and observation of the core are possible at all times through the vertical water shield.

The rotary specimen rack is a doughnut-shaped watertight "lazy susan" which rests in a well in the top of the graphite reflector. The lazy susan rotates around the core and is used for isotope production. Forty test-tube-like containers in this facility may be loaded with samples through a watertight tube from the bridge above. Samples may be inserted and removed while the reactor is operating at full power.

A high-speed pneumatic transfer system permits research with extremely short-lived radioisotopes. The in-pile terminus of this system is located in the outer ring of fuel element positions.

A 1.33-inch-ID water-filled thimble for specimen irradiation is located in the central grid position. A hexagonal section can be removed from the center of the upper grid plate for the insertion of specimens up to 4.4 inches in diameter into the region of highest flux. (The central thimble must be removed from the core, and six fuel elements comprising the B-ring must be relocated in the outer ring of the core.) Two triangular-shaped holes have been provided in the upper grid plate so that specimens up to 2.4 inches in diameter may be inserted. If these facilities are not being used, three fuel elements (one D- and two E-ring) may be placed in each cutout. Both the top and bottom grid plates contain foil insertion holes for making flux measurements. Additional experimental tubes can be easily installed in the core to provide additional facilities for high-level irradiations or in-core experiments.

Instrumentation is provided to monitor, indicate, and record the neutron flux and power level and their rates of changes. Three modes of operation are possible: Mode 1, steady-state operation, with manual or servo control to 1000 kw; Mode 2, square-wave operation (transient reactivity insertions to reach a desired steady-state power essentially instantaneously) at power levels between 300 and 1000 kw; and Mode 3, transient (pulse) operation to 1,600,000 kw.

A fission chamber, with a motor drive to remove it from the core during high-power operation, provides low-level startup log-count-rate information. Two compensated ion chambers read out on linear and log recorders and also provide the signal for period indication during steady-state operation. During transient operation these chambers are disconnected. An ion chamber provides high-level linear-power information during square-wave operation and drives the circuits that read out peak pulse power on the linear recorder during transient operation. A servo is provided to automatically control reactor power in both steady-state and square-wave operation.

Bulk-water temperature, fuel element temperature, and water cooling system inlet and outlet temperatures are displayed on the console. In addition to the reactor control instrumentation, an interlock system is provided to prevent reactor operation unless prescribed safety conditions have been met.

The reactor core is cooled by natural convection of the demineralized water in the reactor pool. The water system provided with the TRIGA Mark I reactor takes water from the pool and pumps it through filtering, demineralizing, and cooling units and then back into the pool. The system consists of a 350-gpm pump, a filter, a demineralizer, a 1000-kw shell-and-tube heat exchanger, and associated aluminum piping and valving. Instruments are provided for measuring conductivity and pressure drop across the filter.

A portion of the 350 gpm pump discharge is diverted through a high-velocity jet above the core. The resulting circulation pattern reduces the dose rate at the pool surface resulting from the nitrogen-16 formed in the coolant water as it passes through the core.

5.2. FUEL-MODERATOR ELEMENTS

The 1 megawatt TRIGA Mark I reactor uses fuel-moderator elements in which a zirconium-hydride moderator is homogeneously com-

bined with partially enriched uranium fuel. The active section of this fuel-moderator element, [REDACTED] contains approximately 8.5 weight-% uranium enriched to 20% in U^{235} . The hydrogen-to-zirconium atom ratio of the fuel-moderator material is approximately 1.7 to 1. To facilitate hydriding, a 0.18-inch-diameter hole is drilled through the center of the active section; a zirconium rod is inserted in this hole after hydriding is complete. As shown in Fig. 5-2, graphite slugs, 3.47 inches in length and 1.4 inches in diameter, act as top and bottom reflectors.

The active fuel section and top and bottom graphite slugs are contained in a 0.02-inch-thick stainless-steel can. The stainless steel can is welded to the top and bottom end fittings. The top end fitting is grooved and specially shaped to fit and lock into a fuel handling tool. The top end fitting also incorporates a triangular spacer block that positions the top of the element in the top grid and yet provides passages for cooling water flow through the grid. The bottom end fitting fits into the beveled holes of the bottom grid plate and supports the entire weight of the element. The approximate over-all weight of the element is [REDACTED] pounds, and the average U^{235} content is about [REDACTED] grams. Serial numbers scribed on the top end-fixture or spacer block are used to identify individual fuel elements.

The initial reactor core loading of stainless-steel-clad fuel elements will produce a cold, clean, excess reactivity of 4.9% $\delta k/k$. Included in this number are one instrumented element, and three fueled control-rod followers. Graphite-filled dummy elements occupy all the remaining holes.

One fuel-moderator element is instrumented with three thermocouples embedded in the fuel. As shown in Fig. 5-3, the sensing tips of the fuel

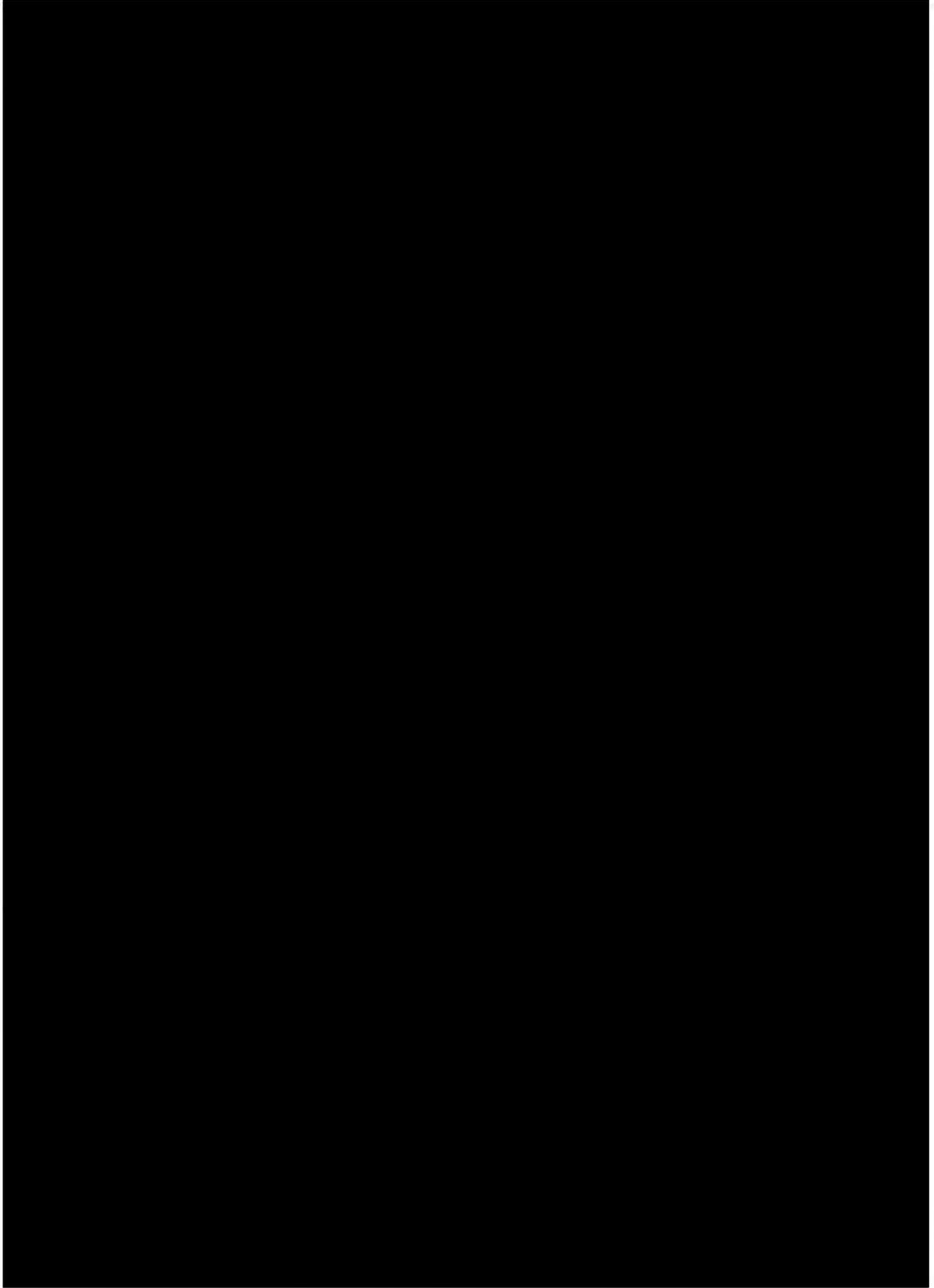


Fig. 5-2--Stainless-steel-clad fuel-moderator element

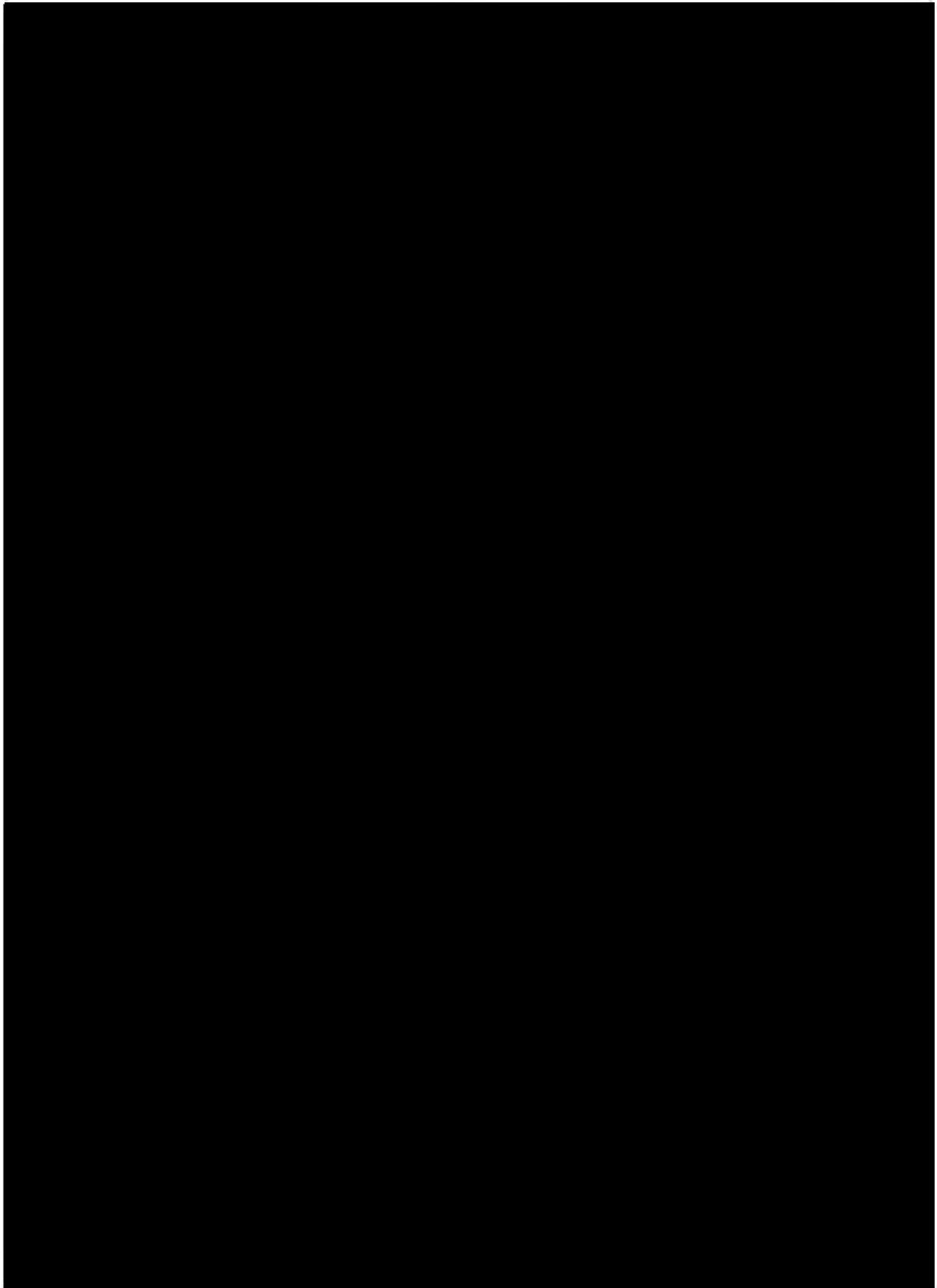


Fig. 5-3--Instrumented stainless-steel-clad fuel-moderator element

10-foot lengths of 1/2-inch-OD tubing connected by Swagelok unions to provide a watertight conduit carrying the leadout wires above the water surface in the reactor pool. In all other respects, the instrumented fuel-moderator element is identical to the standard one.

5.3. CONTROL RODS

5.3.1. Control Rod Descriptions

Three motor-driven control rods--one safety, one shim, and one regulating--control reactor power during steady-state operation. These control rods pass through and are guided by the top and bottom grid plates. The regulating rod is located in the C-ring, and the shim and safety rods are located in the D-ring. The exterior of the control rods is a sealed stainless steel tube approximately 43 inches long and 1-3/8 inches in diameter. The standard control rod is shown withdrawn and inserted in the sketch in Fig. 5-4. For guidance, these rods utilize 1-1/2-inch-diameter holes in the top and bottom grid plates. The upper section of the rod is graphite, the next 15 inches is the neutron absorber (graphite impregnated with powdered boron carbide), the follower section consists of 15 inches of U-ZrH_{1.7} fuel, and the bottom section is 6-1/2 inches of graphite. An aluminum "safety" plate attached to the reflector beneath the lower grid plate prevents the possibility of a control rod, accidentally disconnected from its drive, from dropping out of the core.

The fourth control rod, shown in Fig. 5-5, is the transient rod. It also acts as a safety rod in the steady-state mode of operation and is located in the C-ring. The transient rod assembly is about 37 inches long and is contained in a 1-1/4-inch-OD aluminum tube. The borated graphite poison section is 15 inches long. Unlike the standard rods, the transient rod has an air-filled follower about 21 inches long. The transient rod is guided laterally in the core by a thin-walled aluminum guide tube that passes through the upper and lower grid plates and is screwed into and supported by the aluminum safety plate beneath the grid.

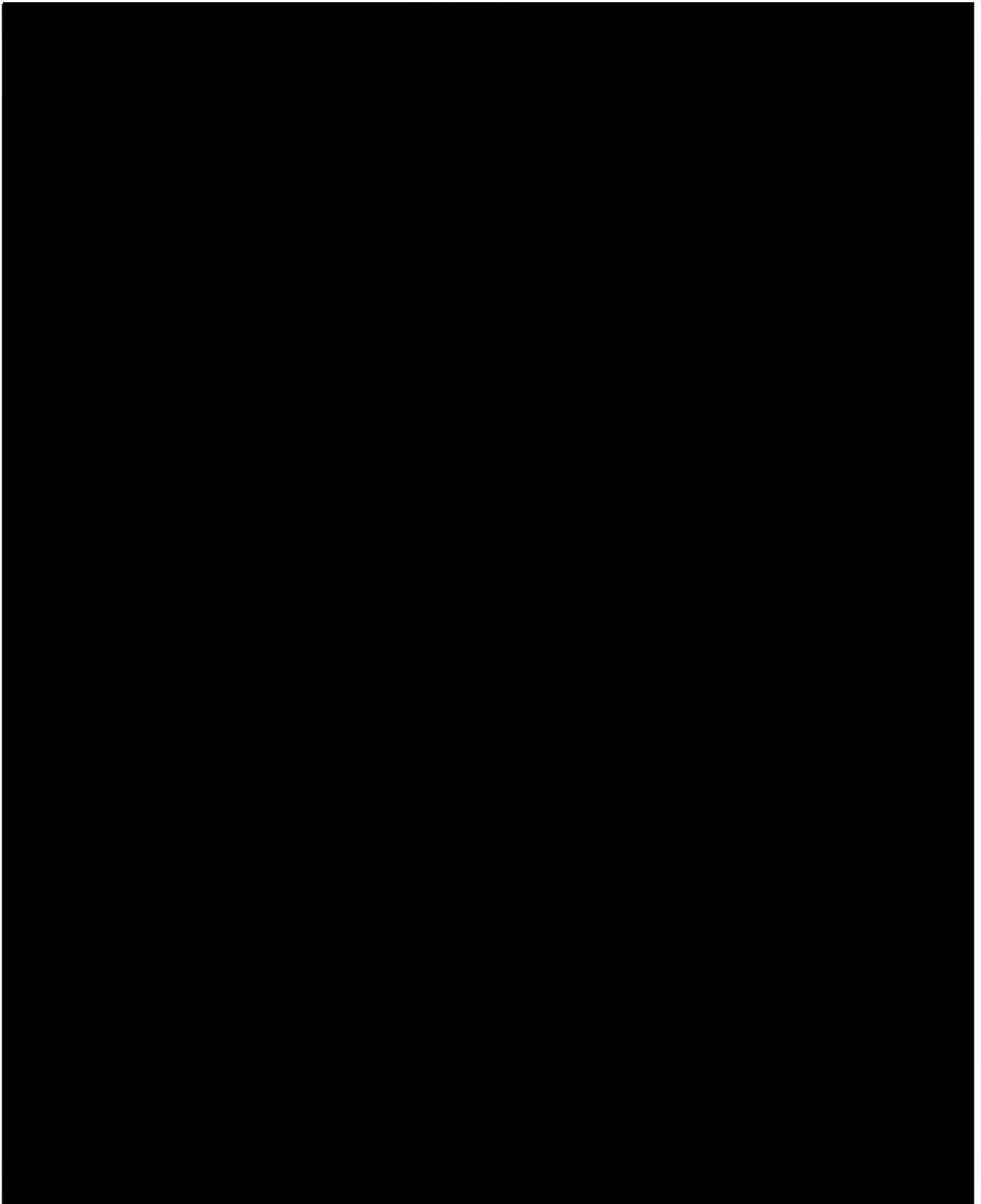


Fig. 5-4 -- Fueled-follower control rod in core

5-11

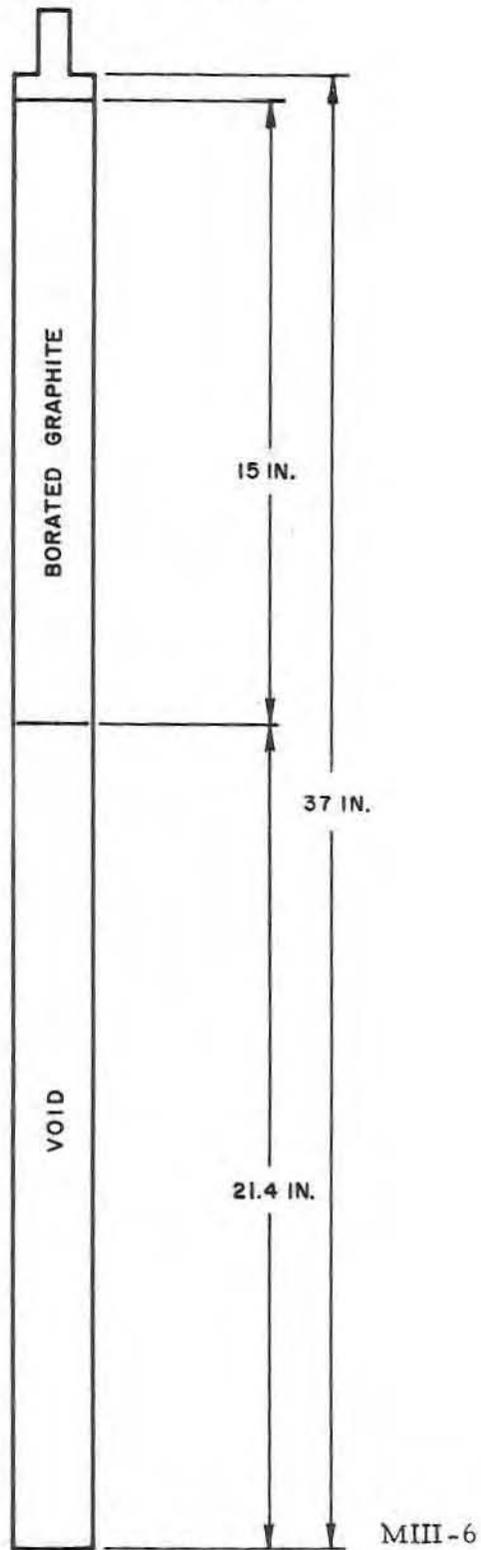


Fig. 5-5--Transient control rod

All control rods have a stroke of approximately 15 inches.

5.3.2. Control Rod Worths

The net change in reactivity that can be caused by the withdrawal of the regulating rod is about 2.7% $\delta k/k$. This amount of change is caused by withdrawal of the poison section and the simultaneous insertion of the fueled follower into the core. The net worth of the shim and safety rods is about 1.8% $\delta k/k$ each. Total net worth of the three standard control rods is about 6.3% $\delta k/k$. The net change in reactivity that can be caused by the operational withdrawal of the transient rod is about 2.1% $\delta k/k$.

5.4. CONTROL ROD DRIVES

The drive assemblies for the four control rods are fastened to a mounting plate located on the bridge. The standard control rods have electrically driven rack-and-pinion drives, and the transient rod has a pneumatic-electromechanical drive.

5.4.1. Rack-and-Pinion Drives

Rack-and-pinion drives, shown in Fig. 5-6, are used to position the shim rod, the regulating rod, and the safety rod. Each drive consists of a single-phase, reversible motor; a magnet rod-coupler; a rack-and-pinion gear system; and a ten-turn potentiometer used to provide an indication of rod position. The pinion gear engages a rack attached to a draw tube supporting an electromagnet. The magnet engages an iron armature attached above the water level to the end of a long connecting rod that terminates at its lower end in the poison rod. The magnet, its draw tube, the armature, and the upper portion of the connecting rod are housed in a tubular barrel. The barrel extends below the reactor water level with the lower end of the barrel serving as a mechanical stop to limit the downward travel of the control rod assembly. Part way down the upper portion of the connecting rod, i. e., just below the armature, is a

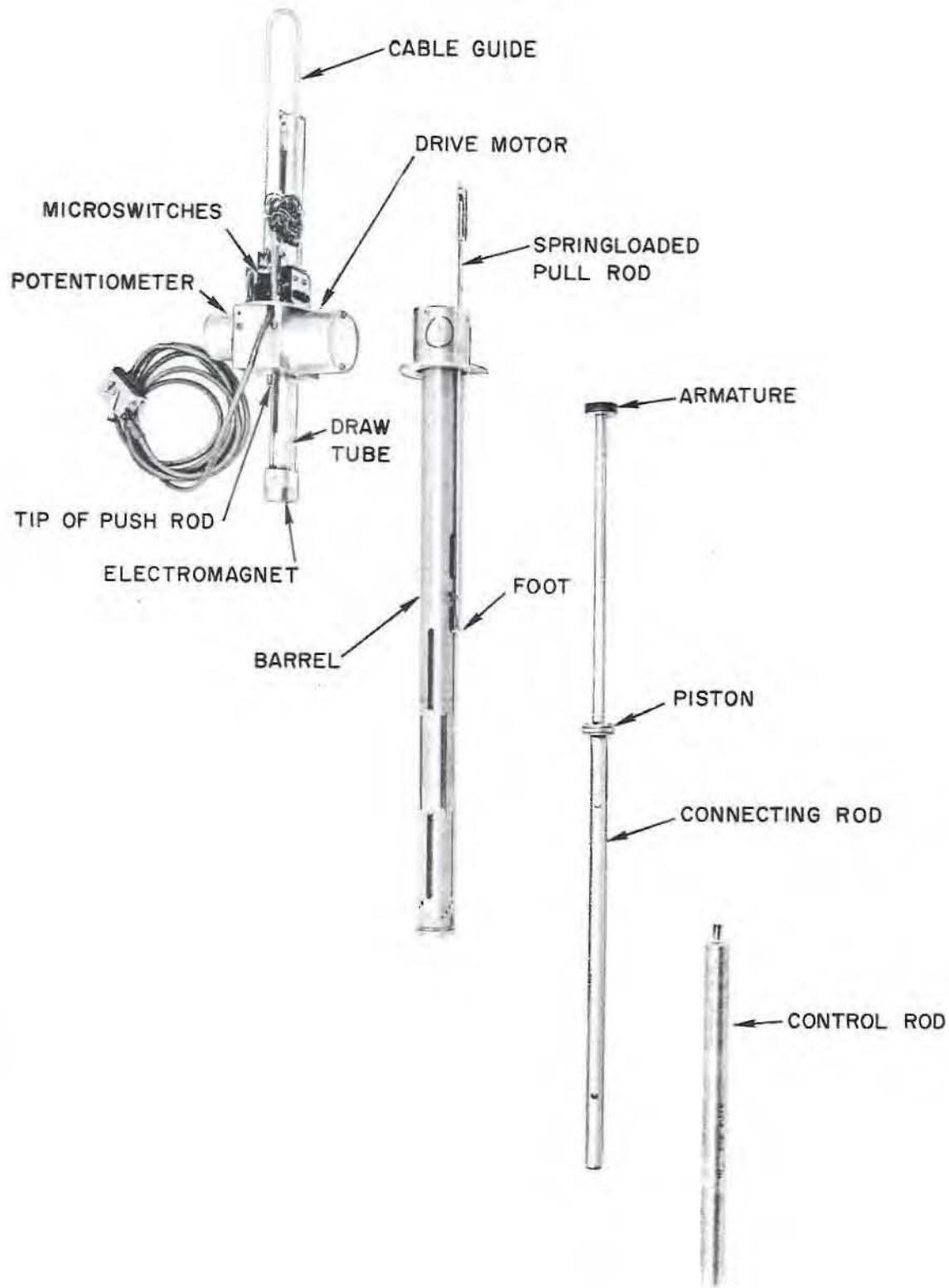


Fig. 5-6--Rack-and-pinion drive

piston that travels within the barrel assembly. Since the upper portion of the barrel is well ventilated by large slotted openings, the piston moves freely in this range; but when the piston is within 2 inches of the bottom of its travel, its movement is restrained by the dashpot action of the graded vents in the lower end of the barrel. This dashpot action reduces bottoming impact when rods are dropped by removal of magnet current during a scram.

Clockwise rotation of the motor shaft raises the draw tube assembly. When the electromagnet attached to the draw tube is energized, the armature and connecting rod rise with the draw tube, and the control rod is withdrawn from the reactor core. When the reactor is scrammed, the electromagnet is de-energized and the armature is released. The armature, connecting rod, and control rod drop by gravitational force to reinsert the neutron poison into the reactor core.

The drive motors for the shim and safety rods are nonsynchronous, single-phase, and electrically reversible; they will insert or withdraw the rods at a rate of approximately 19 inches per minute. Electrical dynamic and static braking on these motors are used to provide fast stops and to limit coasting or overtravel. The regulating rod drive motor is a variable-speed servomotor with a tachometer generator for rate feedback; it will insert and withdraw the rod at a maximum rate of about 24 inches per minute.

Limit switches mounted on each drive assembly stop the rod drive motor at the top and bottom of travel and provide switching for console indicator lights, which show:

1. When the magnet is in the UP position.
2. When the magnet (and thus the control rod) is in the DOWN position.
3. When the magnet is in contact with the control rod armature

5.4.2. Transient-Rod Drive

To allow transient operation, use is made of a pneumatic-electromechanical drive system to eject a predetermined amount of the

transient rod from the core. This drive system is shown in Figs. 5-7 and 5-8.

The pneumatic portion of the pneumatic-electromechanical drive referred to herein as the "transient" rod drive, is basically a single-acting pneumatic cylinder. A piston within the cylinder is attached to the transient rod by means of a connecting rod. The piston rod passes through an air seal at the lower end of the cylinder. Compressed air is admitted at the lower end of the cylinder to drive the piston upward. As the piston rises, the air being compressed above the piston is forced out through vents at the upper end of the cylinder. At the end of its stroke, the piston strikes the anvil of a shock absorber. The piston is thus decelerated at a controlled rate during its final inch of travel. This action minimizes rod vibration when the piston reaches its upper-limit stop.

An accumulator tank mounted on the bridge stores the compressed air that operates the pneumatic portion of the transient rod drive. A three-way solenoid valve, located in the piping between the accumulator tank and the cylinder, controls the air supplied to the pneumatic cylinder. De-energizing the solenoid valve interrupts the air supply and relieves the pressure in the cylinder so that the piston drops to its lower limit by gravity. With this operating feature, the transient rod is inserted in the core except when air is supplied to the cylinder.

The electromechanical portion of the transient rod drive consists of an electric motor, a ball-nut drive assembly, and the externally threaded air cylinder. During electromechanical operation of the transient rod, the threaded section of the air cylinder acts as a screw in the ball-nut drive assembly. These threads engage a series of balls contained in a ball-nut assembly in the drive housing. The ball-nut assembly is in turn connected through a worm-gear drive to an electric motor. The cylinder may be raised or lowered independently of the piston and control rod by means of the electric drive. Adjustment of the position of the cylinder controls the upper limit of piston travel, and hence controls the amount of reactivity inserted for a pulse.

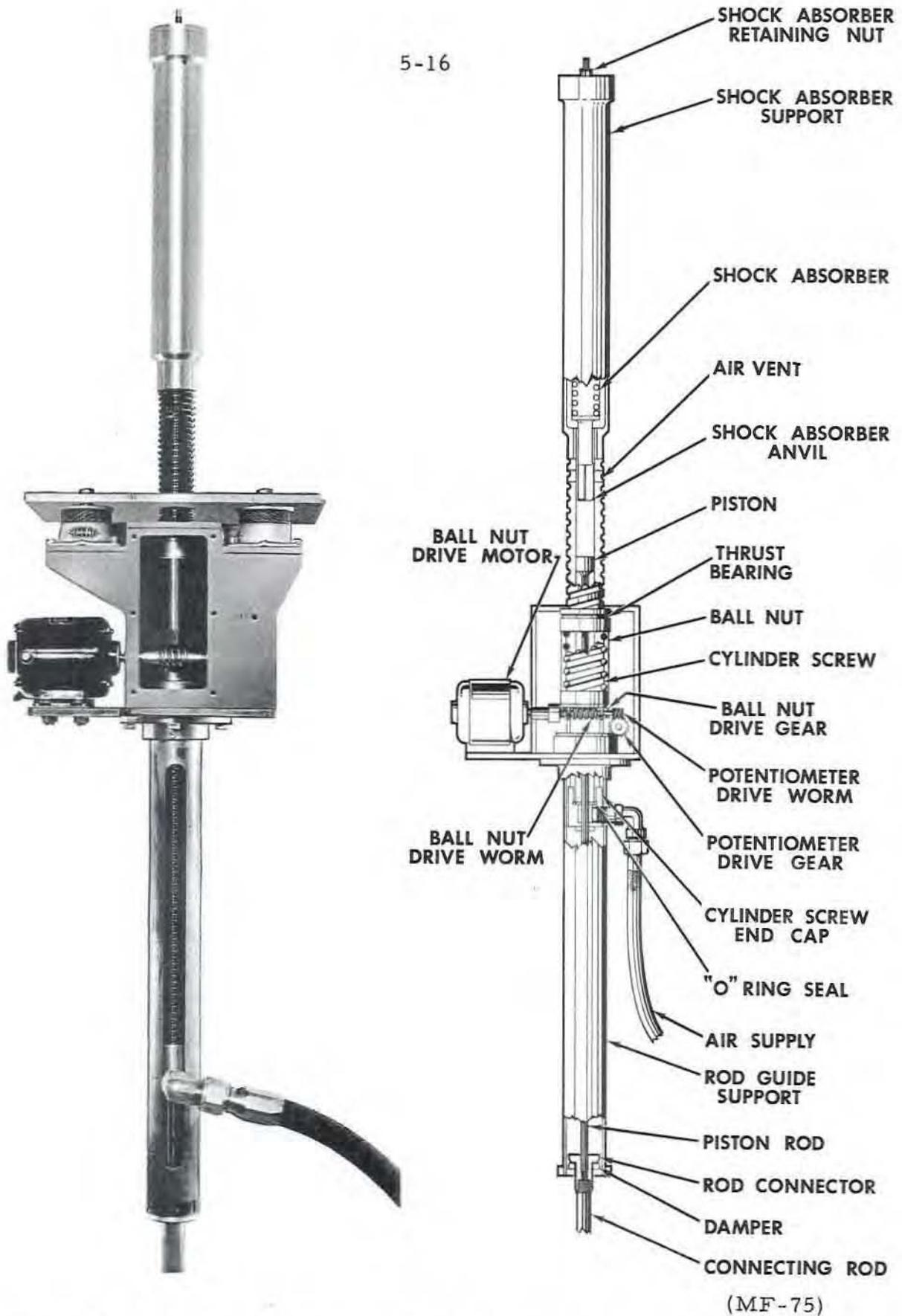


Fig. 5-7--Pneumatic-electromechanical transient-rod drive

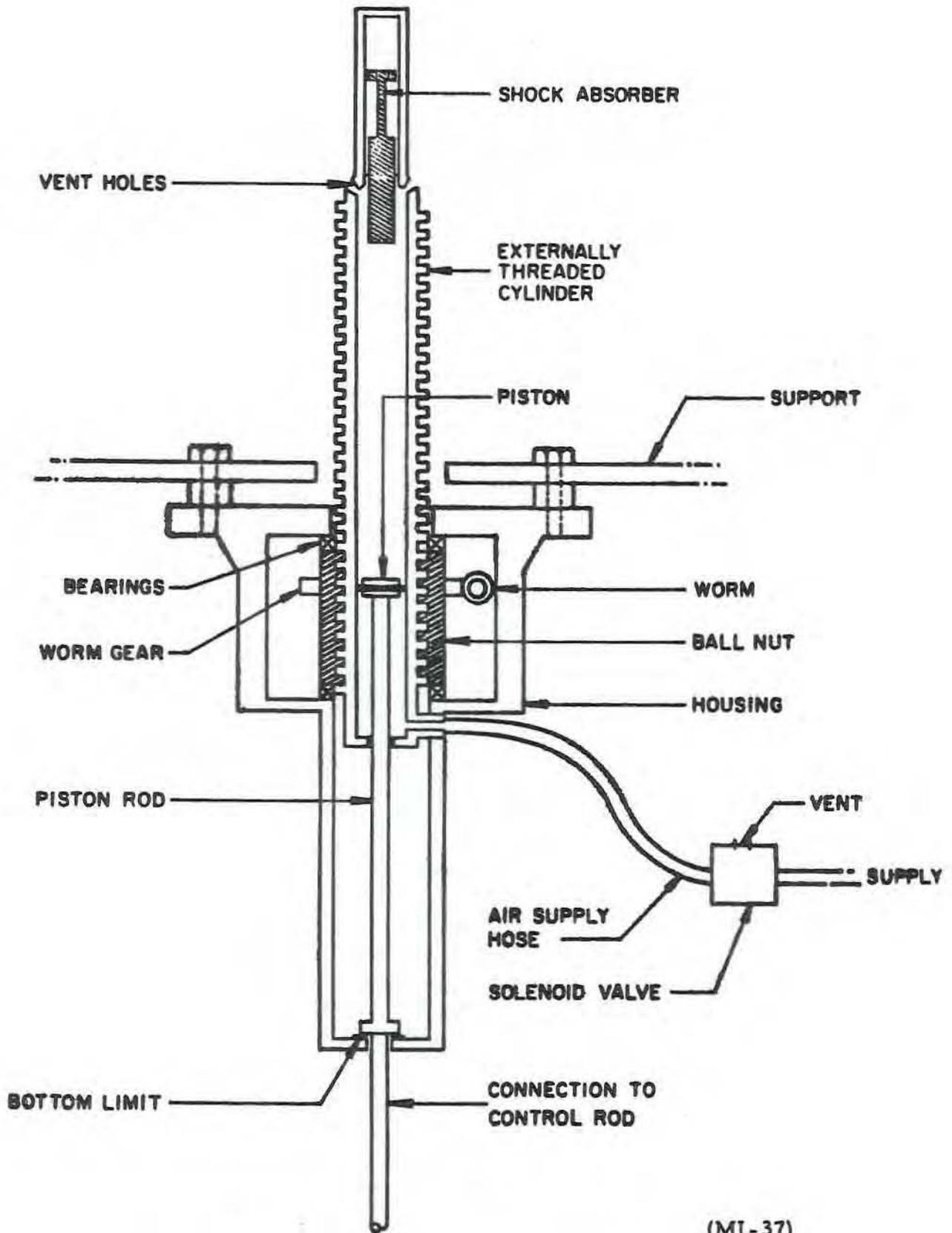


Fig. 5-8 -- Schematic drawing of transient-rod drive

A system of limit switches similar to that used with the standard control rod drives is used to indicate the position of the air cylinder and the transient rod. Two of these switches, the Drive Up and Drive Down switches, are actuated by a small bar attached to the bottom of the air cylinder. A third limit switch, the Rod Down switch, is actuated when the piston reaches its lower limit of travel.

5. 5. CONTROLS AND INSTRUMENTATION

5. 5. 1. General

The TRIGA Mark I reactor operates in three standard modes:

- | | |
|--------|--|
| Mode 1 | Steady-state operation at power levels up to 1000 kw (thermal) |
| Mode 2 | Square-wave operation (transient reactivity insertions to reach a desired steady-state power essentially instantaneously) at power levels between 300 and 1000 kw. |
| Mode 3 | Pulsed operation produced by rapid transient rod withdrawal that results in a step insertion of reactivity to peak powers of 1, 600, 000 kw. |

The reactor is operated from a console that displays all pertinent reactor operating conditions. The console and instrumentation system are largely transistorized.

The control system consists of five power-measuring channels utilizing three ion chambers and one fission counter as well as monitors for bulk-water temperature, fuel temperature, and cooling water inlet and outlet temperatures. Test circuits and calibration signals are provided for the log count-rate, log-n, and linear power-level channels. The rack-and-pinion control-rod drives and the pneumatic-electromechanical transient rod drive are controlled from the console. Manual scram is possible for the control rods individually or as a group.

A selector switch is provided for steady-state, pulsing, or square-wave modes of operation.

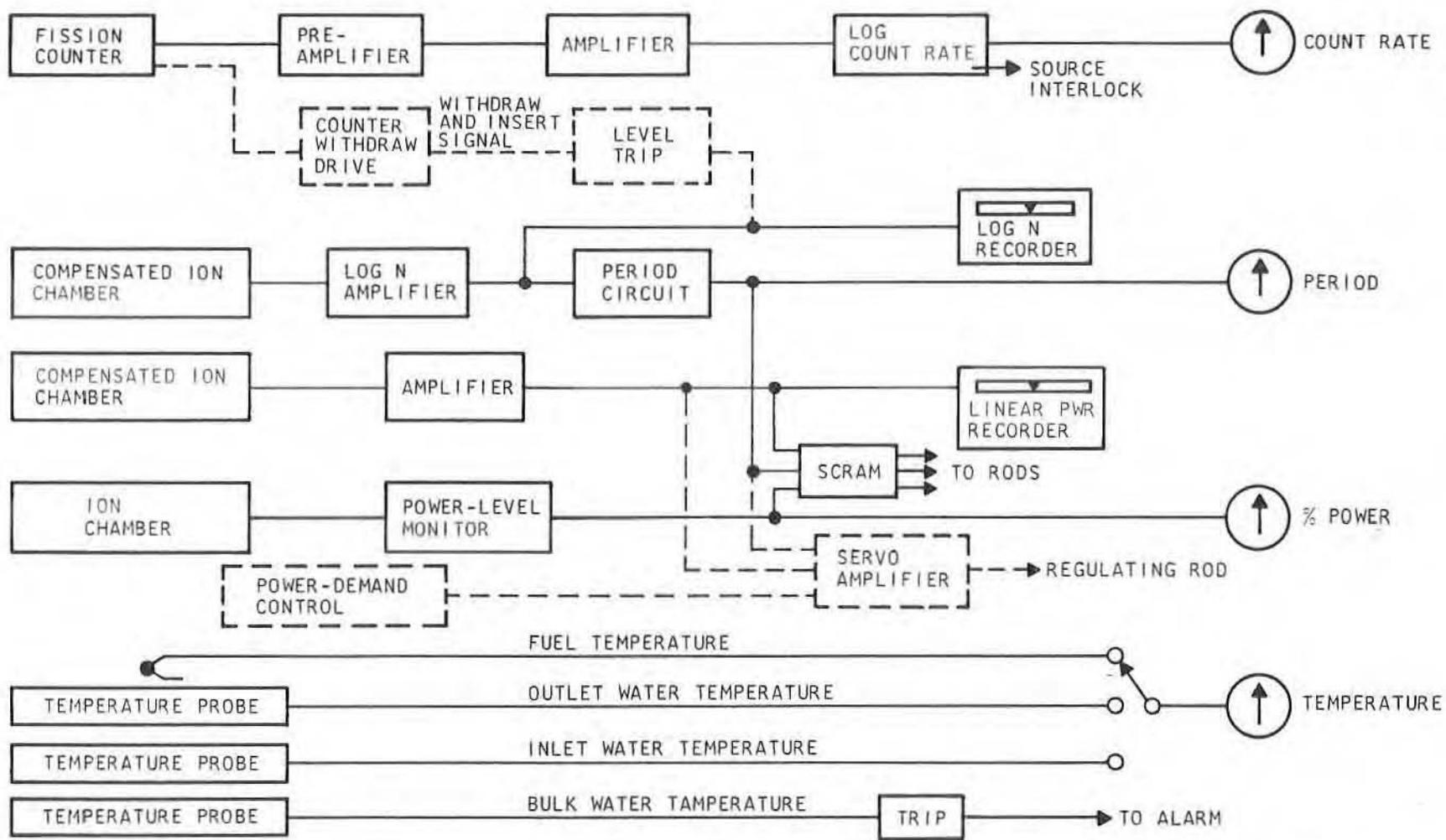
5.5.2. Steady-State Operation

For steady-state operation, the control rods are withdrawn slowly by manual control until the desired power level is reached. A servo may be used to maintain automatically the power constant at the desired level by movement of the regulating rod. Or alternatively, after the reactor is critical, a servo demand power may be selected (up to 1000 kw) and the regulating rod, supplemented automatically by the shim rod, will raise the power to the demand level on a relatively fast period. Once the demand power is achieved, it is maintained by the servo. A diagram of the TRIGA Mark I reactor control system is given for steady-state operation in Fig. 5-9.

The count-rate channel, utilizing a fission counter and transistorized log count-rate chassis, provides power indication over 5 decades from below source level. This channel is provided with a source interlock that prevents rod withdrawal unless source level is above a preset level. At the top of the log count-rate range, the fission counter is automatically withdrawn to prevent excessive burnup by receiving a signal from the log-n channel. Withdrawal and insertion of the fission counters are automatic.

A log-n channel using a compensated ion chamber covers the power range from less than 1 watt to above full power, and is read and recorded by one pen of the 11-inch, dual-channel recorder. A period circuit indicates reactor period from -30 to ∞ to +3 seconds with a scram level adjustable throughout the range from ∞ to +3 seconds.

A linear micromicroammeter channel provides power-level measurement from about 1 watt to full power, with a range switch having two ranges per decade so that measurement of the compensated ion chamber current may be made accurately. The output is read and recorded by the second pen of the dual-channel recorder. A linear-power scram at 110% of full scale is provided on all ranges.



5-20

(MIII-16A)

Fig. 5-9--Block diagram of reactor instrumentation for steady-state operation

A percent-power-level channel operating from an ion chamber indicates power in the range from a few percent to 110% of full power. This circuit provides for an adjustable-level scram within this range.

A bulk-water temperature monitor is equipped with an alarm that sounds if the temperature exceeds a preset value. It is also possible, by means of the selector switch and meter shown in the diagram, to read fuel temperature, and cooling water inlet and outlet temperatures. The unit that monitors bulk water temperature is in operational use in any operating mode.

5. 5. 3. Square-Wave Operation

For some irradiation applications, it is convenient to bring the power level rapidly to a high preset value, hold it there for a period of time, and then drop it rapidly back to a low level, thus producing a square wave of power.

In the square-wave mode of operation, the reactor is first brought to a power level of 1 to 1000 watts in the steady-state mode. The mode switch is then changed to square-wave operation, which will allow a step or rapid rise (within a few seconds) to preset power levels between 300 and 1000 kw. To accomplish this, a preadjusted step reactivity change is made with the transient rod. Then, a high-speed servo automatically inserts additional reactivity required to hold the power level constant at the preset value as the fuel heats up. In this mode the period meter and scram are disconnected, and the range switch on the linear power channel is set at the 1-Mw level. The linear power scram is thus retained at 1.1 Mw. An interlock prevents initiation of the square wave unless the range switch is on the 1-Mw setting.

5. 5. 4. Pulsing Operation

After a moderate power level of less than 1000 watts in the steady-state operating mode is reached, the mode switch is changed to the pulsing mode so that the reactor can be pulsed. When the switch is turned to the

pulsing mode, the normal neutron channels are disconnected and a high-level pulsing chamber is connected to read out the peak power of the pulse on the recorder several seconds after the pulse is completed. Also, changing the mode switch to pulsing removes an interlock that prevents application of air to the transient rod unless the transient rod cylinder is in the full "in" position and thus allows pulsing to take place. Furthermore, only the transient rod can be moved during pulsing mode operation. In this mode, the transient rod is reinserted after a preset time delay. In addition, fuel temperature is recorded during pulsing. These channels are indicated in Fig: 5-10.

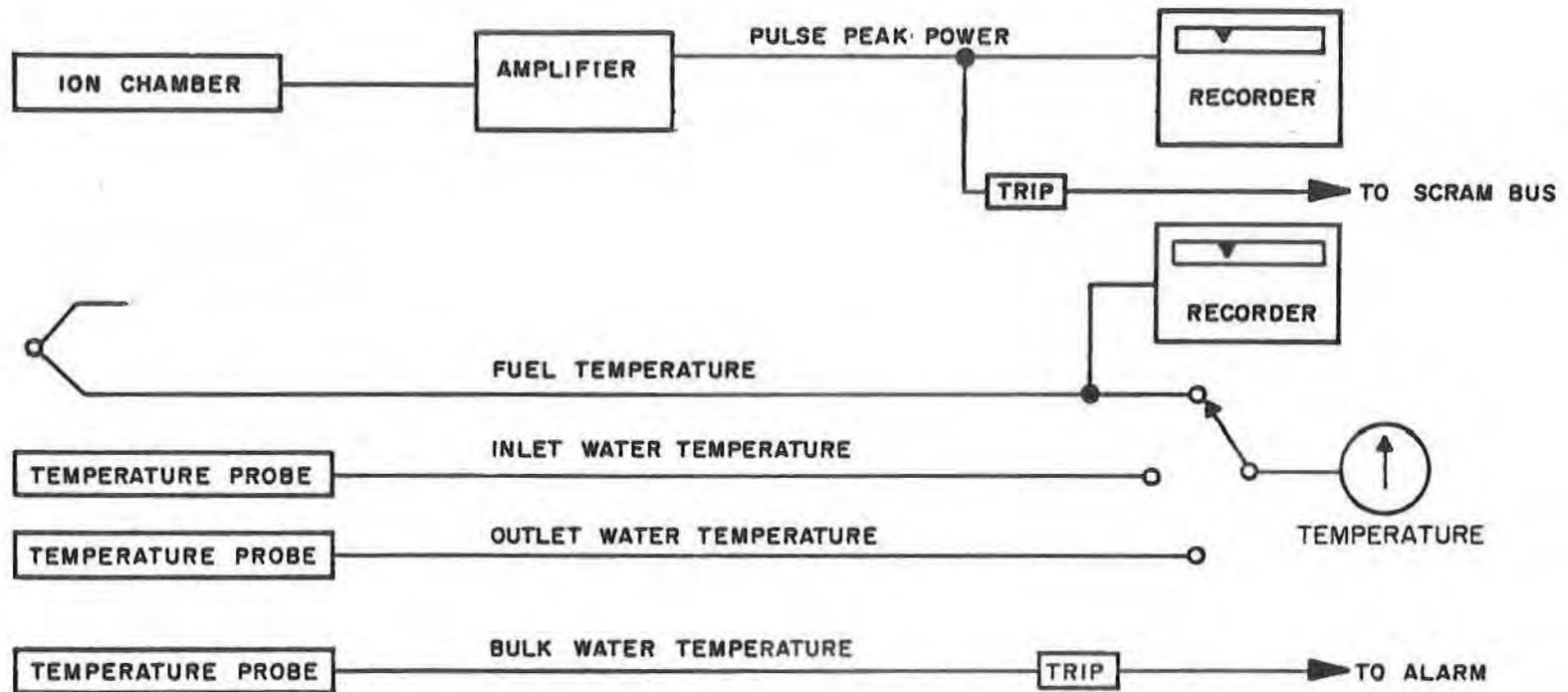
5.5.5 Rod Control

The three standard rod drives and the transient rod drive are controlled by switch-light pushbuttons, which not only control the up or down motion of the rods but also annunciate their extreme positions. In addition, transistorized position indicators with digital readout accurate to 0.1% are provided for all the standard rods and the transient rod. Following a scram, all rack-and-pinion-driven rods are reinserted automatically.

5.5.6 Safety Devices

Scrams:

1. Power level channel, compensated ion chamber, micromicroammeter.
2. Power level channel, ion chamber, adjustable by operator from 20 to 110% of full power.
3. Reactor period channel, adjustable period scram between ∞ and 3 seconds.



(M III - 17A)

Fig. 5-10--Block diagram of reactor instrumentation for pulsing operation

4. Manual.
5. Ion-chamber power supply failure.
6. Console power circuit failure.

Interlocks:

1. To assure minimum source strength before control rods can be withdrawn.
2. To prevent withdrawal of two control rods simultaneously on manual control.
3. To assure that pulse cannot be performed with reactor power above 1 kw.
4. To assure that range switch is on 1-Mw range before initiating the square wave or a pulse.
5. To prevent application of air to the transient rod in the steady-state mode unless the transient rod cylinder is fully inserted.
6. To prevent movement of any rod except the transient rod in pulsing mode.

5.6. NEUTRON SOURCE AND HOLDER

The neutron source used with the TRIGA Mark I reactor is a mixture of polonium and beryllium, and has a nominal strength of ■ curies to provide the necessary neutron level during reactor startup. The source is cylindrical in shape and is doubly encapsulated to assure leak tightness. The holder for the neutron source is a cylindrical aluminum container which fits into either of two diametrically opposed holes between the F- and G-rings of the top grid plate. A flange at the upper end of the container rests on the top grid plate and provides support and proper positioning of the source.

5.7. REFLECTOR

The reflector is a ring-shaped block of graphite that surrounds the core radially. It is 11-3/16 inches thick radially, with an inside diameter of 20-7/8 inches and a height of 28-7/8 inches. The graphite is protected from water penetration by a leak-tight welded aluminum can.

A "well" in the top of the graphite reflector is provided for the rotary specimen rack. This well is also aluminum-lined, the lining being an integral part of the aluminum reflector can. The rotary specimen rack is a self-contained unit and does not penetrate the sealed reflector at any point.

The reflector assembly rests on an aluminum platform at the bottom of the tank, and provides the support for the two grid plates and the safety plate. Four lugs are provided for lifting the assembly.

5.8. GRID PLATES AND SAFETY PLATE

The top grid plate is an aluminum plate 5/8 inches thick (3/8 inches thick in the central region) that provides accurate lateral positioning for the core components. The plate is supported by a ring welded to the top inside surface of the reflector container and is anodized to resist wear and corrosion.

One-hundred-twenty-two (122) holes, 1.505 inches in diameter, are drilled through the top grid plate in six circular bands to locate the fuel-moderator and graphite dummy elements, the control rods, and guide tubes, and the pneumatic transfer tube. (See Fig. 5-11). A 1.505-inch-diameter center hole accommodates the central thimble. Small holes at various positions in the top grid plate permit insertion of foils into the core to obtain flux data.

A hexagonal section can be removed from the center of the upper grid plate for the insertion of specimens up to 4.4 inches in diameter into the region of highest flux; this requires prior relocation of the six

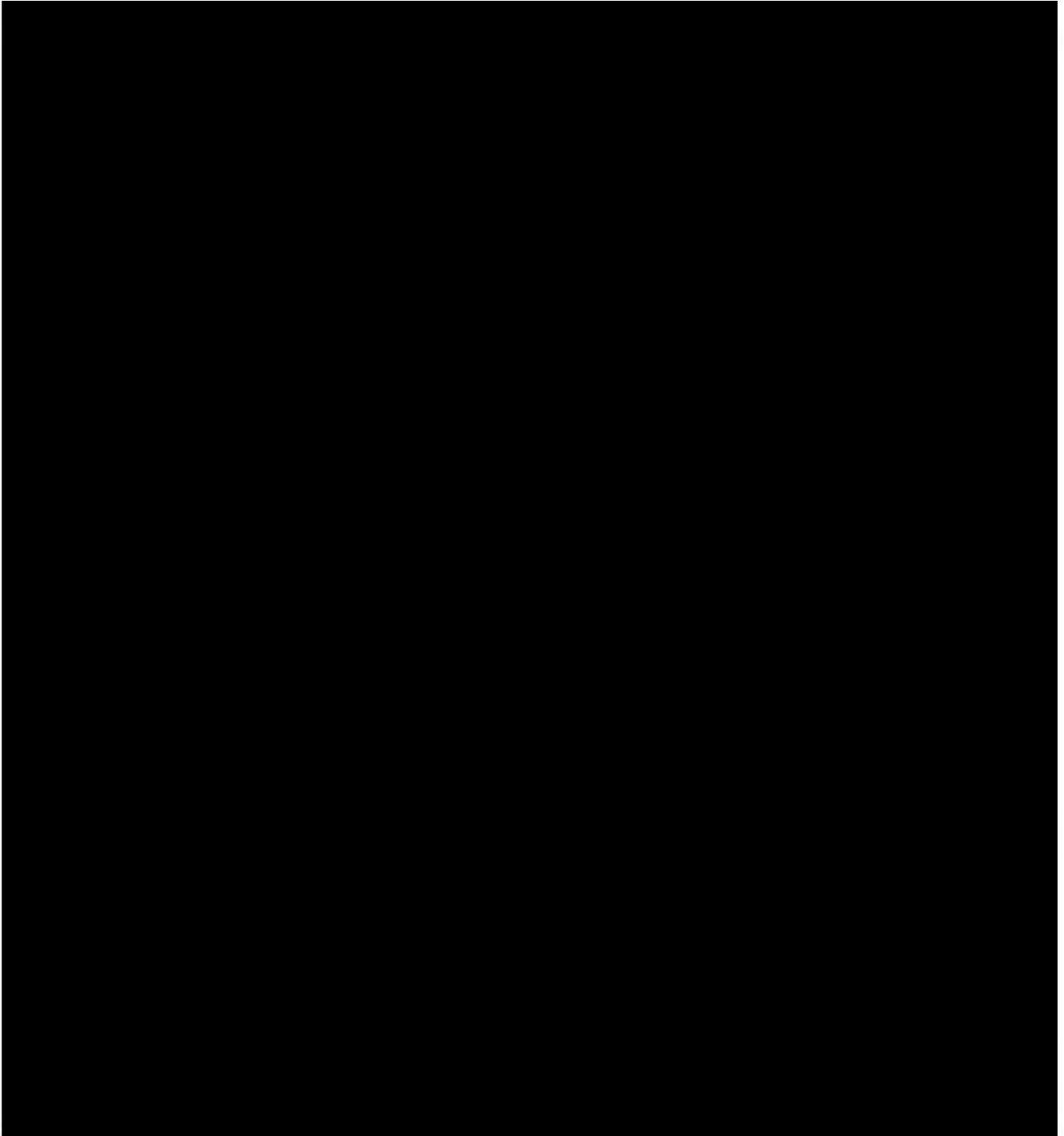


Fig. 5-11--Grid array

fuel elements from the B ring to the outer portion of the core and removal of the central thimble. This removable section will not be used initially; a separate license amendment will be obtained prior to its use.

Two generally triangular-shaped sections are cut out of the upper grid plate. Each encompasses two F and one E ring holes. When fuel elements are placed in these locations, their lateral support is provided by a special fixture. With the fuel element support removed, there is room for inserting specimens up to 2.4 inches in diameter.

Two 3/8-inch-diameter holes between the F and G rings of the grid plate locate and provide support for the source holder at alternate positions.

The differential area between the triangular-shaped spacer blocks at the top of the fuel element and the round holes in the top grid plate permits passage of cooling water through the plate.

The bottom grid plate is an aluminum plate 3/4-inch thick which supports the entire weight of the core and provides accurate spacing between the fuel-moderator elements. Six pads welded to a ring which is, in turn welded to the reflector container, support the bottom grid plate.

Holes in the bottom grid plate are aligned with fuel element holes in the top grid plate. They are countersunk to receive the adaptor end of the fuel-moderator elements, the adaptor end of the control rod guide tube, and the adaptor end of the pneumatic transfer tube.

A central hole 1.505 inches in diameter in the lower grid serves as a clearance hole for the central thimble. Eight additional 1.505-inch-diameter holes are aligned with upper grid plate holes to provide passage of fuel-follower control rods. Those holes in the bottom grid plate not occupied by control rod followers are plugged with removable fuel element adaptors that rest on the safety plate. These fuel element adaptors are solid aluminum cylinders 1.5 inches in diameter by 17 inches long. At the lower end is a fitting that is accommodated by a hole in the safety

plate. The upper end of the cylinder is flush with the upper surface of the bottom grid plate when the adaptor is in place. This end of the adaptor has a hole similar to that in the bottom grid plate for accepting the fuel element lower end fitting. With the adaptor in place, a position formerly occupied by a control rod with a fuel follower will now accept a standard fuel element. The adaptor can be removed with a special handling tool.

The safety plate is provided to preclude the possibility of control rods falling out of the core. It is a 1/2-inch-thick plate of aluminum welded to the extension of the inner reflector liner about 16 inches below the bottom grid plate.

5.9. NEUTRON SOURCE AND HOLDER

A curie polonium-beryllium neutron source will be used for startup. It will be located in a cylindrical aluminum container which fits into either of two holes located between the F and G rings of the top grid plate. A flange at the upper end of the container rests on the top grid plates and provides support and proper positioning of the source.

5.10. REACTOR BRIDGE

The four control rod drives, the fission chamber drive, and the loading and drive mechanisms for the isotope production facility are supported by a bridge that spans the reactor pool. The steel bridge consists of 2 heavy channel beams with a 1-inch steel plate between them and is designed to support the weight of a 3,000-pound cask. The bridge is approximately 18 inches wide by 10 feet long.

The remaining portion of the reactor pool is covered with hinged aluminum grates. Clear plastic sheets are attached to the underside of the grates to give clear visual access to the core and to prevent material from falling into the pool. The grates can support the weight of people and apparatus and yet can be raised to give access to the core.

5. 11. REACTOR POOL

The reactor pool is approximately 8 ft in diameter and 24 ft, 10 inches deep. The entire pool is lined with a welded aluminum structure embedded in reinforced-concrete and surrounded by earth.

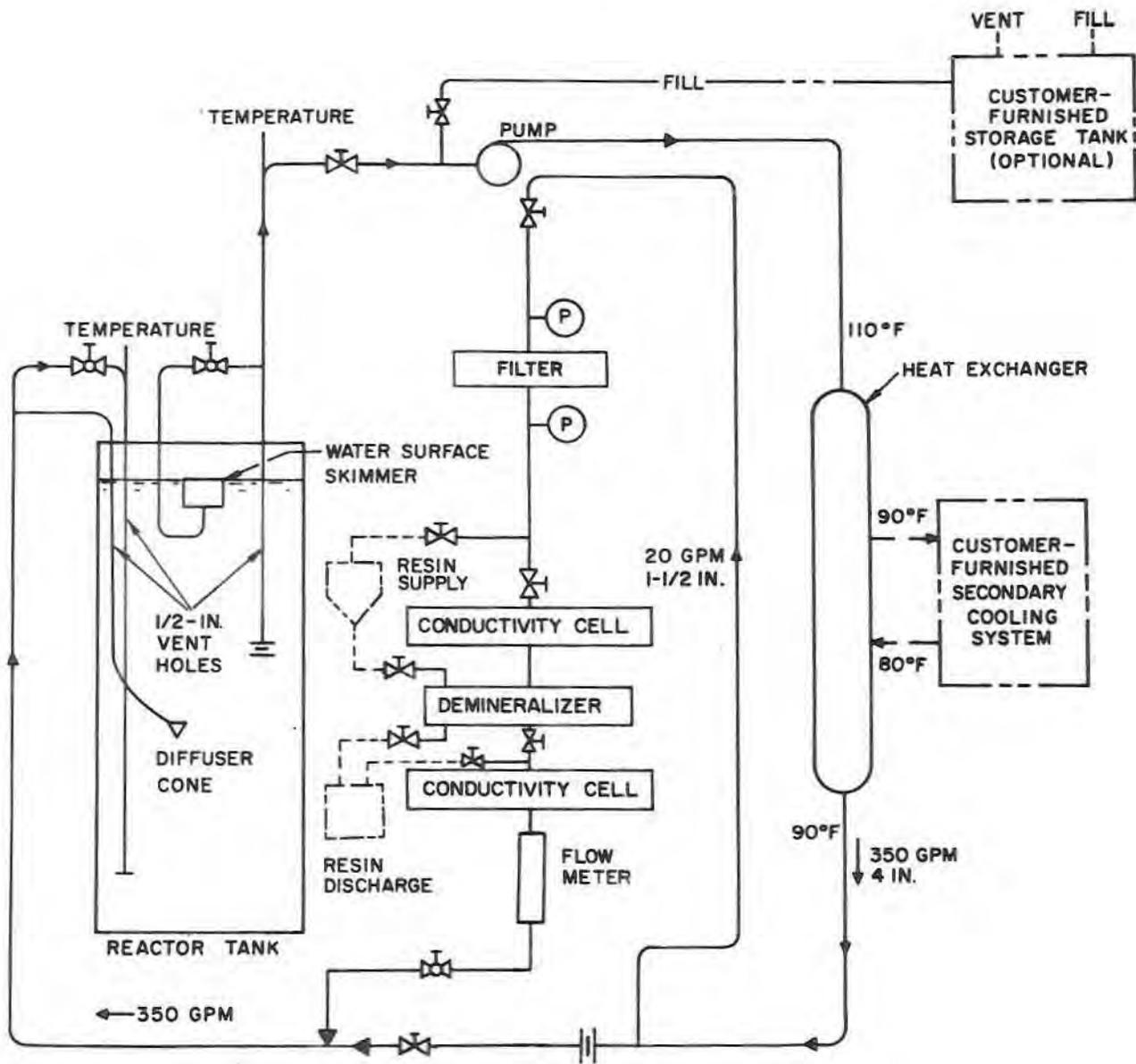
5. 12. WATER COOLING AND PURIFICATION SYSTEM

The water system provided with the TRIGA Mark I reactor serves four functions:

1. Maintains low conductivity of the water to minimize corrosion of reactor components, particularly the fuel elements.
2. Reduces radioactivity in the water by removing nearly all particulate and soluble impurities.
3. Maintains optical clarity of the water.
4. Provides a means of dissipating the heat generated in the reactor.

The water system pump discharges some of the circulating water through a diffuser nozzle aimed down and across the rising convective current of core cooling water. This arrangement creates a thorough diffusion of the Nitrogen 16 into a greater volume of water and permits further decay of the N-16 within the pool by increasing its transport time from the reactor core to the water surface. The remainder of the cooling water is discharged close to the bottom of the tank near the core to improve circulation.

The system consists principally of a pump, fiber cartridge filter, mixed-bed-type demineralizer, flow meter, and 1000-kw shell-and-tube heat exchanger. System components are connected primarily by either 1-1/2- or 4-inch-diameter aluminum piping, as indicated on the schematic diagram shown in Fig. 5-12.



- LEGEND**
-  THROTTLING VALVE
 -  SHUT-OFF VALVE
 -  ORIFICE
 -  PRESSURE GAGE

M III-8A-USGS

Fig. 5-12--Water system schematic

6. IRRADIATION FACILITIES

6.1. IN-POOL AND IN-CORE IRRADIATIONS

Specimens may be exposed by placing them in the reactor pool adjacent to the core by using suitable watertight containers. If very-high-level irradiations are required, specimens can be inserted directly into the highest flux region of the core in a central thimble, or by relocating the six B-ring fuel elements to outer grid positions and removing the central thimble and the 4.4-inch-diameter hexagonal section in the top grid plate. Two other facilities, which can accept samples up to 2.4 inches in diameter, are available by removing three fuel elements, one from the D-ring and two from the E-ring. Or smaller-diameter (less than 1-1/2 inch) specimens can be inserted in special tubes or containers by removing a fuel element from any desired grid location.

6.2. ISOTOPE-PRODUCTION FACILITY

The isotope-production facility permits dry irradiation of specimens and production of isotopes immediately external to the core. The facility consists primarily of five components: (1) the watertight rotary specimen rack assembly, which surrounds the core; (2) the specimen-removal-tube; (3) the tube-and-shaft assembly; (4) the drive-and-indicator assembly on the reactor bridge; and (5) the specimen-lifting assembly, which is used for the insertion and removal of specimen containers. Loading and unloading of the isotope-production facility can be done while the reactor is operating.

The rotary specimen rack assembly consists of an aluminum rack that holds specimens during irradiation, and an outer ring-shaped

seal-welded aluminum housing. The entire assembly rests in a well in the graphite reflector. The rack, which can be rotated inside the housing, supports 40 evenly-spaced aluminum tubes, which are open at the top and closed at the bottom. The tubes serve as receptacles for 1.12-inch-OD specimen containers. The rack can be rotated manually or electrically, and the orientation is controlled from the drive on the bridge. Motion is transmitted through a drive shaft to a stainless-steel-sprocket and chain drive in the rotary-specimen-rack housing.

Specimen containers are inserted into and removed from the 40 specimen tubes through the specimen-removal tube, which has an internal diameter of 1.33 inches. The specimen-removal tube extends from the rotary-specimen-rack housing to the reactor bridge, and is offset by means of large-radii tube bends to avoid direct-line radiation streaming from the core.

The tube-and-shaft assembly is a sealed straight tube that encloses the drive shaft and connects the rotary-specimen-rack housing with the drive-and-indicator assembly on the bridge. Since this tube is in a straight line from the reactor, shielding is provided by plastic enclosed within the tubing.

The drive-and-indicator assembly is located on the bridge. It includes an indicator dial with 40 divisions (one for each rack position), a crank for manual rotation of the specimen rack, a motor and slip-clutch for electric operation, and a locking handle. To rotate the rotary specimen rack inside its housing, the locking handle must be lifted and the crank rotated. When the indicator pointer is properly positioned at the desired specimen rack position, the locking handle can be lowered into a corresponding positioning hole. This locks the gear train in the drive box on the bridge for loading or unloading of specimens. After the rack is loaded, it can be rotated electrically to provide uniform exposure of specimens.

A fishing-pole-type apparatus is used for inserting specimen containers in, or removing them from the rotary specimen rack. This pole enables the operator to keep isotopes at a safe distance and provides maximum flexibility during handling. An electric cable attached to the reel serves as a hoisting cable for the specimen container and a power conductor for actuating the specimen pickup tool. This pickup assembly is a small, solenoid-operated, scissors-like device that fits into the upper end of the specimen container. The pickup solenoid is actuated from a button on the fishing pole.

The irradiation specimen container is cylindrical in shape and consists of a body and a cap. The cap is designed to accommodate the specimen pickup assembly. The inside length of the container is 3.81 inches, and the inside diameter is 0.81 inch. The over-all length of 5.36 inches permits free passage of the container through the curved portion of the removal tube. The usable space in each container is about 2 cubic inches. Since two containers may be inserted in each rotary rack position, the total capacity is 82 specimen containers, or about 164 cubic inches. The polystyrene container is suitable only for experiments of short duration. For longer-term experiments, an aluminum container is recommended.

6.3. PNEUMATIC TRANSFER SYSTEM

Short-lived radioisotopes are produced in a pneumatic transfer system, which rapidly conveys a specimen to and from a position in the outer ring of the reactor core.

The system includes a specimen capsule (rabbit), a blower-and-filter assembly, a valve assembly, a core terminus assembly, a transfer box, up to 3 receiver/sender assemblies, timer and control assemblies, and such items as tubing, flexible hoses, and fittings. The polyethylene capsule provides a space for irradiating specimens 0.56 inch in diameter and about 4 inches long.

Tubing from the blower extends both to the terminus in the core and to the receiver/sender unit in the laboratory. Injection and ejection of the specimen capsule are by means of a vacuum maintained by the blower. The terminus in the reactor core is located in one of the outer ring holes of the grid plate and supported, as are the fuel elements, by the bottom grid. When the specimen capsule is injected into the core terminus, it comes to rest in a vertical position approximately at the midplane of the core.

The system is controlled from any one of its timer and control assemblies, which are located by the receiver/sender assemblies in the laboratory and may be operated either manually or automatically. With automatic control, the specimen capsule is ejected from the core after a predetermined length of time. Four solenoid-operated valves control the airflow. Since the system is always under a negative pressure, any leak will be into the system. All the air from the pneumatic system is drawn through a filter before it is discharged into the building exhaust system for venting radioactive argon. One or more of these systems may be installed in the core.

6.4. CENTRAL THIMBLE

The central thimble located in the center of the core, provides space for the irradiation of small samples at the point of maximum flux. It also makes possible the extraction of a highly collimated beam of neutron and gamma radiation.

The thimble is an aluminum tube 1-1/2 inches in outside diameter with a wall thickness of 0.083 inch. It extends from the bridge straight down through the central hole of the removable hexagonal section in the top grid plate and through the lower grid plate, terminating with a plug in the lower end. The central thimble is supported at its lower end by the safety plate situated about 16 inches beneath the lower grid plate.

Removable rings, located on the tube just above and below the removable hexagonal grid section, support the section and ensure its proper vertical placement in the top grid plate.

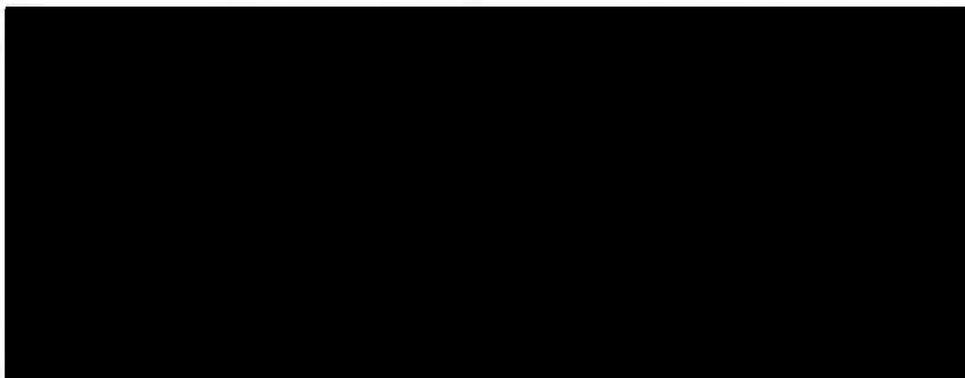
The shield water in the thimble is removed to make the beam available for experimentation by attaching a special blowout cap to the top of the tube and forcing the water out with air pressure. A hole located in the plug at the bottom of the tube permits the water, under air pressure, to escape. The top of the tube is normally covered with a cap containing a simple relief valve. This relief valve, together with the hole in the end plug, ensures that when not in use the central thimble is filled with shielding water.

7. PROTOTYPE PERFORMANCE CHARACTERISTICS AND REACTOR PARAMETERS

7.1. INTRODUCTION

An experimental program was carried out with the TRIGA Mark III prototype reactor at San Diego to establish the performance characteristics and to demonstrate the safety features of the 1000 kw TRIGA reactor. The prototype reactor was loaded to criticality with stainless-steel-clad high-hydride fuel-moderator elements for the first time in December of 1961. Since that time the core has been operated for thousands of megawatt hours and has undergone over 5000 transients at powers as high as 8500 Mw. Steady-state operation has been at powers up to 1.5 Mw. Specifications for the fuel in the prototype reactor are given in Table 7.1.

Table 7.1
TRIGA MARK III PROTOTYPE REACTOR FUEL-
MODERATOR SPECIFICATIONS



U ²³⁵ enrichment (%)	20
Hydrogen-to-zirconium ratio	1.68 (avg)
Cladding material	304 stainless steel
Cladding thickness, (mils)	20

The Mark I fuel elements are identical to those for the prototype core as listed in Table 7.1, except that the U^{235} has been raised to 8.5 weight-percent, and, consequently, the weight of the U^{235} in each element is about █ grams.

7.2. FUEL ELEMENT WORTHS

The worth of the fuel elements is dependent on the position they occupy in the core. Measurements indicating the worths of fuel elements in the six ring positions of the TRIGA Mark III prototype reactor are given in Table 7.2.

Table 7.2
FUEL ELEMENT WORTH COMPARED TO WATER
FOR EACH CORE RING

Ring	Maximum Number of Positions in Ring	Worth, (\$)	Worth, (% $\delta k/k$)
B	6	1.13	0.79
C	12	0.92	0.64
D	15	0.62	0.41
E	24	0.45	0.31
F	30	0.29	0.20
G	36	0.22	0.15

7.3. STEADY-STATE PARAMETERS

Various steady-state operating parameters have been measured and analyzed for the 91-element core loading shown in Fig. 7-1. All fuel positions in the B- through F-rings were occupied, and four elements were in the G-ring.

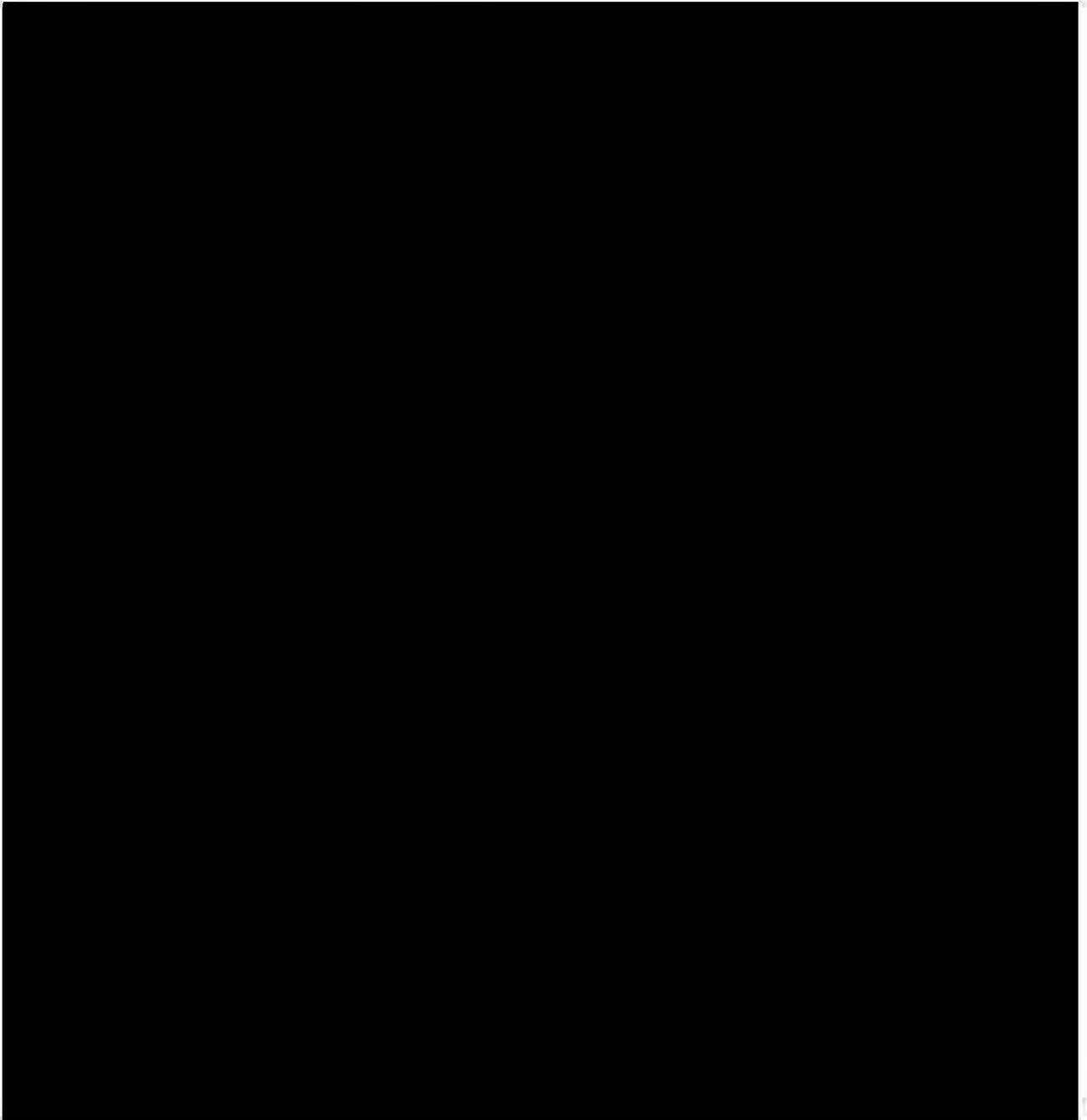


Fig. 7-1 -- Fuel loading diagram - prototype reactor

7.3.1. Critical Loading

The TRIGA Mark III prototype core was loaded to criticality in its initial operating condition, i. e. , with a poisoned follower on the transient rod and aluminum followers on other control rods. To achieve criticality, 79 fuel-moderator elements comprising a total of [REDACTED] kilograms of U^{235} were required.

7.3.2. Power Coefficient

Measurements of the "power coefficient" (loss of reactivity at various power levels) have been made at several intervals during the pulse history of the core (2,500 pulses to date). Some change in values occurs with pulsing. The loss in reactivity at 1 Mw was measured to be ~\$2.80 for the new core with less than 10 pulses; this value increased somewhat to a value of ~\$3.25 at 1 Mw after 2,400 pulses of 1800 Mw amplitude. The changes observed are due to a slight loosening of the clad due to thermal expansion of the fuel meats in going from 20°C pre-pulse temperature to ~400°C after-pulse temperature. The loosened cladding reduces thermal conductivity and results in a higher fuel temperature. The change in reactivity loss at various powers is most rapid during the first few pulses. After approximately 100 pulses, the reactivity reached a value nearly as large as that value quoted for operation after 2400 pulses. Extrapolation of the reactivity loss at 1 Mw versus number of pulses indicates that only after a very large number of pulses (>8,000) will the reactivity loss approach \$4.00.

Figure 7-2 shows the reactivity loss versus reactor power for the prototype Mark III core after 2,400 pulses.

7.3.3. Fuel Temperatures

A series of measurements has been made of the fuel temperature at various positions within the core at powers up to 1 Mw. Observed temperatures are highest at the center of the core; therefore, the maximum fuel temperature is measured in the innermost or B-ring fuel positions.

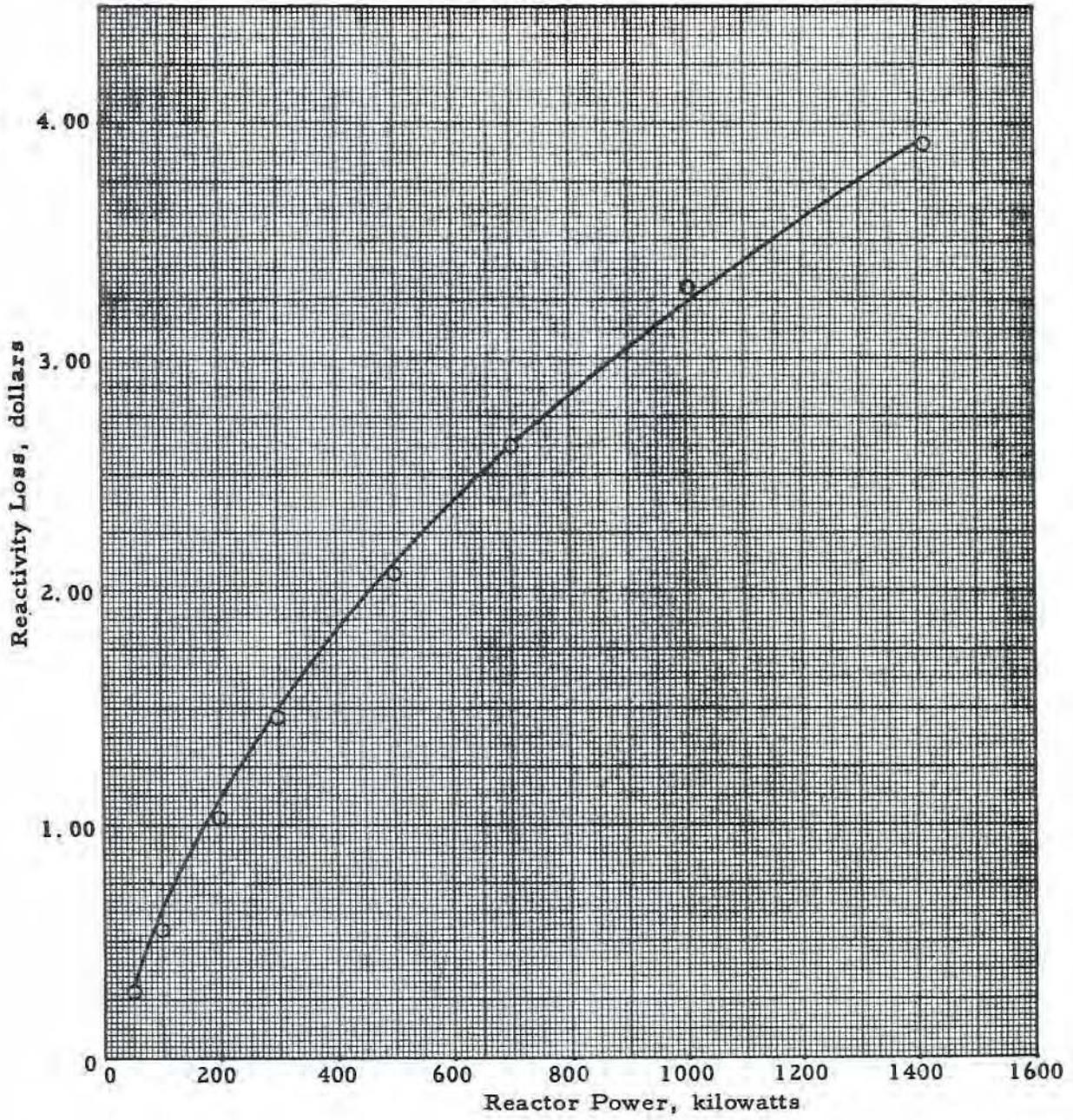


Fig. 7-2--Reactivity loss versus reactor power--prototype reactor

Figure 7-3 shows the temperature measured in the B-ring by means of a thermocouple-equipped element for powers up to ~ 1.5 Mw. The maximum fuel temperature at 1 Mw is $\sim 350^{\circ}\text{C}$ above pool ambient ($\sim 20^{\circ}\text{C}$) and is very much below the temperature limits for the $\text{ZrH}_{1.7}$ fuel material.

7.3.4. Isothermal Temperature Coefficient (Bath Coefficient)

Results of measurements of the effect of varying the coolant water temperature over a large range indicates that the resulting reactivity changes are small. Figure 7-4 is a plot of the change in available excess reactivity (relative to excess at 20°C) for bath temperatures from 20° to 60°C . Reactor power was less than 10 watts to achieve isothermal conditions. The coefficient is slightly positive with a net gain in available reactivity of ~ 11 cents at 60°C relative to the 20°C excess reactivity. The average coefficient (0.27 cents/ $^{\circ}\text{C}$) is small enough to be considered essentially negligible for normal operating conditions.

7.4. PULSE PARAMETERS

Extensive measurements have been made of the various parameters relating to the pulsing operation of the reactor. The most important of these are given below for step insertions of reactivity up to $2.1\% \delta k/k$ ($\$3.00$).

7.4.1. Period

During pulsing operation the reactor is placed in a super-prompt-critical condition. The asymptotic period is inversely related to the prompt reactivity insertion. Figure 7-5 shows the results of plotting the reciprocal of the measured period versus the prompt reactivity insertion. Since the period data must be obtained from an oscillographic recording of the reactor power versus time at a portion of the pulse before fuel temperature limiting effects have begun, the accuracy of the measurements is not as good as for other parameters. The scatter of points about a straight line in Fig. 7-5 is due entirely to this difficulty. As can be seen, the minimum period obtained for reactivity insertions of $\$3.00$ is ~ 3 msec.

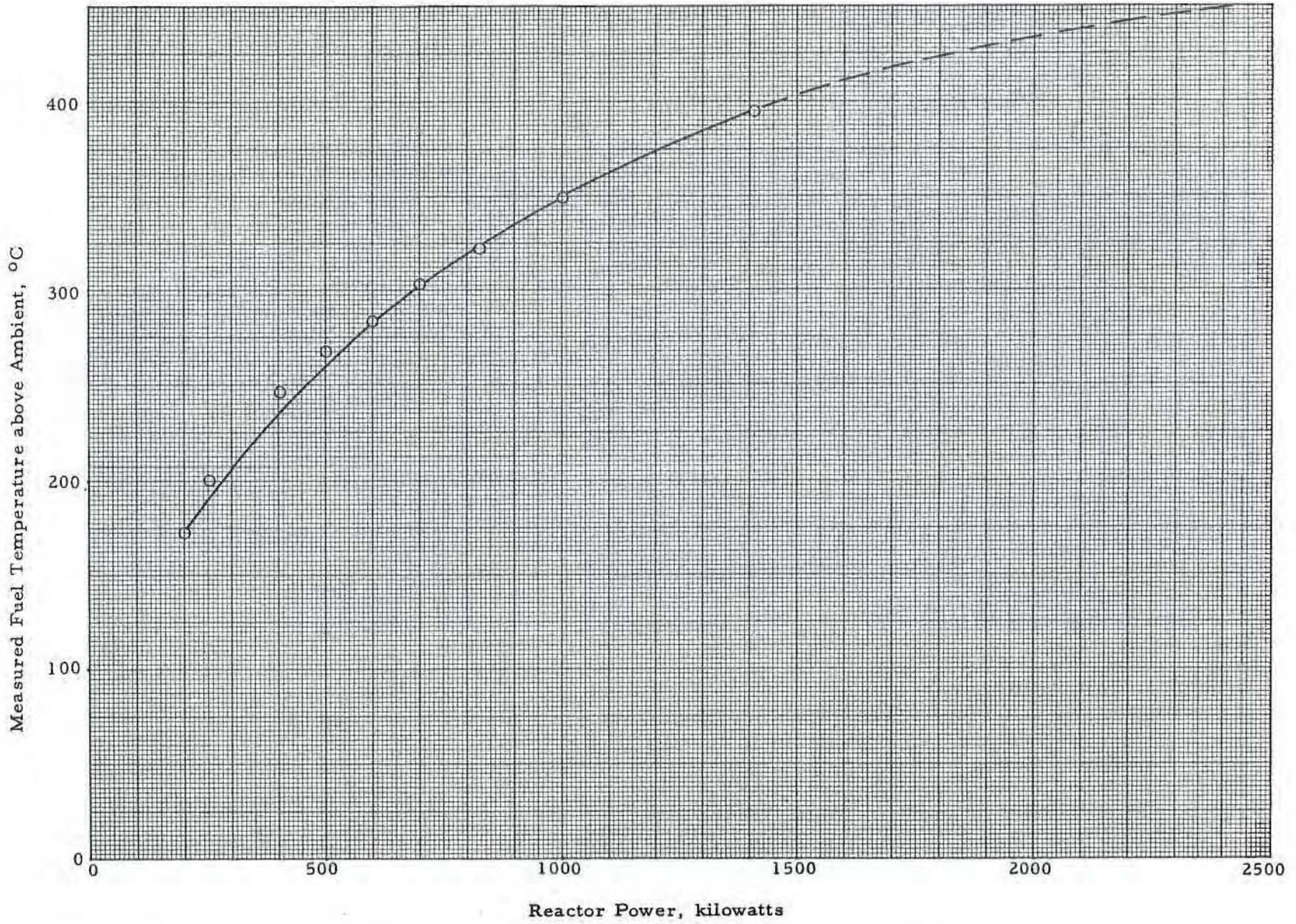


Fig. 7-3 -- Measured fuel temperature above ambient versus reactor power - prototype reactor

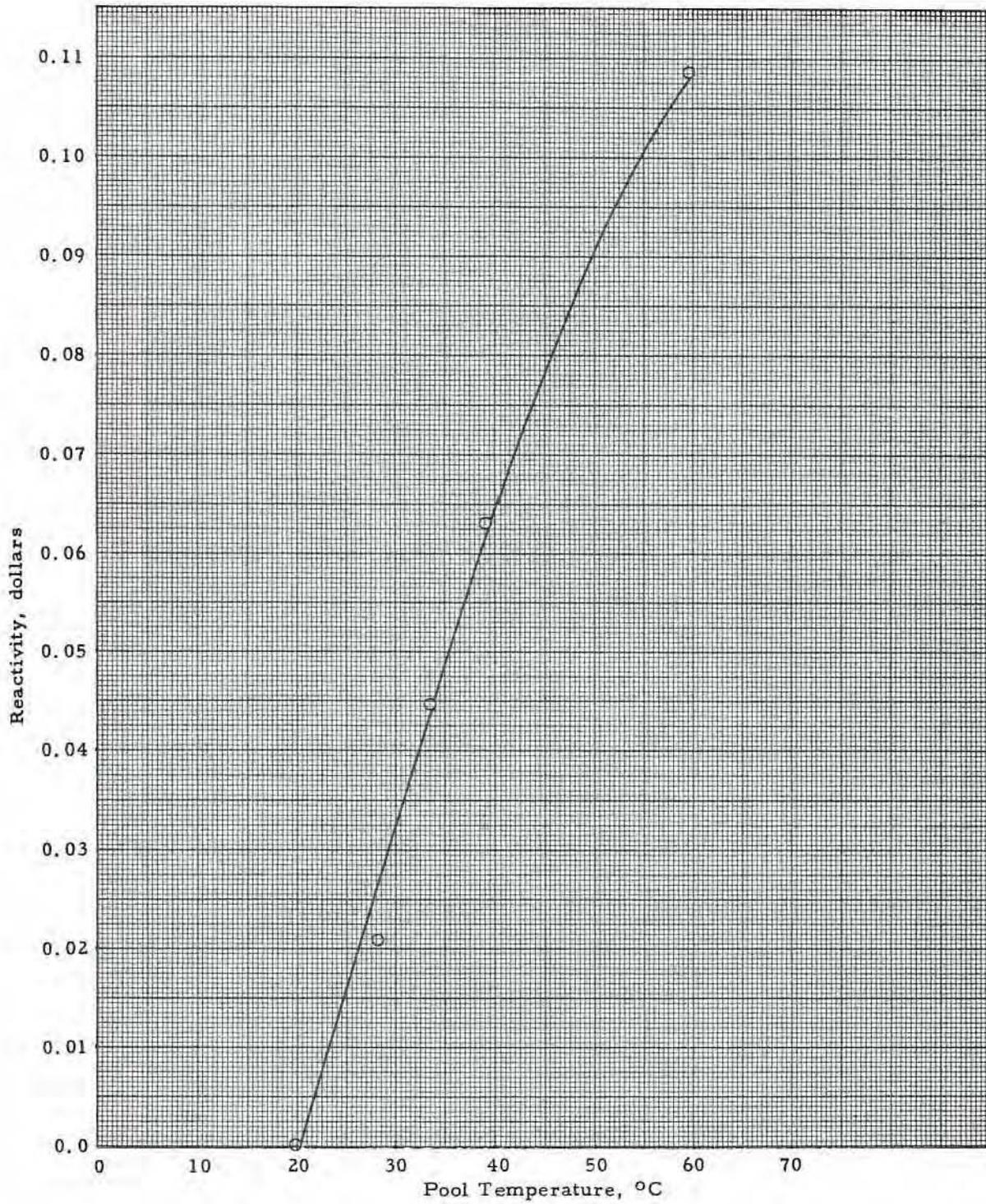


Fig. 7-4 -- Reactivity versus pool temperature - prototype reactor

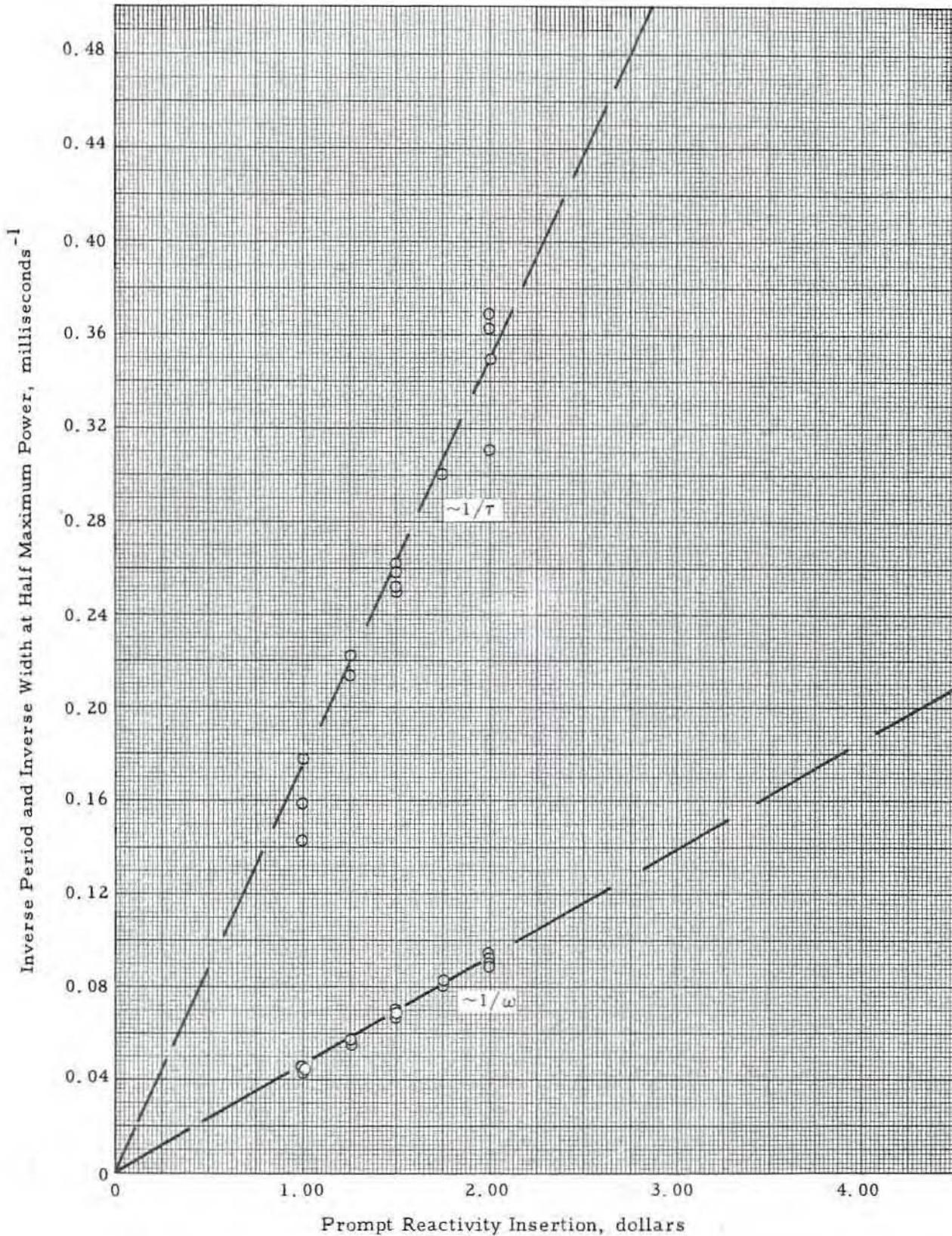


Fig. 7-5 -- Inverse period and inverse width at half maximum power versus prompt reactivity insertion - prototype reactor

7.4.2. Pulse Width

The width of the power pulse is most conveniently described as the time interval between half-power points. Also shown in Fig. 7-5 is a plot of the reciprocal of the measured width versus prompt reactivity insertion, and Fig. 7-6 shows the linear relationship between peak power and $(1/\text{width})^2$.

7.4.3. Peak Power

Figures 7-7, 7-8, and 7-9, show the interrelationship between maximum transient power, pulse widths, and period. When considered together, these plots serve to describe the general features of the Mark III core performance in pulsing modes. For a given core configuration, the peak power, integral power in the prompt burst, and width of the pulse, are determined by the reactivity insertion made. It can be seen from the plots that the peak power is controllable over a rather wide range since this parameter is very nearly proportional to $(\delta k/k - \$1.00)^2$. Pulse width and integral powers, on the other hand, are approximately linear functions of reactivity insertions above prompt critical so that their range is more limited.

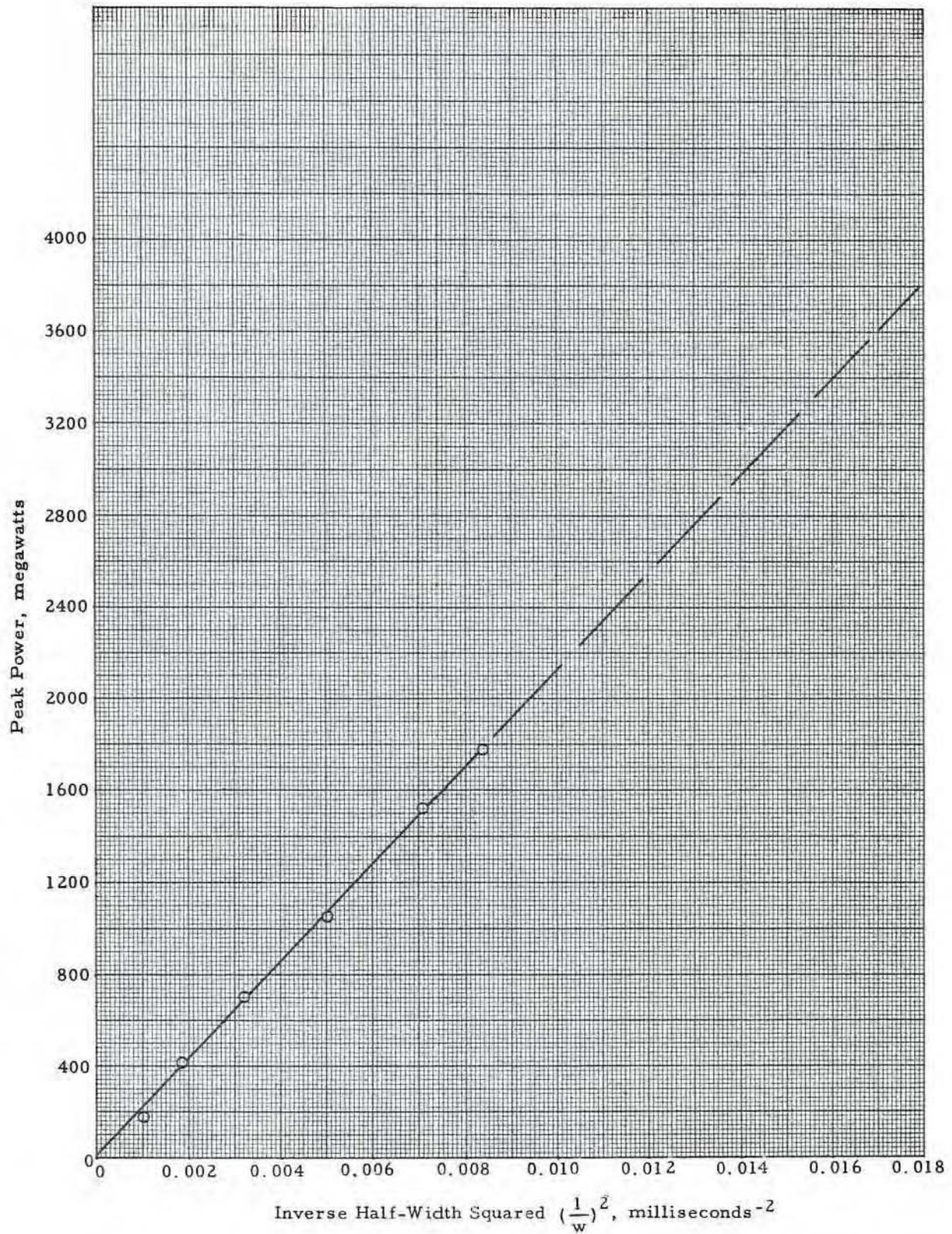


Fig. 7-6 -- Peak power versus inverse half-width squared - prototype reactor (100-element core)

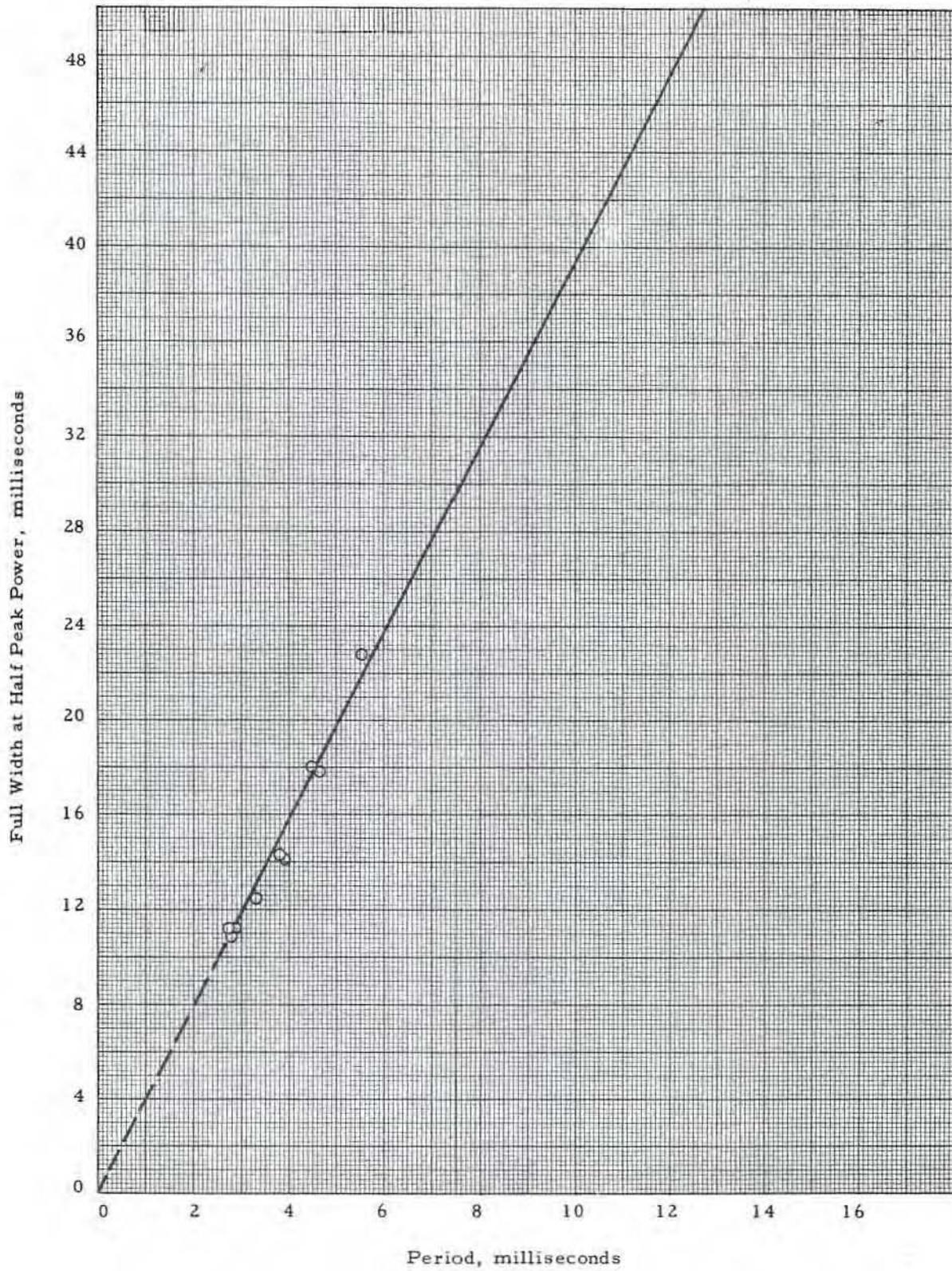
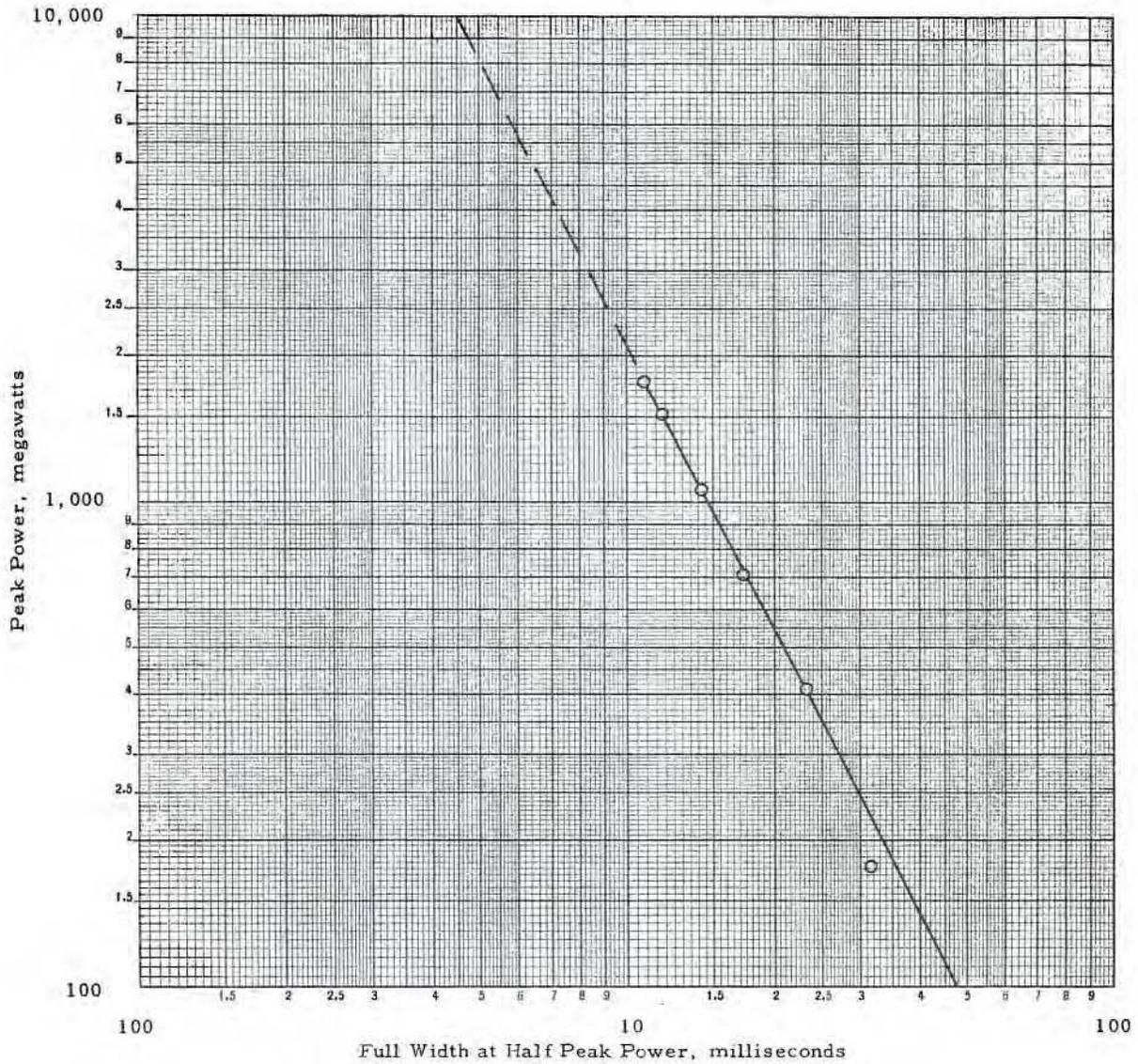


Fig. 7-7 -- Full width at half peak power versus period - prototype reactor



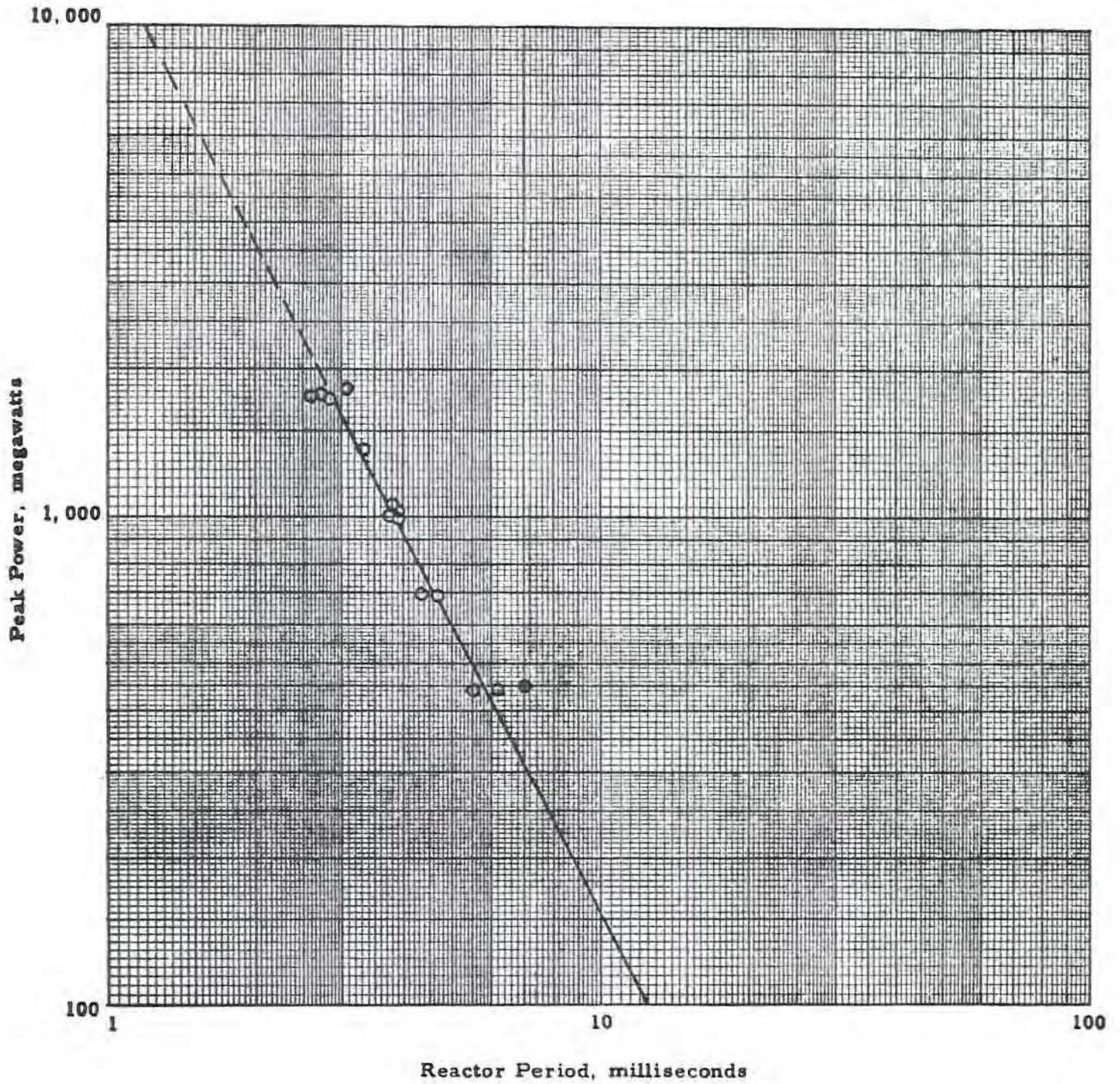


Fig. 7-9 -- Peak power versus reactor period - prototype reactor
(100-element core)

8. HAZARDS SUMMARY

8.1. REACTIVITY REQUIREMENTS ASSOCIATED WITH REACTOR OPERATION

The following reactivities associated with operation of the reactor are based on measurements made on the Torrey Pines TRIGA Mark III reactor, extrapolations from the TRIGA Mark F and Mark II reactors, and calculations. These values are for a core loaded with 70 fuel elements including the fueled followers on three control rods.

The worth of fuel as compared to water in each ring of the grid as measured in the prototype TRIGA Mark III reactor is given in Table 7.2.

To operate at 1 Mw, $\sim 2.3\%$ $\delta k/k$ will be required for overcoming the power coefficient of reactivity and about 1.7% $\delta k/k$ for overcoming the equilibrium xenon poisoning.

The reactivity changes caused by the various experimental facilities are shown in Table 8.1. These values are estimates based on measurements made on the prototype Mark III reactor.

Table 8.1

EFFECTS OF EXPERIMENTAL FACILITIES ON REACTIVITY

Experimental Facility	Worth, $\% \delta k/k$
Pneumatic transfer tube	-0.07
Rotary specimen rack	-0.18

It can be seen from the worths of the fuel elements given in Table 7.2 that in the case of grossly improper fuel loading procedures such as the

insertion of a fuel element into the core while the reactor is operating at a steady-state power that the maximum reactivity that could be introduced in a single action would be \$1.13. Since step additions of reactivity greater than this value are made on a routine basis when the reactor is pulsed, the addition of one fuel element (or even two or three) into the critical reactor core would present no hazard.

8.2. PRODUCTION AND RELEASE OF RADIOACTIVE GASES

8.2.1. Experimental Facilities

In the TRIGA Mark I reactor installation, the pneumatic transfer tube, and the rotary specimen rack contain air. In addition, the main reactor room is of course also filled with air that is in contact with the reactor pool. Of the radioisotopes produced in these air cavities, argon-41 is the most significant with respect to hazards and nitrogen-16 is considerably less significant.

At prolonged 1000-kw operation with no ventilation, the argon-41 activity in the pneumatic transfer tube is $4 \times 10^3 \mu\text{c}$, and in the rotary specimen rack it is $8.6 \times 10^5 \mu\text{c}$. Measurements at the Torrey Pines TRIGA Mark I reactor and the Illinois TRIGA Mark II reactor indicate no measurable argon-41 release from the rotary specimen rack. The release of argon-41 from the reactor pool water is discussed below in Section 8.2.2. The release to the atmosphere is discussed in Section 8.2.4.

In an experiment performed with the University of Illinois' 100-kw TRIGA Mark II reactor, air from the pneumatic tube and the rotary specimen rack was discharged into the reactor room and the argon-41 concentration in the room was found to be well below $2 \times 10^{-6} \mu\text{c}/\text{cm}^3$, which is the maximum permissible concentration (MPC) for restricted areas. The experiment consisted of operating the reactor at 100 kw for several hours, discharging the air from the pneumatic tube into the reactor room, and lowering and raising of a sample container in the access tube to pull out

some of the air from ten different positions of the rotary specimen rack. A scintillation crystal placed at the height of the top of the reactor, between the reactor and building exhaust, detected no increase over the background in the region 1 to 1.5 Mev. Sensitivity was checked with a cobalt-60 source, which gave a counting rate of 10 times the background for a flux of 8 gammas/cm²-sec. According to this calibration, the gamma flux due to argon-41 release was less than 8 gammas/cm²-sec. Converting this surface source of activity to an equivalent volume source indicates that the argon-41 concentration in the reactor room was less than 0.3% of the MPC. In the case of 1000-kw operation for the TRIGA Mark III reactor, the argon-41 concentration in the room should still be less than 3% of the MPC if the rotary specimen rack and pneumatic transfer tube were operated in this same fashion.

In summary, the argon-41 produced in the reactor cavities is not considered hazardous to the operating personnel. Leakage of air from these cavities into the reactor room is so small that the argon-41 activity is not detectable during 100-kw operation even when some of the air is deliberately discharged into the room. Activity from argon-41 may be only at the threshold of detection at the 1000-kw operating level.

8.2.2. Release of Argon-41 from Reactor Water

The argon-41 activity in the reactor pool water results from irradiation of the air dissolved in the water.

The following calculations were performed to evaluate the rate of argon-41 escaping from the reactor pool water into the reactor room. The calculations show that the argon-41 decays while in the water, and most of the radiation is safely absorbed in the water. The changes in argon-41 concentration in the reactor, in the pool water external to the reactor, and in the air of the reactor room are given by

$$V_1 \frac{dN_1^{41}}{dt} = V_1 \bar{\Phi} N_1^{40} \sigma^{40} - N_1^{41} (v_1 + V_1 \bar{\Phi} \sigma^{41} + \lambda^{41} V_1) + N_2^{41} v_1, \quad (1)$$

$$V_2 \frac{dN_2^{41}}{dt} = -\lambda^{41} N_2^{41} V_2 + v_1 (N_1^{41} - N_2^{41}) - (f_{2 \rightarrow 3} N_2^{41} V_2 - f_{3 \rightarrow 2} N_3^{41} V_3) , \quad (2)$$

$$V_3 \frac{dN_3^{41}}{dt} = (f_{2 \rightarrow 3} N_2^{41} V_2 - f_{3 \rightarrow 2} N_3^{41} V_3) - N_3^{41} (\lambda^{41} V_3 + q) , \quad (3)$$

where subscript 1 = Reactor region (water region in core),

subscript 2 = Reactor tank water region external to the reactor,

subscript 3 = Reactor room region ,

superscript 40 = Argon-40,

superscript 41 = Argon-41,

superscript A = Argon-40 plus argon-41,

V = Volume of region, cm^3 ,

N = Atomic density, atoms/cm^3 ,

λ = Decay constant, sec^{-1} ,

σ = Absorption cross section, cm^2 ,

q = Volume flow rate from reactor room exhaust
(cm^3 -sec),

v_1 = Volume flow rate through region No. 1 (cm^3 -sec),

$\bar{\Phi}$ = Average thermal neutron flux in Region No. 1,
(n/cm^2 -sec),

$f_{i \rightarrow j}$ = Fraction of argon-41 atoms in region i that escape
to region j per unit time, sec^{-1} .

To estimate the volume flow rate of the water in the reactor, the following equation is used:

$$v_1 = \frac{Q}{C_p \delta T \rho} , \quad (4)$$

where v_1 = Volume flow rate of the water through the core,

Q = Reactor power = 10^6 watts,

C_p = Specific heat of water ~ 4.19 watt-sec/g- $^{\circ}\text{C}$,

δT = Temperature rise across the core, $\sim 75^\circ\text{C}$ (conservatively high),

ρ = Exit water density = 0.958 g/cm^3 .

Thus,

$$v_1 = \frac{10^6}{(4.19)(75)(0.958)} \frac{\text{cm}^3}{\text{sec}} = 3.3 \times 10^3 \text{ cm}^3/\text{sec} .$$

Equation (1) can be reduced to

$$V_1 \frac{dN_1^{41}}{dt} = V_1 \bar{\Phi} N_1^{40} \sigma^{40} - (N_1^{41} - N_2^{41}) v_1$$

by considering the following numbers

$$\begin{aligned} v_1 &= 3.3 \times 10^3 \text{ cm}^3/\text{sec}, \\ V_1 &= 1.45 \times 10^4 \text{ cm}^3, \\ \bar{\Phi} &= 1.2 \times 10^{13} \text{ n/cm}^2\text{-sec}, \\ \sigma^{40} &= 0.060 \times 10^{-24} \text{ cm}^2, \\ \lambda^{41} &= 1.06 \times 10^{-4} \text{ sec}^{-1}, \end{aligned}$$

to show that $v_1 + V_1 \bar{\Phi} \sigma^{40} + \lambda V_1 \approx v_1$.

During equilibrium conditions the three equations reduce to:

$$V_1 \bar{\Phi} N_1^{40} \sigma^{40} = (N_1^{41} - N_2^{41}) v_1 \quad , \quad (5)$$

$$N_2^{41} \left[\lambda^{41} V_2 + f_{2 \rightarrow 3} V_2 \right] = (N_1^{41} - N_2^{41}) v_1 + f_{3 \rightarrow 2} N_3^{41} V_3 \quad , \quad (6)$$

$$N_3^{41} \left[\lambda^{41} V_3 + q + f_{3 \rightarrow 2} V_3 \right] = f_{2 \rightarrow 3} N_2^{41} V_2 \quad . \quad (7)$$

Combining equations (5) and (6) gives

$$N_2^{41} = \frac{V_1 \bar{\Phi} N_1^{40} \sigma^{40}}{\lambda^{41} V_2 + f_{2 \rightarrow 3} V_2} + \frac{f_{3 \rightarrow 2} N_3^{41} V_3}{\lambda^{41} V_2 + f_{2 \rightarrow 3} V_2} \quad , \quad (8)$$

which inserted into equation (7) for N_2^{41} yields

$$N_3^{41} \left[\frac{\lambda^{41} V_3 + q + f_{3 \rightarrow 2} V_3}{f_{2 \rightarrow 3} V_2} - \frac{f_{3 \rightarrow 2} V_3}{\lambda^{41} V_2 + f_{2 \rightarrow 3} V_2} \right] = \frac{V_1 \bar{\Phi} N_1^{40} \sigma^{40}}{\lambda^{41} V_2 + f_{2 \rightarrow 3} V_2} \quad (9)$$

The values of constants in equation (9) are

$$\begin{aligned} V_2 &= 4.02 \times 10^7 \text{ cm}^3 \text{ (8-1/2 ft by 25 ft),} \\ V_3 &= 3.12 \times 10^8 \text{ cm}^3 \text{ (21 ft by 36-1/2 ft by 16 ft),*} \\ q &= 1.73 \times 10^5 \text{ cm}^3/\text{sec} \\ \sigma^{40} &= 0.53 \times 10^{-24} \text{ cm}^2 \times \frac{1}{\sqrt{\pi}} = 0.47 \times 10^{-24} \text{ cm}^2, \end{aligned}$$

which leaves N_1^{40} , $f_{2 \rightarrow 3}$, $f_{3 \rightarrow 2}$, and N_3^{41} to be evaluated. The saturated concentration of argon in water according to Henry's law is

$$X = \frac{P}{K},$$

where X is the mole fraction of argon in water; P is the partial pressure in mm of Hg of argon above the water; and K is Henry's constant, which for argon is 2.9×10^7 mm of Hg per mole fraction of gas in solution at 30°C . Since the argon content of air is 0.94% by volume, the partial pressure of argon above the water is

$$0.0094 (650 \text{ mm} - 23 \text{ mm, vapor pressure of water}) = 5.89 \text{ mm of Hg}.$$

This yields 6.77×10^{15} argon atoms per cc of water.

An estimate of the parameter $f_{2 \rightarrow 3}$ (that is the fraction of argon atoms in the pool water that escape each second) can be obtained by examining the mobilities of ions in dilute solution. Most ions have velocities of the order of 3 to 8×10^{-4} cm/sec under a potential gradient of 1 volt per cm.† Since the argon atom will not have the advantage of being affected

*The room volume has been reduced by 10% to account for the presence of equipment, etc.

†Daniels, F., Outlines of Physical Chemistry, Wiley and Sons, New York, 1948, p. 414.

by a potential gradient, its velocity should be less than 3×10^{-4} cm/sec. Therefore, only the argon atoms within 3×10^{-4} cm of the pool surface will be in a region in which the argon atoms can leave the water within any given second. Actually, even this source volume is still too large. Nevertheless, it gives an upper limit for the fraction of the total argon atoms that can leave the water per second.

$$f_{2 \rightarrow 3} = \frac{3 \times 10^{-4}}{\text{water height}} = \frac{3 \times 10^{-4}}{7.31 \times 10^2} = 4.1 \times 10^{-7} \text{ sec}^{-1} .$$

During equilibrium conditions and assuming no difference in the rates of escape fractions for argon-40 and -41, the number of argon atoms that escape from the water into the air equals the number of argon atoms that enter the water from the air, i. e. ,

$$f_{2 \rightarrow 3} N_2^A V_2 = f_{3 \rightarrow 2} N_3^A V_3 , \quad (10)$$

where $N_3^A = 2.1 \times 10^{17}$ argon atoms/cm³ of air $\simeq N_3^{40}$,

$N_2^A = 6.77 \times 10^{15}$ argon atoms/cm³ of water $\simeq N_1^{40}$.

Solving for $f_{3 \rightarrow 2}$ gives

$$f_{3 \rightarrow 2} = f_{2 \rightarrow 3} \frac{N_2^A V_2}{N_3^A V_3} = 1.70 \times 10^{-9} \text{ sec}^{-1} . \quad (11)$$

Since $\lambda^{41} > f_{2 \rightarrow 3} > f_{3 \rightarrow 2}$, equation (9) reduces to

$$N_3^{41} = \frac{V_1 \Phi N_1^{40} \sigma^{40}}{(\lambda^{41} V_3 + \bar{q})} \frac{f_{2 \rightarrow 3}}{\lambda^{41}} = 1.04 \text{ atoms/cm}^3 , \quad (12)$$

Solving for N^{41} yields 0.48 atoms/cm^3 . This corresponds to a concentration of argon-41 activity of

$$A^{41} = \frac{\lambda^{41} N^{41}}{C} = \frac{1.06 \times 10^{-4} \times 1.04}{3.7 \times 10^4} = 0.297 \times 10^{-8} \mu\text{c/cm}^3 , \quad (13)$$

where A^{41} = Argon-41 concentration, $\mu\text{c}/\text{cm}^3$,

C = Conversion factor from disintegrations/sec to μc .

This is below the MPC recommended by the U. S. Atomic Energy Commission for unrestricted areas ($4 \times 10^{-8} \mu\text{c}/\text{cm}^3$).

The argon-41 activity discharge rate from the reactor room is obtained by multiplying the activity concentration by q, the value of air discharged per unit time, that is

$$\frac{A^{41}}{q} = 0.297 \times 10^{-8} \times 1.73 \times 10^5 = 0.513 \times 10^{-3} \mu\text{c}/\text{sec}. \quad (14)$$

8.2.3. Nitrogen-16 Activity in Reactor Room

The cross section threshold for the oxygen-16 (n, p) nitrogen-16 reaction is 9.4 Mev; however, the minimum energy of the incident neutrons must be about 10.2 Mev because of center of mass corrections. This high threshold limits the production of nitrogen-16 since only about 0.1% of all fission neutrons have an energy in excess of 10 Mev. Moreover, a single hydrogen scattering event will reduce the energy of these high-energy neutrons to below the threshold. The measured effective cross section for oxygen-16 (n, p) nitrogen-16 reaction and a fission neutron spectrum is $1.85 \times 10^{-29} \text{ cm}^2$. * This value agrees well with the value obtained from integrating the effective cross section over the entire fission spectrum.

The concentration of nitrogen-16 atoms per cm^3 of water as it leaves the reactor core is given by

$$N^N = \frac{\Phi_v N^O \sigma^O}{\lambda^N} \left[1 - e^{-\lambda^N t} \right] \quad (15)$$

where N^N = Nitrogen-16 atoms per cm^3 of water,

Φ_v = Virgin fission neutron flux $\approx 10^{13}$ n/cm²-sec at 1000 kw,

N^O = Oxygen atoms per cm^3 of water = 3.3×10^{22} atoms/cm³,

σ^O = Absorption cross section of oxygen = $2 \times 10^{-29} \text{ cm}^2$,

* Henderson, W. J., and P. R. Tunncliffe, "The Production of N-16 and N-17 in the Cooling Water of the NRX Reactor," NSE, 1958, pp. 145-150.

$$\lambda^N = \text{Nitrogen-16 decay constant} - 9.35 \times 10^{-2} \text{ sec}^{-1},$$

t = Average time of exposure in reactor.

The average exposure time in the reactor is given by

$$t = \frac{V_c}{v_1}, \quad (16)$$

where V_c is the core water volume exposed to flux Φ_v , and v_1 is the volume flow rate through the core (see Section 8.2.2). Thus,

$$t = \frac{1.45 \times 10^4 \text{ cm}^3}{3.3 \times 10^3 \text{ cm}^3/\text{sec}} = 4.4 \text{ sec}.$$

Solving for N^N from equation (15), one obtains 2.4×10^7 nitrogen-16 atoms per cm^3 of water leaving the core. With a flow of $3.3 \times 10^3 \text{ cm}^3/\text{sec}$, the rate of nitrogen-16 leaving the core is therefore 7.9×10^{10} atoms/sec.

In the TRIGA Mark I, the measured transport time for the water to travel the 16 feet from the reactor core to the surface of the tank is 42 seconds when the reactor is operating at 100 kw. To a first approximation the velocity of the rising water is proportional to the density difference between the pool water and the heated water from the core, that is,

$$v = K(\rho_O - \rho_{\text{exit}}) \quad (17)$$

Thus the velocity of the rising water column for the TRIGA Mark III reactor can be estimated from

$$v_{\text{III}} = \frac{v_1(\rho_O - \rho_{\text{exit III}})}{(\rho_O - \rho_{\text{exit I}})}, \quad (18)$$

where the subscripts I and III refer to TRIGA Mark I and Mark III reactors, respectively.

The water leaving the TRIGA Mark I core at 100 kw is at 45°C with a density of 0.9902 g/cm^3 . The water leaving the TRIGA Mark III core at 1000 kw is conservatively assumed to be at 100°C for this calculation

with a density of 0.9583 g/cm^3 . Thus, from equation (18):

$$v_{\text{III}} = \frac{16 \times 30.5 \text{ cm}}{42 \text{ sec}} \frac{0.99705 - 0.95835}{0.99705 - 0.99021} = 65.6 \text{ cm/sec} .$$

The transport time for nitrogen-16 through the 20 ft of water above the TRIGA Mark III reactor is then

$$t_{\text{rise}} = \frac{20 \times 30.5 \text{ cm}}{65.6} = 9.3 \text{ sec} . \quad (19)$$

This assumes that the nitrogen-16 rises straight up toward the pool surface. In practice, however, the nitrogen-16 is slowed down by the interruption of the vertical convective currents from the discharge of water through the diffuser nozzle across the core.

Measurements at the prototype TRIGA Mark III reactor at 1000 kw indicate that the diffuser results in a hold-up time increase by a factor of 3, so that the rise time is closer to 25 seconds. In 25 seconds, the nitrogen-16 decays to 0.096 of its initial value. Thus, the number of nitrogen-16 atoms that reach the water near the pool surface is about 7.6×10^9 atoms/second.

Only a small portion of the nitrogen-16 atoms present near the pool surface is transferred into the air of the reactor room. When a nitrogen-16 atom is formed, it appears as a recoil atom with various degrees of ionization. For high-purity water ($\sim 2 \mu\text{mho}$), practically all of the nitrogen-16 combines with oxygen and hydrogen atoms of the water. Most of it combines in an anion form, which has a tendency to remain in the water.* It is assumed that at least one-half of all ions formed are anions. Because of its 7.4-second half-life, the nitrogen-16 will not live long enough to attain a uniform concentration in the tank water. With the diffuser in operation, one can assume that the nitrogen-16 atoms will be dispersed in the 1 ft of water at the top of the pool directly above the core. Actually, they are more likely to be dispersed over a wider area in the pool and

*Mittl, R. L., and M. H. Theys, "N-16 Concentrations in EBWR," Nucleonics, March 1961, p. 81.

will decay before this lateral movement is completed. In the area directly above the core, the dominant contribution to the dose rate is the direct radiation from the core. The interest from a hazard point of view is then the number of nitrogen-16 atoms escaping into the air and diffusing away from the area above the core.

The maximum fraction of nitrogen-16 atoms that can escape from the water to the air per second can be estimated similarly to the case of argon (see Section 8.2.2.). Thus,

$$f_{2 \rightarrow 3}^N \leq \frac{1}{2} \frac{3 \times 10^{-4} \text{ cm/sec}}{30 \text{ cm}} = 0.5 \times 10^{-5} \text{ sec}^{-1} , \quad (20)$$

where it is assumed that one-half of the ions formed, namely the anions, remain in the water.

Thus the number of nitrogen-16 atoms entering the air is given by

$$f_{2 \rightarrow 3}^N N^N V = \frac{f_{2 \rightarrow 3} (7.6 \times 10^9)}{f_{2 \rightarrow 3} + \lambda^N} = \frac{0.5 \times 10^{-5} (7.6 \times 10^9)}{0.5 \times 10^{-5} + 9.35 \times 10^{-2}} =$$

$$4.06 \times 10^5 \text{ atoms/sec} . \quad (21)$$

The nitrogen-16 concentration (assuming no mixing with the rest of the room air) in the volume immediately above the mechanism bridge, 3 meters high and having a 36-in. -diameter, is

$$A = \frac{4.06 \times 10^5}{3.7 \times 10^4 \cdot 300 \text{ cm} \cdot \pi \cdot 18^2 \cdot (2.54)^2} = 5.6 \times 10^{-6} \mu\text{c/cm}^3 . \quad (22)$$

The dose rate from this concentration in this volume is small compared to the dose rate from the core itself.

In the rest of the room, the activity is affected by dilution, ventilation, and decay. Thus, the rate of accumulation of nitrogen-16 in the

room as a whole is given by

$$\frac{d(V_3 N_3^{16})}{dt} = S - (\lambda^N + q/V_3) N_3^{16} V_3 \quad (23)$$

where S = Number of nitrogen-16 atoms entering the room from the pool per second,

$$V_3 = \text{Volume of the reactor room} = 3.12 \times 10^8 \text{ cm}^3,$$

$$q = \text{Volume flow rate from the reactor room exhaust} = 1.73 \times 10^5 \text{ cm}^3/\text{sec}.$$

For saturation conditions,

$$V_3 B_3^{16} = \frac{S}{(\lambda^N + q/V_3)} = \frac{4.06 \times 10^5}{9.35 \times 10^{-2} + 0.471 \times 10^{-3}} = 4.31 \times 10^6 \text{ nuclei} \quad (24)$$

This corresponds to an activity concentration of $3.49 \times 10^{-8} \mu\text{c}/\text{cm}^3$, which is negligible. Thus, everywhere but directly above the core, the nitrogen-16 activity presents no hazard. Above the core, the direct activity of the core is dominant.

8.2.4. Routine Release of Argon-41 to Atmosphere

Expressions for the concentration of argon-41 in the exposure room were presented earlier. Although higher concentrations will prevail with stagnant air in the exposure room, air will not be discharged to the atmosphere under these conditions. When the room is being ventilated, the concentration will be a few percent of the stagnant-air value.

When the activity is released from the building, the concentration on the ground must be less than the nonoccupational maximum permissible concentration (MPC) in unrestricted areas set by the U. S. AEC, which is $4 \times 10^{-2} \mu\text{c}/\text{m}^3$ for argon-41. One appropriate method for determining whether this value is met is to use the following equation for predicting the cloud concentration at ground level at the building.

$$X(0, 0, 0) = \frac{1}{(0.5) A \bar{u}} (Q) \quad (25)$$

where $X(0, 0, 0)$ = Concentration at the building, $\mu\text{c}/\text{m}^3$,

Q = Activity release rate, $\mu\text{c}/\text{sec}$,

\bar{u} = Mean wind speed, 1 msec ,

A = Cross sectional area of reactor room, 31.2 m^2 .

Applying a reasonable duty cycle for the operation of the facility, that is 6 hours per day and 5 days per week of 1000-kw operation, yields an average argon-41 discharge rate of

$$\bar{Q} = \frac{6}{24} \cdot \frac{5}{7} (0.513 \times 10^{-3}) = 0.913 \times 10^{-4} \mu\text{c}/\text{sec} . \quad (26)$$

Substitution of $0.913 \times 10^{-4} \mu\text{c}/\text{sec}$ for Q in equation (25) gives a value of $X(0, 0, 0) = 5.85 \times 10^{-6} \mu\text{c}/\text{m}^3$.

8.3. HANDLING OF RADIOACTIVE ISOTOPES

Because there can be intense radiation fields from radioactive isotopes produced by this reactor, reactor operations must be supervised by individuals trained in the detection and evaluation of radiological hazards. For instance, activation calculations indicate that the reactor is capable of producing an equilibrium concentration of approximately 10^5 curies of cobalt-60 in the rotary specimen rack using only 20 rack positions. Since this production is distributed in 20 samples, the maximum amount of activity that can be withdrawn at one time is 5×10^3 curies. This constitutes such an intense source of radiation that manual handling is out of the question. If such isotope production is attempted, the operating agency should obtain shielded isotope-handling equipment to reduce the radiation dosage to an acceptable level.

Remote handling equipment will not be necessary if the radioactivity of a sample is limited to a radiation level associated with 10 curies of cobalt-60. From such a source removed from the reactor the dose rate to persons within 2 meters would be 58 mr/min. A person exposed to this dose rate for about 2 minutes would receive his weekly permissible dose of 100 mr. Naturally,

the source limit will depend upon the removal procedure used by the operating agency; however, such a limit should be made known to the operating personnel for evaluating the potential exposures when removing a sample from the rotary sample rack. A radiation detector will need to be used to monitor the activity of the sample as it is removed from the reactor.

The maximum potential hazard from isotope production is from an accident in which this radioactivity is released into the air of the reactor room. Such an accident would require a very special set of conditions. There would have to be either a spill during the removal from the reactor, or a reaction (perhaps chemical) of sufficient strength to disperse the samples from the rack. In addition, the radioactive sample would have to be in gaseous form or of particle size sufficiently small to be dispersed as an aerosol before a hazard could exist external to the reactor room. If such a set of conditions should occur, the radioactivity released would still be less than that assumed for the fuel cladding break. Hence, in the unrestricted area outside the reactor room, the hazard associated with isotope production will be less than that produced by a fuel element failure. This assumes that the radioactive isotope has approximately the same body organ uptake as that of the released fission products.

8.4. REACTOR POWER PULSING

The reactor loading will be a maximum of 4.9% $\delta k/k$ (47.00) excess reactivity above a cold, clean, critical and compact condition. A compact condition is a core loaded with all of the innermost lattice positions containing fuel and with the uranium-235 content (according to the metallurgical analysis of the fuel elements) progressively smaller with increasing distance from the core center. Thus, any rearrangement of the fuel will result in a lower excess reactivity.

Although the maximum excess reactivity is 4.9% $\delta k/k$, the transient control rod will be limited to 2.1% $\delta k/k$. The reactivity that is controlled by the other rods, however, through a number of remote possibilities, could

permit the power pulse to be initiated while at a high power or on a period. The expected hazards caused by pulsing the reactor from other than a low power are discussed in Section 8.7 under excursion accidents.

On the basis of operating experience with TRIGA reactors, it is concluded that a 2.1% $\delta k/k$ step addition of reactivity from low power presents no hazard. TRIGA reactors have been pulsed safely thousands of times with 2.1% $\delta k/k$ additions and many times with reactivity additions up to 3.2% $\delta k/k$. Under normal pulsing conditions no hazards exist to either the operating personnel or the public, since the shielding affords protection from a 1000-kw reactor and the possible pulse frequency is equivalent to only a 100-kw reactor.

A typical plot of power pulse and the fuel temperature for a 2.1% $\delta k/k$ addition with a core containing 100 fuel elements is shown in Fig. 8-1. The resulting power pulse attains a peak power of about 2000 Mw with a reactor period of 2.8 msec and a total energy release during the pulse of about 23 Mw-sec. Fuel temperatures are limited to less than 500°C.

8.5. FUEL ELEMENT CLADDING FAILURE

The release of radioactivity by aqueous corrosion was determined experimentally to be 100 micrograms of uranium-zirconium hydride per day per square centimeter of unclad fuel element exposed to the water during shutdown conditions. Therefore, in the event of a rupture of the fuel element cladding, only the gaseous fission products that have collected in the space between the fuel material and the cladding will be released.

The hazards associated with a failure of the fuel element cladding and consequent fission product contamination of the reactor pool water and the reactor room air have been calculated. The water activity would reach a level of $0.9 \mu\text{c}/\text{cm}^3$ from the failure of one fuel element, and the airborne activity in the reactor room would attain a uniform concentration of $0.6 \times 10^{-2} \mu\text{c}/\text{cm}^3$.

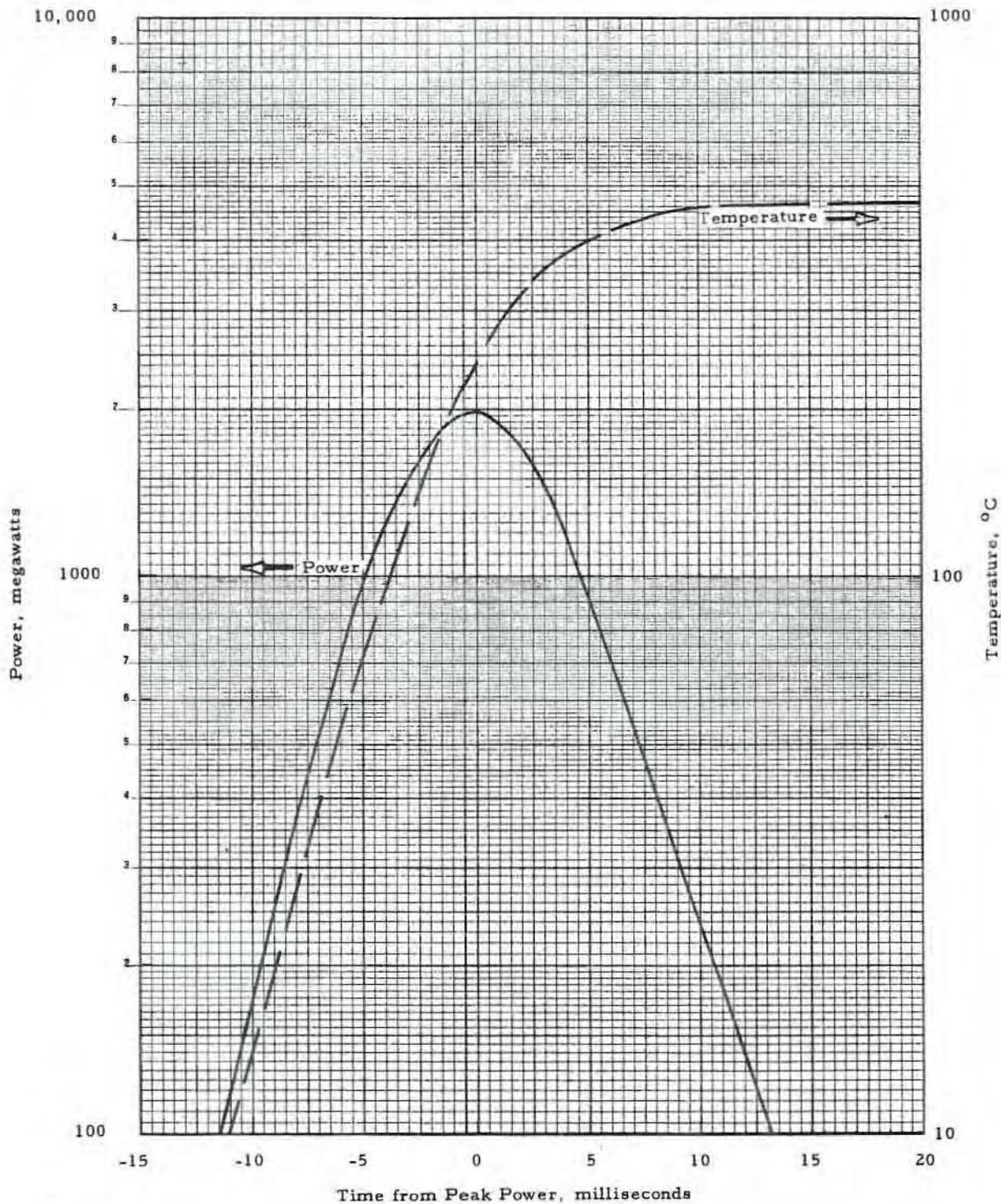


Fig. 8-1--Calculated peak power and fuel temperature versus time for \$3.00 pulse—TRIGA Mark III reactor

Fission Product Release Fraction

The quantity of gaseous fission products produced in the fuel element was determined by the use of Blomeke and Todd data* for infinite operation at 1000 kw. Table 8.2 is a compilation of the volatile gamma emitters in a fuel element loaded with [redacted] grams of U^{235} and exposed to an effective thermal flux of 0.83×10^{13} n/cm²-sec. (The Blomeke and Todd data are for a 2200-m/sec fission cross section.) The activities listed are the saturation levels for the fuel element with the highest power density.

For the purpose of this analysis, the gaseous fission products have been separated into two groups. Group one consists of the gaseous isotopes that for all practical purposes remain in the reactor tank water. In this group are the bromine and iodine isotopes. According to water boiler reactor data, less than 10% of the fission product gases with half-lives of less than a minute escape from the reactor solution. Evidently, such a short lifetime prevents the diffusion of the isotopes to the surface and their escape from the solution. In addition, the isotopes with short half-lives are normally beta emitters with negligible gamma emission. Group two consists of the insoluble volatiles, krypton and xenon isotopes, with half-lives greater than a minute. They are the major source of gamma activity in the reactor room from a cladding failure because of their low solubility in water.

The sum of the saturated gamma activities of the volatile fission products is 0.64×10^4 curies in the fuel element with the highest power density. Most of this activity will be retained in the lattice structure of the fuel. Of the activity released from the fuel, approximately two-thirds will stay in the water and one-third will escape into the air.

* Blomeke, J. O., and Mary F. Todd, "Uranium-235 Fission-Product Production as a Function of the Thermal Neutron Flux, Irradiation Time, and Decay Time," Oak Ridge National Laboratory Report ORNL-2127, August 1957 - November 1958.

Table 8.2

GASEOUS FISSION PRODUCTS IN B-RING FUEL ELEMENT
1000 SATURATED ACTIVITY-117 ELEMENT CORE

Nuclide	Decay Constant (hr ⁻¹)	Inventory (curies/element)
Group I		
Br 83	3.02×10^{-1}	59.01
Br 84	1.31×10^0	135.52
Br 84m	6.95×10^0	2.79
Br 85	1.39×10^1	183.52
Br 87	4.49×10^1	330.00
I 129	4.60×10^{-12}	122.97
I 131	3.58×10^{-3}	356.64
I 132	3.07×10^{-1}	540.72
I 133	3.34×10^{-2}	796.80
I 134	7.93×10^{-1}	933.84
I 135	1.04×10^{-1}	726.24
I 136	2.90×10^1	381.36
	Total Iodines	3859.2
	Total Group I	4575.7
Group II		
Kr 83m	3.66×10^{-1}	59.02
Kr 85m	1.59×10^{-1}	183.50
Kr 85	7.67×10^{-6}	36.88
Kr 87	5.35×10^{-1}	333.36
Kr 88	2.50×10^{-1}	456.48
Kr 89	1.31×10^1	566.88
Kr 90	7.55×10^1	637.44
Kr 91	2.54×10^2	379.44
Xe 131m	2.41×10^{-3}	3.56
Xe 133m	1.26×10^{-2}	19.19
Xe 133	5.50×10^{-3}	797.52
Xe 135m	2.67×10^0	217.89
Xe 135	7.60×10^{-2}	588.72
Xe 137	1.07×10^1	72.62
Xe 138	2.45×10^0	674.40
Xe 139	6.08×10^1	697.68
Xe 140	1.56×10^2	732.48
	Total Group II	6457.0

To determine the release fraction of fission products from the fuel element, the following experiment was performed with a special TRIGA fuel element. The element was fabricated with a tube connecting the space between the cladding and the fuel with a charcoal-filled cold trap at the surface of the reactor tank. All of the fission products gases that accumulated in the gap were then collected in the charcoal trap by purging the system with helium. Analysis of the amount of radioactive gases in the gap and the total inventory in the fuel permitted the determination of the fraction of fission product gases that escaped from the uranium-zirconium hydride material into the gap.

In this experiment the special element was exposed to a thermal flux of 2.1×10^{12} n/cm²-sec for an hour. The activity of Xe¹³³ present in the element after this hour was calculated from the work of Bolles and Ballou.* In this work the number of Xe¹³³ nuclei present after the fission of 10,000 U²³⁵ nuclei is given as a function of time. Integrating these data over a time of 1 hr gives 4.9×10^{14} Xe¹³³ nuclei/fission. In the 1-hr irradiation 2.7×10^{17} atoms of U²³⁵ fissioned. Thus, the number of Xe¹³³ nuclei present at the end of the irradiation was $(4.9 \times 10^{-4})(2.7 \times 10^{17}) = 1.3 \times 10^{14}$ nuclei, or a total activity of 5.4×10^3 microcuries. The analysis of the Xe¹³³ activity that had collected in the gap in the 1-hr gave 78 microcuries. Therefore the fraction of Xe¹³³ that diffuses into the gap between the fuel and the cladding is

$$\frac{\text{Xe}^{133} \text{ in gap}}{\text{Xe}^{133} \text{ inventory in fuel}} = \frac{7.8 \times 10^1}{5.4 \times 10^3} = 1.4 \times 10^{-2} \quad (27)$$

In this analysis it is assumed that the fractions of various gaseous isotopes produced that collect in the gap are the same as that determined for Xe¹³³. Since it is possible for the cladding rupture to occur at a higher

* Bolles, R. C., and N. E. Ballou, "Calculated Activities and Abundances of U²³⁵ Fission Products," U. S. Naval Radiological Defense Laboratory Report US-NRDL-456, August 30, 1956.

fuel temperature, for instance, during a power transient, the release fraction has been increased by a factor of 2 to approximate the possible difference in fuel temperatures. Therefore, the release fraction is taken as 2.8×10^{-2} .

Water Activity

As discussed previously, 0.46×10^4 curies is the total inventory of soluble gaseous fission products. Multiplying by the release fraction gives 1.28×10^2 curies as the released gamma activity that remains in the water from one fuel element. Since the volume of water in the reactor pool is $4.02 \times 10^7 \text{ cm}^3$, the activity concentration is $3.18 \mu\text{c}/\text{cm}^3$. In 24 hours, the activity would decrease to $0.49 \mu\text{c}/\text{cm}^3$. The activity remains moderately high because of the small decay constant of iodine-131 and -133. The demineralizer can be used to remove the soluble volatile fission products.

External Exposure from Fission Product Releases in the Reactor Room

The amount of insoluble volatiles released into the reactor room from a cladding failure is 6.45×10^3 curies (the inventory of insoluble volatiles in the hottest fuel element) times 2.8×10^{-2} (the release fraction), or 180 curies of gamma emitters. The external exposure received by a person in the reactor room from this release of fission products will be dependent upon the fission product distribution in the reactor room, the exit path, and time to exit. An example has been selected to indicate possible exposures from the release of fission products from one fuel element.

In this example, the fission products have escaped from the tank water and are uniformly dispersed into the reactor room of $3.12 \times 10^8 \text{ cm}^3$. The activity concentration becomes

$$\frac{A}{V} = \frac{180 \times 10^6}{3.12 \times 10^8} = 5.77 \times 10^{-1} \frac{\mu\text{c}}{\text{cm}^3} = 2.08 \times 10^4 \frac{\gamma}{\text{sec-cm}^3} . \quad (28)$$

The maximum dose rate in the reactor room after the uniform dispersal can be calculated by assuming the room is equivalent to a hemisphere of 5.3-meter radius, i. e. ,

$$D = \frac{S_v (1 - e^{-\Sigma R})}{2\Sigma C} , \quad (29)$$

where D = Dose rate, mr/min,

S_v = Photons emitted per sec per cm^3 of air ($2.08 \times 10^4 \text{ } \gamma/\text{sec-cm}^3$),

Σ = Attenuation coefficient for air,

R = Outer radius of hemisphere ($5.3 \times 10^2 \text{ cm}$),

C = Flux to dose rate conversion.

The average gamma energy is 0.7 Mev for the insoluble volatiles, and for 0.7 Mev, $\Sigma = 3.5 \times 10^{-5} \text{ cm}^{-1}$ and $C = 4.2 \times 10^4 \text{ } \gamma/\text{cm}^2\text{-sec}$ per mr/min. Substitution of these values in the above equation gives a maximum dose rate of $1.25 \times 10^2 \text{ mr/min}$.

Internal Exposures from Breathing Fission Product Cloud

If a person should breath the release of insoluble volatiles from the fuel element, the critical organ becomes the lungs and the beta-emitting nuclides become more important than the gamma emitters. This situation is different from external exposures wherein betas are insignificant unless they are deposited on a person's clothing. Table 8.3 lists the activity of the beta-emitting nuclides that escape from the reactor tank water during a rupture in the fuel cladding. The activities were calculated in a fashion similar to that of the released gamma activity.

A person standing directly over the reactor during the release of fission products could receive an internal exposure to his lungs of about 23 rads, providing the fission products come out of the water immediately and are not dispersed in the reactor room air. However, because the fission products do not come out of the water all at once and because the air in the room is not stagnant, sufficient time exists for a person to evacuate

Table 8.3
 BETA ACTIVITY IN REACTOR ROOM AIR RESULTING
 FROM FUEL CLADDING RUPTURE

Mass Number	Nuclide	β energy, (Mev)	Decay constant (sec^{-1})	Activity, (curies)	Fraction of total activity
85m	Kr	0.236	4.41 (-5)	5.12	0.043
87	Kr	1.01	1.48 (-4)	9.33	0.078
88	Kr	0.33	6.95 (-5)	12.77	0.105
89	Kr	1.3	3.63 (-3)	15.87	0.132
133	Xe	0.115	1.52 (-6)	22.33	0.184
135	Xe	0.30	2.11 (-5)	16.48	0.136
137	Xe	1.33	2.96 (-3)	20.33	0.167
138	Xe	1.0	6.79 (-4)	18.88	0.155
(Total)				121.11	1.000

the room before breathing a significant amount of beta-emitting nuclides. This conclusion assumes that the evacuation alarm sounds with the release of fission products and that the person leaves the reactor platform within a minute of the release.

Calculations have been made of the internal exposure from breathing the fission products after they have been dispersed uniformly in the reactor room. This situation approximates the condition of a person leaving the reactor room after the fission products have been dispersed. In the case in which the person spends a short time in the reactor room (compared to the shortest half-life of the beta emitters, i. e., about 2 minutes), his lungs should be filled with the concentration existing in the room, which for the reactor room is

$$\frac{A}{V} = \frac{1.21 \times 10^8}{3.12 \times 10^8} = 0.387 \mu\text{c}/\text{cm}^3 \quad (30)$$

Since there are approximately 3 liters of air in the lungs, the beta activity in the lungs would be 1.16×10^3 microcuries on leaving the reactor room.

The beta exposure to the lungs is calculated from the following formula:

$$\text{Dose} = \frac{\text{ACR}}{m} \sum_i^8 \left[\frac{f_i E_i}{\lambda_i} \left(1 - e^{-\lambda_i t} \right) \right] \text{rads (beta)} \quad (31)$$

where A = Activity in the lungs on leaving the reactor room, μc ,

C = Conversion factor $(3.7 \times 10^4 \beta/\text{sec} - \mu\text{c})(1.6 \times 10^{-6} \text{ erg/Mev}) / 100 \text{ ergs/g-rad}$,

R = Retention factor

m = Mass of lungs (1000 grams),

f_i = Fraction of total activity ,

E_i = Energy of beta from nuclide i, Mev,

λ_i = Decay constant,
 t = Time of exposure .

Only a fraction of the inhaled activity will be retained in the lungs. One might expect that all the volatiles will be exhaled; unfortunately, however, gases have a tendency to collect on dust particles. A 1/8 retention factor is customarily used for the inhaled activity. The decay constant, λ_1 is given by the sum of the radioactive decay constant and the biological release constant ($6.7 \times 10^{-8} \text{ sec}^{-1}$). For an inhalation of 1.16×10^3 microcuries, the maximum integrated exposure to the lungs is 0.16 rad. The limiting biological exposure criteria for an accident or one exposure may be considered to be 50 r for the lungs. *

Release of Fission Product Cloud to Atmosphere

Although the release of fission products from the reactor would cause the shutdown of the ventilation system and the closing of the buildings, it is appropriate now to examine the hazards associated with such a release from the reactor room to the atmosphere. Because of the difficulty in specifying both the rate of release of fission products from the stack as a function of the time and the time interval over which the rate of release occurs, a slightly different diffusion equation from the one in Section 8.2.4 is used to facilitate the calculation of the hazard. The rate of release Q, is changed to the total activity released, Q' , and the activity concentration, X, is changed to the exposure, X' . The maximum exposure equation becomes:

$$X'(0, 0, 0) = \frac{Q'}{0.05 A\bar{u}} \quad , \quad (32)$$

where the units of X' are ($\mu\text{c-sec}/\text{m}^3$), Q' is (μc), and the other terms have the same units as given in Section 8.2.4. With this substitution, the exposure term X' is the integral of the activity concentration while the

* Marley, W. G., and T. M. Fry, "Radiological Hazards from an Escape of Fission Products and the Implications in Power Reactor Locations," Proc. Inter. Conf. on Peaceful Uses of Atomic Energy, Paper 394, Geneva, 1955.

activity exists in that location. The calculated maximum exposure value is conservative, since in equation (32) the wind is assumed to be constant in direction and speed during the release interval. Changes in the wind direction and speed during the release would increase the dispersion of the cloud and, therefore, decrease the maximum exposure.

The total activity in the reactor room is 180 curies, i. e., Q^1 equals 1.80×10^8 microcuries. The effective release of this activity at ground level (see Section 8.2.4) gives an exposure of $1.15 \times 10^7 \mu\text{c-sec/m}^3$. Within 250 m, the integrated concentration would be reduced to less than $6 \times 10^5 \mu\text{c-sec/m}^3$ under stable atmospheric conditions.

The lowest value of acceptable concentration for the Xe and Kr in an unrestricted area is that for Kr^{87} , $2 \times 10^{-3} \mu\text{c/cm}^3$. Integrated over a 1-yr period, this gives an acceptable exposure of $6.3 \times 10^5 \mu\text{c-sec/m}^3$. Thus, the integrated exposure from the release of the Xe and Kr in a fuel element is 18 times higher than the acceptable level for Kr^{87} to the general public, but, under very unfavorable atmospheric conditions, it is below that level within 250 m.

It should be noted that in the more than 10 reactor-years of operation of TRIGA reactors at Torrey Pines, only 4 fuel element failures have occurred with aluminum-clad elements. During the more than 3 years of operation of the prototype TRIGA Mark III reactor at Torrey Pines, there has been one failure of the stainless-steel high-hydride fuel elements. Thus, it seems more than reasonable to average the exposure over 1 year.

8.6. LOSS OF REACTOR POOL WATER

Although the total loss of reactor pool water is considered to be an extremely improbable event, calculations have been made to determine the maximum fuel temperature rise resulting from such a loss of coolant. Results indicate that if the complete water loss in the core occurs immediately after the reactor has been shut down from infinite operation at 1000 kw, the maximum temperature of the fuel, and consequently the

temperature of the stainless steel cladding, is less than 800°C . At this temperature, the equilibrium hydrogen pressure for $\text{U-ZrH}_{1.7}$ fuel material plus the pressure exerted by trapped air and fission product gases is about 140 psi. This pressure produces a stress of about 5100 psi whereas, the yield stress for the stainless steel cladding is $>12,000$ psi at 800°C . (At this temperature the ultimate strength is almost 25,000 psi.) Therefore, the fission products will be retained in the fuel elements.

It should also be noted that although the only mechanism considered herein for heat removal after the water loss is through natural convection of air through the core, some heat will be removed by conduction to the grid plates and by radiation.

Method of Calculation

Use was made of a two-dimensional, transient heat transport computer code, entitled RAT, developed at General Atomic, for calculating the maximum temperature in the core after a water loss.

It was assumed that at the time at which the water was lost, the temperature distribution in the B-ring fuel element was equal to the temperature distribution in a central fuel element during steady-state operation of the reactor at 1000 kw.

It was also assumed that the reactor had been operating for an infinite time at 1000 kw with only 74 elements in the core. The rate of energy release in the B-ring element was determined from consideration of the energy deposition of fission product gammas and betas only. The energy release from delayed neutrons is relatively small (about 5500 watt-sec total in a central element) and has an average decay constant of about 0.08 sec^{-1} .

The after-shutdown power density (in Btu/hr-ft^3) in the B-ring fuel

element is given by:

$$\frac{q}{V} = 0.1 P \frac{P}{V_f} \cos \left[0.78 \frac{\pi}{L} \left(x - \frac{L}{2} \right) \right] \times \left\{ \left[t + t_o + 10 \right]^{-0.2} - 0.87 \left[t + t_o + 2 \times 10^7 \right]^{-0.2} - 0.5 \right\} \quad (33)$$

where P = Peak-to-average power density in the core = ~ 2.0 ,

P = Operating reactor power = 3.41×10^6 Btu/hr (1000 kw),

V_f = Volume of the fuel in the core = 1.06 ft^3 ,

L = Length of the fuel = 1.25 ft

x = Distance measured from the bottom of the fuel element, ft

t = Time after the core is exposed to the air, sec,

t_o = Time from shutdown to the time the core is exposed, sec.

Equation (33) is a modification of the Untermeyer-Weill formula that matches the work of Stehn and Clancy* from 10 sec to about 5×10^4 sec after shutdown. It is also conservatively assumed that all the energy produced by fission product decay in the elements is deposited in the element.

While the decay gammas and betas are raising the fuel element temperature, the flow of air between the fuel elements will be removing heat and attempting to lower the fuel temperature. The air velocity through a central channel can be determined by setting the frictional pressure loss equal to the buoyancy. Entrance and exit losses will be negligible (between 2 and 5%) compared with friction losses

$$\delta P (\text{buoyancy}) = \delta P (\text{friction}) . \quad (34)$$

The term on the left is given by

$$\delta P (\text{buoyancy}) = (\rho_o - \rho_1) \frac{L}{2} , \quad (35)$$

* Stehn, J. R., and E. F. Clancy, Fission Product Radioactivity and Heat Generation, Paper No. 1071, Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, United Nations, Geneva, Switzerland, September 1958.

where L is the length of the channel and ρ_0 and ρ_1 are the entrance and exit air densities, respectively.

Since the frictional pressure drop calculations for laminar flow in non-circular channels are incorrect when expressed in terms of the hydraulic radius, the pressure drop for the TRIGA reactor must be predicted by other means. The method selected was to convert the free-flow area into an annulus around the fuel element. With an annular space of inner diameter D_1 and outer diameter D_2 , the frictional pressure drop becomes

$$\delta P \text{ (friction)} = \frac{32 \mu v L}{g \left[D_2^2 + D_1^2 - \frac{(D_2^2 - D_1^2)}{\ln(D_2/D_1)} \right]} \quad (2) \quad (36)$$

where μ = Viscosity of air, lb/hr-ft,

v = Velocity of the air, ft/hr,

L = Length of the fuel element, ft,

D_1 = Fuel element diameter, ft,

$D_2 = D_1 + 2b$, ft.

The term b in D_2 should be the effective separation distance between the B-ring element and those in the C-ring. The use of the minimum separation distances as b would yield too large a pressure drop and is considered too restrictive. The use of the average separation distance, based on the free flow between the B-ring element and the C-ring, would yield too low a pressure drop since the pressure drop is not a linear function of the separation distance. As an approximation, b is taken as the mean of the two values, i. e. ,

$$b = \frac{1}{2} (0.01130 + 0.04930) = 0.0303 \text{ ft} \quad (37)$$

For the channel between the B- and C-ring, the equation for the pressure balance (34) becomes

$$(\rho_0 - \rho_1) = 6.88 \times 10^{-5} \bar{\mu} \bar{v} , \quad (38)$$

where $\bar{\mu}$ is the average viscosity of the air (lb/hr-ft) in the channel and is a function of the entrance and exit temperatures, \bar{v} is the average air velocity (ft/hr) in the channel, and the entrance and exit densities are ρ_0 and ρ_1 (lb/ft³).

The mass flow rate of air in the channel is

$$w = \bar{v} \bar{\rho} A_c , \quad (39)$$

where $\bar{\rho}$ is the average density of air in the channel, and A_c is the flow area associated with the channel. Combining equations (38) and (39) the mass flow rate becomes

$$w = 405 (\rho_0 - \rho_1) \bar{\rho} / \bar{\mu} . \quad (40)$$

Assuming that the average properties in equation (40) are the average of the properties at the entrance and exit, equation (40) becomes

$$w = 405 \frac{(\rho_0^2 - \rho_1^2)}{(\mu_0 + \mu_1)} . \quad (41)$$

Over the range of temperatures of interest, the properties of air have been approximated by linear equations. Thus,

$$\rho = \frac{1}{2.5 \times 10^{-2} T} \text{ lb/ft}^3$$

and

$$\mu = (0.01135 + 0.6017 \times 10^{-4} T) \text{ lb/hr-ft} , \quad (42)$$

where T is the temperature in ^oRankine.

Using these expressions in equation (41) one obtains

$$W = \frac{3.24 \times 10^5 (T_1^2 - T_0^2)}{(T_1^2 T_0^2) [0.01135 + 0.30085 \times 10^{-4} (T_1 + T_0)]} \text{ lb/hr} \quad (43)$$

The determination of the amount of heat removed by this air flowing past the element rests on the evaluation of a heat transfer coefficient. For a free-standing cylinder cooled by natural circulation, the heat transfer coefficient is given, conservatively, by

$$h = 0.531 \frac{k_f}{L} [\text{GrPr}]^{0.25}, \quad (44)$$

where k_f is the thermal conductivity of the air film at temperature T_f (Btu/hr-ft- $^{\circ}$ F), Gr is the Grashof number, and Pr is the Prandtl number. Because the fuel element is surrounded by adjacent elements, the flow will probably not be laminar, even at low Reynolds numbers, and the heat transfer correlation should be better than that assumed, perhaps by as much as a factor of 2.

Again, over the temperature range of interest, one can write for the thermal conductivity and specific heat of the air

$$k_a = (0.0009 + 0.26 \times 10^{-4} T) \text{ Btu/hr-ft-}^{\circ}\text{F}$$

and

$$C_{pa} = 0.240 \text{ Btu/lb-}^{\circ}\text{F}$$

with temperature, T, in $^{\circ}$ R. Using the values from equations 42 and 45,

in equation (44) one obtains

$$h = 50.16 \left[\left\{ 0.0009 + 0.13 \times 10^{-4} (T_w + T_a) \right\}^3 \left[1.25 \times 10^{-2} (T_w + T_a) \right]^2 \right. \\ \left. \left[0.5 (T_w + T_a) \right] \left[0.01135 + 0.30085 \times 10^{-4} (T_w + T_a) \right]^{-1} (T_w - T_a) \right]^{0.25} \quad (46)$$

where T_w is the wall temperature, T_a is the bulk air temperature, and T_f is the average of the two, all in $^{\circ}\text{R}$.

For the purpose of this analysis the heat transfer along the graphite end reflectors was neglected, and the fuel element was considered to be a 1.25-ft-long, 0.1225-ft-diameter cylinder of U-ZrH_{1.7} with specific heat and thermal conductivity given by:

$$C_{pf} = (26.3 + 0.0245 T) \text{ Btu/ft}^3 \text{ } ^{\circ}\text{F} \text{ ,}$$

and

$$k_f = (10.7 - 6.42 \times 10^{-4} T) \text{ Btu/hr-ft-}^{\circ}\text{F} \text{ ,} \quad (47)$$

where T is the local temperature in $^{\circ}\text{R}$. The temperature drop in the clad and any gap was ignored because it is in the 1 to 5 $^{\circ}\text{C}$ range when the fuel reaches maximum temperature. For the computer program, the fuel element was divided into five radial and five axial regions, and the temperature in each region was computed as a function of time after complete water loss.

In Fig. 8-2, the maximum fuel temperature calculated is plotted as a function of the time after the loss of cooling water. The loss of water in the core was assumed to take place immediately at shutdown at which time the maximum fuel temperature is 375 $^{\circ}\text{C}$.

To determine the pressure exerted on the cladding by released hydrogen, fission products, and air trapped in the fuel can, the conservative assumption was made that the entire fuel meat is at the peak fuel temperature, i. e. , 780 $^{\circ}\text{C}$.

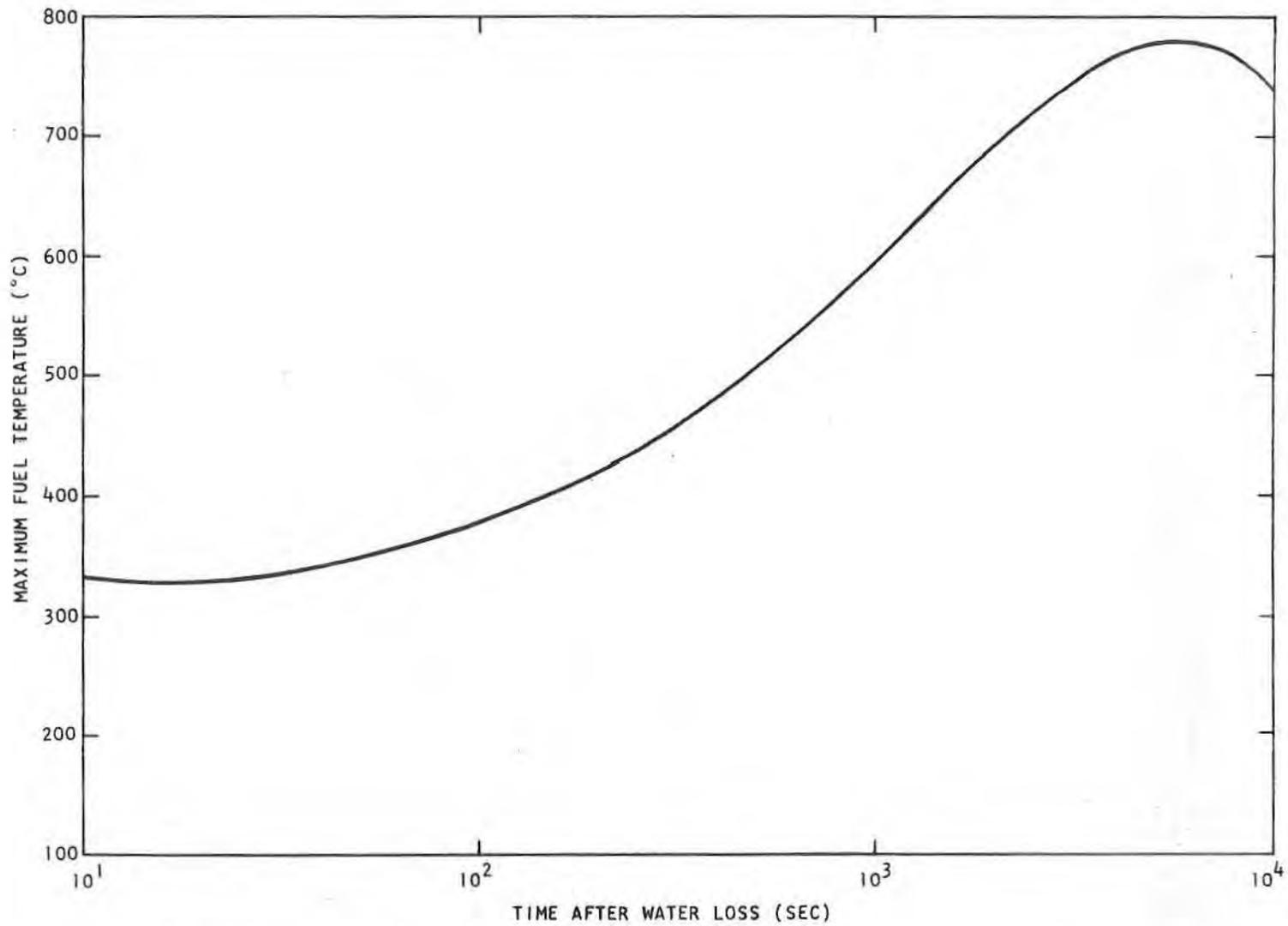


Fig. 8-2--Maximum Fuel Temperature vs Time After Instantaneous Coolant Loss

The total number of fission product nuclei released to the gap between the fuel and clad was determined from Blomeke and Todd, and the results of the experiment described in Section 8.5. The total quantity of Br, I, Kr, and Xe released to the gap in the B-ring fuel elements after continuous operation at 1000 kw for 500 days will be

$$N_i = 0.028 \times 3.444 \times 10^{21} = 0.964 \times 10^{20} \text{ atoms} \quad (48)$$

The number of gram-atoms in the gap is

$$n_{fp} = \frac{0.964 \times 10^{20}}{6.02 \times 10^{23}} = 1.601 \times 10^{-4} \text{ gram-atoms} \quad (49)$$

The partial pressure exerted by the fission products gases is

$$P_{fp} = n_{fp} \frac{RT}{V} \quad (50)$$

where, initially, the volume, V , is a 1/8 in. space between the end of the fuel and the top reflector end piece. This is quite conservative because the graphite reflector pieces have a porosity of 20% and the fission product gases can expand into the graphite. The initial volume, then, is

$$V_0 = \pi r^2 h = \pi (1.82)^2 0.317 \text{ cm}^3 = 3.30 \text{ cm}^3 \quad (51)$$

where the radius of the fuel, r , is 1.82 cm and the space width, h , is 1/8-inch (0.317 cm).

Thus, the initial pressure exerted by all the fission product gases is

$$P_{fp} = \frac{1.601 \times 10^{-4}}{3.30} RT = 0.485 \times 10^{-4} RT \quad (52)$$

The partial pressure of the air in the fuel element is

$$P_{air} = \frac{RT}{22.4 \times 10^3} = 4.46 \times 10^{-5} RT \quad (53)$$

and the total pressure exerted by the air and fission products is

$$P_r = \left(1 + \frac{P_{fp}}{P_{air}}\right) P_{air} = 1 + \left(\frac{0.485 \times 10^{-4}}{4.46 \times 10^{-5}}\right) P_{air} = 2.087 P_{air} , \quad (54)$$

Also,

$$P_{air} = 14.7 \frac{T}{273} \frac{V_O}{V} , \quad (55)$$

where P_{air} is in lb/in.², and T is in °K.

The equilibrium hydrogen pressure over the U-ZrH_{1.7} fuel material is simply a function of the fuel temperature. At a temperature of 780°C, the hydrogen pressure is about 2.05 atmospheres or 30 psi (see Fig. 8-3).

The total gas pressure at the maximum fuel temperature of 780°C is

$$P = P_H + P_r = 30 + 2.087 (14.7) \left(\frac{780 + 273}{273}\right) \frac{3.31}{V} = 30 + \frac{392}{V} \text{ psi} . \quad (56)$$

The tangential stress in the fuel element cladding when subjected to an internal pressure, P, is

$$S = Pr/t , \quad (57)$$

where r is the radius of the fuel element can (1.82 cm) and t is the wall thickness (0.051 cm) or

$$S = 35.7 P . \quad (58)$$

Under elastic conditions, the strain in the cladding is

$$\frac{\delta r}{r} = \frac{S}{E} , \quad (59)$$

where E is the modulus of elasticity. At a temperature of 780°C, $E = 19.7 \times 10^6$ psi for 304 stainless steel. *

* Steels for Elevated Temperature Service, United States Steel Corporation.

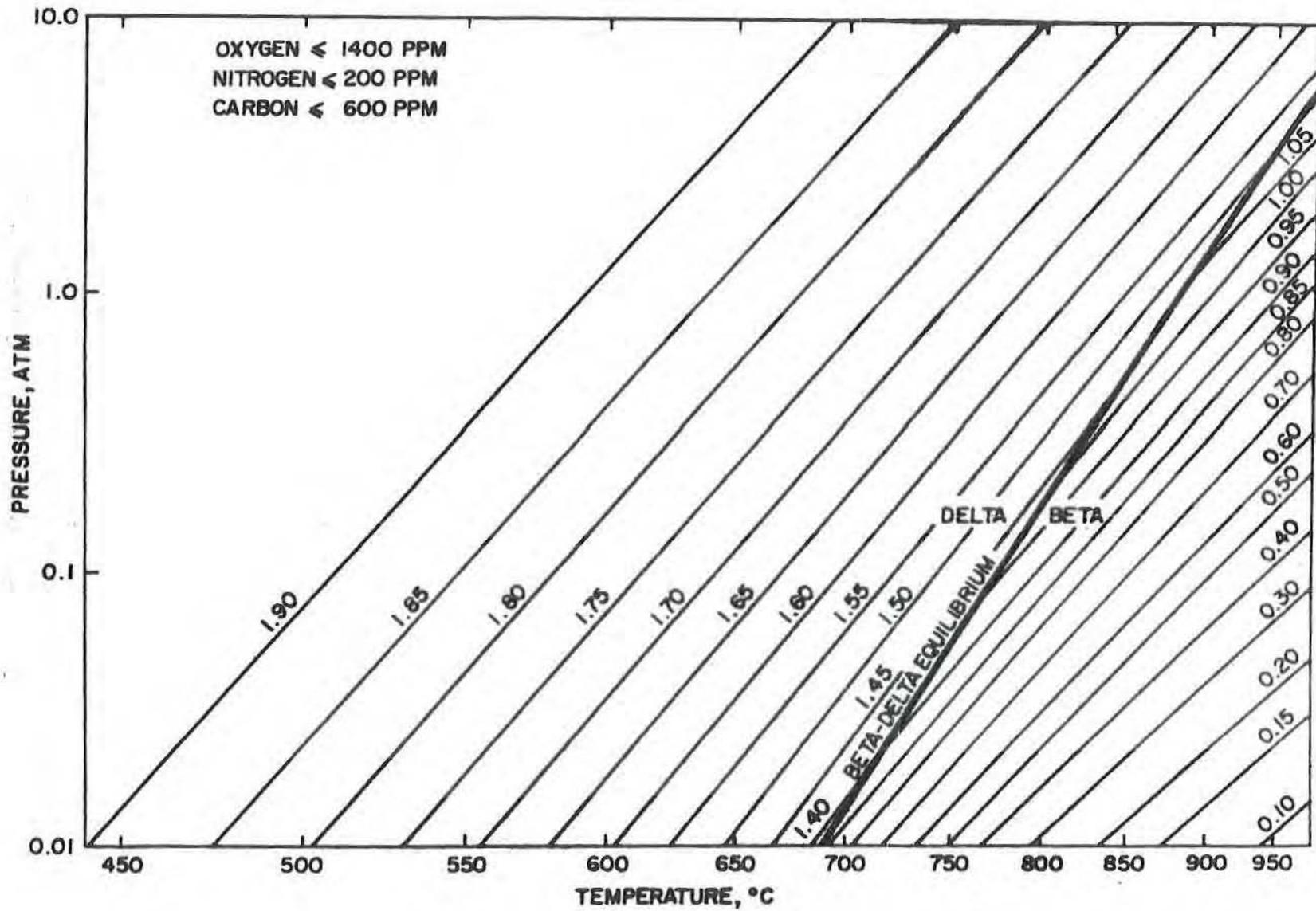


Fig. 8-3 --Hydrogen Pressure vs Temperature

Therefore,

$$\begin{aligned}\delta r &= \frac{Sr}{E} = \frac{1.82S}{19.7 \times 10^6} = 0.924 \times 10^{-7} S \\ &= 0.924 \times 10^{-7} (35.7 P) = 33.0 \times 10^{-7} P .\end{aligned}\quad (60)$$

Considering only the volume bounded by the active fuel and clad, the increased volume available for the released gases is then

$$V_1 = 2\pi (1.82) (38.0) \delta r = 436 \delta r \text{ cm}^3 ,$$

and, as

$$\delta r = 33.0 \times 10^{-7} P, \quad (61)$$

$$V_1 = 436 (33.0 \times 10^{-7} P) = 1.44 \times 10^{-3} P .$$

The total pressure then, is

$$P = \left(30 + \frac{392}{V} \right) = \left(30 + \frac{392}{V_o + V_1} \right) = \left(30 + \frac{392}{3.30 + 1.44 \times 10^{-3} P} \right) . \quad (62)$$

Rearrangement gives

$$1.44 \times 10^{-3} P^2 + (3.30 - 30 \times 1.44 \times 10^{-3}) P - (30 \times 3.30 + 392) = 0 ,$$

or

$$P^2 + 2.262 \times 10^3 P - 3.410 \times 10^5 = 0 . \quad (63)$$

The solution of this equation gives

$$P = 142 \text{ psi} .$$

The stress in the cladding is

$$S = 35.7 P = 35.7 (142) = 5070 \text{ psi} . \quad (64)$$

From the United States Steel handbook "Steels for Elevated Temperature Service," the yield stress for 304 stainless steel at 780°C is 12,300 psi and the ultimate strength is 24,800 psi.

Consequently, it is concluded that, subsequent to the loss of cooling water after prolonged operation at 1000 kw, the release of hydrogen from the fuel and the expansion of air and fission product gases in the gap between fuel and cladding will not result in the rupture of the fuel element cladding.

8.7. REACTIVITY ACCIDENT

The limitation on the size of the reactivity insertion permitted for routine operation of the aluminum-clad fuel for TRIGA reactors was determined by the temperature at which the $U-ZrH_{1.0}$ fuel undergoes the phase transition from alpha to beta phase, which results in dimensional changes in the fuel material. This temperature limitation was determined to be $530^{\circ}C$. Since the stable gamma phase of the $U-ZrH_{1.7}$ fuel material utilized in the TRIGA Mark III reactor does not undergo such a phase transition, the pulsing limits for these elements are set by the hydrogen equilibrium pressures within the fuel element. This pressure is a function of temperature and must not exceed the rupture stress of the fuel element cladding. For the 0.02-in. -thick stainless-steel cladding, the rupture pressure has been measured to be 1800 psi at $100^{\circ}C$. The fuel temperature at which the equilibrium hydrogen pressure will be 1800 psi is about $1000^{\circ}C$. The peak fuel temperature occurring during 1-Mw steady-state operation (about $350^{\circ}C$) and resulting from the routine reactivity insertions of 2.1% $\delta k/k$ from zero power (about $450^{\circ}C$) are both well below the limit.

The reactivity accident considered here would take place in the following manner. Initially, the reactor is "cold-clean" with all control rods inserted. The reactor is loaded with 4.9% $\delta k/k$, and the transient rod worth is 2.1% $\delta k/k$. This accident envisages someone deliberately violating the operating license and several interlocks and scrams.

In the first step, the operator slowly withdraws all the control rods except the transient rod until all the rods are completely out and the reactor is operating at a high steady power. From Fig. 8-4, which shows the steady-state power and fuel temperatures as a function of compensated reactivity

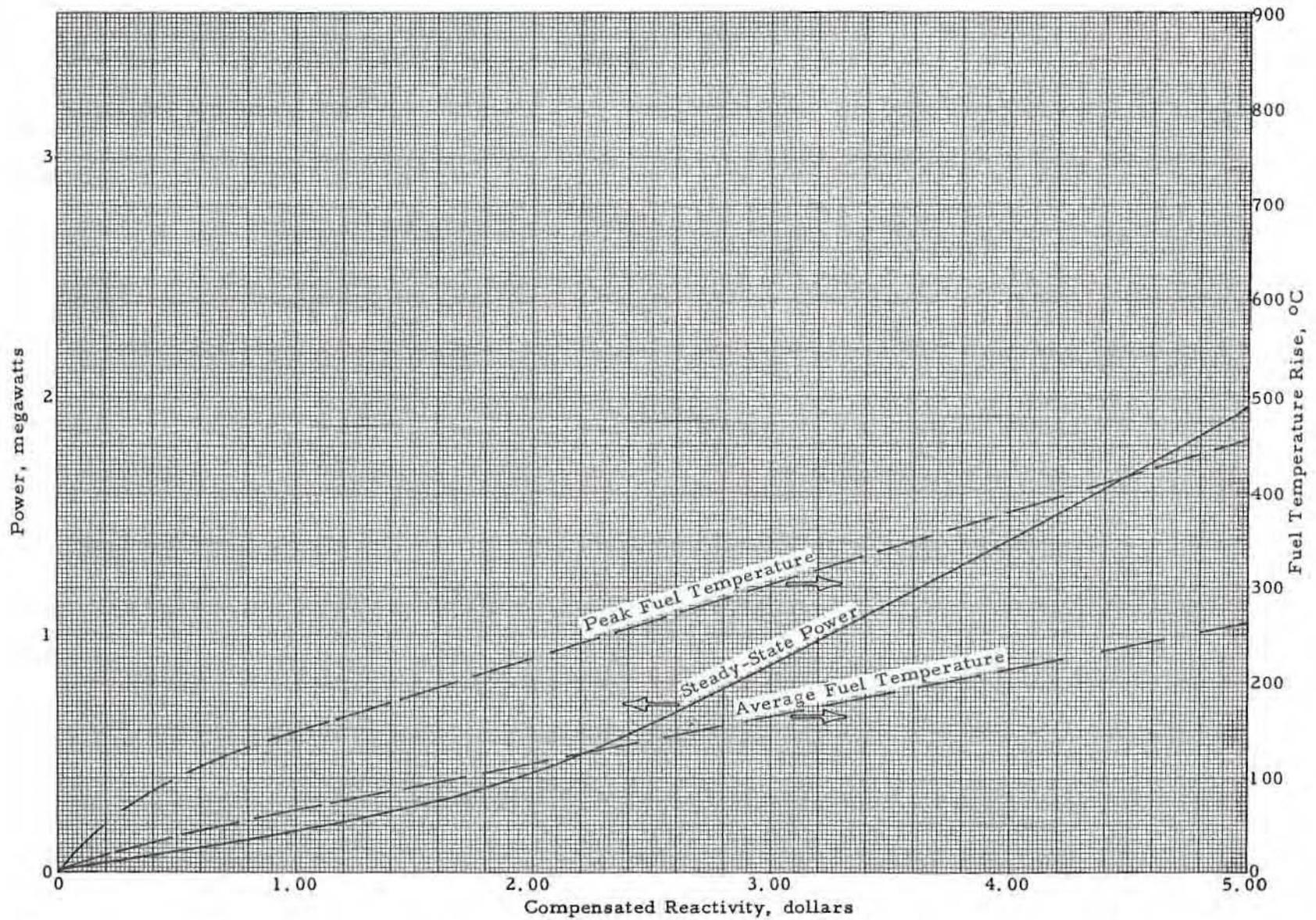


Fig. 8-4 -- Steady-state power and fuel temperature rise from 25°C versus compensated reactivity - TRIGA Mark III prototype reactor

as measured at the prototype TRIGA Mark III reactor, the steady-state power and fuel temperatures for a compensated reactivity of \$4.00 (that is $4.9\% - 2.1\% = 2.8\% = \4.00) is 1.4 Mw, 405°C peak fuel temperature, and 237°C average fuel temperature.

The second step in this accident takes place as the operator inserts the full worth of the transient rod by ejecting it from the hot reactor operating at 1.4 Mw. This prompt insertion of $2.1\% \delta k/k$ results in an average temperature rise in the core of 233°C and a peak temperature rise of 399°C . Thus, the average and peak temperatures at the conclusion of the pulse from 1.4 Mw are then 470° and 804°C , respectively, well below the maximum temperature permitted. The equilibrium hydrogen pressure over the $\text{ZrH}_{1.7}$ fuel resulting from this type of reactivity insertion would be about 60 psi, well below the rupture pressure of the fuel element clad (i. e., 1800 psi).

The computations leading to these conclusions are presented below.

The Fuchs-Nordheim model for reactor dynamics yields the coupled set of differential equations:

$$\lambda \dot{P} = (\rho - \alpha T) P \quad , \quad (65)$$

$$C \dot{T} = P - P_0 \quad , \quad (66)$$

with $C = C_0 + \gamma T$ (67)

where P = Power level (P_0 = initial power), w,

ρ = Reactivity above prompt critical,

α = Magnitude of the negative temperature coefficient, $^{\circ}\text{C}^{-1}$,

T = Temperature (average over fuel) above the equilibrium temperature at P_0 $^{\circ}\text{C}$,

λ = Prompt neutron lifetime, sec,

C = Heat capacity of the fuel in the core, w-sec/ $^{\circ}\text{C}$,

C_0 = Heat capacity at the equilibrium temperature corresponding to P_0 ,

γ = Rate of change of heat capacity with temperature, w-sec/ $^{\circ}\text{C}^2$.

The above lumped parameter system neglects heat transfer and delayed neutron effects and averages space and neutron energy variations so that all coefficients are assumed constant. Combining equations

$$\frac{dP}{dT} = \frac{(\rho - \alpha T)(C_0 + \gamma T)P}{\ell(P - P_0)} \quad (68)$$

Integrating, using the condition that $T = 0$ when $P = P_0$, yields

$$\ell \left[(P - P_0) - P_0 \ln \left(\frac{P}{P_0} \right) \right] = T \left[\rho C_0 + (\gamma \rho - \alpha C_0) T/2 - \frac{\alpha \gamma T^2}{3} \right] \quad (69)$$

Maximum (or minimum) temperatures occur when

$$\frac{dP}{dT} = 0 = P - P_0,$$

or when

$$P = P_0.$$

Then,

$$T \left[\rho C_0 + (\gamma \rho - \alpha C_0) \frac{T}{2} - \frac{\alpha \gamma T^2}{3} \right] = 0 \quad (70)$$

The roots of this equation are

$$\bar{T} = 0 = \frac{-3}{8} (\sigma - 1) \left\{ 1 \pm \left[1 + \frac{16}{3} \frac{\sigma}{(\sigma - 1)^2} \right]^{1/2} \right\} T_f \quad (71)$$

where $\sigma = \alpha C_0 / \gamma \rho$,

$$T_f = 2\rho / \alpha$$

and the positive sign is taken when $\sigma < 1$. Values of the parameters are

$$\begin{aligned} \alpha &= 1.34 \times 10^{-4} \text{ } ^\circ\text{C}^{-1} \\ C_0 &= \left[9.80 \times 10^4 + 1.92 \times 10^2 (T_{ss} - 25) \right] \text{ watt-sec/}^\circ\text{C (117 element core),} \\ \gamma &= 1.92 \times 10^2 \text{ watt-sec/}^\circ\text{C}^2, \end{aligned}$$

T_{ss} = Fuel temperature at P_0 .

For the TRIGA Mark III reactor, Fig. 8-4 shows that operating with a compensated reactivity of \$4.00 (2.8% $\delta k/k$), the power level, P_0 , is 1.4 Mw; the average fuel temperature, $\bar{T}_{ss} = (212 + 25)^\circ\text{C}$; and the peak fuel temperature is $\hat{T}_{ss} = (380 + 25)^\circ\text{C}$. Then

$$\begin{aligned} C_0 &= \left[9.80 \times 10^4 + 1.92 \times 10^2 (\bar{T}_{ss} - 25) \right] \text{w-sec}/^\circ\text{C} \\ &= 9.80 \times 10^4 + 1.92 \times 10^2 \cdot 212 = 1.39 \times 10^5 \text{w-sec}/^\circ\text{C} . \end{aligned}$$

The insertion of \$3.00 by ejection of the transient rod gives a value for $\rho = (\$3.00 - \$1.00) (0.7 \times 10^{-2}/\$) = 1.4 \times 10^{-2}$,

$$\sigma = \frac{1.34 \times 10^{-4} \cdot 1.39 \times 10^5}{1.92 \times 10^2 \cdot 1.4 \times 10^{-2}} = 6.91 ,$$

and

$$T_f = \frac{2 \cdot 1.4 \times 10^{-2}}{1.34 \times 10^{-4}} = 208.9^\circ\text{C} .$$

Thus,

$$\begin{aligned} \bar{T} &= \frac{-3}{8} (6.91 - 1) \left\{ 1 - \left[1 + \frac{16}{3} \frac{6.91}{(6.91 - 1)^2} \right]^{1/2} \right\} 208.9^\circ\text{C} \\ &= -2.21 \left\{ 1 - [1 + 1.05]^{1/2} \right\} 208.9^\circ\text{C} = -2.21 (-0.43) 208.9 = 198.5^\circ\text{C} . \end{aligned}$$

Therefore, at the conclusion of the pulse, the average fuel temperature will be

$$\bar{T}_f = \bar{T} + \bar{T}_{ss} = 198.5 + 212 + 25 = 435.5^\circ\text{C} .$$

To determine the maximum temperature in the hottest fuel element, the average energy release is determined and then multiplied by the peak-to-average power ratio to obtain the maximum energy release in the center element. Then one returns to the energy-temperature equation to determine the maximum temperature.

Let \bar{E} = the energy necessary to raise the average core temperature from \bar{T}_{ss} to \bar{T}_f , then

$$\begin{aligned}\bar{E} &= \int_0^{\bar{T}_f - \bar{T}_{ss}} C dt = \int_0^{\bar{T}_f - \bar{T}_{ss}} \left[C_{\bar{T}_{ss}} + \gamma T \right] dt = \left[C_{\bar{T}_{ss}} T + \frac{\gamma}{2} T^2 \right]_0^{\bar{T}_f - \bar{T}_{ss}} \\ &= C_{\bar{T}_{ss}} (\bar{T}_f - \bar{T}_{ss}) + \frac{\gamma}{2} (\bar{T}_f - \bar{T}_{ss})^2 = \left[9.80 \times 10^4 + 1.92 \times 10^2 (\bar{T}_{ss} - 25) \right] \\ &\quad (\bar{T}_f - \bar{T}_{ss}) + \frac{1.92 \times 10^2}{2} (\bar{T}_f - \bar{T}_{ss})^2 = \left[9.80 \times 10^4 + 1.92 \times 10^2 (212) \right] \\ &\quad (198.5) + \frac{1.92 \times 10^2}{2} (198.5)^2 = 1.39 \times 10^5 (198.5) + 0.0990 \times \\ &\quad \times 10^2 (198.5)^2 = 2.76 \times 10^7 + 0.39 \times 10^7 = 31.5 \times 10^6 \text{ w-sec} . \quad (72)\end{aligned}$$

The peak-to-average power ratio for the core during a pulse of this magnitude is about 2, so the energy release corresponding to the peak power is $(2)(31.5 \times 10^6) = 63 \times 10^6$ w-sec.

Rewriting the equation for the energy (72)

$$C_{\bar{T}_{ss}}^{\hat{}} (\hat{T}_f - \hat{T}_{ss}) + \frac{\gamma}{2} (\hat{T}_f - \hat{T}_{ss})^2 - \hat{E} = 0 .$$

Thus,

$$\hat{T}_f - \hat{T}_{ss} = \frac{-C_{\bar{T}_{ss}}^{\hat{}}}{\gamma} + \left[\left(\frac{C_{\bar{T}_{ss}}^{\hat{}}}{\gamma} \right)^2 + \frac{2\hat{E}}{\gamma} \right]^{1/2}$$

and,

$$\begin{aligned}C_{\bar{T}_{ss}}^{\hat{}} &= 9.80 \times 10^4 + 1.92 \times 10^2 (\hat{T}_{ss} - 25) = 9.80 \times 10^4 + 1.92 \times 10^2 \cdot 380 \\ &= 1.71 \times 10^5 \text{ w-sec/}^\circ\text{C} .\end{aligned}$$

Then,

$$\begin{aligned}\hat{T}_f - \hat{T}_{ss} &= \frac{-1.71 \times 10^5}{1.92 \times 10^2} + \left[\left(\frac{1.71 \times 10^5}{1.92 \times 10^2} \right)^2 + \frac{2 \cdot 63 \times 10^6}{1.92 \times 10^2} \right]^{1/2} \\ &= -8.91 \times 10^2 + \left[7.94 \times 10^5 + 6.56 \times 10^5 \right]^{1/2} \\ &= -8.91 \times 10^2 + 1.205 \times 10^3 = 314^\circ\text{C} ,\end{aligned}$$

and

$$\hat{T}_f = 314^\circ\text{C} + \hat{T}_{ss} = 314 + 380 + 25 = 719^\circ\text{C} .$$

8. 8. RADIATION LEVELS ABOVE REACTOR AFTER COOLING WATER LOSS

Even though the possibility of the loss of the shielding water is remote, calculations have been performed to evaluate the radiological hazard associated with this type of accident. The radiation dose rates given in Table 8. 4 are based on the assumption that the reactor has been operating for a long period of time at 1000 kw prior to the loss of all the shielding water. Time listed in the table is measured from shutdown. The first location considered is the top of the reactor pool, 20 ft above the unshielded reactor core. The second is also at the top of the reactor pool, but shielded from the direct radiation and subjected only to the scattered radiation from a thick concrete ceiling 9 ft above the top of the pool. The assumption of a thick concrete ceiling maximizes the reflected radiation dose.

An individual who does not expose himself to the core directly could work for about 4 hrs at the top of the reactor pool 1 day after shutdown, without receiving a dose in excess of the dose permitted by AEC regulations for a calendar quarter. The 4 hours would be sufficient time to view the interior of the pool with a mirror and to make emergency repairs.

Table 8.4
CALCULATED RADIATION DOSE RATES FOR
LOSS OF REACTOR POOL WATER ACCIDENT

Time after shutdown	Direct radiation, (r/hr)	Scattered radiation, (r/hr)
10 seconds	1.0×10^4	2.6
1 day	1.2×10^3	0.30
1 week	5.4×10^2	0.14
1 month	1.4×10^2	0.04

Method of Calculation

The core, shut down and drained of water, was treated as a bare cylindrical source of 1-Mev photons of uniform strength. Its dimensions were taken to be equal to those of the active core lattice. The source strength as a function of time was determined from Perkins and King's data* on fission product decay. No accounting was made of sources other than fission product decay gammas (i. e., activation gammas from the steel cladding and the aluminum grid plates) or of attenuation through the fuel element end pieces and the upper grid plate. The first of these assumptions is optimistic, the second conservative; the net effect is conservative. The conservative assumption of a uniformly distributed source of 1-Mev photons was balanced by not assuming any buildup in the core.

The direct dose rate at a point outside and on the axis of a cylindrical source is given by:

$$D_d = \frac{S_v}{K} \int_0^{x_c} \int_0^{r_c} \frac{2\pi r dr dx e^{-\mu_c z}}{4\pi R^2}$$

where S_v = Source strength in photons (1 Mev)/cm³-sec,
 K = Flux-to-dose conversion factor = 5.77×10^5 photons/cm²-
 sec per rad/hr,

$2\pi r dr dx$ = dV = Cylindrical volume element,

r_c = Core radius, cm,

x_c = Core height, cm,

R = Distance from volume element to receiver, cm,

z = Slant penetration in core $(x/a+x)R$, cm,

a = Distance from top of core to receiver,

μ = Core attenuation coefficient = 0.207 cm^{-1} .

*Perkins, J.F., and R. W. King, Reactor Handbook, Vol. III, Part B "Shielding," p. 32, 1962.

For distances far from the core (i. e., for $a \gg r_c$ and x_c) the above expression reduces to

$$D_d = \frac{S_v r_c^2}{4\mu_c a^2} (1 - e^{-\mu_c x_c})$$

The scattered dose rate was calculated from

$$D_s = 6.03 \times 10^{23} \rho \frac{Z}{A} \frac{I_0 C}{K(E)x^2} \frac{1}{\mu_0 + \mu_1 (\cos \theta_0 / \cos \theta_1)} \frac{\delta\sigma}{\delta\Omega} *$$

where ρ = Density of scattering material (concrete) = 2.3 g/cm³,

$\frac{Z}{A}$ = Ratio of average atomic number of atomic mass of the scatterer (=0.5),

$I_0 C$ = Incident current times cross section of beam, photons/sec,

$K(E)$ = Photon current to dose rate conversion for photon of energy E,

E = Energy of scattered photon, Mev,

x = Distance from scattering point to detector, cm,

μ_0, μ_1 = Attenuation coefficient in scatter for incident and scattered photons, cm⁻¹,

θ_0, θ_1 = Incident and scattered angle (measured from the normal to the scatterer),

$\delta\sigma/\delta\Omega$ = Differential Klein-Nishina scattering cross section, cm²/electron-steradian.

It was assumed that all of the source photons that "saw" the top of the reactor pool were incident normally to the concrete roof (i. e., $\theta_0 = 0$) at a point directly over the core, thus

$$I_0 C = S_0 \omega ,$$

* Stephenson, R., Introduction to Nuclear Engineering, 2nd Edition, McGraw-Hill Publishing Co., New York, 1958. p. 213.

where

$$S_0 = \frac{\pi r_c^2 S_v}{\mu_c} ,$$

$$\omega = \frac{1}{2\pi} \left\{ \sin^{-1} \left[\frac{y_0^2 (r_0^2 - x_0^2) + r_0^2 + x_0^2}{(r_0^2 + x_0^2) (r_0^2 + y_0^2)} \right] - \frac{\pi}{2} \right\} ,$$

and r_0 = Distance from the core to the top of the pool (~21 ft),

x_0 = Half-width of the pool (~5 ft),

y_0 = Half-length of the pool (~10 ft),

S_v , r_c , μ_c , have already been defined.

The energy of the scattered photons is given by

$$E = \frac{E_0}{1 + \frac{E_0}{0.51} (1 - \cos\theta)} ,$$

where E_0 is the incident photon energy (1 Mev) and θ is the scattering angle = $\pi - (\theta_0 + \theta_1)$.

The differential scattering cross section is given by

$$\frac{\delta\sigma}{\delta\Omega} = \frac{r_e^2}{2} \left[\frac{E}{E_0} - \left(\frac{E}{E_0} \sin\theta \right)^2 + \left(\frac{E}{E_0} \right)^3 \right] ,$$

where r_e is the classical electron radius = 2.818×10^{-13} cm.

9. NUCLEAR CENTER ORGANIZATION AND PROCEDURES

9.1 ORGANIZATION AND RESPONSIBILITY

The organization which shall be established for the administration and operation of the nuclear reactor at the Denver Federal Center is shown in Figure 9-1. Most of the personnel that will be utilized in setting up this organization are already employed by the U. S. Geological Survey in other capacities.

9.1.1 Reactor Facility Administration

The facility administration will be responsible for and in complete charge of the reactor facility. This responsibility will include establishing budgets and policy, reporting to the AEC and accepting responsibility for the safe and legal operation of the facility. The administration will be a committee composed of representatives of the Water Resources Division, Geologic Division, and others as may be determined by the Director of the Geological Survey. The facility administration will be guided by the Health Physicist and the Reactor Operations Committee. The Health Physicist will furnish advice on the radiological safety of any operation connected with the reactor facility, and the Operations Committee will review the safety of the reactor and associated experiments.

9.1.2 Reactor Facility Supervisor

This position will be filled by someone with experience in the supervision and operation of a reactor facility. He will be responsible for all safety and operational aspects of the facility. He will determine scheduling and review all experiments, and will work in close liaison with the Health Physicist and the Reactor Operations Committee. He will also be a licensed Senior Reactor Operator and will direct the Reactor Operator Training Program. He will be in administrative charge of all

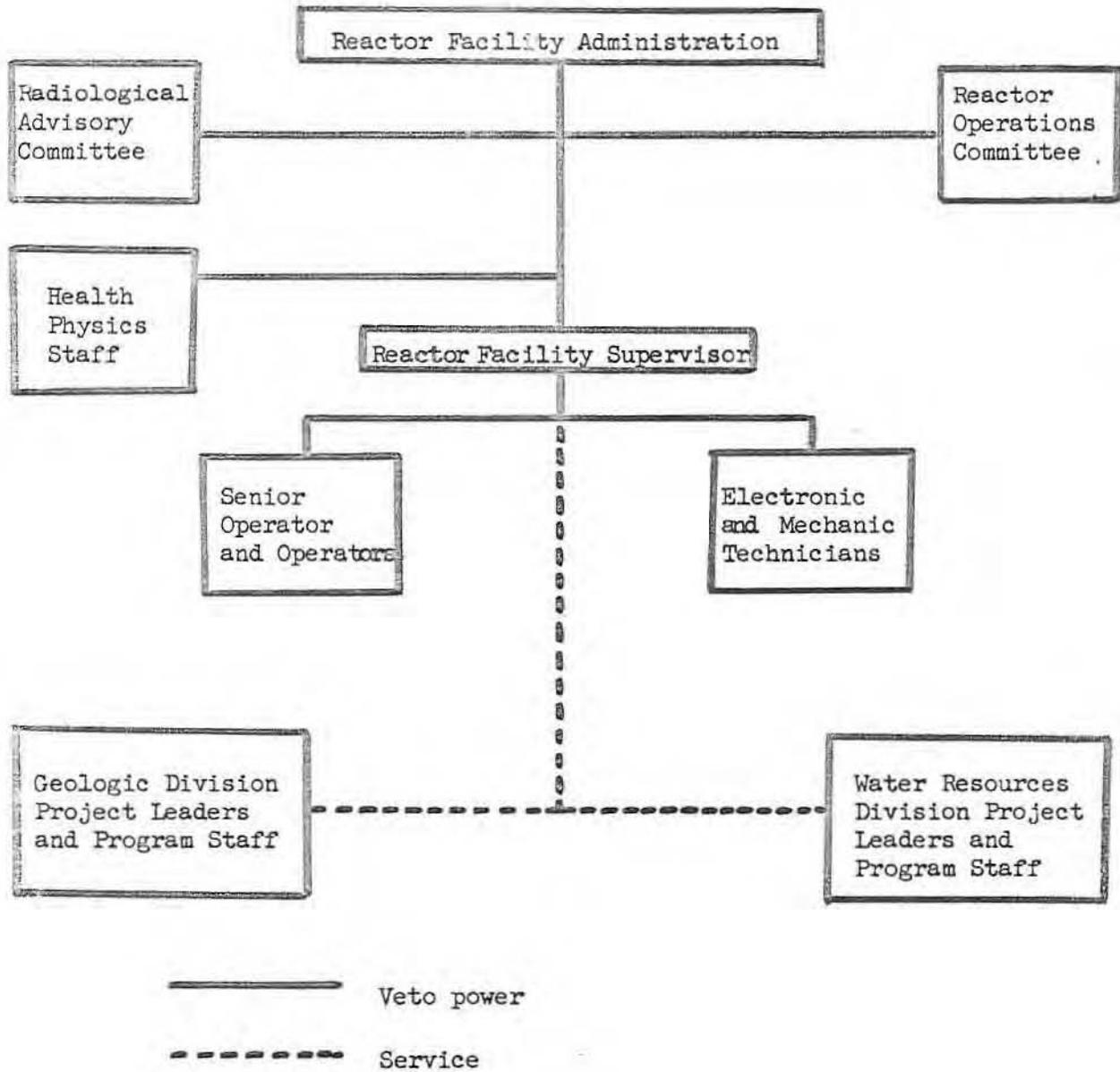


Fig. 9-1. Organization Chart for Administration and Operation of U. S. G. S. Reactor

operators and technicians. He will be responsible for maintenance of reactor records and physical inventory.

9.1.3 Senior Operators and Operators

All reactor operators will be licensed by the U. S. Atomic Energy Commission to operate this reactor. The first reactor operator will be trained to operate the TRIGA reactor, probably at the General Atomic facility in San Diego, California. All reactor operators will assist in the training of new operators prior to their examination for the reactor operating license.

9.1.4 Electronic and Mechanic Technicians

The reactor maintenance personnel will perform routine maintenance and repair work on the reactor and its associated equipment.

9.1.5 Health Physicist Technician

A Health Physicist Technician will be assigned to the Nuclear Science Building. He will advise the Reactor Facility Supervisor on operations in the facility and be responsible for radiation safety in all Nuclear Science Building laboratories. The Health Physicist Technician will also supervise the transfer of all radioactive materials. The Health Physicist Technician would derive his veto authority from the Radiological Safety Committee.

9.1.6 Radiological Advisory Committee

The Radiological Advisory Committee advises on the radiological safety of any operation of the Geological Survey at the Denver Federal Center involving radioactive materials. This committee presently exists and is comprised of Geological Survey personnel at the Denver Federal Center. This committee is available for consultation whenever the Health Physicist Technician or reactor supervisor deems it advisable.

The Committee enforces the regulations in Appendix 9-I which are a part of the Survey's AEC Broad License S-1399-8.

9.1.7 Reactor Operations Committee

The Reactor Operations Committee provides an additional measure of security in the reactor operation. It will review the hazards associated with the operation of the Nuclear Reactor and experiments. It is anticipated that this Committee will be composed of Geological Survey employees, personnel from National Laboratories of the AEC and faculty members from universities with reputable nuclear engineering departments. This Committee will be consulted when there is any probability of a hazard due to operation of the reactor or an experiment involving the reactor and whenever else the Reactor Supervisor deems it advisable.

9.1.7.1 Committee Jurisdiction

The domain of jurisdiction of the Committee is the safety evaluation of incore nuclear reactor experiments and the periodic review and evaluation of the physical integrity of the core. The Committee will be concerned with those experiments which by their unusual nature, inherent hazard or unprecedented complexity could endanger health, life, and property in and about the Nuclear Reactor Facility. These experiments are further identified in Section 9.2.4 as Class II.

It is not the purpose of the Committee to evaluate or consider conventional administrative or operational functions at the Nuclear Reactor Facility. Neither is it the purpose of the Committee to assess hazards which are not directly associated with the immediate reactor core environs or fuel storage vaults. However, should a particular reactor-oriented experiment contain unusual dangers in addition to those listed above, which

in the event of an accident could cause considerable unfavorable public reaction against nuclear reactors in general, this experiment must be submitted to the Committee for evaluation. Approval of an experiment by the Committee, however, does not obligate the reactor staff to carry out the experiment.

Generally, any experiment for which adequate precedent already exists to guarantee safety would not be referred to the Committee. Such experiments are further characterized in Section 9.2.4 as Class I. It is expected that the day-to-day in-core experiments will proceed by approval of a member of the senior reactor staff. Questionable experiments are to be referred by or through the Reactor Facility Supervisor to the Reactor Operations Committee for approval. Records are to be kept by the reactor staff listing all in-core experiments for periodic review by the entire Committee to insure that the intent and function of the Operations Committee is being maintained.

The Committee will aid the reactor staff with the review and evaluation of any indications that changes are taking place in the core which influence its integrity.

If a situation exists in the reactor which indicates a possibility of unsafe operation, the Committee by a majority vote, can recommend that the reactor be shut down. The reactor staff retains the authority to shut down the reactor without such a recommendation. In the event of a difference of opinion between the staff and the Committee, the Committee recommendation to shut down will govern. The matter will be referred immediately to the Office of the Director, U. S. Geological Survey for appropriate action.

Startup after a shut-down recommended by the Committee shall have the Committee's approval prior to such startup.

The reactor staff has the responsibility to report immediately or cause to be reported to the AEC in writing any indication or occurrence of a possible unsafe condition relating to the reactor. This action can be taken without concurrence of the Committee, however, all committee members should be advised of the action.

9.1.7.2 Structure of the Committee

The Reactor Operations Committee is to be composed of not less than three members, including the chairman. The Reactor Supervisor shall be a member of the Committee but shall not be chairman. All members of the Committee are to be knowledgeable in subject matter related to reactor operations.

The Committee is to be appointed by the Director, U. S. Geological Survey. No definite term of service is specified but should a vacancy occur in the Committee, the Director is to appoint a replacement subject to approval of the candidate by a majority of the other remaining members of the Committee.

9.1.7.3 Details of Meetings

Meetings will be called by the Chairman whenever an experiment is being delayed awaiting approval. Quarterly meetings will be held to review experiment records of the reactor and to discuss physical integrity of the core with the reactor staff. Records of all experiments will be made available to the Committee for examination. Two-thirds of the members of the Committee will constitute a quorum.

9.2 ADMINISTRATIVE CONTROLS AND PROCEDURES

9.2.1 Introduction

This section describes the controls and procedures pertinent to the operation of the facility. All new operations will be thoroughly reviewed by the Reactor Supervisor who may also refer these operations to the Reactor Administrator, the Reactor Operations Committee, and the Radiological Advisory Committee for comment. Existing operating procedures will be reviewed periodically to bring them up to date. The controls and procedures are based on predicted operating conditions of the reactor facility. If these conditions should change, the Reactor Staff will advise the Reactor Operating Committee and make recommendations for improving operating procedures. Operating procedure changes can be made if at least two-thirds of the Reactor Operations Committee agree with the recommendations.

9.2.2 Reactor Facility Access

Normal access to the facility is via the area control center where personnel radiation monitoring equipment will be issued. After receiving a radiation dosimeter, the individual passes through a door to the reactor room which is normally closed. In case of some emergency, the laboratory can be evacuated through the truck access door which will normally be closed and locked. Normal exit from the reactor facility is through the door leading to the area control center. All personnel who enter the reactor room will be required to wear radiation monitoring devices.

9.2.3 Reactor Operations

A Senior Reactor Operator in charge will be designated for each reactor operation. He may be also the reactor operator, or he may be otherwise engaged so long as he is continuously on call within the building. However, at least

one other person besides the reactor operator must be present in the room at all times during reactor operation.

Before each day's reactor operations are begun and at the end of each operating day, a checklist similar to the one shown in Figure 9-2 will be completed. Each month a reactor inspection will be made to insure that the reactor is in proper operating condition. Changes in reactor core loading in reactor experiments will be recorded in the Reactor Log Book.

REACTOR STARTUP CHECKLIST

Operator _____ Supervisor _____ Date _____

Radiation Detectors

Area Monitor calibrated _____ Alarm set _____ Background _____
 Stack Gas Monitor calibrated _____ Alarm set _____ Background _____
 Particulate Monitor calibrated _____ Alarm set _____ Background _____

Auxiliary Systems

Water: Demineralizer Flow Rate _____ GPM Forced water flow rate _____ GPM
 Temperature _____ °C Conductivity _____ μmhos Background _____
 Transient Rod: Upper limit _____ cm Worth \$ _____ Air pressure _____ psi
 Lazy Susan check _____

Reactor

Visual Inspection _____ All rods down _____

Scram checks

Each rod raised
 from down limit
 and scrammed by
 one of following:

- Scram Circuits
- (1) Manual
 - (2) Linear recorder
 - (3) % power
 - (4) Period
 - (5) Fuel temp # 1
 - (6) Fuel temp # 2
 - (7) Keithley Ramp.

Control Rods

Safety _____

Shim _____

Regulating _____

Transient _____

Instrument Calibrations

Count rate channel _____
 Period _____
 Linear recorder _____
 Log recorder _____

Today's Power Calibrations

% Power _____ Kw
 Keithley amp. _____ Mw
 Linear recorder _____ Mw
 Log recorder _____ Mw

Adjustable scram, alarm, interlock levels properly set _____
 Source Induced Scram Safety raised and scrammed by source _____

Chamber and Instrument Sensitivity

Maximum readings with:

	Today	Source Adjacent Previous	Source In Today
Linear recorder	_____	_____	_____

REACTOR SHUTDOWN CHECKLIST

All rods down _____ Water temp. _____ °C Area monitors operative _____
 Visual inspection of reactor _____
 Console power off, keys removed from console _____
 Lazy Susan: Loaded _____ Empty _____
 Irradiation Request Form No(s). for samples _____

 Reactor Operator

Fig. 9-2 -- Startup and Shutdown Checklists

9.2.4 Authorization of Experiments

All experiments proposed for the reactor will be either Class I or Class II experiments. The classification of the proposed experiments will be the responsibility of the Reactor Supervisor.

Class I experiments include all experiments that have been run previously or that are minor modifications to a previous experiment. The Reactor Supervisor has the authority to approve the following:

1. Experiments for which there exists adequate precedence for assurance of safety.
2. Experiments which represent less than that amount of reactivity worth necessary for prompt criticality.
3. Experiments in which any significant reactivity worth is stable and mechanically fixed; i.e., securely fastened or bolted to the reactor structure.

Class II experiments include all new experiments and major modifications of previous experiments. These experiments must be reviewed and approved by the Reactor Operations Committee before being run. The Radiological Safety Committee may also be consulted. These include:

1. In-core experiments which involve, in an unstable form, reactivity worth greater than that necessary to produce a prompt critical condition in the reactor core, i.e., deliquescent or high vapor pressure poisons.
2. Experiments, involving corrosive chemicals, pressures or temperatures, which if failure should occur could endanger the safety of the reactor core.

3. Dynamic experiments which could introduce appreciable reactivity worth into the reactor by failure or malfunction. Included in this group are circulation systems which operate in or at the core and by which if a failure occurred, the core could be damaged.
4. Experiments which are dynamically coupled to the reactor core and together function as a system, i.e., to measure nuclear absorption cross sections, or study transient responses.
5. Experiments which interfere in any way with the normal function of any of the reactor safety circuits.
6. Experiments which could produce radiation levels sufficient to cause serious personnel radiation injury.
7. Experiments which by their unusual hazard could produce conventional injury or death and thereby reflect unfavorably on reactor operations.

Both classes of experiments will require submittal of the information described in the following paragraph, primarily to provide adequate data to make a hazards analysis.

9.2.4.1 Irradiation Authorization Form

The information form (fig. 9-3), which is subject to change is to be provided with each sample or group of virtually identical samples. When the format provided is not adequate, additional information will be added.

A. General

1. The form may be handwritten, if done legibly and permanently.
2. A form will be completed for each separate capsule or experiment, unless virtually identical items are to be irradiated on the same day.

FIGURE 9-3 U.S.G.S. TRIGA REACTOR

IRRADIATION AUTHORIZATION FORM

1. Name _____
2. Organization _____
3. Phone _____
4. Exposure Date _____
5. Sample Identification _____
- No. _____
6. Permit No. _____
7. Reference _____
8. Encapsulation method _____
9. Nuclear Conditions _____
10. Location _____
11. Chemical Reactions _____
12. Thermal Analysis _____
13. Reactivity worth: Normal _____ Maximum _____
14. Special Procedures _____

B. Specific - Numbers below correspond to items on the Irradiation Authorization Form.

1 to 4 - these should be obvious.

5. Sample Identification - this space can be used to record any identification desired by experimenter or facility and the number of such samples.

6. Permit No. - Issued by Radiation Advisory Committee under terms of U.S.G.S. Broad License for Byproduct Materials.

7. Reference - If experiment is within the limits of one previously approved, item 8 through 18 need not be completed, but positive reference to the earlier approval must be given here.

8. Encapsulation Method - Describe completely.

9. Nuclear Conditions - Give dose required in most general manner possible.

10. Location - Give location desired in reactor, and any special environment or experimental facility.

11. Chemical Reactions - Any tendency which the sample may have to decompose, offgas, or react in any way should be given.

12. Thermal Analysis - Heating problems generally do not occur in the Triga reactor. However, where a high density material is to be irradiated which will be thermally insulated, or where a fueled sample is involved, evidence should be presented that excessive temperatures will not be reached.

13. Reactivity Worth - "Normal" refers to that associated with insertion of the experiment; "Maximum" means the change that could result from a conceivable failure.
14. Special Procedure - Give any special requirements.
15. Columns A, B, and C must be completed by the experimenter. Facility personnel will provide assistance in completing D and E if required, although the experiment may be expedited if the experimenter does this work. Where complex combinations of material are involved, discussion with the reviewer may simplify this item.
16. Technical Specifications impose limits on the quantity of activity which may be produced in any one capsule. However, depending on the nature of the experiment, a great variety of approaches can be used to demonstrate compliance. This section should be used to give the basis for approval in this regard.
17. Reactor Operations Committee and Radiological Advisory Committee Review - These reviews are required when new experiments or modifications of experiments are made and when radiation intensities approach limits imposed by the user permits (Item 6). In this space the reviewer will either indicate that such review is not required, or make reference to the Reactor Operations Committee document which authorizes the experiment.
18. Special Requirements - The reviewer will give any special tests or procedures required, such as a reactivity check.

19. Approved by - Approval will be given by a person designated by the Reactor Operations Committee to be an authorized reviewer.

9.2.5 Authorization of Reactor Modifications

Approval for a modification of the reactor within the technical specifications can be obtained from the Reactor Supervisor. Other modifications will require authorization from the U. S. Atomic Energy Commission.

9.3 HEALTH PHYSICS

9.3.1 Radiation Monitoring

All situations in which there is a potential for contamination with radioactive material or exposure to ionizing radiation will be monitored. Monitoring systems operated on 115v ac power will be connected to the emergency power supply to provide continuous measurements of radiation intensities. The monitoring program shall be under the direction of the Health Physicist and consists of the following types:

a. Personnel. A film badge service will be provided to the laboratory through the Health Physicist. The badges shall have both neutron and gamma sensitive films and shall normally be processed on a monthly basis; however, immediate processing will be available in cases of suspected overexposure. Pocket dosimeters shall be used to supplement film badges. These dosimeters will be used for occasional visitors and will be issued by the reactor control office under the direction of the Health Physicist.

b. Area. Areas where the radiation field might reach a hazardous level will be monitored with remote area monitoring detectors. These detectors will be permanently-mounted gamma sensitive Geiger-Muller tubes or ionization chambers. The readout from these detectors will be on meters and/or visual alarms located at the console and the Health Physics Office. These alarms will not be interlocked with the ventilation system. Areas which will be monitored on a continuing basis are: the top of the reactor, the reactor console, the entrance to the reactor laboratory, and the room exhaust air after it passes through the absolute filter bank. Other areas in the Nuclear Science Building will

be monitored with systems described in the specifications in Appendix 9-II.

c. Stack Gas. One of the remote area monitoring system detectors will be positioned in the plenum behind the room exhaust filters to detect releases of radioactive gases which might occur in the event of a serious accident. This monitor will be interlocked with the emergency ventilation system described in Section 4.5 and will provide the actuating signal for closing the butterfly valves. The readout will be near the reactor console.

Another stack gas monitor will be located in the exhaust duct. This will be a sampling type of instrument for detecting very small increases in the radioactive gas concentration. A continuous readout will be provided for recording the response of the radiation detector.

d. Portable. The laboratory will be supplied with a variety of portable monitoring equipment, including Geiger-Müller thin window detectors, alpha proportional counters, BF_3 fast and slow neutron counters, and tritium ion chambers.

e. Equipment. All equipment used with the reactor will be monitored for contamination and activation. No radioactive material will be removed from the laboratory without permission of the Health Physicist and according to the appropriate 10 CFR regulations and the Radiological Advisory Committee's recommendations. All radioactive material will be stored in one of the areas designated for this purpose, which include the reactor pool, the deep water pool outside of the reactor, and the Radioisotope Storage Room.

9.3.2 Waste Disposal

Disposal of radioactive wastes from the U. S. Geological Survey shall be made through the Rocky Flats facilities operated by the Dow Chemical Company. Collection, packaging and labeling of wastes shall be in accordance with AEC regulations. The Health Physicist will be responsible for proper waste disposal procedures. Individuals involved in experimental work will be responsible for advising the Health Physicist concerning the type and amount of activity they will have for disposal. Generally, liquid and solid wastes will be collected and stored in the Radioisotope Storage Room prior to delivery to Rocky Flats. Fuel storage racks will be located on the side of the reactor pool and will conform to the limitations imposed by the Technical Specifications in regard to criticality. A deep water tank is provided in the Radioisotope Storage Room for the storage of coffins or shipping casks prior to shipment.

9.3.3 Radiation Areas

All areas containing radiation or contamination for which precautionary measures are required will be periodically monitored. Survey reports will be maintained by the Health Physicist as permanent records in the Health Physics Office.

9.3.4 Health Physics Training

All individuals associated with the health physics functions of the facility will be given instruction in the following subjects:

- a. The rules and regulations of the AEC, the State of Colorado, and the U. S. Geological Survey.
- b. Personnel dosimetry, film badge assignment, and time of pick up.
- c. Survey instrumentation, types of instruments, when and where to use.
- d. Contamination control and reporting.

- e. Waste disposal, containers to be used, labeling, storage and collection.
- f. Emergency procedures, types of accidents that are possible, meaning of alarms, and steps to be taken.

9.4 EMERGENCY PLANS AND PROCEDURES

The most probable causes of emergencies which will require implementation of emergency procedures are fire and release of anomalous radioactivity. Any or all of the following will be notified immediately, depending on the particular emergency situation:

1. Denver Federal Center Fire Department will be called on all fires and on radioactivity releases which might require special equipment or additional personnel for traffic control, building evacuation, or security.
2. Reactor Facility Supervisor and Senior Operator will be notified on all emergency situations.
3. Reactor Health Physicist will be notified on all situations which might affect reactor operations.

9.4.1 Fire

The reactor will not be operated during an abnormal fire within the Nuclear Science Building. Ventilation fans and nonessential electrical circuits in the reactor facility will be turned off. Carbon dioxide extinguishers will be used when possible. If fuel-element loading is in progress, all special nuclear material will be returned to storage areas. The person in charge of the reactor will be responsible for determining the appropriate action if a fire occurs outside of the Nuclear Science Building.

9.4.2 Release of Radioactivity

If monitoring systems indicate an abnormal release of radioactivity, the operator will immediately scram the reactor and shut down the ventilation system. Radiation levels greater than 5 R/hr will require evacuation of all personnel from the reactor area. Levels less than 5 R/hr will require evacuation of all

nonessential personnel while the cause of the indicated release is investigated. A report on the occurrence, including cause, remedial action, recommendations for preventing similar problems, and personnel exposures will be submitted by the Reactor Facility Supervisor to the Reactor Facility Administration.

APPENDIX A
License R -
TECHNICAL SPECIFICATIONS FOR THE

U.S. Geological Survey TRIGA REACTOR

Date: AT

~~The U. S. Geological Survey~~

The dimensions, measurements, and other numerical values given in these specifications may differ from measured values owing to normal construction and manufacturing tolerances, or normal accuracy of instrumentation.

A. Definitions

1. Shutdown

The reactor, with fixed experiments in place, shall be considered to be shut down (not in operation) whenever all of the following conditions have been met: a) the console key switch is in the "off" position and key is removed from the console and under the control of a licensed operator (or stored in a locked storage area); b) sufficient control rods are inserted so as to assure the reactor is subcritical by a margin greater than 0.7% β_k/k cold, without xenon; c) no work is in progress involving fuel handling or refueling operations or maintenance of its control mechanisms.

2. Steady State Mode (SS)

Steady state mode shall mean operation of the reactor at power levels not to exceed 1 megawatt utilizing the scrams in Table I and the interlocks in Table II. However, for the purpose of testing

the 110% full power safety circuits, an exception shall be made to allow the reactor to be operated at power levels not to exceed 1.15 megawatts during the testing period.

3. Pulse Mode

Pulse mode shall mean operation requiring the use of the scrams in Table I and the interlocks in Table II to assure that no more than one rod is pneumatically withdrawn to produce power pulses.

4. Square Wave Mode (SW)

Square wave mode shall mean operation requiring use of the scrams in Table I and the interlocks in Table II to assure that no more than one rod is pneumatically withdrawn and that the transient power level does not exceed 1.1 megawatt and the steady state power does not exceed 1 megawatt.

5. Operable

A system or component shall be considered operable when it is capable of performing its intended functions. ~~in its normal manner.~~

6. Experiment

Experiment shall mean: ^(a) any apparatus, device, or material installed in the core or experimental facilities (except for underwater lights, fuel element storage racks and the like) which is not a normal part of these facilities ^{(b) any operation ~~designed~~ to measure reactor parameters or characteristics.}

7. Experimental Facilities

Experimental facilities shall mean ^{The} rotary specimen rack, vertical tubes, pneumatic transfer systems, central thimble, and in-pool irradiation facilities.

8. Reactor Safety ^{Systems} ~~Circuits~~

Reactor safety ^{systems,} ~~circuits~~ shall mean those ~~circuits~~, including their associated input circuits, which are designed to initiate a reactor scram.

B. Site

The minimum distance from the center of the reactor pool to the boundary of the exclusion area shall be 25 feet.

C. Reactor Building

1. The reactor shall be housed in a closed room designed to restrict leakage. The minimum free volume in the reactor room shall be 13,000 cubic feet.
2. All air or other gas exhausted from the reactor room and from associated experimental facilities during reactor operation shall be released to the environment at a minimum of 21 feet above ground level.

D. Reactor Pool and Bridge

1. ~~Corrective action shall be taken if during reactor operation the~~ ^{The reactor shall not be operated if the} pool water level is less than 16 feet above the top grid plate. The bulk pool temperature shall be monitored while the reactor is in operation and ~~corrective action shall be taken~~ ^{the reactor shall be shut down,} if the temperature exceeds ~~120°F.~~ ^{60°C.} The reactor core shall be cooled by natural convective water flow.
2. The pool water shall be sampled for conductivity at least weekly. Conductivity averaged over a month shall not exceed 5 micromhos per centimeter.

E. Reactor Core

1. The core shall be an assembly of TRIGA Mark III stainless steel clad fuel-moderator elements arranged in a close-packed array except for (1) replacement of single individual elements with in-core irradiation facilities or control rods; (2) two separated experiment positions in the D through E rings, each occupying a maximum of three fuel element positions. The reflector (excluding experiments and experimental facilities) shall be water or a combination of graphite and water.
 2. The ~~maximum available~~ excess reactivity above cold critical, ^{not exceed} without xenon, shall ~~be~~ ^{be} 4.9% $\delta k/k$ with experiments in place.
 3. Fuel temperatures near the core midplane in either the B or C ring of elements shall be ^{continuously} recorded during the pulse mode of operation. The reactor shall not be operated with this measured fuel temperature greater than 800°C.
 4. ^{change} The reactor shall be pulsed semi-annually with an insertion of at least 1.5% $\delta k/k$ to compare fuel temperature measurements and peak power levels with those of previous pulses of the same reactivity value.
 5. Each standard fuel element shall be checked for transverse bend and longitudinal elongation after the first 100 pulses of any magnitude and then after every 500 pulses or annually, whichever comes first.
- The limit of transverse bend shall be 1/16-inch over the total length of the cladding portion of the element (excluding end fittings). The

limit on longitudinal elongation shall be 1/10-inch. The reactor shall not be pulsed with elements which have been found to exceed these limits.

Any element which exhibits a clad break as indicated by a measurable release of fission products shall be located and removed from service before continuation of routine operation.

F. Control and Safety Systems

1. The standard control elements shall have scram capability and the poison section shall contain borated graphite, ~~B₄C powder~~, or boron and its compounds in solid form as a poison in an aluminum or stainless steel clad.
2. The control elements shall be visually inspected at least once every two years. If indication of significant distortion or deterioration is found, the element(s) will be replaced.
3. Only one pulsing control element may be used in the core. The poison section of this element shall contain borated graphite or boron and its compounds in a solid form as a poison in an aluminum or stainless steel clad. The pulse rod shall be designed to release and fall upon initiation of a scram signal. The maximum reactivity worth of the element fully inserted by the drive in relation to fully withdrawn shall be equal to or less than 2.9% $\delta k/k$.
4. A pulse will only be initiated when the reactor is at a power less than 1 kw. Pulsed reactivity insertion shall not exceed 2.1% $\delta k/k$.
5. The minimum shutdown margin (with fixed experiments in place) provided by operable control elements (including the transient rod) in the cold condition, without xenon, with the most reactive of the operable control elements withdrawn shall be ^{0.4%} ~~0.2%~~ k/k .

6. The maximum rate of reactivity insertion associated with movement of a standard rod shall be no greater than $0.20\% \delta k/k/sec.$

7. The type and minimum number of safety ^{Systems} ~~circuits~~ which shall be operable for reactor operation are shown in Table I.

8. The type and minimum number of interlocks which shall be operable for reactor operation are shown in Table II.

9. The reactor instrumentation channels and safety ^{Systems} ~~circuits~~ for the intended modes of operation as listed in Table I shall be ~~checked~~ ^{verified} to be operable at least once each day the reactor is operated unless the operation extends continuously beyond one day, in which case the ^{operability} ~~instrumentation channels~~ need only be ~~checked~~ ^{verified} prior to beginning the extended operation.

10. Following maintenance or modification of the control or safety

11. systems, an operational ^{test} ~~check~~ or calibration of the associated system shall be performed before the affected system is to be considered operable.

12. The tests listed below shall be performed at least once semi-annually,

with the exception that if the reactor is operating continuously, the tests shall be performed after the first shutdown if this occurs more than six months after the previous tests:

a. A functional ^{test} ~~check~~ of all reactor interlocks.

b. Verification that all control element drop times are less than one second (two seconds for pulse element). If drop time is found to be greater than this, the element shall not be considered operable.

12. A licensed reactor operator shall be present during maintenance of the reactor control and safety systems. 1-1-7

c. A test of the power level safety circuits.

d. A functional test of the ventilation system interlocks, ~~the~~ ^{the} ~~shyl.~~ ^{shyl.}

~~The linear power level channel shall be calibrated at least annually~~

e. ~~by~~ ^A thermal power calibration of the linear power channel, ~~annually~~

G. Radiation Monitoring

1. The radiation levels within the reactor laboratory shall be monitored by at least one area radiation monitor during reactor operation or when work is done on or around the reactor core or experimental facilities. The monitor shall have a readout and provide a signal which actuates an evacuation alarm. During short periods of repair to this monitor, reactor operations may continue while a portable gamma-sensitive ion chamber is utilized as a temporary substitute.
2. A continuous air monitor with readout and audible alarm shall be operable in the reactor room when the reactor is operating.
3. The alarm set points for the above radiation monitoring instrumentation shall be ^{verified} ~~checked~~ at least once a week. This instrumentation shall be calibrated at least once a year.

H. Fuel Storage

1. All fuel elements or fueled devices shall be rigidly supported during storage in a safe geometry (k_{eff} less than 0.8 under all conditions of moderation).
2. Irradiated fuel elements and fueled devices shall be stored in an array which will permit sufficient natural convection cooling such that the fuel element or fueled device temperature will not exceed design values.

I. Administrative Requirements

1. The facility shall be under the direct control of the ~~Facility Director~~ ^{Reactor Supervisor}.
He shall be responsible to the Reactor Facility Administrator ^{or} ~~at~~ for safe operation and maintenance of the reactor and its associated equipment. He or his appointee shall review and approve all experiments and experimental procedures prior to their use in the reactor. He shall enforce rules for the protection of personnel against radiation.
2. A Reactor Operations Committee shall review and approve safety standards associated with the operation and use of the facility. Its jurisdiction shall include all nuclear operations in the facility. The Committee or subcommittees thereof shall monitor reactor operations at least quarterly.
3. Written instructions shall be in effect for:
 - a. ~~Checkout~~ ^{Testing} and calibration of reactor operating instrumentation and control, control rod drives, and area radiation monitors and air particulate monitors.
 - b. Reactor startup, routine operation and reactor shutdown.
 - c. Emergency ~~evacuation~~ ^{and abnormal conditions, including evacuation, reentry and recovery.}
 - d. Fuel loading or unloading.
 - e. ^{Control rod removal and replacement.}

J. Experiments

- f. ^{Maintenance operations which may affect reactor safety.}
1. Prior to performing any new reactor experiment, the proposed experiment shall be evaluated by a person or persons appointed by the Facility Director to be responsible for reactor safety. He shall consider the experiment in terms of its effect on reactor operation and the possibility and consequences of its failure, including, where

significant, consideration of chemical reactions, physical integrity, design life, proper cooling, interaction with core components, and reactivity effects. He shall ^{determine whether,} ~~conclude that,~~ in his judgement, the experiment by virtue of its nature and/or design ^{does} ~~should~~ not constitute a significant threat to the integrity of the core or to the safety of personnel. Following a favorable evaluation and prior to conducting an experiment, he ^{shall} ~~must~~ sign an authorization form containing the basis for the favorable evaluation.

2. ^{A favorable} ~~The~~ evaluation of ^{an} experiment shall conclude that failure of the experiment will not lead to a direct failure of a fuel element or of other experiments.
3. No new experiment shall be performed until the proposed experimental procedures for that experiment or type of experiment has ^{and approved} been reviewed by the Operations Committee.
4. The following limitations on reactivity shall apply to all experiments:
 - a. The reactivity worth of any individual in-core experiment shall not exceed \$3.00.
 - b. The total ^{absolute,} reactivity worth of in-core experiments shall not exceed \$5.00. This includes the potential reactivity which might result from experimental malfunction, experiment flooding or voiding, removal or insertion of experiments.
 - c. Experiments having reactivity worths greater than \$1.00 shall be securely located or fastened to prevent inadvertent movement during reactor operations.

5. Experiments containing materials corrosive to reactor components, compounds highly reactive with water, potentially explosive materials, liquid fissionable materials shall be doubly encapsulated.
6. Explosive materials such as gun powder, dynamite, TNT, nitro-glycerine, or PETN in quantities greater than 25 milligrams shall not be irradiated in the reactor or experimental facilities without out-of-core tests which shall indicate that with the containment provided no damage to the reactor or its components shall occur upon detonation of the explosive. Explosive materials in quantities less than 25 milligrams may be irradiated without out-of-core tests provided that the pressure produced in the experiment container upon detonation of the explosive shall be ~~calculated~~^{shown} to be less than the design pressure of the container.
7. Experiment materials, except fuel materials, which could off-gas, sublime, volatilize or produce aerosols under (a) normal operating conditions of the experiment or reactor, (b) credible accident conditions in the reactor or (c) possible accident conditions in the experiment shall be limited in activity such that if 100% of the gaseous activity or radioactive aerosols produced escaped to the reactor room or the atmosphere, the airborne concentration of radioactivity averaged over a year would not exceed the limits of Appendix B of 10 CFR Part 20.
9. If the effluent from an experiment facility exhausts through a filter installation designed for greater than 99% efficiency for 0.3 micron particles, the assumption ~~may~~^{shall} be used that 10% of the aerosols produced can escape.

10. For materials whose boiling point is above 130°F and where vapors formed by boiling this material could escape only through an undisturbed column of water above the core, the assumption ^{shall} ~~may~~ be used that ~~only~~ 10% of these vapors can escape.
11. Each fueled experiment shall be controlled such that the total inventory of iodine isotopes 131 through 135 in the experiment is no greater than 1.5 curies and the maximum strontium-90 inventory is no greater than 5 millicuries.
12. If a ^{container} ~~capsule~~ fails and releases material which could damage the reactor fuel or structure by corrosion or other means, physical inspection shall be performed to determine the consequences and need for corrective action. *The results of the inspection and any corrective action taken shall be reviewed by The Reactor Operations Committee and determined to be satisfactory before operation of the reactor is resumed.*

Systems
 TABLE I - MINIMUM REACTOR SAFETY CIRCUITS

Originating Channel	Set Point	Mode in Which Effective		
		SS	Pulse	SW
1. Linear	110% of full power	X		X
2. Percent Power	110% of full power	X		X
3. Scram button on console	Manual	X	X	X
4. Preset Timer	Less than or equal to 15 seconds		X*	

*Does not scram reactor - only drops transient rod following a pulse.

TABLE II - MINIMUM INTERLOCKS

Action Prevented	Mode in Which Effective		
	SS	Pulse	SW
1. Control element withdrawal with less than two neutron induced counts per second on the startup channel	X		
2. Simultaneous manual withdrawal of two control elements, including the pulse rod	X		
3. Simultaneous manual withdrawal of two control elements excluding the pulse rod			X
4. Initiation of pulse above 1 kw		X	
5. Application of air to transient rod unless cylinder is fully inserted	X		
6. Withdrawal of any control element except transient rod		X	

The minimum safety circuits and interlocks are required to be operable prior to reactor startup. However, failure of any safety circuit or interlock while the reactor is in operation is permissible, provided that immediate corrective action is taken.

Appendix 1-II

OUTLINE OF TRIGA MARK I REACTOR

ACCEPTANCE TEST PROCEDURES

The following is a brief outline of the procedures usually followed in performing acceptance tests on the TRIGA reactor. By following these procedures, all necessary instrument checks and nuclear calibrations will be performed.

1. Pre-Critical test.

a. Functional Tests of Mechanical Equipment

This includes verification of proper installation and performing operational tests on the isotope production facility, pneumatic system, thermal column, beam ports, exposure room, bridge, water system, fuel element handling tool and other related equipment.

b. Checkout of Instrumentation and Control System

This includes a complete checkout of the control console by GA electronics personnel and ensures that all circuits are calibrated and operating properly. All interlocks and scram initiations are shown to operate satisfactorily. Additionally, the pneumatic and standard rod drives are checked for proper operation during rod withdrawal and scram situations. The fission and ion chambers are shown to be sensitive and responsive to neutrons.

A Keithley micromicroammeter and a scaler are used as a temporary addition to the console instrumentation to provide more information during initial start-up.

2. Initial Loading of Fuel Elements to Criticality Using Inverse Multiplication Data.

Loading proceeds according to standard procedures used on the many TRIGA reactors in operation. At the end of each loading step (the number of elements loaded in each step is determined by $1/2$ the number projected for criticality, or one, whichever is greater), data is recorded under the headings shown in Figure 1.* The first four loading steps proceed with all control rods withdrawn. The remainder of the loading steps proceed with the shim rod inserted and the other rods withdrawn. A plot of the number of fuel elements loaded versus the inverse multiplication is maintained and extrapolated to predict criticality. The weight of uranium in each fuel element varies slightly, so the elements are positioned selectively to group the heavier elements at the core's center. All inner ring grid plate holes will be filled with fuel elements, control rods or other core components. The source is removed temporarily upon reaching criticality to verify criticality.

3. Tests to be Performed Upon Reaching Criticality.

a. Approximate Calibration of Control Rods by Rod Drop Method.

This is only a rough approximation used to give some indication of the rods' values.

b. Adjustment of Excess Reactivity by Addition of Fuel

Elements and Preliminary Control Rod Calibration by the Period Method.

Additional fuel is added to the core. The outer ring, G-ring, of the core (see the diagram in Figure 2*) will initially contain the pneumatic system terminus and graphite dummy elements. The F-ring will contain graphite dummy elements and enough adjacent fuel elements to provide the specified excess reactivity. A preliminary differential control rod calibration, using measured periods and the In-hour curve (see Figure 3*), is made during the loading of the excess reactivity.

- c. Final Calibration of Control Rods by the Period Method and Determination of Final Excess Reactivity.

These accurate calibration curves are calculated and plotted after the final core configuration has been established. They are used throughout the rest of the start-up tests.

- d. Calibration of Period Meter

Using the rod calibration curves, a predetermined amount of excess reactivity is inserted. Using the In-hour curve, the period corresponding to that insertion can be determined and the period meter adjusted accordingly.

4. Tests to be Performed at Higher Powers

- a. Check of Instrument Linearity by Going to Full Power in Decade Steps

Console instrumentation is compared with the reference Keithley micromicroammeter readings.

- b. Power Calibration by the Method of Rate of Rise of Water Temperature

Reactor heat added to the water at a constant rate raises the water temperature X degrees per hour. This rate of rise compared with the rate of rise for a known kilowatt heat input (determined by electric heaters in the reactor tank) establishes the reactor power level.

5. Acceptance Tests at Rated Power

a. Demonstration of Performance and Reliability of Reactor System by Operation at Rated Power

b. Radiation Survey to Demonstrate Adequacy of Reactor Shielding
Readings will be taken at different positions at the top of the reactor tank over the water, on the bridge, at the beam tubes, thermal column and exposure room doors, step in the shield, and various other points on the shield.

c. Demonstration of Pulsing Performance of Reactor System

The purpose of these measurements is to determine the pulse characteristics and to check the nv circuit's calibrated extrapolation from full steady state power. A high-speed recorder, calibrated potentiometer and a Keithley micromicroammeter are used as auxiliary instrumentation. The high-speed recorder is used to give a graphic picture of the pulse, the shape of which can be compared with pulses from other TRIGA reactors. The peak power shown on the high-speed recorder is compared with the peak power information displayed on the console recorder to assure accurate calibration. Temperature information from thermocouples embedded in the fuel elements can be read out on both recorders and compared. Pulse data are recorded under the headings shown in

Figure 4.* The reactor will be brought to peak pulsed power in gradual steps, and at each step, the observed pulse shape will be compared to previous TRIGA data as a prerequisite to further increase in reactivity insertion.

* This Figure is included in G.A. Tech. Bull. No. 122, but omitted here.

APPENDIX 1 - III

Partial list of scientific and technical personnel expected to be associated with the reactor operation or utilization of reactor-produced radioisotopes. Key members of the operating staff have not been employed, but personnel on this list may be included in committees in the administrative organization.

BRADLEY, WENDELL A., Electronics Development Technician

Northeastern Okla. Jr. College, 1938-1939 (General Science)
National Schools, 1950-1952 (Automotive and Allied Mechanics)
Colorado Technical Institute, 1960-1961 (Industrial Electronics)

Technical Experience

Gamma-ray logging instrument operator (uranium exploration)
U.S.G.S., 1950-1952

Field supervision of instrument operators and maintenance of
electronics and mechanical equipment, U.S.G.S., 1952-1958

Physical Science Technician, U.S.G.S., 1958-1961

Electronics Development Technician, U.S.G.S., 1961-

Technical Interests

Development, maintenance, calibration of radiation detection
systems including Geiger and scintillation detectors; scalars,
ratemeters, and multichannel pulse-height analyzers.

BUNKER, CARL M., Geophysicist

B.S., Univ. of Dayton (Ohio), 1951 (Geology)

Professional Experience

Physicist (optics), U. S. Air Force, 1951-1952
Geophysicist, U. S. Geol. Survey, 1952-present

Professional Interests

Application of gamma-radioactivity measurements to geologic studies, including gamma-ray logging, gamma-absorption density measurements, neutron-activation analysis of rocks and minerals, natural and induced radioisotope measurements by gamma-ray spectrometry.

CLAASAN, HANS C., Chemist

M. S., Montana State Univ., 1964 (Chemistry)

B. S., Univ. of Colorado, 1962 (Chemistry)

Professional Experience

Teaching fellowship, Univ. of Colo. 1964-1965.

Analytical Chemist, U.S.G.S., Denver, March 1965.

Professional Interests

Development and use of methods for determining radionuclide content of water and earth material samples. Development of procedures for analyses of tritium at levels as low as 3 picocuries per milliliter.

DOOLEY, JOHN RAYMOND, JR., Physicist (Nuclear)

M. S., Univ. of Denver, 1951 (Math. Phys. and Education)

B. S., Regis College (Denver), 1949 (Math. Phys. and Philosophy)
Univ of Denver, 1943-1944 (Elect. Engr.)

Professional Experience

Training instructor, physics, Lowry Air Force Base, 1951-1952

Physicist (Staff Member), Sandia Corp., 1952-1953

Physicist, atomic, U. S. Geol. Survey, 1953-1954

Physicist, nuclear and molecular, U. S. Geol. Survey, 1954-1958

Physicist, nuclear, U. S. Geol. Survey, 1958-

Professional Interest

Research on the physical, radioactive, and solid state properties of rocks, minerals, waters, and gases of geologic significance. Radium and radium-daughter analyses.

GOLDBERG, MARVIN C., Chemist

B. A., Univ. of Colorado, 1955 (Chemistry)

Professional Experience

1955, Chemist, U. S. Bureau of Mines, Tucson

1956-58, Military duty-5th Army, Medical Lab., Biochemical analyses

1958, Analytical chemist, Colorado Assaying Co., Graduate courses,
Univ. of Denver.

1959, Research Assistant, Univ. of Colo.

1961 to present, Analytical Chemist, U.S.G.S.

Professional Interests

Gamma spectrometry analyses and research for determination of
pesticides and other organics in water.

JANZER, VICTOR J., Research Chemist

B. S., Clarkson College of Technology, 1948 (Chemistry)

Professional Experience

Analytical Chemist, Wright Aeronautical Division, 1948-50 and 1952-54.

Military duty 1950-52.

Graduate Assistant and graduate study in chemistry and geology
Univ. of Colorado 1954-57.

Research Chemist, U.S.G.S., 1957 to date.

Professional Interests

Research regarding ion-exchange mechanisms and exchange properties of clay minerals and fluvial sediments; hydrologic transport of radioactive isotopes through earth materials; determination of radionuclide occurrence and behavior in surface and ground water.

JENNE, EVERETT A., Soil Scientist

Ph.D., Oregon State Univ., 1960 (Soil Science)

M. S., Univ. of Nebraska, 1953 (Soil Science)

B. S., Univ. of Nebraska, 1952 (Soil Science)

Professional Experience

Post Doctoral Fellow, National Science Foundation grant project,
"Rheology of non-ideal impressions," Oregon State Univ., 1960-62.
Soil Scientist (Research), U.S.G.S., 1962 to date.

Professional Interests

Research regarding cation-exchange phenomena of soils and clay minerals and the kinetics of sorption-desorption reactions of heavy metals in the hydrologic-geologic environment. Research involves use of radioactive tracers and radiation analysis instruments.

JOHNSON, JESSE O., Analytical Chemist

Geological Engineer degree, Colorado School of Mines, 1941

Professional Experience

Physical Science Aid, Chemistry, U.S.G.S., 1954
Chemist, analytical, U.S.G.S., 1955 to date.

Professional Interests

Development and use of methods for analysis of radium, uranium
and radioisotopes of other elements in water and earth materials.

ROSHOLT, JOHN N., Chemist

Ph. D., Univ. of Miami (Florida), 1963 (Oceanography)
M. S., Univ. of Miami (Florida), 1961 (Marine Science)
B. S., Univ. of Colorado, 1948 (Chemical Engr.)

Professional Experience

Chemist, U. S. Geol. Survey, 1948-present

Professional Interests

Radioelement analysis, radioactivity in deep-sea sediments,
radioactive disequilibrium studies by alpha- and mass-
spectrometry.

SZABO, BARNEY J., Chemist

M. S., Univ. of Miami (Florida), 1966 (Oceanography-Chemistry)

B. S., Univ. of Miami (Florida), 1961 (Chemistry-Math)

Univ. of Sopron, 1948-1950 (Mining Engr.)

Professional Experience

Chemist, Institute of Marine Science, 1957-1966

Chemist, U. S. Geol. Survey, 1966 to present

Professional Interests

Chemical oceanographic research, radioisotope dating, trace element analysis of sea water, sediment and marine organisms, geochemistry of radioactive elements in marine environment.

TATSUMOTO, MITSUNOBU, Chemist

Ph. D., Tokyo Univ. of Literature and Science, 1957 (Geochemistry)
Tokyo Univ. of Literature and Science, 1958 (Chemistry)
Tokyo Higher Normal School, 1945 (Physics and Chemistry)

Professional Experience

Research Associate, Tokyo Univ. of Education, 1958-1957
Lecturer, Tokyo Univ. of Education, 1957-1962
Post-doctoral Research Fellow, Univ. of Calif. (La Jolla)
1957-1958
Post-doctoral Research Fellow, Agric. and Mech. College of
Texas, 1959
Post-doctoral Research Fellow, Calif. Inst. of Technology,
1959-1962
Chemist, U. S. Geol. Survey, 1962 to present

Professional Interests

Isotopes in upper mantle, lead isotopic composition of rock.

WERSHAW, ROBERT L., Geochemist

Ph.D., University of Texas, 1963, (Geology)
M. S., Calif. Inst. of Tech., 1959, (Geochemistry)
B. S., Texas Western College, 1957, (Geology)

Professional Experience

Geochemist, U.S.G.S., 1963 to date.

Professional Interests

Research regarding detergents, pesticides and other organic compounds in water, involving tracer studies employing radioactive isotopes.

Appendix 9-I
Radiological Advisory Committee
Radiation Protection Procedures

April 1, 1963

Radiological Advisory Committee

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RADIOLOGICAL ADVISORY COMMITTEE

0.0 The Radiological Advisory Committee was established December 7, 1959 to ensure and enhance safety against radiation hazards within the purview of the Rocky Mountain Survey Committee of the U. S. Geological Survey, Denver Federal Center, Denver, Colorado and its associated field projects.

Membership

1.0 The Radiological Advisory Committee, hereinafter referred to as the Committee, shall be composed of at least two members from each Survey Division that utilized ionizing radiation in its various projects. Such members must meet A.E.C. requirements. In addition, the Radiation Safety Officer (Health Physicist) shall be a permanent member of the Committee and shall act as its Executive Officer. Alternates are designated for the Committee members to act in their absence.

Functions and Responsibilities

2.0 It is the responsibility of the Committee to insure the safe use of radioactive materials and ionizing-radiation-producing devices and compliance with current Federal and State regulations.

2.1 The Committee operates under a "Broad" By-products Material License granted by the U. S. Atomic Energy Commission. Under the conditions of this license, and to meet its responsibilities, the Committee shall evaluate and approve safe uses of radiation. The Committee is not to restrict the use of radiation, rather, it is to be concerned with the task of maintaining the safety of personnel, operations and facilities involved with the use of ionizing radiation.

The Committee shall not restrict its scope to the safe use of licensed material only. It is concerned with all radiological hazards which include naturally occurring radioisotopes and radiation devices. (e.g. - X-ray machines, neutron generators, etc.)

2.2 The Committee shall exercise this authority throughout the U. S. Geological Survey, Denver, Colorado and its associated field projects.

2.3 The Committee shall be available for advice and assistance on radiological problems and shall make information available to interested parties concerning the A.E.C. regulations under which it operates.

2.4 The Committee shall maintain records necessary to complete its responsibilities and to comply with Federal and State regulations. These records will include the following:

- a) radiation inventory and disposal records
- b) radiation survey and personnel monitoring records
- c) Committee records of actions including application approvals, conditions and recommendations, etc.

2.5 The Executive Officer shall prepare and hold these records in the form of a permanent file. These records will be available for inspection to the Atomic Energy Commission and to interested individuals.

2.6 A copy of the minutes of the Committee meetings will be furnished to each member, to the Chairman of the Rocky Mountain Survey Committee, the Denver Survey Safety Committee, and the Washington Safety Committee. The original shall be retained by the Executive Officer.

Administrative Procedures

The Committee shall meet regularly, as may be required, for the purpose of accepting, reviewing and acting upon proposed uses of radiation and other matters of business that may occur. Business between sessions may be handled by letter or by special meetings. Special meetings may be called at any time by the Chairman or Acting Chairman.

3.1 Officers of the Committee shall be the Chairman, elected in January of each year, and the Executive Officer, a permanent post held by the Health Physicist. The Chairman shall call and conduct meetings. In his absence, the Executive Officer shall preside.

3.2 A quorum of 80% of the members constitutes a valid Committee action. When a vote of the Committee is required, a minimum approval of two-thirds of the members present will be necessary.

Applications for Radiation Usage

4.0 Each existing or proposed project within the U. S. Geological Survey, Denver, Colorado that intends to use or generate ionizing radiation must receive the approval for such usage from the Committee in writing.

4.1 Existing projects that are currently licensed under other A.E.C. Specific By-Product Material licenses that will be incorporated into the broad license may be granted Committee approval, based in part on the Health Physicist's recommendations. The approval will be effective on the starting date of the "Broad" license under which the Committee operates. The need for an initial application by such a project to the Committee is waived.

4.2 The Committee shall approve an application for radiation usage or generation when it has determined that potential or real radiation hazards are safely controlled in accordance with current Federal and State regulations, and that the individuals or groups involved are qualified to safely handle the radiation.

4.3 All individuals or groups desiring Committee approval must submit sufficient information to enable the Committee to make an adequate evaluation. Particular emphasis will be placed upon the following items which must be included in the application:

- a) The amount, form and use of the radiation to be used or generated.
- b) A list of safety procedures and precautions that have been or will be instituted prior to the start of the project to insure the compliance of Federal and State regulations.
- c) The location or place of use.
- d) The names of the individuals responsible for safe usage and regulations compliance.
- e) Experience and training of the individuals that will supervise and/or use the radiation.
- f) Radiation protection, detection and monitoring equipment and facilities to be used.
- g) Where practical, an estimate of the waste to be generated by the project and the manner in which the waste is to be controlled.

(AEC form #313 offers an excellent guide to follow. Copies are available from the Executive Officer.)

4.4 Conditions for safe use of radiation made by the Committee in granting approval to a radiation usage application will be binding upon the individuals responsible for the project. Non-compliance with these conditions shall be grounds for Committee action to suspend such project activity until compliance can be reestablished and reapproved by the Committee.

- 4.5 In the event that the Committee does not approve an application the individual may submit a new application. Upon the request of any individual, a review of the Committee refusal shall be submitted to the Washington Safety Committee for recommendations by that group.
- 4.6 Any project activity suspended by the Committee for non-compliance of safety conditions may appeal such stoppage to the Committee. Appeals shall be made within 60 days of such action.
- 4.7 The effective time period of Committee approval of a project activity will be based on the needs of that project. However, the Committee shall not have authority to grant approval time periods for uses of by-product materials beyond the expiration date of the A.E.C. Broad license under which it operates. In addition, a review of each current Committee approval will be conducted by the Committee annually.
- 4.8 Renewals of projects previously granted Committee approval will be made by the Committee, providing that past safety measures and appropriate regulations have been maintained. Applications for renewal shall be made to the Committee in writing, describing the project.
- 4.9 Amendments may be made at any time to existing project approvals by the individuals concerned, by applying to the Committee. Applications for amendments shall be submitted to the Committee in writing and shall contain applicable information referred to in part 4.3 (a - g) of this document.
- 4.10 All correspondence that requires Committee action will be placed on the Committee Agenda in the order that it is received.

Procurement of Radioactive Material

No project shall procure or possess radioactive materials, including "classified" material, without obtaining project approval by the Committee. The requisition form shall contain the signature of the Executive Officer for purposes of insuring compliance with existing maximum possession limits, prior to the placement of the order by the purchasing officer. In the absence of the Executive Officer, the Chairman of the Committee shall sign the requisition.

Inventories

6.0 Current inventories of radioactive materials and devices are necessary to comply with current Federal regulations and/or to insure the safety of Survey personnel and property. The Executive Officer shall prepare and maintain the inventory file.

6.1 An inventory shall be made twice a year unless otherwise designated by the Committee. The inventory shall include the following information:

- a) location
- b) ownership and the name of the person to whom the source was issued.
- c) type, form and strength of the sources including the date of activity measurement). Radioactive waste must be included.
- d) A.E.C. license number, if any.

6.2 Throughout the year, the Executive Officer shall be informed of subsequent radioactive material arrivals or disposals. This will enable a running inventory of Survey possession amounts and will help to prevent exceeding maximum possession limits contained in current licenses.

RADIATION SAFETY OFFICER

7.0 The Bureau Safety Officer will act as the Radiation Safety Officer and will be referred to as the Health Physicist.

7.1 The Health Physicist shall be responsible for supplying the common services necessary for a safe radiation program within the U. S. Geological Survey, Denver, Colorado and its associated field projects. This responsibility shall include the following:

a) To hold the position of Executive Officer of the Radiological Advisory Committee and the duties associated with that position.

b) To maintain a central file of records of the use, storage and disposal of radioactive materials as required by the Atomic Energy Commission. This will include radiation inventory records, personnel exposure records, etc.

c) To conduct radiation surveys at times necessary to insure that work is being conducted in a safe manner in accord with recognized procedures contained in the Committee project approval and current Federal regulations.

d) To determine and direct procedures to be followed in any radiological emergency. This may include direction of personnel involved in the emergency to be attended by the Survey Health Unit for observation and treatment.

7.2 The Health Physicist, on instruction from the Committee, shall have authority to suspend any project handling radioactive materials in a manner that violates personnel safety conditions set forth by the Committee and contained in current regulations. (10CFR20, etc). The Health Physicist shall make a detailed report of such action to the Committee. It then becomes the responsibility of the Committee to act upon the suspension and the appeal, if forthcoming.

7.3 To facilitate direct responsibility for radiation safety over field projects, the Health Physicist shall appoint a competent individual to act for the Health Physicist to insure the safe use of radiation during the time that the project is in the field. Such individual shall have the same authority and responsibilities for the radiation safety program of the particular field project as the Health Physicist.

7.4 In addition to responsibilities, the Health Physicist provides certain services. These include:

- a) advice
- b) procedural supervision when necessary
- c) calibration of monitoring equipment
- d) the conduct of surveys on request of the radioactive material users as an aid in conducting a program
- e) special Committee requests relative to the radiological health situation within the Survey.

Radiation Protection Procedures

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RADIATION PROTECTION PROCEDURES

General Instructions

1.0 The procedures and instructions contained herein shall be followed by all personnel handling or using radioactive materials or devices. These procedures are necessary for the following reasons:

- a) To insure the safety of personnel and facilities
- b) To obtain the maximum amount of information from the application of radiation
- c) To retain our privilege of employing radiation which is an integral part of our program

1.1 Although the sources of radiation in our possession are used for scientific investigations, the concern for the health and safety of both user and non-user personnel must be greater than the interest in achieving scientific objectives.

1.2 Radiation must not be used in a manner that would constitute a real or potential hazard to personnel and property greater than currently accepted limits. There are several sources of information that recommend acceptable limits. The most important ones are: The Federal Radiation Council, U. S. Atomic Energy Commission, National Committee on Radiation Protection and Measurements (NCRP) and the International Commission on Radiological Protection (ICRP). The basis for the formulation of these RADIATION PROTECTION PROCEDURES are the current regulations and recommendations contained in the Code of Federal Regulations, Chapter 10, parts 20 and 30. These will serve as guidelines in setting up a radiological safety program that will adequately serve the needs of the U. S. Geological Survey, Denver, Colorado. From the previously mentioned sources, the following personnel exposure limits have been adopted by the Radiological Advisory Committee:

DOSE LIMITS PER CALENDAR QUARTER

- A) Whole Body: (Head & trunk, active blood-forming organs, lens of eye, gonads).....1.25 rem
- B) Hands & forearms, feet & ankles.....18.75 rem
- C) Skin of whole body.....7.50 rem

1.3

An individual may be permitted to be exposed to no more than three rem to the whole body in any one calendar quarter providing the whole body dose, when added to the accumulated occupational dose to the whole body, shall not exceed $5(N - 18)$ rems, where N is the individual's age at this last birthday. To be permitted to receive up to three rems in any one calendar quarter, the individual's occupational accumulated dose must be computed on A.E.C. form # 4.

1.4

Exposure to minors (under age 18) is discouraged. However, no minor shall be permitted to be exposed to doses greater than 1/10 those listed above.

1.5

Under emergency conditions, an individual may be permitted to be exposed to a whole body dose of 25 rems. This shall be permitted only once in the person's lifetime.

1.6

For the accepted concentration limits of radionuclides in air and water, refer to 10 CFR 20.

1.7

If it becomes necessary to handle radiation in a manner not described herein, consult your supervisor for further instructions. Your supervisor will then contact the Health Physicist for advice on safe usage.

Responsibilities of Supervisors

2.0

The individual who supervised work which utilizes radiation has in addition to his usual duties, the following responsibilities:

- A) To be aware of the nature and hazards of ionizing radiation

- B) To enforce the personnel rules of this manual.
- C) To see that necessary equipment for working safely with radiation is available to and used by personnel. (The Health Physicist may be consulted for specific recommendations.)
- D) To immediately report all radiological health emergencies to the Health Physicist.
- E) To inform new employees of radiation protection. This applies to all individuals who have occasion to enter in areas which are classed as Radiation Areas. The Health Physicist will aid in this education if requested.
- F) To see that radioactive materials used under his supervision are handled and disposed of in accordance with the provisions contained in the Radiological Advisory Committee project approval.

Handling Radioactive Materials

Sealed Sources

- 3.0 External radiation levels must be kept to a minimum at all times. Sealed sources and radiation producing devices both enter into the category of external sources of radiation.
- 3.1 For maximum safety, the following principles should be applied:
 - A) Time: Keep the exposure time to a minimum
 - B) Distance: Keep as great a distance as practical from the source of radiation.
 - C) Shielding: Utilize the maximum practical amount of shielding between the source and personnel that is commensurable with project activity.
- 3.2 When handling sealed radioactive sources, avoid having the source come in contact with any part of the body. Use tweezers or tongs to handle the source and when practical, hold it at arm's length to place it at a greater distance.
- 3.3 Take care not to subject the sealed source to excessive mechanical or thermal shock.

- 3.4 A leaky or fragmented source can present a real health hazard. For this reason, each sealed source of activity greater than one microcuries shall be leak-tested by the Health Physicist at intervals not exceeding six months or whenever leakage is suspected.
- 3.5 The Health Physicist shall conduct the leak-tests in accordance with A.E.C. regulations. A source will be considered leaky or contaminated if 0.005 microcuries or more of removable beta and/or gamma activity is detected. A pure alpha emitting source will be considered leaking or contaminated if the alpha count is ten times background or 500 dpm/100 cm², whichever is lower.
- 3.6 The A.E.C. shall be notified of all leaky or contaminated sources by the Health Physicist in accordance with applicable regulations.
- 3.7 The Health Physicist shall maintain a record of all leak-tests. These records shall be available for inspection.

Open Sources

- 4.0 Radioactive materials that are not sealed and are greater in activity than the levels for which only a general license is required (see 10 CFR 30) shall be considered open sources. Open sources present additional problems of contamination and human internal deposition that are not present with sealed sources.
- 4.1 The following safety precautions shall be followed when working with open sources:
- A) No eating in usage areas. Smoking may also be prohibited.
 - B) Wear rubber gloves when handling open sources. Other protective apparel may be specified.
 - C) Do not pipette by mouth...use syringes, rubber bulbs or other mechanical devices

- D) After handling open sources, wash hands and arms before handling any object that goes to the mouth, eyes or nose.
- E) When possible, work with open sources inside exhaust hoods or glove boxes.
- F) Keep or transport high level open sources in such a manner as to prevent spillage or breakage (e.g.,: double containers, trays to contain possible spills, etc.)

Control of Contamination

- 5.0 Cleanliness and orderliness are two of the most important methods of curbing contamination. Keep the work area neat and clean, free from equipment and materials not necessary in the operation.
- 5.1 Trays, preferably of stainless steel, covered with absorbent material are recommended to collect contaminants in the event of a spill. If contamination is suspected, contact the Health Physicist. The Health Physicist will survey the area and persons involved to determine the degree of contamination and to institute proper clean-up procedures.
- 5.2 Any area emitting greater than two millirem/hour at the surface is considered contaminated. An area emitting over 10 millirem/hour at the surface is seriously contaminated. In the case of alpha emitters, an area shall be considered contaminated if the count is $500 \text{ dpm}/100 \text{ cm}^2$ or 10 times background, whichever is lower.
- 5.3 After coming in contact with open sources and before leaving the work area, check clothing and body for contamination. Each employee is responsible for his own contamination check.
- 5.4 Equipment such as glassware that comes in contact with open sources should be kept separate from other equipment. Once used for radioactive substances, it should not be used for other work and should not be sent to other areas until it is demonstrated to be free from contamination. It is recommended that a separate storage cabinet or

area be provided for such equipment.

Surveys

- 6.0 Periodic surveys shall be conducted by the Health Physicist or his aides to detect areas of radiation hazard and contamination. A record of such surveys will be maintained by the Health Physicist.
- 6.1 An area in which a whole body radiation dose of greater than 5 millirem/hour if the person were continually present or 100 millirem in any 5 consecutive days shall be considered to be a radiation area and shall be so posted.
- 6.2 An area in which a person could receive 2 millirem/hour or greater if continually present or 100 millirem in seven consecutive days shall be considered a restricted area for purposes of radiation protection. It shall be restricted to authorized personnel, only.
- 6.3 An area in which a person, if continually present, could receive greater than 100 millirem/hour shall be considered a high radiation area and shall be posted as such.

Personnel Monitoring

- 7.0 Any person working in a restricted area and is expected to receive greater than 25% of his quarterly dose limit shall be monitored. Other individuals may be monitored at the discretion of the Health Physicist.
- 7.1 Personnel monitoring is accomplished through the use of film badges. In high radiation areas or when working with large sources of radiation, the person should also wear two condenser-type self-reading dosimeters. The use of self-reading dosimeters provides an instant check on the cumulated radiation dose and helps to prevent overexposure if it is read several times during the operation. Two

dosimeters worn simultaneously negates any charge loss due to a faulty dosimeter. Charge loss results in high readings, therefore, if a difference in dosimeter readings occurs, the lower reading is to be taken as most accurate.

7.2 A record of personnel exposures as determined by film-badge reports shall be maintained by the Health Physicist. The results of these reports are available to individuals upon request.

7.3 If an overexposure occurs, the Health Physicist shall inform the individual, investigate the circumstances and shall not permit the individual to return to radiation work until his accumulated radiation exposure is again within limits. Overexposures shall be reported to the Radiological Advisory Committee by the Health Physicist.

7.4 The Health Physicist may require individuals to take periodic bioassay tests, especially upon discovery of high radiation doses, accidents or other radiological emergencies.

Storage and Shipment

8.0 Radioactive materials shall be stored in such a manner as to reduce the associated radiation field to an acceptable level. Radiation signs shall be conspicuously posted to inform individuals of the hazard that is present.

8.1 When not in use, the containers in which radioactive materials are present, shall be kept in an area not readily accessible to casual personnel and shall be properly labeled in accordance with 10 CFR 20.

8.2 Radioactive gases or materials with radioactive gaseous daughters shall be stored in gas-tight containers and must be kept in areas with good ventilation.

8.3 Open sources should be stored in such a manner as to contain the material in case of spillage or breakage.

8.4 Radioactive materials to be shipped outside of the Federal Center shall comply with A.E.C. and I.C.C. shipping regulations. The Health Physicist shall provide advice, information and packing supervision to persons who desire it.

8.5 When radioactive materials or radiation producing devices are received, at the Federal Center, the Survey Shipping department shall notify the Health Physicist. He shall determine if he needs to be present to supervise the opening and initial handling of the material.

Waste Disposal

9.0 The disposal of radioactive wastes must be controlled for the safety of Survey personnel and the general public and to protect Survey property. Disposal shall conform to Federal and State regulations, especially those contained in 10 CFR 20. Disposal shall be controlled by the Health Physicist.

9.1 Radioactive wastes shall be considered to be solid materials contaminated to 1 millirem/hour or more and radioactive solutions no longer desired.

9.2 Each area creating radioactive waste shall have at least one radioactive waste container. The container shall be properly marked.

9.3 Radioactive wastes of short half-life may be held until the activity is insignificant.

9.4 Other radioactive waste shall be periodically transferred to Dow Chemical Company, Rocky Flats, Colorado for disposal.

9.5 A record shall be kept of all radioactive waste disposals, by the Health Physicist.

9.6 Future conditions may warrant a different mode of waste disposal. The Health Physicist shall be responsible for determining the proper procedure in accordance with AEC and other Federal and State regulations.

Accidents and Emergencies

10.0 The term "radiological health emergency" shall apply to any incident pursuant to the use of radiation which produces an overexposure to an individual or which produces contamination of personnel and/or areas. This includes minor as well as major incidents.

10.1 Notify your supervisor and/or the Health Physicist immediately in the event of any radiological health emergency.

10.2 Upon notification, the Health Physicist shall take immediate steps to remedy the situation.

10.3 The individuals causing contamination have the responsibility of performing the major portion of the decontamination. This must be done under the supervision of the Health Physicist.

10.4 Of prime consideration in an emergency is the confining of contamination to as small an area as possible and keeping personnel exposures to a minimum. This can be accomplished by the following procedures:

- A) Do not allow newcomers to enter the area unprepared.
- B) Permit no one to leave the contaminated area without clean footwear or being monitored to demonstrate no contamination. An exception will be made in cases to prevent over-exposure.

- C) Prevent the flow of activity with liquid carriers, raised barriers (putty, etc.), apply absorbers, seal cracks in the floor, desk tops, etc.
- D) If contamination is air-borne and a respirator is not readily available, hold breath or cover nose and mouth with a wet handkerchief, close windows and doors, shut off fans or air-conditioners and leave the area as quickly as possible.
- E) Do not turn on ventilating equipment.
- F) Remove contaminated clothing and place it in a receptacle designated for this purpose.

10.5 If open wounds are contaminated, immediately flush the wound with water, preferably under a running tap. The Health Physicist may elect to use additional measures but immediate flushing with water is usually most effective.

10.6 If ingestion of radioactive materials has occurred, make the person vomit. Repeated vomiting should be induced and the person should drink 2 - 4 glasses of lukewarm water between inducements.

10.7 Emergencies involving high levels of radiation should be monitored at all times to prevent overexposure of individuals taking part in emergency procedures. The Health Physicist may designate an individual to be responsible for the specific monitoring task.

10.8 The Health Physicist will maintain a record of all radiological health emergencies and the action that was taken. If AEC regulations require, he shall notify the AEC in conformance with the applicable regulations.

APPENDIX 9-II

RADIATION MONITORING SYSTEM

16A-1. SCOPE. This section of these specifications includes the furnishing and installation of a complete radiation monitoring system as specified herein and as shown on the drawing.

16A-2. GENERAL.

a. The purpose of the radiation monitoring system is to provide radiation safety for the occupants of the Nuclear Science Building and to control radiation intensity levels in different parts of the building where excessive radiation intensity might interfere with the scientific program. To meet these objectives, the system must provide visual and audible indications of the radiation intensity at each detector to warn personnel in the immediate vicinity of the detector when the radiation intensity has exceeded an acceptable level. At the same time, an indication of excessive radiation must be provided to the health physicist or other designated individual at a central monitoring station.

b. The monitoring system supplied shall detect nuclear radiation, measure quantitatively the amount of radiation, and provide visual and audible warning when any or all detectors are subjected to radiation intensities exceeding levels preset by the user.

c. All components specified herein, plus any others required to provide an operating system, shall be supplied by the Contractor. Installation may be accomplished by an electrical Subcontractor under direct supervision and control of the Contractor. The Contractor shall supply complete circuit diagrams, instructions, and all pertinent information required to facilitate the installation and to repair malfunctions in any part of the system.

d. The specifications contained herein and related to components are to be considered as minimum; the system and components must meet or exceed the performance requirements. The components must be compatible electronically to operate as a system.

e. The system must operate on 115 volt, 60 cycle power. In the event of an external power failure, the system must return automatically to proper operating conditions when the external power is restored.

f. The detection and monitoring system is, for the purpose of these specifications, divided into two groups hereafter referred to as System A and System B. The primary differences in the two systems are that (1) System A measures only gamma radiation at lower intensities than may be measurable with System B, (2) the components for each detection station of System A must be constructed in a single package, and in System B the detectors are separated from their monitoring packages, and (3) the components at each detection station in System A must be constructed in an esthetically acceptable package.

g. All detectors and monitoring units are to be semi-permanently mounted on walls and laboratory benches or suspended from the ceiling. Construction of the components shall permit convenient withdrawal of modules from racks or disengagement from brackets to service or adjust components of the system.

h. All warning lamps, lights, or indicators must be clearly visible at a distance of at least 30 feet when fluorescent lighting incident on them is as much as 500 foot-candles.

i. All audible warning devices must emit a distinctive tone, warble, or other sound which can be heard clearly above office and laboratory background noise, and at a distance of at least 30 feet.

j. Bids are acceptable only on an "all or none" basis. The successful bidder will be responsible for supplying, installing and adjusting all components and demonstrating that the components and the system are operational after installation.

k. The system and its components shall be guaranteed for a period of one year after installation.

16A-3. SYSTEM A.

a. System A shall consist of 14 each gamma-radiation detecting stations and 14 each indicator units at a central monitoring station in room 152.

b. Each detecting station shall include the following:

- (1) A gamma-ray detector.
- (2) A meter for indicating radiation intensity in a range of at least 0.1 to 100 milliroentgens per hour in at least 5 logarithmic decades. The meter shall include an alarm trip which is mechanically adjustable throughout the meter range to actuate internal and external warning devices when the radiation intensity exceeds a preset level.
- (3) A visual warning indication that the detecting station is operational.
- (4) A visual indication, in addition to the meter, showing when the radiation intensity exceeds a preset limit.
- (5) An audible warning indication of radiation intensities exceeding a preset limit.
- (6) Electrical connections to provide signal outputs for operating remote meters, recorders, visual alarms, and audio alarms.
- (7) High- and low-voltage power supplies, as required, which operate on 115 V 60 cycle external power.

(8) Preamplifiers and impedance matching circuits, as required.

c. Each central station unit shall include a small clearly-visible, red indicator lamp activated simultaneously with the visual indicator at the detector station showing when the radiation intensity exceeds a preset limit.

d. The bid shall include, as an additive item, an audible alarm system at the central station which is actuated when the radiation intensity at any of the detection stations exceeds the preset limit.

e. The general locations for the detectors are shown on the attached plan. The unit at each detecting station near entryways shall be mounted as close as possible to the power outlet in such a way as to prevent being obscured by, or interfering with, the doors. Installations shall provide at least 6 feet 4 inches between the floor and the base of the detecting stations unit. Installations in rooms 143 and 160 shall be made at the ends of laboratory benches nearest the room entrances. One unit is to be installed on a wall in the basement in the general location shown; the specific location will be determined at the time of installation.

The central monitoring station shall be installed in room 152. The individual indicators shall be assembled in a panel board, rack, or chassis and mounted on the wall in the area indicated on the contract drawing.

f. The bidder shall specify the electrical characteristics and type of cable to be used to connect the units at the detecting station and the central station. The bid shall include, as an additive item the cost of a 1,500 foot continuous length of this connecting cable.

16A-4. SYSTEM B.

a. System B shall consist of 3 each gamma-ray detectors, 2 each neutron detectors and associated monitoring units mounted in the vicinity of the detectors and at the central monitoring station.

- b. Each detection station shall include the following:
- (1) A gamma-ray or neutron detector as indicated on the contract drawing.
 - (2) Nearby monitoring units including:
 - (a) A meter for indicating on at least 3 logarithmic decades radiation intensity related to a range of 1.0 to 1,000 milliroentgens per hour for gamma-ray detectors and related to a range of about 0.3 to 3,000 millirem per hour for neutron detectors. The meter shall include an alarm trip mechanically adjustable throughout the meter range to actuate internal and external warning devices.
 - (b) A visual indication that the detecting station is operational.
 - (c) A visual warning indication, in addition to the meter, showing when the radiation intensity exceeds a preset limit.
 - (d) An audible warning indication of radiation intensities exceeding a preset limit.
 - (e) Electrical connections to provide signal outputs for operating remote meters, recorders, visual alarms, and audio alarms.
 - (3) High- and low-voltage power supplies, as required, which operate on 115 V 60 cycle external power. These may be located at the central station instead of at the nearby station.
 - (4) Instrument panel boards, racks, or chassis as required to contain the components and for wall mounting.
 - (5) Preamplifiers and impedance matching circuits as required.

d. The general locations for the detectors are shown in the contract drawing. The detectors are to be at least 13 feet from the floor in room 149, and at least 10 feet from the floor in rooms 151 and 153. They are to be suspended from the ceiling; therefore, construction of the detector housings will require a design amenable to attachment on vertical pipe or tubing. The nearby monitoring units shall be mounted in the general vicinities indicated and at least 6 feet 4 inches from the floor. Visual warning lights shall be clearly visible from the immediate vicinity of the detectors.

e. The bidder shall submit with his bid the electrical characteristics and type(s) of cable to be used to connect the detectors with the nearby monitoring units, and to connect the nearby monitoring units with the central station. The bid shall include, as an additive item, the cost of 175 feet of cable in a continuous length for connecting detectors and nearby monitoring units; and 150 feet of cable in a continuous length for connecting nearby and central monitoring units.

16A-5. COMPLETION OF CONTRACT. The Contractor shall notify the Regional Director, Public Buildings Service, General Services Administration, Building 41, Denver Federal Center, Denver, Colorado 80225, in writing when all work under this contract has been completed. Final billings, statements or invoices will not constitute written notice of completion.

END OF SPECIFICATIONS