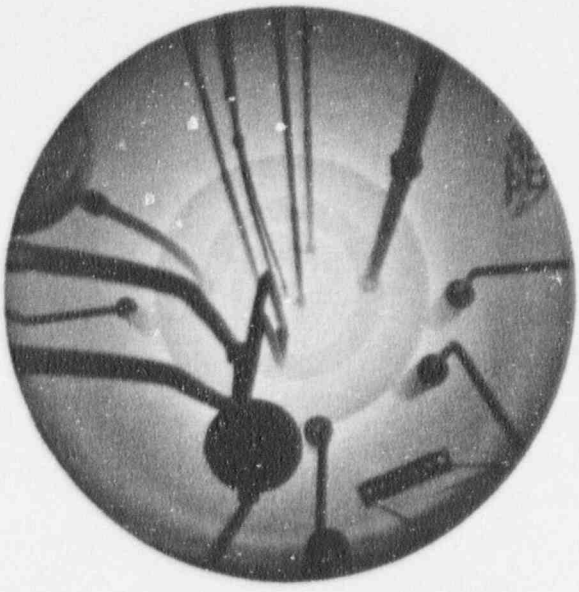


NUCLEAR REACTOR LABORATORY

TECHNICAL REPORT

THE UNIVERSITY OF TEXAS
COLLEGE OF ENGINEERING
DEPARTMENT OF MECHANICAL ENGINEERING

1978 Annual Report

of

The University of Texas at Austin
Nuclear Reactor Laboratory

January 1, 1978 - December 31, 1978

E. L. Draper, Jr., Director
D. E. Klein, Director
J. A. Burack, Supervisor

Taylor Hall 104
512-471-5136

March 1978

TABLE OF CONTENTS

List of Figures

Figure No.

List of Tables

I. INTRODUCTION

This report has been prepared by the staff of the Nuclear Reactor Laboratory (NRL), The University of Texas at Austin, to satisfy the reporting requirements of U.S.E.R.D.A. Contract Number At-(40-1)-3919 and 10 CFR 50.59. This report covers the period from January 1, 1978 to December 31, 1978.

The NRL is presently equipped with a 250 kW TRIGA Mark III nuclear reactor, a 2200 curie Cobalt-60 irradiator, a 15 kV Cockcroft-Walton 14 MeV neutron generator, four Californium-252 neutron sources, a subcritical assembly, and a computer coupled neutron activation analysis laboratory.

The major changes made to the laboratory during this period were:

1. Dr. E. Linn Draper, Jr., resigned as Director of this facility on September 1, 1978. Dr. Dale E. Klein assumed the position.
2. The Automatic Power control system for the TRIGA Reactor was decommissioned due to its subnormal mode of operation.
3. The modification to the pool pumping/purification system which was started last year was completed and the reactor pool was cleaned for the first time since installation.
4. Replacement of a Shim Rod magnet was necessitated by an open circuit occuring in the original. Design changes by General Atomic since production of the original magnet resulted in a loss of 3/8 inch of travel in the Shim Rod.
5. The original Hazards Summary Report and its amendment of 1963

were combined and rewritten to reflect all the changes and modifications to this facility since its inception.

6. Purchase of a new power level detector to replace the spare one being used presently.

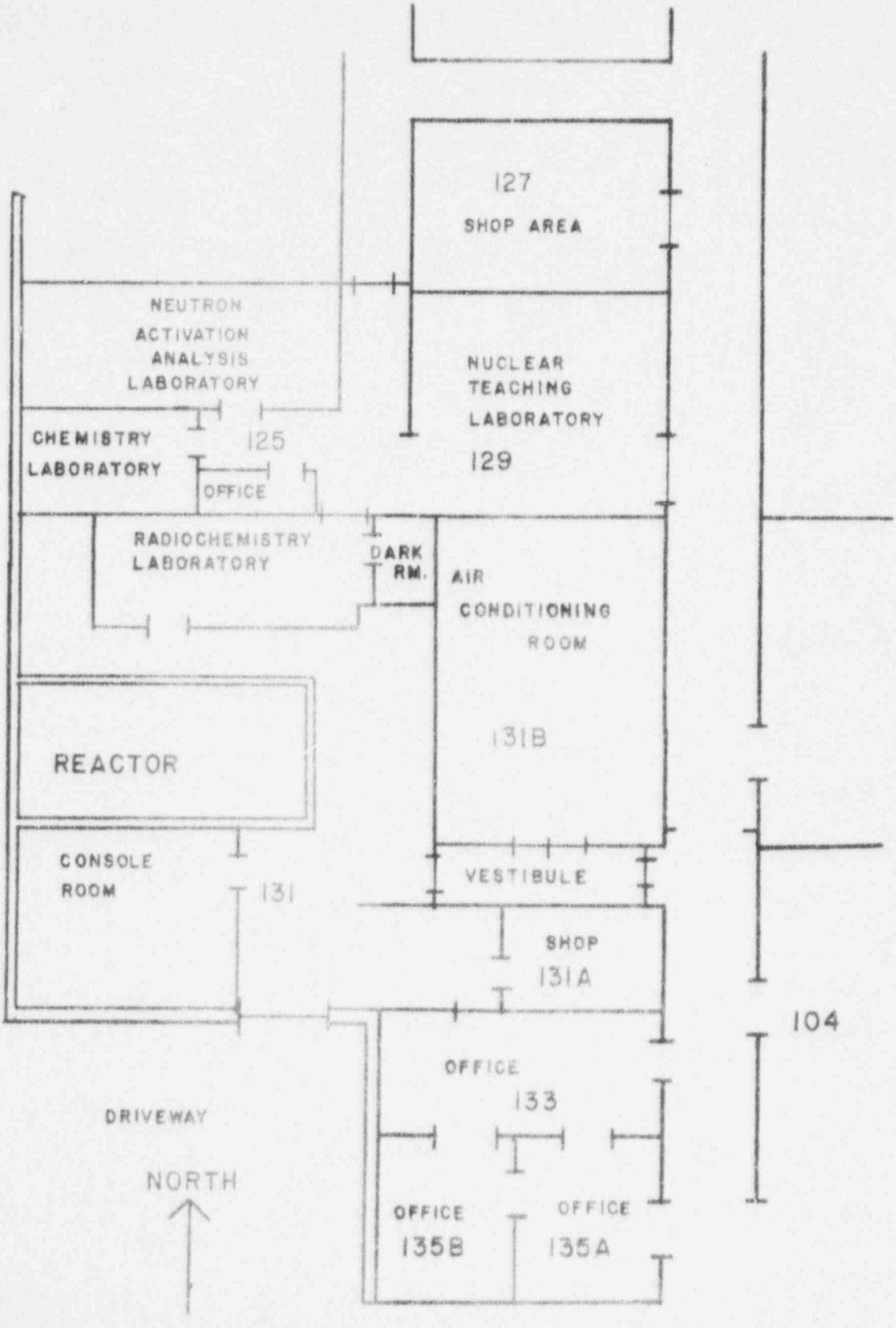


FIGURE 1— FLOOR PLAN OF REACTOR LABORATORY COMPLEX.

II. LABORATORY ADMINISTRATION

A. Organization

The present organizational chart of the Nuclear Reactor Laboratory is presented in Figure 4.

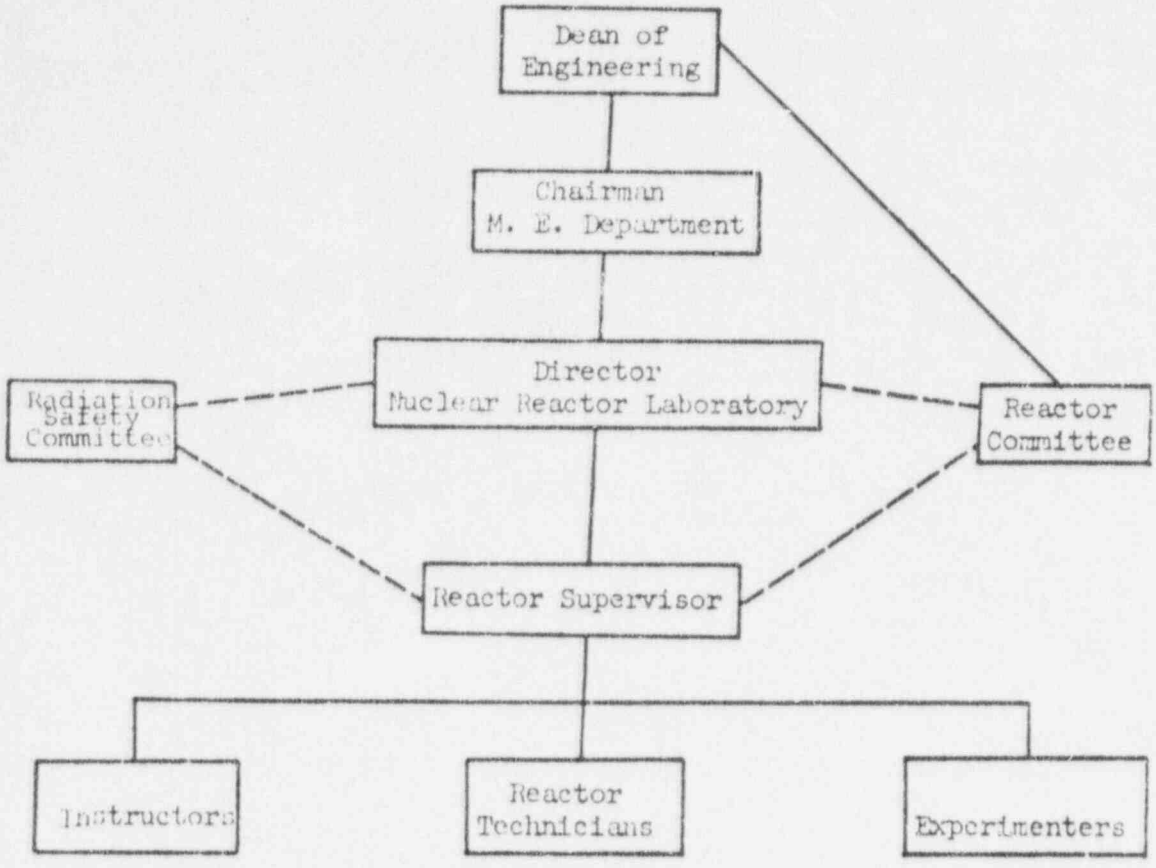
B. Personnel

The following is a list of personnel of the Nuclear Reactor Laboratory for the period January 1, 1977 to December 31, 1977.

Laboratory Administration and Staff

Bauer, T. L.
+Burack, J. A., Supervisor
Campbell, B.
Cornwell, J. B.
Dao, Tuan
Davidson, J. W.
+Draper, E. L., Jr., Director (terminated)
Ebrahimi, M.
Ganthner, S.
Garvel, L. M.
Hamann, J.
Haynes, M. (terminated)
Hertel, N. E.
Hodge, S.
Hsu, D.
Klein, D. E. (Director)
Koen, B. V.
Krause, M.
Kunimoto, Y.
*Meredith, F. A. (terminated)
Murphie, B.
Murphy, R.
Parish, T. A. (terminated)
Pradzynski, A. H.
Razzaque, M.
Rosello, C
Sanders, T.
Sharif-Homayoun, A.
Simov, Rangel
*Smith, R. D.
Taylor, G.
Tomlin, B. B., Chief Technician
Yang, Sam

+ -- Licensed Senior Operator
* -- Licensed Operator



LINE OF RESPONSIBILITY

CONSULTATION AND VETO POWER

ORGANIZATIONAL CHART

FIGURE 2

C. Reactor Committee

1. Committee Composition

Regular Members:

Dr. Joe Ledbetter
Dr. Harris Marcus (Chairman)
Dr. Dale Klein
Dr. E. Linn Draper, Jr.
Mr. Jack Marvin

Ex-Officio Members:

Dr. H. G. Rylander, Chairman, Department of Mechanical Engineering
Dr. E. L. Draper, Jr., Director, Nuclear Reactor Lab
Mr. H. W. Bryant, University Radiation Safety Officer
Mr. J. A. Burack, Supervisor, Nuclear Reactor Lab
Dr. D. E. Klein

2. Meeting Frequency

The Reactor Committee met four times during the calendar year of 1978.

3. The Nuclear Reactor Committee Annual Report for September 1977 to August 1978 is included on the following pages.

D. Radiation Safety Committee

1. Committee Composition

Dr. E. L. Powers, Chairman
Dr. L. O. Morgan
Dr. Joanne Ravel
Mr. H. W. Bryant, Ex-Officio

2. Meeting Frequency

The Radiation Safety Committee met twice during the calender year of 1978.

NUCLEAR REACTOR COMMITTEE ANNUAL REPORT

September 1977 to August 1978

Function, Activities, and Membership

The Nuclear Reactor Laboratory staff and facility are administered through the Department of Mechanical Engineering of the College of Engineering at the University of Texas at Austin. The Nuclear Reactor Committee is a requirement of the Nuclear Regulatory Commission and its function is to insure that the nuclear reactor and its associated facilities are operated in a professional and safe manner.

This objective is met primarily by regularly scheduled quarterly meetings of the Nuclear Reactor Committee. These meetings include a review of the activities of the reactor staff, faculty, and associated graduate students; a tour of the laboratory facilities by a member or members of the committee; and a reporting of the minutes which includes any unusual occurrences. The committee also reviews and accepts or rejects any special experiments or requested changes in the reactor technical specifications and/or operating procedures. Minutes of the committee meetings are recorded and reviewed. The committee does not involve itself with the day-to-day activities of the reactor facility, as these responsibilities lie with the associated faculty and professional staff of the Nuclear Reactor Laboratory.

Membership on the committee consists of two categories:

members and ex-officio members. The committee members are appointed by the Dean of the College of Engineering, usually after conferring with the Director of the Nuclear Reactor Laboratory and the Chairman of the Department of Mechanical Engineering. The members are normally faculty members having some knowledge of nuclear engineering and reactor operation, but have no direct involvement in the operation of the reactor facility. In recent years one or two students have been added to the committee. The members of the committee as of August 1978 are: Professors H.L. Marcus (Committee Chairman and on the Mechanical Engineering faculty), Joseph Ledbetter (Civil Engineering), Dale Klein (Mechanical Engineering), and Jack Marvin (student in Chemical Engineering). The ex-officio members are: the Chairman of the Mechanical Engineering Department (H.G. Rylander), the Director of the Nuclear Reactor Laboratory (E. Linn Draper), the Reactor Supervisor (J. Burack), and the University Radiation Safety Officer (T. Bryant).

Activities and Comments

The recorded activities of the Nuclear Reactor Committee and those items of business pertinent to the committee are included in the minutes of the quarterly meetings, copies of which are attached. These include minutes for the meetings of October 31, 1977; January 30, 1978; May 1, 1978, and

July 27, 1978. Specific items of interest include: (1) Review of the March 9-10, 1978, on-site visit of the Nuclear Regulatory Commission and of the April 15, 1978, response letter clarifiying their comments, and (2) review and approval for unresolved safety questions a modification to one of the automatic shut-down functions. In addition, other repairs were approved, as itemized in the minutes of the meetings.

The full-time permanent faculty associated with the nuclear area is now four, with the addition of Dale Klein and Nolan Hertel (January 1979). This group plus the temporary members of the faculty brings the area beyond the minimal level for a reasonable research effort.

The most significant problem in the nuclear reactor area in supporting the faculty is associated with the non-safety related equipment for both the teaching and research activities of the laboratory. These include (1) The Ge-Li drifted detectors, and (2) the minicomputer interfacing for activation analysis. In the not too distant future the reactor console electronics will have to be upgraded for the sustained efficient operation of the reactor.

Beyond the upgrading of the equipment, the committee sees no significant problems associated with the operation of the reactor.

Harris L. Marcus
Chairman
Nuclear Reactor Committee

III. LABORATORY DEVELOPMENT

A. Organization

1. Dr. Dale E. Klein assumed the position of Director when Dr. Draper resigned on September 1, 1978.

B. Nuclear Reactor Laboratory

The Nuclear Reactor Laboratory is an important part of the Nuclear Engineering program at the University of Texas.

The Nuclear Reactor Laboratory's central feature is a Mark I TRIGA thermal fission reactor. Originally licensed by the Atomic Energy Commission to operate at 10 KW in 1963, the Nuclear Reactor and the Reactor Laboratory have been updated over the past few years and the research capabilities of the Laboratory are now quite diverse.

In 1968, the facility license was amended to allow the TRIGA reactor to operate at a steady state power level of 250 KW and pulsed power of 250 MW which increased experimental capabilities tremendously. The acquisition of Lithium drifted Silicon and Germanium solid state radiation detectors along with dual Nuclear Data 1024 Channel multichannel analyzers coupled to a Data General Nova 800 minicomputer, into which the outputs of the solid state detectors are sent, has made possible rapid, accurate analyses of large quantities of data.

Other experimental devices available at the Laboratory are a Subcritical Assembly, a 2200 Curie Cobalt-60 irradiator, a Nuclear Chicago Sodium Iodide coincidence counting system, a neutron beam irradiation facility and numerous proportional and Geiger-Mueller detection and counting systems.

The Nuclear Reactor and the Reactor Laboratory are also used to teach the fundamentals of reactor operation. Students from all over the state as

well as other organizations and groups have toured the Nuclear Reactor Laboratory. Approximately 1000 persons tour the Laboratory each year.

The use, operation, regulation, security, and monitoring of the Nuclear Reactor Laboratory is controlled by the United States Nuclear Regulatory Commission, the Nuclear Reactor Committee of the University of Texas, the Director of the Nuclear Reactor Laboratory and the Radiation Control Board of the State of Texas.

The use, operation, security and radiation monitoring of the Nuclear Radiation Laboratory is controlled by the Director of the Nuclear Reactor Laboratory and the Radiation Safety Officer of the University of Texas at Austin.

C. Nuclear Radiation Laboratory

The Nuclear Radiation Laboratory at Balcones Research Center has been heavily utilized by the students and staff of the Nuclear Engineering Program at the University of Texas at Austin for the past several years.

The initial use of the building was to house and operate a 14 MeV Texas Nuclear neutron generator. To facilitate experiments utilizing the neutron generator, a large quantity of graphite was obtained, machined and formed into a 8 ft. cube into which was inserted the drift tube of the neutron generator. This device is a very flexible one for investigating neutron migration in matter. When the Nuclear Reactor Laboratory acquired three (3) Californium-252 sources from Louisiana State University in 1973, the Nuclear Radiation Laboratory became the logical place for their storage and use due to the space and availability of the already installed neutron shielding around the neutron generator. Thus, with the acquisition of Cf^{252}, students and staff can perform experiments utilizing not only the high energy neutrons

from the neutron generator but fission spectrum neutrons from Cf^{252}. In addition to the neutron generator and the Californium sources, other smaller radioactive sources are also used within the confines of the Nuclear Radiation Laboratory.

Large amounts of neutron shielding material have been installed in and around the neutron generator cavity and at selected spots on the exterior of the building to protect other experimentalists and the environment from abnormal radiation levels. Because of the posibility of high radiation doses within the confines of the neutron generator cavity, the Nuclear Radiation Laboratory is a limited access building and all external doors are wired into the controls of the generator so that any unwarranted entry automatically shuts down the machine.

D. Subcritical Facility

In association with ME 361F, the undergraduate Nuclear Engineering Laboratory, the subcritical assembly has continued to provide a facility with which the student can perform measurements that are related to fundamental reactor parameters. Such concepts as age, diffusion, and buckling are demonstrated in individual experiments performed by each student.

E. Neutron Activation Analysis Facilities

The Nuclear Analytical Laboratory has provided support for more than twenty separate projects ranging from analyses for criminal investigation to student laboratory support for advanced classes in chemistry, zoology, physics, and engineering. Scientific articles based upon the results of unsponsored research by this laboratory have been published or accepted for publication in several journals and proceedings, and have been presented at conferences of the state, national and international level.

IV. Laboratory Operation and Utilization

A. Reactor

1. Operation

During this reporting period the reactor operated for a total of 171.9 hours while supporting numerous nuclear engineering and operation courses, research, and other related activities.

2. Utilization

Tables 1 and 2 show the utilization of the reactor for the year and comparisons of this data with that of previous years.

3. Courses Offered Using the Reactor

Table 3 lists the courses offered at The University of Texas at Austin which utilized the reactor and associated facilities.

4. Maintenance

During 1978 most maintenance was limited to routine and adjustments; however, in March of 1978 a loss of magnet power to the Shim Rod occurred. It was determined that there was an open circuit in the magnet. A replacement magnet was purchased from General Atomic. This magnet was 3/8 inch longer than the original so that 3/8 inch of travel on the Shim Rod was lost. The reactivity of the Shim Rod thus decreased by $ .08.

5. Inadvertant Reactor Scrams

During 1978 there were 14 inadvertant shutdowns as compared with 9 in 1977. Table 4 itemized these "SCRAMS" and compares this year's data to previous years'.

TABLE 1

COMPARISON OF PREVIOUS UTILIZATION DATA

Year	Total Hours Reactor In Operation*	Total Burn-up (kW-hrs)	Number of Samples Irradiated
1967-66**	104.5	251	63
1966-67	150.0	595	202
1967-68***	342.6	28,168	2449
1968-69	260.8	49.985	1452
1969-70	222.0	36,477	1640
1970-71	262.5	53,912	2990
1971-72	222.8	38,624	1946
1973	318.6	45,794	1347
1974	226.1	27,641	778
1975	206.095	20,450	363
1976	135.74	11,312	468
1977	139.29	7,509	164
1978	171.9	26,870	178

* Includes experimental setup time, maintenance, etc.
** 1965 was the first year the utilization data were maintained.
*** Reactor upgraded from 10 to 250 kW during this academic year.

TABLE 2

NUCLEAR REACTOR LABORATORY PERFORMANCE DATA, 1978

	Total Hours Reactor In Operation*	Total Burn-up (kW-hrs)	Number of Samples Irradiated
First Quarter 1978	24.65	55	3
Second Quarter 1978	39.7	1,126	24
Third Quarter 1978	23.67	3,908	31
Fourth Quarter 1978	83.8	11,781	120
TOTAL	171.82	26,870	178

* Includes experimental setup time, maintenance, etc.

TABLE 3

COURSES UTILIZING THE REACTOR AND ASSOCIATED FACILITIES

Course Number	Course Description
ME. 397	This course was a project course in which the experiments for ME. 397M were documented for use as an aid to the students.
ME. 361F	Introductory Nuclear Laboratory Designed as an introduction to radioactive decay, activation, and radiation measurement.
CH. 376K	A senior level course in instrumental analytical methods which utilizes the Nuclear Reactor Laboratory Facilities to study Neutron Activation Analysis.
CE. 390L	Environmental Analysis - This is a graduate course in environmental analysis for environmental health engineers. Students use neutron activation analysis and energy dispersive X-ray emission spectrometry to analyze samples for toxic contaminants.
ME. 389R	A special projects laboratory course for foreign nuclear engineers.
ME. 379M	A reactor operations course, open to all students. Designed to allow students with an interest in nuclear power to learn basic reactor theory and participate in actually operating a nuclear reactor.

TABLE 4

REACTOR SCRAMS -- INADVERTANT

Intentional	--	3
Operator Error	--	11
Instrument Error	--	0
Power Outage	--	0
Safety	--	0
Total	--	14

TABLE 5

COMPARISON OF YEARLY INADVERTANT SCRAMS

'63	'64	'65	'66	'67	'68	'69	'70	'71	'72	'73	'74	'75	'76	'77	'78
10	9	3	4	3	31	15	11	13	6	10	4	7	5	9	14

V. LABORATORY INSPECTIONS

A. An inspection by members of the Nuclear Regulatory Commission Region IV office regarding operations occurred during March, 1978, in which four items of non-compliance were found.

VI. PUBLIC SERVICE ACTIVITIES

These activities include those other than research projects or assistance provided in the pursuit of research which are covered in Section VII of this report. In this section the subjects are symposiums, lectures, presentations, and tours.

A. Summer High School Science Teachers Symposium

Funded by the Electric Utility Companies of Texas, this program was designed to familiarize high school science teachers with the theory and technology of enrrgy resources. During the summer of 1978, 32 teachers participated in the program.

B. Lectures and Presentations

On numerous occasions during 1978 the NRL staff gave talks on subjects including: "Nuclear Reactor Safety," "Nuclear Engineering and Society," and "Research and Development of Energy Resources."

C. Tours and Radiation Monitoring

During the calendar year of 1978, 952 persons visited the lab. Most of these persons represented educational, civic, or industrial organizations, while others were part of formal engineering laboratory groups. All persons working in or around the laboratory are provided with personal radiation monitoring devices while tour group members are randomly monitored. Measurable exposure to each of the above persons was less than .1 Rem in all cases. The highest exposure to any one person this year was .05 Rem.

VII. Research Projects

A. Funded Research Activities

The Nuclear Reactor Laboratory is pursuing research in numerous areas which are sponsored by the organizations listed below.

1. The Texas Atomic Energy Research Foundation is funding engineering studies pertinent to the design and construction of electric power plants based on controlled thermonuclear fusion. These studies treat the interaction of radiation in a blanket region which acts as a heat generator for use with conventional steam turbine equipment. Environmental effects of larger power generating stations are being evaluated. Fuel resources for fusion reactors are being examined and measurements of neutron interactions with typical materials are being made.

2. The U.S. Energy Research and Development Administration has provided research support by providing reactor fuel for the operation of the University of Texas at Austin TRIGA reactor.

3. The Electric Utility Companies of Texas have sponsored Summer High School Science Teachers Symposium, a program designed to familiarize these teachers with the theory and technology of energy resources.

4. The Electric Power Research Institute sponsors research on the feasibility of reducing the hazard of long-lived radioactive waste from fission reactors. The investigations involve the evaluation of a variety of conceptual fusion reactor blankets and a combination of fission products.

B. Assistance to University Teaching and Research Progress

1. Advanced Analytical Chemistry - Ch 376K

Personnel: J. Holcomb, Chemistry, U.T. Austin
A. H. Pradzynski, Nuclear Reactor Laboratory

Undergraduate students in chemistry performed quantitative determinations of several trace elements in samples by instrumental neutron activation analysis. This was part of a course designed to acquaint senior level students with various instrumental methods of chemical analysis.

2. Short Course on "Nuclear Radiation Application" was offered in April 1978.

Director: E. L. Draper, Jr.

Personnel: A. H. Pradzynski
T. L. Bauer

The short course was offered to scientists and engineers interested in using nuclear radiation in their research. Enrollment was 19 persons from chemistry, archaeology, pharmacy, geology, geophysics, health science and radiation safety. They come from UT campus in Austin, Balcones Research Center, Geophysics Laboratories in Galveston, UT Health Science School in Dallas and ISOTEX-private company in Houston.

3. Activation Analysis of Marine Sediments and Manganese Nodules for Rare Earths Elements

Personnel: S. K. Addy
A. H. Pradzynski

Activation analysis was used in a research project performed at the Geophysics Laboratoy of the UT Marine Science Institute in Galveston. Sixty samples and standards were irradiated and counted after different decay times. Results were reported in a paper presented at a scientific meeting in Port Aransas.

4. Activation Analysis of Ferro-Alloys

Personnel: A. H. Pradzynski

In cooperation with University of Monterey, Mexico samples of Ferro-alloys were analyzed.

C. Other Analyses

1. Analyses of Bronze Artifacts for Major and Minor Elements

Personnel: A. H. Pradzynski
D. Hamilton, TARL, UT Austin

Bronze and other artifacts were analyzed by x-ray fluorescence spectrometry.

D. Unsponsored Research

1. Analysis of Uranium and Thorium in Water by X-ray Fluorescence Spectrometry

Personnel: A. H. Pradzynski

Previous work on U and Th determination by APDC precipitation and X-RF spectrometry was extended in order to use the method for control of contamination of the environment around uranium mining sites in Texas. Results were presented at the Annual Meeting of Texas Association for Radiation Research in Houston, Texas, November 4-5, 1977.

E. Graduate Research

1. Neutron Interaction Studies

a) Fission Devices

E. Linn Draper, Jr. - P.I.
Y. Kunimoto - Research Assistant (MS Candidate)

Toxicities of Fuel Cycle Waste Streams

An evaluation is underway of the toxicity of potential waste streams at each step of six nuclear fuel cycles. The objective is to identify fuel cycles for which long term hazard can be made acceptably small and to emphasize the importance to toxicity of potential waste streams which have received little attention to date.

b) Fusion Devices

i) Systems Analysis of Nuclide Transmutation

E. Linn Draper, Jr. - P.I.
J. W. Davidson - Research Assistant (PhD Candidate)

A system including light water reactors, reprocessing, partitioning, and transmutation devices is being evaluated to ascertain required neutron fluences and chemical separation capabilities to reduce effluent toxicity to a few times uranium ore toxicity.

ii) Radiation Hazards from Structural Activation of Fusion Reactors

E. Linn Draper, Jr. - P.I.
R. D. Smith - Research Assistant (PhD Candidate)

Comparisons are being made of the potential hazard from structural activation of a variety of proposed materials in fusion devices. Several first wall replacement strategies are being evaluated on the basis of their risk to both workers and the public.

iii)Fusion-Fission Hybrid Safety Analysis

Dale Klein - P.I.
Yueh-Sam Yang - Research Assistant (PhD Candidate)

An examination is being made of the heat generation and removal characteristics of a fusion hybrid blanket. Consideration will be given to transient as well as steady state operation.

c) Accelerator Devices

i) Fissile Fuel Production

E. Linn Draper, Jr. - P.I.
Stephen Ganthner - Research Assistant (MS Candidate)

For many years high current accelerators have been proposed as neutron sources to breed fissile material from either uranium-238 or thorium-232. Recent policy decisions on the deployment of breeder reactors and the possible advent of inexpensive accelerators have renewed interest in this concept. This work is aimed at establishing the technical requirements for success.

ii) Heat Removal Considerations

E. Linn Draper, Jr. - P.I.
M. Razzaque - Research Assistant (PhD Candidate)

The removal of heat from the target/blanket region of a 300 mA, 1 GeV accelerator while retaining an appropriate neutron population is being evaluated.

2. Radioactive Waste Repository Siting

a) Public Acceptance

S. Lopreato, E. Linn Draper, Jr. - P.I.
W. Murphie - Research Assistant (MS Candidate)

An examination will be made of the ways in which people acquire technical information and how that information can most effectively be conveyed.

b) Design and Economics

i) Temperature Considerations in Waste Repositories

Dale Klein - P.I.
James Hamann - Research Assistant (MS Candidate)

An evaluation is being made of the temperature distribution in geologic repositories for spent fuel and reprocessed wastes in a

variety of formations as a function of canister spacing and time after emplacement.

ii) Spent Fuel Disposal Charges

E. Linn Draper, Jr. - P.I.
T. Sanders - Research Assistant (MS Candidate)

The Department of Energy has proposed to accept, store, transport, and dispose of spent nuclear fuel assemblies for a fixed fee. The assumptions of quantities, delivery schedules, cost of facilities and pricing methodology are being evaluated critically.

iii) Environmental Impacts of Repository Effluents

E. Linn Draper, Jr. - P.I.
Sharif-Homayoun - Research Assistant (MS Candidate)

The pathways by which radionuclides might be transported from the repository to man through the geosphere and biosphere are being evaluated and calculations will be made for representative sites.

3. Gas Cooled Fast Reactor Studies

a) Temperature Profile and Pressure Drop Around GCFR Spacer Grids

Dale Klein - P.I.
G. Taylor - Research Assistant (MS Candidate)

Recent experimental data indicate an increased pressure drop around grid spacers for rough rods as compared to smooth rods. Since the fuel rod surface for the GCFR will be roughened, an accurate analysis of these grid spacers is needed. A thermal hydraulic computer code (SAGAPO) has recently been developed at Karlsruhe, Germany. This code will be modified to consider the recently observed pressure drop characteristics. The results of this study will be used in the Core Flow Test Loop (CFTL) being built at ORNL.

b) Temperature and Pressure Drop Studies at the ORNL CFTL

Dale Klein - P.I.
S. Hodge - Research Assistant (PhD Candidate)

The recently developed code (SAGAPO) described above has some severe restrictions that prevent its application to analysis of the Core Flow Test Loop (CFTL). One of these restrictions involves the fuel rod diameter. The CFTL will have larger diameter bundle support rods and this prevents the thermal-hydraulic analysis using SAGAPO. This project involves the modification of SAGAPO to include a new method of channel modeling. Preliminary results on a new modeling method for obtaining the parameters look extremely promising and have important implications on present transformation methods. After SAGAPO has been modified, the results will be applied to the CFTL and checked with the U.S. computer version COBRA4.

c) Experimental Measurement of Local Heat Transfer Values

Dale Klein - P.I.
J. Percy - Research Assistant (MS Candidate)

A method is being devised for the interpretation of data for a single roughened surface placed inside a rectangular wind tunnel. An aluminum test base with removable roughness elements has been designed and fabricated. Local heat transfer measurements will be obtained using Rdf heat flux sensors. Heat will be generated by electrical resistance underneath the aluminum test base. After the determination of these local heat transfer values surrounding the roughness elements the result will be applied to the roughness design for the GCFR.

4. Nuclear Analytical Techniques

Dale Klein - P.I.
M. Aliy - Research Assistant (MS Candidate)

A preconcentration method for field sampling of trace elements in water is being developed. Specifically this technique will be applied to uranium concentrations in water. This method can be used as an indication for potential uranium deposits as well as environmental monitoring at a mining site.

TABLE 6

RESEARCH FUNDING

Texas Atomic Energy Research Foundation	$ 40,000
ERDA -- Californium-252 Loan Program	----
Center for Energy Studies	32,713
Southwest Research Institute	23,736
TOTAL	$ 96,449

VIII. Publications From Nuclear Reactor Laboratory

1. K.L. Gilbert, "Neutron Flux Mapping of a Subcritical Reactor Core with a Polyethylene Reflector", Masters Thesis, Physics Department, The University of Texas, 61 pp., June 1961.

2. T.T. Doss, "Neutron Density Distribution in and Unreflected Subcritical Reactor Core", Masters Thesis, Physics Department, The University of Texas 63 pp., June 1961.

3. J.M. Norwood, "The Point Source Transport Solution for the Position and Velocity Dependent Neutron Distribution in a Spherical Body of Non-Multiplying Material", Masters Thesis, Physics Department, The University of Texas, 75 pp. June, 1962.

4. R.W. Reed, "Effect of a Cadmium Control Rod on the Neutron Density in a Subcritical Reactor", Masters Thesis, Physics Department, The University of Texas, 85 pp., August 1962.

5. P. Berananda, "Neutron Flux Distribution of a Subcritical Reactor Core with a Graphite Reflector", Masters Thesis, Physics Department, The University of texas, 40 pp., January 1962.

6. T.A. Fredericks, "Thermal Neutron Flux Distribution Due to a Plane Wave Source in and Unreflected Reactor Assembly", Masters Thesis, Physics Department, The University of Texas, 107 pp., August 1963.

7. D.G. Martin, "Film Detector for a Neutron Spectrometer", Masters Thesis, Physics Department, The Univerisyt of Texas, June 1963.

8. M.L. West II, "Flux Decay Rate in a Reflected Subcritical Reactor", Masters Thesis, Physics Department, The University of Texas, 55 pp., August 1963.

9. R.D. O'Dell, "The Angular Thermal Neutron Spectrum in the Vicinity of the Interface Between Two Media", Ph.D. Dissertation, The University of Texas, 186 pp., January 1965.

10. B.E. Thompson, Sr., "Cryogenic Neutron Spectra Measurements Across a Discontinuity in Temperature and Properties", Ph.D. Dissertation, The University of Texas, 149 pp., January 1965.

11. R.S. Kolflat, "An Experimental Approach to the Study of Nucleonic Fundamentals", Masters Thesis, Mechanical Engineering Department, The University of Texas, 190 pp., May, 1965.

12. D.G. Anderson, "A Technique for Recoil Fission Fragment Range Measurements", Health Physics, Vol. 12, No. 4, 558, April 1966.

13. J.B. Whitworth, "A Cryogenic Irradiation Device for the TRIGA Mark I Nuclear Reactor", Masters Thesis, Mechanical Engineering Department, The University of Texas, 52 pp., January 1967.

14. D.A. Pullen, " A Rod Oscillator Design and Zero Power Transfer Function Measurement for a TRIGA Mark I Reactor," Masters Thesis, Mechanical Engineering (Nuclear Engineering), The University of Texas, January, 1967.

15. G.F. Malan, "Transfer Function Analysis of Temperature and Xenon Feedback in Coupled-Core Nuclear Reactor Systems", Masters Thesis, Mechanical Engineering Department, The University of Texas, 69 pp., August 1967.

16. M.G. Stevenson, "Investigations on the Macroscopic Nodal Approach to Space-dependent Nuclear Reactor Kinetics", Ph.D. Dissertation, The University of Texas, 203 pp., August 1968.

17. F.H. Antunez-Castillo, "Gamma Radiation Dosimetry Techniques and Application to Mapping of The University of Texas Cobalt-60 Irradiation Facility", Masters Thesis, Physics Department, The University of Texas, 60 pp., September 1968.

18. R. Valiente, "Neutron Radiography with the University of Texas TRIGA Nuclear Reactor", Masters Thesis, Physics (Nuclear Engineering), The University of Texas, August 1968.

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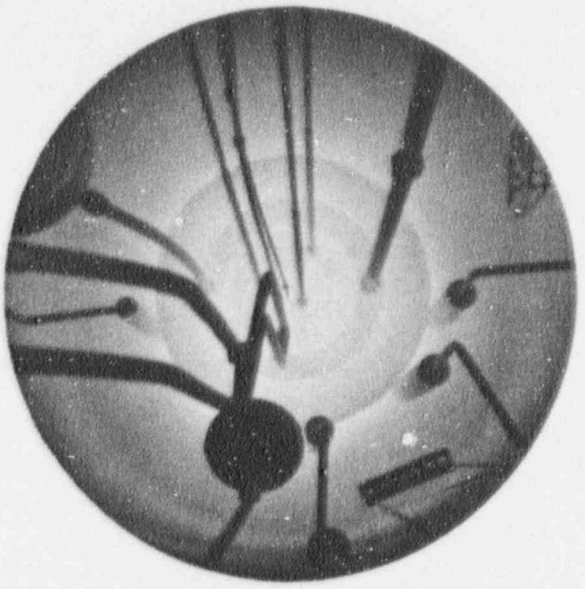

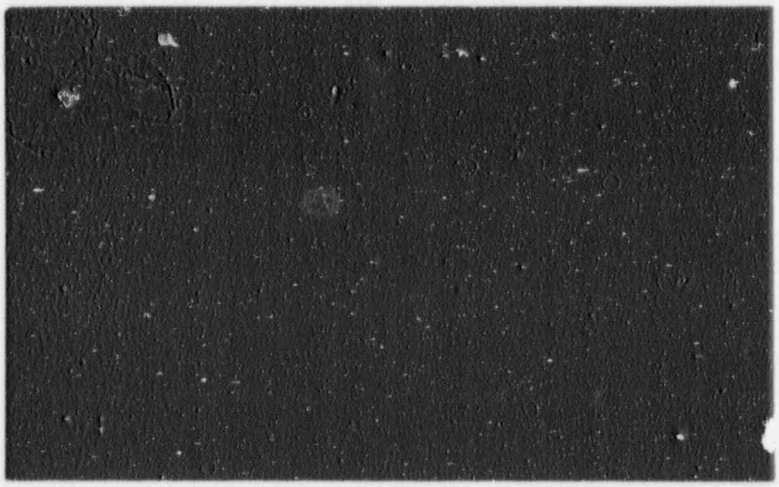

NUCLEAR REACTOR LABORATORY

TECHNICAL REPORT

THE UNIVERSITY OF TEXAS
COLLEGE OF ENGINEERING
DEPARTMENT OF MECHANICAL ENGINEERING

TECHNICAL INFORMATION
AND
HAZARDS SUMMARY REPORT
FOR A
TRAINING, RESEARCH, AND ISOTOPE-PRODUCTION
250 KW, PULSING NUCLEAR REACTOR
AT
THE UNIVERSITY OF TEXAS
AUSTIN, TEXAS 78712

December 1978

CONTENTS

FIGURES

FOREWORD

An analysis of the hazards that might arise while operating the TRIGA Mark I pulsing reactor is presented in this report. In order to assure that the analysis is conservative, it has been assumed that the reactor has been operating for a long time at a constant power level of 250 kw. Also, the design of the relatively inaccessible below-ground tank has been based upon an assumed steady-state power of 250 kw.

The operational safety of the reactor results from an inherent self-regulating characteristic of the core. This self-regulation is obtained from the prompt negative temperature coefficient, which is a property of the homogeneous uranium-zirconium-hydride fuel-moderator elements that form the reactor core. The inherent safety of the reactor is supplemented by careful procedures adopted for the operation of the reactor.

On the basis of the hazards evaluation presented in this report, it can be assumed that operation of the reactor will not cause undue risk to the health and safety of the staff or of the public.

1. INTRODUCTION

This report is an update and modification of the Hazard Summary Report originally submitted in support of an application by The University of Texas at Austin for a Construction Permit and a Utilization Facility License to permit the installation and operation of a training, research, and isotope-production reactor (TRIGA-I) that was purchased from the General Atomic Division of General Dynamics Corporation. It includes the specifications for this reactor and an evaluation of possible hazards.

The reactor core, of a type developed by General Atomic*, uses zirconium hydride as the moderator. A large number of reactors of this type have been installed in both the U.S. and abroad in Universities and other research facilities. Many of these have been in continuous service for more than 15 years; the first was installed in 1958. The University of Texas at Austin reactor went critical 1963.

A comprehensive test program and long experience have conclusively demonstrated this reactor's inherent safety, which is due to its prompt negative temperature coefficient of reactivity. The reactor installed at The University of Texas is equipped with conventional safety rods and electronic scram equipment.

Initial loading and start-up of The University of Texas reactor was performed by General Atomic personnel. University of Texas personnel were trained at General Atomic in the operation and maintenance of the reactor. During the past years a continuous training program for new personnel at the University of Texas at Austin has been in existence.

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*General Atomic is now a joint venture of Gulf Oil Co. and a subsidiary of Royal Dutch Shell.

The TRIGA family of reactors was developed by General Atomic Division of General Dynamics Corporation for use by universities and research institutions as a general-purpose research and training facility.

The TRIGA-I pulsing reactor utilizes the TRIGA core with pulsing fuel-moderator elements. The reactor is licensed for steady-state operation at a power level of 250 kw and for routine pulsed operation. The total loading of the TRIGA-I reactor system is 1.8% δ k/k ($2.41) excess reactivity above a cold, clean, critical, compact condition. Pulsed operation will be limited to rapid insertions of reactivity of up to $2.00. For this discussion, a pulse is defined as the step insertion of an amount of excess reactivity between 0.75% and 1.5% δ k/k.

From past experience, it is concluded that the TRIGA reactor can be operated at 250 kw steady-state power and can be pulsed up to a 1.5% δ k/k transient without hazard to persons using the reactor or to the general public, provided normal precautions for handling radioactive materials are observed.

In addition, General Atomic has demonstrated by direct experiment on its prototype reactor that the TRIGA-type reactor is so stable that even rapid insertion of 3.1% δ k/k reactivity causes no hazard to individuals near the reactor and in the surrounding area. After a power transient, the prompt negative temperature coefficient of the reactor limits the total energy release to a safe level.

2. LOCATION AND GENERAL FEATURES OF THE SITE

2.1. General Location

The reactor is situated near the center of the main campus of The University of Texas, City of Austin, Travis County, Texas (see Figures 1 and 2). The reactor is located in a laboratory room in the mechanical engineering building, Taylor Hall (see Figures 5 and 6).

The campus is located on the high point of a gentle knoll. The reactor is located on the east slope and about fifty feet below the crest of the knoll and about forty feet above Waller Creek, two blocks away. The reactor site is approximately 1.5 miles east, north, and west of branches of the Missouri Pacific, Missouri Kansas Texas, and Texas New Orleans (Southern Pacific) railroads, 0.8 miles north of the Texas State Capitol building, 2.0 miles southwest of the Austin Municipal Airport, eight miles northwest of Bergstrom Air Force Base, 1.7 miles north of the (Texas) Colorado River, and 0.6 mile west of Interstate Highway 35.

2.2. Surrounding Activities and Population

The 1978 population of Austin is 308,000, and the surrounding Travis County rural population is approximately 30,000. A total of approximately 330,000 people live within a 20-mile radius of the site. It is estimated that there will usually be approximately 40,000 people within a radius of one mile of the reactor site during normal working hours. Population densities are shown in Figures 3 and 4.

Austin is composed primarily of governmental, business, and professional persons and their families, with substantial light industry and practically no heavy industry. The population has roughly doubled since 1960, and it is

MAP OF
AUSTIN AND VICINITY

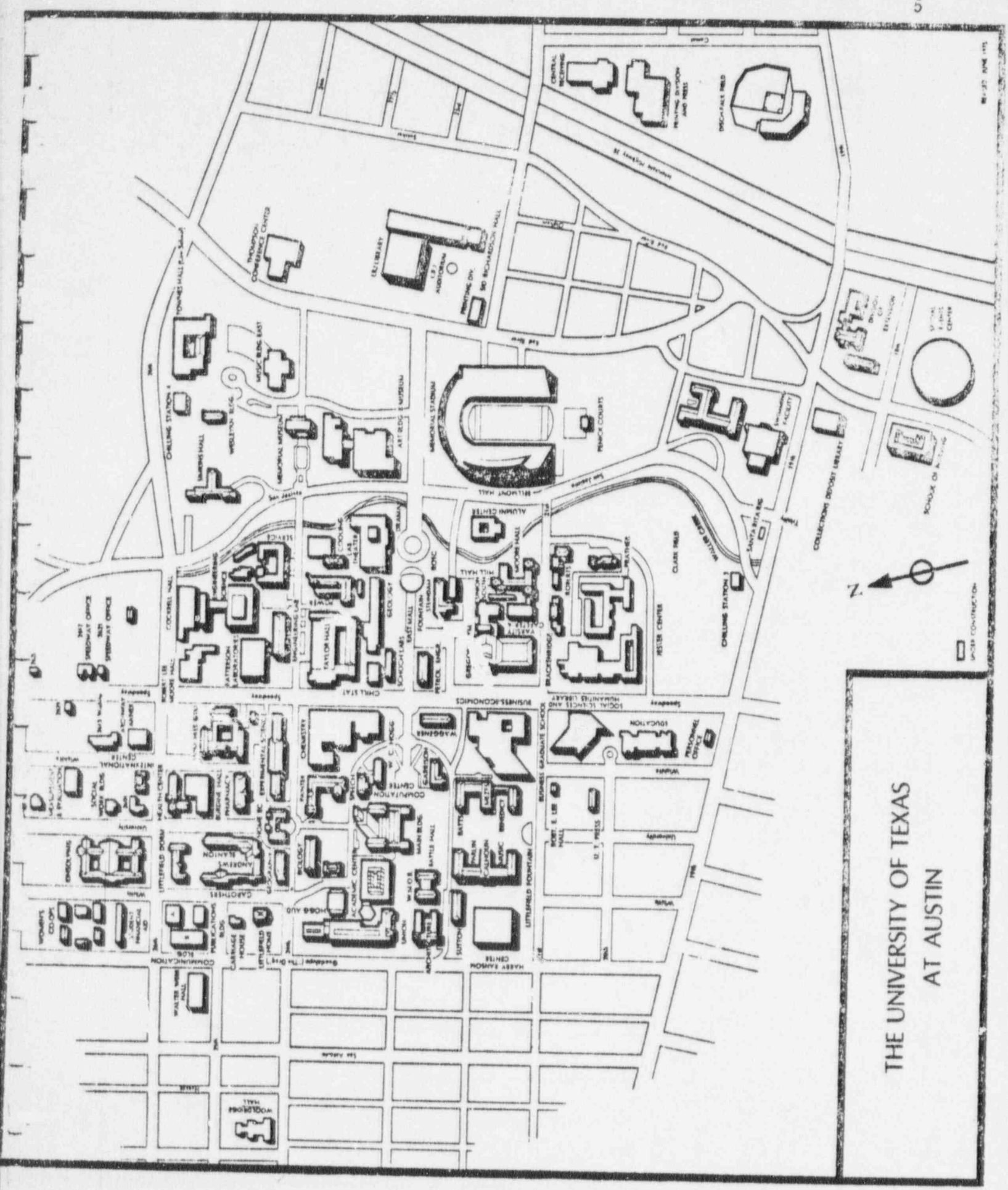


Figure 2

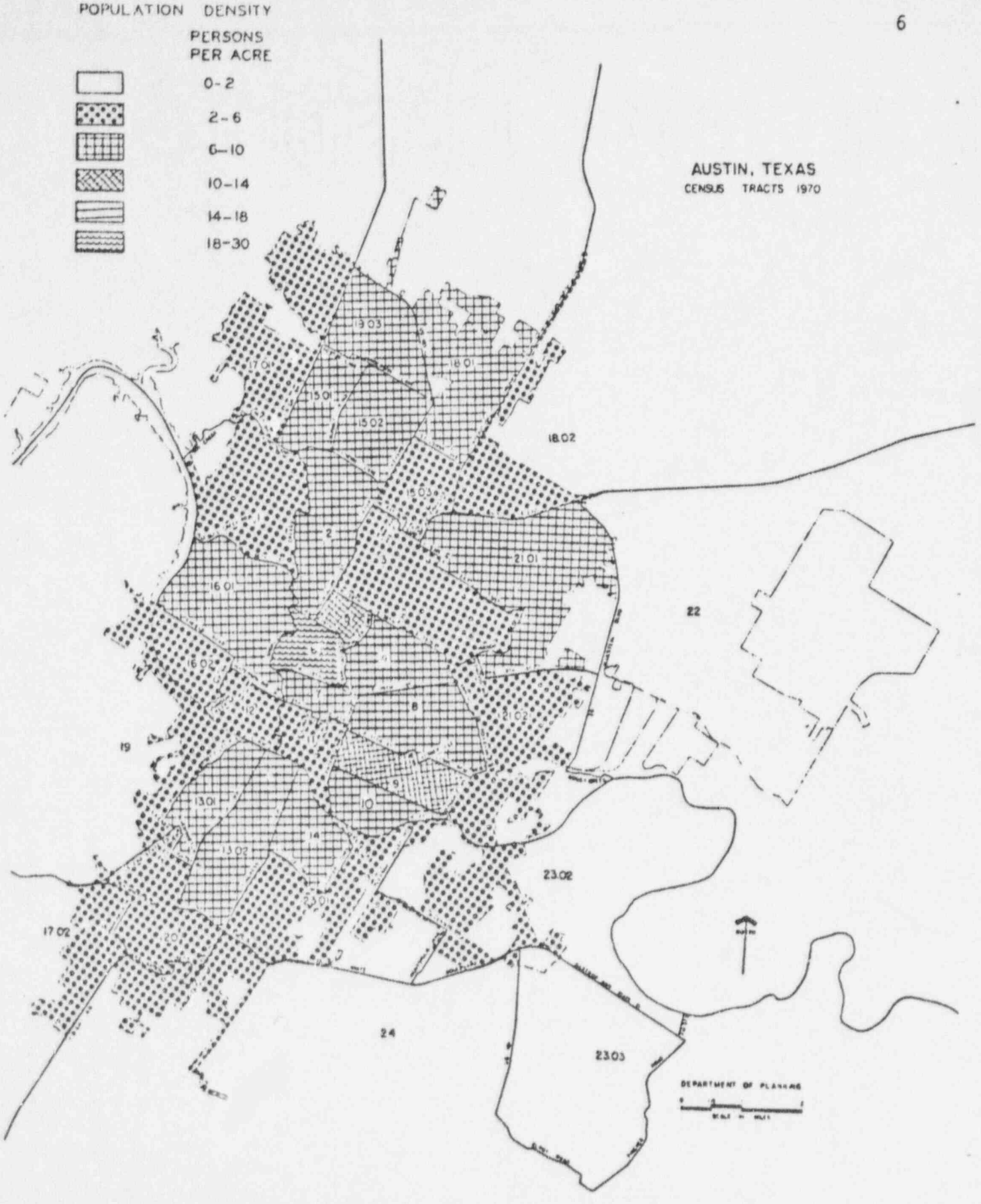


Figure 3

Austin, Texas SMSA Census Tracts: 1970
(Travis County)

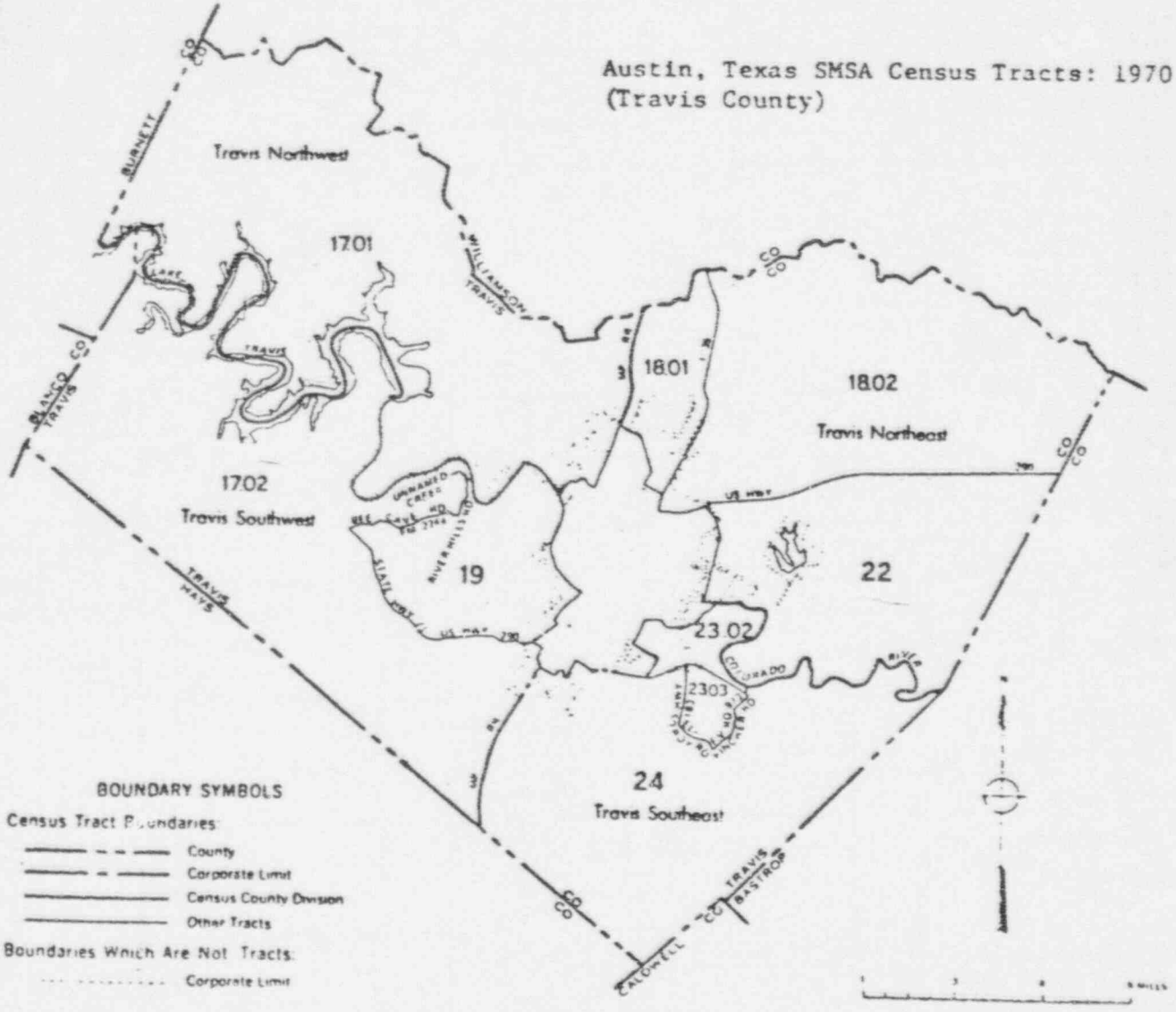
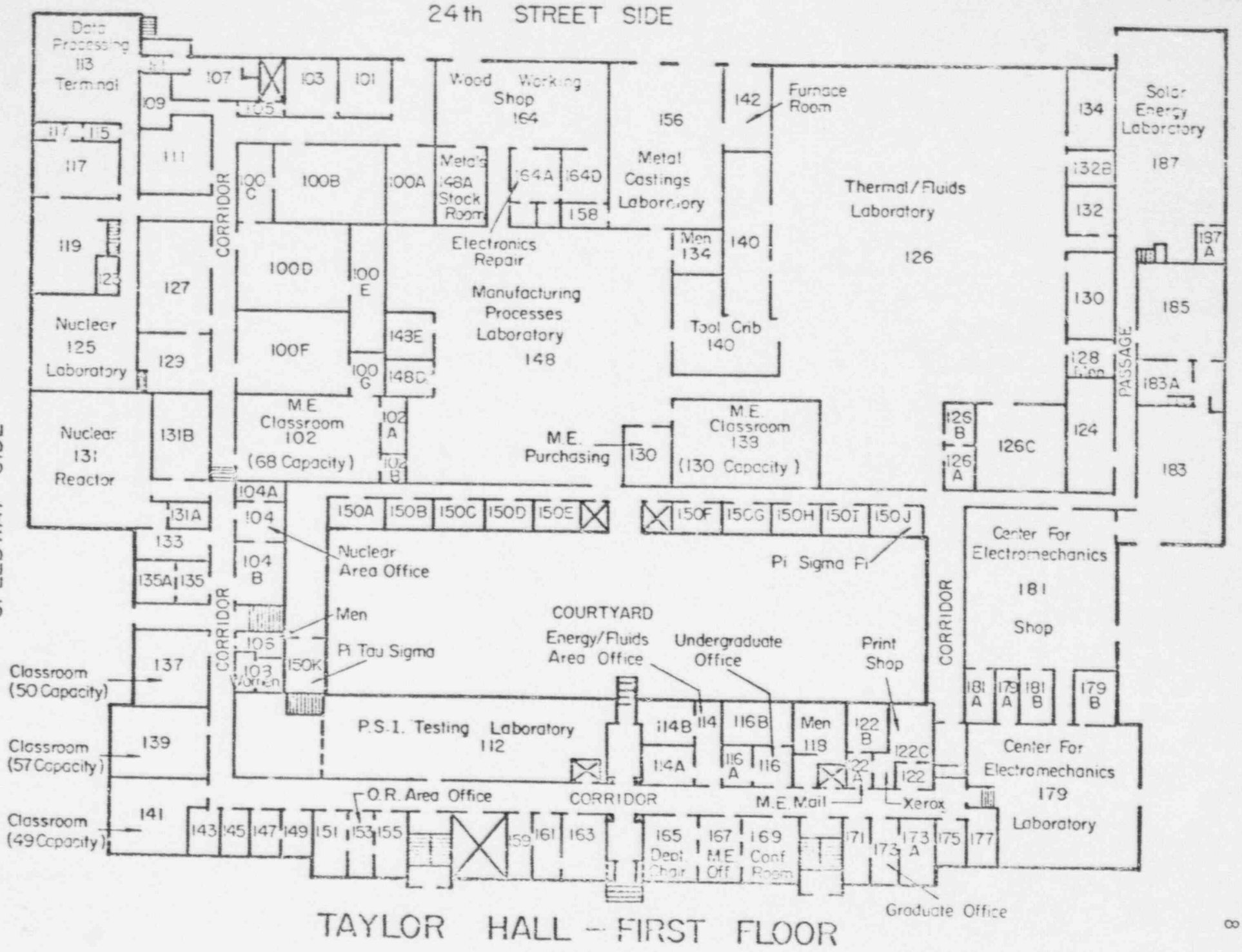


Figure 4



TAYLOR HALL – FIRST FLOOR

Figure 5

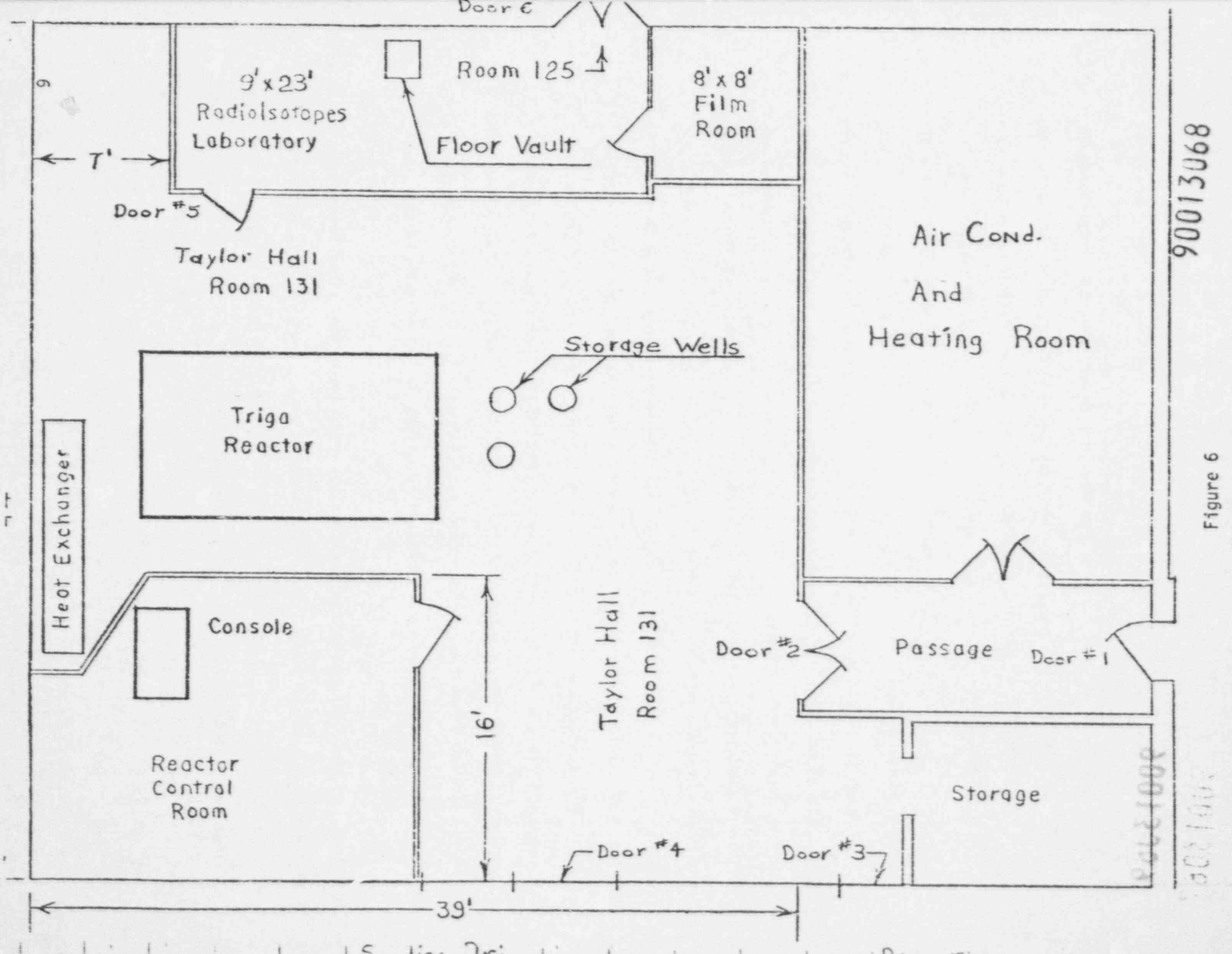

Door C
Room 125
9'x23'
Radiolsotopes
Laboratory
Floor Vault
8'x8'
Film
Room
7'
Door #5
Taylor Hall
Room 131
Air Cond.
And
Heating Room
Storage Wells
Triga
Reactor
Heat Exchanger
Console
16'
Taylor Hall
Room 131
Door #2
Passage
Door #1
Reactor
Control
Room
Storage
Door #4
Door #3
39'
9001306B

Figure 6

expected that the rate of growth and type of activities in the future will be about the same.

2.3. Meteorology

Austin, capital of Texas, is located on the Colorado River where the stream crosses the Balcones Escarpment separating the Texas Hill Country from the Blackland Prairies to the east. Elevations within the City vary from 400 feet to 900 feet above sea level. Native trees include cedar, oak, walnut, mesquite, and pecan.

The climate of Austin is humid subtropical with hot summers. Winters are mild, with below freezing temperatures occurring on an average of less than 25 days each year. Rather strong northerly winds, accompanied by sharp drops in temperature, occasionally occur during the winter months in connection with cold fronts, but cold spells are usually of short duration, rarely lasting more than two days. Daytime temperatures in summer are hot, but summer nights are usually pleasant with average daily minima in the low seventies.

Precipitation is fairly evenly distributed throughout the year, with heaviest amounts occurring in late spring. A secondary rainfall peak occurs in September. Precipitation from April through September usually results from thundershowers, with fairly large amounts falling within short periods of time. While thunderstorms and heavy rains have occurred in all months of the year, most of the winter precipitation occurs as light rain. Snow is insignificant as a source of moisture, and usually melts as rapidly as it falls. The City may experience several seasons in succession with no measurable snowfall.

Prevailing winds are southerly throughout the year. Northerly winds accompanying the colder airmasses in winter soon shift to southerly as these airmasses move out over the Gulf of Mexico.

The average length of the warm season (freeze-free period) is 270 days. Based on data from 1943-1961, the average date of the last occurrence of 32°F in spring is March 3; the average date of the first fall occurrence of 32°F is November 28. Previous records show that 32°F or below has occurred as late as April 13 (1940), and as early as October 26 (1924).

Destructive winds and damaging hailstorms are infrequent. On rare occasions, dissipating tropical storms affect the City with strong winds and heavy rains.

Figures 7a, 7b, and 7c are a summary of meterological data for Austin for a typical year, 1977. Figure 8 shows monthly surface wind roses prepared from a summary of data for five recent years.

2.4. Geology and Hydrology

The northwestern half of Travis county is part of the physiographic province of Texas known as the Edwards Plateau. In Travis County this is a highly dissected plateau with wooded hills rising in some places more than 500 feet above the drainageways. In marked contrast, the southeastern half of the county is gently rolling prairie land which is part of the physiographic province known as the Gulf Coastal Plain. These provinces are separated by the scarp of the Balcones fault zone, which rises 100 to 300 feet above the Coastal Plain. The scarp, however, is not a vertical cliff; it is an indented line of sloping hills leading up from the lower plain to the plateau summit.

Almost the entire county is drained by the Colorado River and its tributaries. Lake Austin and Lake Travis, which are formed by the Tom Miller and Mansfield Dams, respectively, on the Colorado River, are part of the power, flood-control, water-conservation, and recreation project of the Lower Colorado Authority.

Meteorological Data For The Current Year

Station: AUSTIN, TEXAS (# 13958) — MUNICIPAL AIRPORT — Standard time used: CENTRAL — Latitude: 30° 18' N — Longitude: 97° 42' W — Elevation (ground): 597 feet — Year: 1971

Month	Temperatures °F: Averages: Daily maximum	Daily minimum	Monthly	Extremes: Highest	Date	Lowest	Date	Degree days Base 65 °F: Heating	Cooling	Precipitation in inches: Water equivalent: Total	Greatest in 24 hrs	Date	Snow, ice pellets: Total	Greatest in 24 hrs	Date	Relative humidity, pct: Hour 00 (Local time)	Hour 06 (Local time)	Hour 12 (Local time)	Hour 18 (Local time)	Wind: Resultant: Direction	Speed m.p.h.	Average speed m.p.h.	Fastest mile: Speed m.p.h.	Direction	Date	Percent of possible sunshine	Average sky cover, tenths, sunrise to sunset	Number of days: Sunrise to sunset: Clear	Partly cloudy	Cloudy	Precipitation .01 inch or more	Snow, ice pellets 1.0 inch or more	Thunderstorms	Heavy fog, visibility ¼ mile or less	Temperature °F: Maximum: 90° and above	32° and below	Minimum: 32° and below	0° and below	Average station pressure mb
JAN	50.8	32.3	41.6	80	27	19	10	719	0	2.25	0.74	22-25	0.0	0.0		72	80	62	81	01	3.7	9.1	37	NW	9	91	6.6	6	8	17	10	0	1	0	0	19	0	0	0	998.6
FEB	66.0	42.1	54.1	86	24	31	6	908	7	2.58	1.87	10-11	T	T	1	66	76	51	65	29	0.5	9.0	37	NW	23	72	4.7	11	11	6	7	0	0	0	0	2	0	0	0	997.3
MAR	72.4	55.1	63.8	86	15	35	7	148	58	2.18	1.08	2-3	0.0	0.0		67	73	51	47	13	1.8	10.9	30	N	19	93	3.6	7	9	15	7	0	0	0	0	3	0	0	0	991.5
APR	77.4	56.6	67.0	88	3	45	6	20	90	6.08	2.22	15	0.0	0.0		78	89	57	58	15	2.5	8.9	36	NE	16	60	5.6	10	9	11	11	0	0	0	0	7	0	0	0	995.6
MAY	85.9	67.2	75.3	92	31	60	12	0	324	1.24	0.69	20-21	0.0	0.0		86	94	66	64	16	6.6	8.3	31	NW	21	36	8.2	1	8	22	10	0	0	0	0	5	0	0	0	991.9
JUN	92.0	72.4	82.2	94	29	68	8	0	526	1.22	0.56	22-23	0.0	0.0		83	90	55	56	18	6.0	9.1	32	SW	22	76	3.5	7	15	8	7	0	1	0	0.4	26	0	0	0	992.9
JUL	95.7	74.9	85.1	101	31	71	16	0	629	0.21	0.18	17	0.0	0.0		74	48	48	42	18	7.3	8.2	29	E	17	87	3.9	13	17	1	2	0	0	0	0	31	0	0	0	994.9
AUG	97.7	75.8	86.8	101	24	74	31	0	681	0.06	0.04	28	0.0	0.0		71	87	47	42	16	6.1	8.5	25	SE	27	85	4.9	9	17	5	2	0	0	0	0	30	0	0	0	992.5
SEP	94.7	73.4	83.9	99	5	66	15	0	572	3.10	1.34	12-13	0.0	0.0		77	87	52	48	18	5.2	9.0	38	NE	19	75	4.4	11	14	5	9	0	0	1	0	27	0	0	0	993.9
OCT	82.9	65.7	71.4	95	1	42	13	16	227	1.19	1.05	21-22	0.0	0.0		68	79	49	47	14	1.9	7.5	33	N	11	87	4.7	14	9	8	4	0	0	1	0	2	0	0	0	995.6
NOV	72.5	50.8	61.7	84	16	35	10	136	40	1.69	0.75	8	0.0	0.0		75	92	58	59	32	1.7	8.9	32	NW	9	55	5.2	12	6	12	7	0	1	2	0	0	0	6	0	993.9
DEC	65.1	42.0	53.6	84	4	25	22	354	7	3.34	0.22	28-29	0.0	0.0		59	88	47	48	33	0.3	9.6	29	N	9	95	5.1	14	6	11	4	0	0	4	0	0	0	6	0	995.3
YEAR	79.1	58.2	68.7	101	AUG 24	19	JAN 10	1701	3141	27.14	2.22	APR 15	T	T	FEB 1	73	83	54	51	17	2.6	8.8	38	NE	SEP 19	65	5.4	115	129	121	76	0	29	26	119	0	23	0	994.3

Normals, Means, And Extremes

Month	Temperatures °F: Normal: Daily maximum	Daily minimum	Monthly	Extremes: Record highest	Year	Record lowest	Year	Normal Degree days Base 65 °F: Heating	Cooling	Precipitation in inches: Water equivalent: Normal	Maximum monthly	Year	Minimum monthly	Year	Maximum in 24 hrs	Year	Snow, ice pellets: Maximum monthly	Year	Maximum in 24 hrs	Year	Relative humidity pct: Hour 00 (Local time)	Hour 06 (Local time)	Hour 12 (Local time)	Hour 18 (Local time)	Wind: Mean speed m.p.h.	Prevailing direction	Fastest mile: Speed m.p.h.	Direction	Year	Pct. of possible sunshine	Mean sky cover, tenths, sunrise to sunset	Mean number of days: Sunrise to sunset: Clear	Partly cloudy	Cloudy	Precipitation .01 inch or more	Snow, ice pellets 1.0 inch or more	Thunderstorms	Heavy fog, visibility ¼ mile or less	Temperatures °F: Max: 90° and above	32° and below	Min: 32° and below	0° and below	Average station pressure mb
(a)				36		36					36		36		36		36		36		16	16	16	16	36	15	36	36		36	36	35	36	36	35	36	36	36	16	16	16	16	
J	60.0	39.3	49.7	90	1971	-2	1949	483	8	1.88	7.94	1968	0.04	1971	3.44	1965	7.0	1944	7.0	1944	73	79	60	57	9.8	5	47	N	1962	49	6.2	9	6	16	8	4	1	0	0	0	997.9
F	63.8	42.8	53.3	93	1954	7	1971	344	10	3.09	6.39	1958	0.28	1954	3.73	1958	6.0	1966	6.0	1966	70	78	56	50	10-12	5	37	N	1947	53	6.0	9	6	12	8	4	2	0	0	0	996.8
M	70.7	48.2	59.5	98	1971	18	1948	223	32	1.89	4.98	1945	T	1972	2.68	1951	2.0	1965	2.0	1965	71	79	55	48	11-0	5	44	W	1957	55	6.1	9	8	14	7	4	3	1	0	0	993.4
A	79.0	58.2	68.6	98	1963	35	1973	44	152	3.49	9.95	1957	0.10	1961	3.86	1942	0.0		0.0		77	84	59	54	10-7	5SE	44	NE	1957	53	6.3	7	8	15	8	3	1	1	0	0	991.1
M	85.2	65.1	75.2	99	1967	43	1954	0	310	3.97	9.98	1963	0.81	1960	4.87	1975	0.0		0.0		81	88	60	57	9.8	5SE	49	N	1972	57	6.2	7	11	13	8	5	1	1	0	0	993.3
J	91.7	71.4	81.6	103	1955	53	1970	0	498	3.19	11.43	1961	T	1967	8.50	1964	0.0		0.0		79	88	55	53	9.5	S	49	SE	1956	60	5.2	8	15	7	6	0	4	8	20	0	0	992.7
J	95.4	73.7	84.6	109	1954	64	1970	0	608	1.88	8.40	1961	0.00	1962	5.46	1961	0.0		0.0		74	87	50	46	8.5	S	43	SE	1969	76	4.8	11	14	6	5	0	4	9	27	0	0	990.9
A	93.9	73.5	84.7	106	1953	61	1957	0	611	2.20	8.90	1974	0.00	1952	4.68	1945	0.0		0.0		74	87	51	46	8.1	S	53	NW	1969	75	4.7	11	14	6	5	0	5	9	28	0	0	991.5
S	89.4	68.4	78.9	103	1956	41	1942	0	417	3.68	8.11	1942	0.07	1947	6.74	1973	0.0		0.0		79	87	53	50	8.0	S	43	NE	1961	67	5.0	11	11	8	7	0	4	6	19	0	0	993.1
O	81.3	58.9	70.1	96	1956	32	1957	39	197	3.02	12.31	1980	T	1952	7.22	1960	0.0		0.0		76	85	55	55	8.1	S	47	NW	1967	65	4.6	13	9	9	6	0	3	2	3	0	0	995.1
N	70.2	48.0	59.1	91	1951	27	1976	203	38	2.04	7.91	1946	T	1970	5.09	1974	1.0	1959	1.0	1939	76	82	57	58	9.2	S	48	NW	1951	57	5.2	11	7	12	7	0	3	0	0	0	993.5
D	63.0	41.6	52.3	90	1955	16	1950	399	5	2.22	9.91	1944	T	1950	4.02	1953	T	1971	T	1971	73	80	59	58	9.2	S	49	NW	1956	51	5.9	10	6	13	7	0	1	4	0	0	995.8
YR	78.8	57.4	68.1	109	JUL 1954	-2	JAN 1949	1797	2908	32.49	12.31	OCT 1980	0.00	JUL 1962	7.22	OCT 1960	7.0	JAN 1944	7.0	JAN 1944	75	84	56	53	9.3	S	37	N	FEB 1947	61	5.5	116	113	134	82	6	41	24	102	1	23	0	994.3

Means and extremes above are from existing and comparable exposures. Annual extremes have been exceeded at other sites in the locality as follows: Maximum monthly precipitation 23.78 in September 1971; maximum precipitation in 24 hours 19.03 in September 1921; maximum monthly snowfall 9.7 in November 1937; maximum snowfall in 24 hours 9.7 in November 1937.

(a) Length of record, years, through the current year unless otherwise noted, based on January data.
(b) 70° and above at Alaskan stations.
* Less than one half.
T Trace

NORMALS - Based on record for the 1941-1970 period.
DATE OF AN EXTREME - The most recent in cases of multiple occurrence.
PREVAILING WIND DIRECTION - Record through 1963.
WIND DIRECTION - Numerals indicate tens of degrees clockwise from true north. 00 indicates calm.
FASTEST MILE WIND - Speed is fastest observed 1-minute value when the direction is in tens of degrees.

Figure 7a

Average Temperature

Year	Jan	Feb	Mar	Apr	May	June	July	Aug	Sept	Oct	Nov	Dec	Annual
1928	52.2	59.2	64.6	68.5	74.6	82.2	84.7	84.8	78.6	72.7	57.0	53.6	69.6
1929	53.1	51.9	61.2	69.0	78.0	82.6	85.1	84.0	81.2	71.9	58.4	55.8	69.3
1940	48.6	52.8	61.8	66.8	74.5	77.7	82.1	83.9	76.7	70.0	58.7	53.6	66.4
1941	53.4	51.0	55.0	66.8	75.0	79.6	83.6	85.1	80.8	74.2	58.2	53.8	68.4
#1942	46.6	51.8	54.4	67.4	74.4	82.6	82.0	82.7	74.6	68.3	62.8	53.5	67.4
1943	49.4	59.6	59.8	70.6	76.0	81.1	83.8	85.6	78.4	68.3	56.7	49.2	67.5
1944	48.4	53.8	58.6	67.6	71.8	81.0	84.4	84.4	77.6	69.3	59.8	48.0	67.2
1945	50.8	54.8	65.8	66.0	74.2	81.4	83.0	83.6	78.8	67.0	63.2	49.8	68.2
1946	48.6	55.0	67.3	69.6	74.6	79.1	83.8	85.8	76.6	70.8	59.1	54.9	68.2
1947	48.0	47.0	56.8	66.1	74.0	82.7	84.0	83.3	79.7	76.2	58.4	52.8	67.1
1948	48.7	50.7	58.0	71.6	76.2	83.7	84.6	85.8	78.4	69.9	58.2	55.4	68.0
1949	45.1	55.3	60.3	65.5	77.6	81.0	85.9	82.7	79.4	69.6	62.6	54.2	68.0
1950	55.7	56.4	59.1	66.0	76.2	80.3	83.4	83.4	78.3	73.6	59.9	52.7	68.8
1951	50.8	54.4	61.8	67.4	74.9	81.8	86.8	87.6	79.2	71.4	56.9	54.0	68.9
1952	54.1	55.1	58.8	64.3	72.8	81.4	85.7	87.1	77.9	69.3	57.3	50.4	68.0
1953	56.7	53.5	66.1	66.4	74.8	84.8	85.1	84.3	78.5	70.9	57.2	48.6	68.9
1954	52.0	59.5	59.4	72.4	75.4	83.5	86.9	86.6	82.4	72.8	59.9	56.3	70.4
1955	51.6	53.8	62.4	72.9	78.0	79.6	84.9	84.5	82.2	71.4	59.2	52.3	69.4
1956	50.8	53.8	61.8	69.1	78.7	84.4	87.0	85.9	81.7	73.7	57.5	56.6	70.2
1957	50.5	60.9	59.5	66.0	73.5	80.9	86.3	86.0	77.0	65.5	59.9	56.3	68.2
1958	48.7	49.8	54.7	66.8	76.0	83.2	85.1	85.8	79.5	67.5	60.7	48.5	67.0
1959	47.7	57.9	55.9	65.1	77.1	82.6	84.2	84.5	80.1	69.5	62.2	53.7	67.4
1960	49.3	47.9	54.3	69.7	75.5	83.4	84.7	83.3	79.5	71.5	60.7	48.5	67.2
#1961	46.0	54.8	63.7	67.5	76.3	79.6	81.6	81.8	78.2	69.8	56.5	52.2	67.3
1962	48.5	50.4	57.9	67.1	76.6	80.3	85.4	86.8	79.7	73.9	59.6	51.0	68.7
1963	43.6	51.4	64.5	72.9	76.9	82.9	86.4	87.5	82.0	75.1	62.6	45.1	69.3
1964	50.3	48.4	60.7	5.6	76.6	81.3	85.2	86.0	79.9	67.6	62.5	52.2	68.5
1965	53.4	49.6	53.6	7.-	7.7	80.7	84.2	84.3	80.5	66.6	64.1	56.0	69.2
1966	45.5	48.9	60.1	69.1	5.9	79.9	84.7	82.2	78.2	68.6	64.2	51.0	67.2
1967	50.9	51.4	67.3	75.9	75.5	83.7	85.2	82.6	79.2	67.7	58.9	49.4	68.6
1968	43.6	47.2	57.3	68.1	74.2	79.5	82.2	84.1	76.0	72.0	55.8	50.4	66.2
1969	53.6	52.6	53.9	69.0	73.7	80.7	86.5	84.9	79.7	68.6	57.9	53.8	67.9
1970	44.0	53.1	55.8	64.7	71.9	79.9	83.8	85.6	79.2	66.7	57.4	56.7	67.1
1971	54.6	54.8	62.1	68.2	76.2	83.7	85.9	81.2	79.3	72.2	60.5	56.4	69.6
1972	51.0	55.7	64.1	73.5	73.5	81.7	82.5	83.1	82.4	70.9	52.8	48.3	68.5
1973	48.9	50.3	63.7	69.1	73.7	78.4	82.9	82.5	79.7	71.1	65.1	52.4	67.4
1974	49.0	56.3	64.5	68.9	77.7	80.1	84.4	81.3	72.5	64.5	60.9	50.7	67.8
1975	53.4	52.2	59.7	67.8	73.7	79.9	82.0	82.6	75.8	70.0	60.4	52.4	67.5
1976	49.5	61.3	67.0	69.0	76.7	79.8	80.2	83.4	74.1	61.3	51.6	48.5	66.2
1977	41.6	54.1	61.3	67.0	75.3	82.2	85.1	86.8	83.6	71.6	61.7	53.6	69.7
RECORD MEAN	50.3	53.6	60.6	67.9	74.7	81.6	84.1	84.2	79.0	69.8	59.1	51.7	68.1
MAX	60.6	64.4	71.8	78.4	84.7	91.7	94.6	93.0	89.4	80.8	69.8	61.9	78.6
MIN	39.9	42.8	49.3	57.4	64.7	71.4	73.5	73.4	68.6	58.7	48.4	41.4	57.8

Heating Degree Days

AUSTIN, TX

Season	July	Aug	Sept	Oct	Nov	Dec	Jan	Feb	Mar	Apr	May	June	Total
1957-58	0	0	0	98	298	275	499	457	319	53	0	0	1499
1958-59	0	0	0	74	184	505	535	346	183	112	0	0	1934
1959-60	0	0	0	38	393	548	492	495	350	17	12	0	2145
#1960-61	0	0	0	28	182	512	582	299	116	80	0	0	1797
1961-62	0	0	0	21	270	404	598	161	243	48	0	0	1745
1962-63	0	0	0	20	172	429	657	374	109	27	4	0	1792
1963-64	0	0	0	6	137	610	651	475	154	20	0	0	1847
1964-65	0	0	0	17	158	413	370	431	367	12	0	0	1770
1965-66	0	0	2	58	76	288	611	449	198	24	11	0	1687
1966-67	0	0	0	31	109	453	446	380	77	6	2	0	1498
1967-68	0	0	7	41	192	477	538	510	265	37	1	0	2068
1968-69	0	0	0	7	294	446	368	365	346	6	1	0	1813
1969-70	0	0	0	69	253	339	693	330	309	47	10	0	2010
1970-71	0	0	1	75	240	231	328	300	171	32	0	0	1400
1971-72	0	0	0	0	84	271	434	295	86	4	0	1	1175
1972-73	0	0	0	46	375	519	586	403	72	145	7	0	2153
1973-74	0	0	0	8	83	387	496	258	104	36	0	3	1376
1974-75	0	0	3	11	271	441	377	356	204	53	0	0	1718
1975-76	0	0	1	27	204	404	476	158	173	19	3	0	1465
1976-77	0	0	0	170	402	506	719	308	148	20	0	0	2273
1977-78	0	0	0	16	136	354							

Cooling Degree Days

Year	Jan	Feb	Mar	Apr	May	June	July	Aug	Sept	Oct	Nov	Dec	Total
1969	25	5	8	132	276	476	668	620	450	186	46	1	2893
1970	8	1	10	164	235	491	589	651	435	134	22	43	2744
1971	14	25	86	157	355	570	656	511	496	229	53	13	3100
1972	6	34	111	269	272	510	547	571	529	239	16	3	3107
1973	1	0	34	78	277	407	551	550	447	205	93	4	2653
1974	6	21	140	158	298	483	606	519	229	157	34	6	2751
1975	21	0	47	141	277	454	536	554	329	187	72	22	2638
1976	0	37	85	119	186	451	479	577	401	61	6	0	2418
1977	0	7	58	90	324	526	629	681	572	227	40	7	3161

Precipitation

Year	Jan	Feb	Mar	Apr	May	June	July	Aug	Sept	Oct	Nov	Dec	Annual
1938	3.95	2.72	1.53	4.90	3.28	3.17	1.34	0.63	2.85	0.24	0.34	1.75	27.03
1939	2.23	1.41	1.09	1.87	3.07	1.00	3.57	1.82	1.64	1.82	2.26	0.80	22.13
1940	0.83	3.73	1.34	3.34	2.17	8.83	0.57	1.77	3.36	4.82	5.01	5.32	42.45
#1941	1.52	3.06	4.64	5.70	3.87	...60	3.37	0.07	0.72	7.12	1.24	2.25	46.16
#1942	0.07	1.46	0.66	5.56	2.05	2.71	3.69	2.16	8.13	9.14	1.98	1.53	34.64
1943	0.80	0.45	2.54	2.66	5.38	1.27	3.91	0.97	3.31	0.38	1.78	1.42	24.74
1944	5.40	3.89	1.83	0.33	9.25	2.01	0.32	4.47	4.84	0.35	4.35	5.91	42.97
1945	2.83	8.94	4.98	4.11	1.76	5.69	1.63	3.74	2.76	3.00	1.47	2.94	40.87
1946	3.76	2.28	2.77	7.92	6.13	1.39	1.48	3.36	6.00	1.62	7.91	2.71	47.28
1947	3.63	0.43	3.28	2.29	3.95	0.13	2.18	2.12	0.01	0.02	4.09	1.89	21.88
1948	0.92	2.71	1.35	1.68	4.48	1.25	2.29	0.27	1.24	1.78	1.34	1.67	20.98
1949	3.93	2.39	2.24	6.91	0.83	3.52	1.95	2.37	3.77	4.38	0.03	4.04	36.34
1950	0.74	3.79	0.80	7.58	4.19	1.98	0.73	0.59	4.77	0.39	0.08	T	25.79
1951	0.51	2.96	3.73	1.04	3.31	6.19	0.19	2.01	6.45	0.93	1.06	0.34	28.98
1952	0.28	1.73	2.25	3.08	4.06	1.88	0.65	0.00	3.26	T	5.36	3.15	27.71
1953	0.63	1.32	1.78	4.69	1.88	1.59	0.51	2.10	2.76	6.38	0.38	5.29	29.68
1954	1.01	0.28	0.27	1.66	2.80	0.68	0.85	1.14	0.82	0.89	0.35	0.61	11.42
1955	1.83	4.24	0.88	0.78	4.69	2.61	2.02	1.92	1.33	0.09	1.40	1.03	22.34
1956	1.65	1.74	0.26	0.56	3.12	0.99	0.11	1.21	0.09	0.84	2.14	2.76	15.41
1957	0.55	3.14	4.58	9.93	7.38	5.25	1.10	T	6.43	8.78	2.95	1.20	51.30
1958	3.09	6.33	2.59	4.24	3.67	2.89	3.42	0.68	6.89	3.18	0.83	1.15	41.02
1959	0.42	2.33	0.23	4.25	1.66	3.30	3.49	4.80	4.37	5.96	1.95	2.11	34.96
1960	1.03	2.36	1.37	1.01	0.81	4.28	2.41	2.60	1.48	12.31	1.90	4.08	35.82
1961	1.27	4.85	0.69	0.10	1.03	11.43	3.40	0.40	3.68	0.91	2.82	0.93	36.47
1962	0.56	0.63	1.19	4.04	1.06	6.24	0.00	4.58	4.75	4.09	0.92	3.47	33.48
1963	0.38	2.83	0.27	3.51	1.32	2.10	0.58	0.88	1.50	0.78	1.57	1.42	17.30
1964	2.83	1.47	1.95	1.47	1.87	7.59	0.65	2.09	6.29	3.74	2.43	0.86	32.97
1965	4.04	9.06	1.30	1.91	9.98	0.89	0.37	1.32	5.48	3.26	2.05	4.28	60.57
1966	1.58	3.23	0.50	3.74	3.15	1.88	0.47	6.22	3.20	0.60	0.11	0.87	25.19
1967	0.29	1.52	1.09	4.44	0.35	T	1.15	3.71	5.71	4.35	4.36	3.43	33.54
1968	7.96	1.64	2.09	1.87	8.75	3.16	3.11	0.74	3.42	0.60	4.91	0.55	38.72
1969	0.40	4.18	3.26	5.04	3.25	2.64	0.12	5.78	1.17	2.85	0.79	4.28	33.89
1970	1.83	5.70	2.47	1.36	6.18	0.29	0.66	1.00	3.82	5.22	T	0.15	30.64
1971	0.04	0.69	0.79	1.07	1.37	1.68	1.23	5.69	2.13	2.02	3.02	4.22	24.98
1972	1.48	0.31	T	1.46	7.88	2.20	2.55	2.53	1.55	2.96	2.82	0.53	26.07
1973	5.42	2.05	2.87	3.09	1.38	4.70	2.95	0.06	7.44	11.11	0.38	0.76	40.46
1974	2.74	0.36	1.54	1.79	3.88	0.21	0.61	6.90	1.24	3.45	7.35	2.00	36.21
1975	1.11	2.30	0.80	3.86	8.14	7.07	2.25	2.54	3.62	2.54	0.32	2.09	36.81
1976	1.16	1.11	2.11	8.13	6.05	3.19	4.71	0.80	2.80	5.92	1.76	2.46	41.23
1977	2.25	2.56	2.18	6.08	1.24	1.28	0.21	0.06	2.10	1.19	1.06	0.34	22.14
RECORD MEAN	2.08	2.48	2.20	3.58	4.24	2.82	2.19	2.22	3.63	3.13	2.37	2.48	33.32

Snowfall

Season	July	Aug	Sept	Oct	Nov	Dec	Jan	Feb	Mar	Apr	May	June	Total
1970-71	0.0	0.0	0.0	0.0	0.0	0.0	0.0	T	0.0	0.0	0.0	0.0	T
1971-72	0.0	0.0	0.0	0.0	0.0	T	T	0.0	0.0	0.0	0.0	0.0	T
1972-73	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.8	0.9	0.0	0.0	0.0	1.7
1973-74	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
1974-75	0.0	0.0	0.0	0.0	0.0	0.0	T	T	0.0	0.0	0.0	0.0	T
1975-76	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
1976-77	0.0	0.0	0.0	0.0	T	0.0	0.0	T	0.0	0.0	0.0	0.0	T
1977-78	0.0	0.0	0.0	0.0	0.0	0.0							
RECORD MEAN	0.0	0.0	0.0	0.0	T	T	0.5	0.4	0.1	0.0	0.0	0.0	1.0

# Indicates a station move or relocation of instruments. See Station Location table.

Record mean values above are means through the current year for the period beginning in 1898 for temperature, 1856 for precipitation, and 1942 for snowfall.

Figure 7b

STATION LOCATION

AUSTIN, TEXAS

Location	Occupied from	Occupied to	Airline distance and direction from previous location	Latitude North	Longitude West	Elevation above Ground at temperature site	Elevation above Ground: Wind instruments	Elevation above Ground: Extreme thermometers	Elevation above Ground: Psychrometer	Elevation above Ground: Thermograph	Elevation above Ground: Tipping bucket rain gage	Elevation above Ground: Weighing rain gage	Elevation above Ground: 8" rain gage	Elevation above Ground: Hygrothermometer	Elevation above Sea level: Pyranometer	Remarks
CITY																
Engineering Building University of Texas	1906	4/01/23		30° 18'	97° 44'	590		40					80			Cooperative station.
Engineering Building University of Texas	4/01/23	4/01/24	1/2 mi. W	30° 17'	97° 44'	560		6					3			Cooperative station.
Engineering Building University of Texas	4/01/24	10/15/26	1/2 mi. NW	30° 17'	97° 44'	570		6					3			Cooperative station.
Room 901, Littlefield Bldg., Sixth & Congress	10/15/26	10/08/36	1-1/2 mi. S	30° 17'	97° 44'	495	148	135	135		128		128			Weather Bureau Office.
Room 408, Federal Court Building, 200 W. 8th St.	10/08/36	2/12/42	1/4 mi. NW	30° 17'	97° 44'	540	79	66	66		60		60			Weather Bureau Office.
Rooms 801-803 Littlefield Building Sixth & Congress	2/12/42	8/16/42	1/4 mi. SE	30° 17'	97° 44'	495	148	135	135		128		128			Weather Bureau Office.
AIRPORT																
Administration Building Municipal Airport	8/16/42	6/30/61	3 mi. NNE	30° 18'	97° 42'	615	32	6	5		3		3			Wind instruments 41 feet until 9/13/57.
Administration Building 3600 Manor Road Municipal Airport	7/01/61	Present	0.7 mi. SE	30° 18'	97° 42'	597	20	b5	b5		3		3	a6		a - Commissioned 7/1/61. b - Discontinued 7/1/61.

Requests for additional climatic information should be addressed to: Director, National Climatic Center, Federal Building, Asheville, N. C. 28801

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Director, National Climatic Center

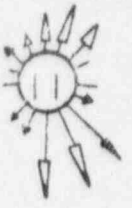
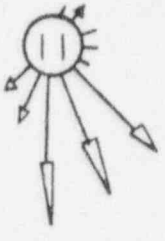
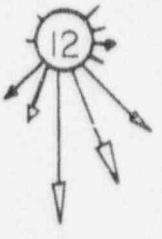
USCOMM-NOAA-ASHEVILLE - 900

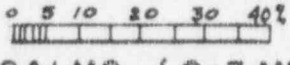
Figure 7c

SURFACE WIND ROSES

N

A FIVE YEAR SUMMARY OF WIND VELOCITY AND DIRECTION BY MONTHS FOR AUSTIN, TEXAS

JANUARY	FEBRUARY	MARCH	APRIL
			
MAY	**JUNE**	**JULY**	**AUGUST**
			
SEPTEMBER	**OCTOBER**	**NOVEMBER**	**DECEMBER**
			

FREQUENCY SCALE:  FROM CIRCUMFERENCE

(13) % OF CALMS (0-3 MPH)

4-12 MPH

OVER 12 MPH

NOTE: A line or arrow pointing south indicates a south wind, i.e., a wind blowing from the south to the north, etc.

FIG. 8

The rocks that crop out in Travis County are primarily of sedimentary origin and of Mesozoic (Cretaceous) and Cenozoic age. They consist largely of limestone, clay, and sand strata which dip southeastward toward the Gulf of Mexico at an angle slightly greater than the slope of the land surface. Therefore, in going from southeast to northwest the outcrops of progressively older formations are encountered, and the rocks lowest in the geologic column have the highest topographic exposure.

The Balcones fault zone, which extends from Williamson County to Uvalde County, extends the full length of Travis County on a line passing through Manchaca, Austin, and McNeil. Here the orderly sequence of formations is replaced by an out-crop pattern controlled by the faults, most of which are normal faults with the down-thrown side toward the coast.

The Hosston, Sligo, and Travis Peak formations are believed to be the oldest and lowest formations beneath Travis County that offer possibilities for ground-water development. These formations contain a number of sands, sandstones, and limestones known locally as the "Trinity sands," which yield water that has a considerable range in quality. Yields of wells tapping the "Trinity sands" have a large range. West of the fault zone the yields generally are small to moderate. Yields are greater in the fault zone and to the east, the largest reported yield being 275 gpm. East of the fault zone the water may flow naturally from wells at low elevations, the largest reported flow being 170 gpm. The water from the "Trinity sands" generally is of suitable quality for domestic and stock use, but in places it is highly mineralized.

The Glen Rose limestone generally yields only small quantities of water. In most places the water becomes more highly mineralized with depth, calcium and magnesium sulfate accounting for most of the increase.

The reactor site rests primarily upon limestone several hundred feet thick, as evidenced by test drillings and well logs in the surrounding area. These formations are essentially impervious, as shown by the high pressure of the water found below 300 to 1000 feet, some wells flowing to the surface without pumps. They slope generally to the south, and little or no useful ground water is found above a depth of 300 feet. There is no well within a mile of the reactor site, and only a few exist within Austin. A complete record (location, depth, use, ownership, etc.) of all wells in Travis County has been obtained from the Texas Board of Water Engineers so that in case of some unforeseen emergency, notification and water testing could be initiated immediately.

The Austin public water system, to which the University is connected, is supplied entirely with filtered river water and is distributed under pressure through a typical underground piping system. The water is continuously tested and controlled under the jurisdiction of the City and State Health Departments. Occasional checks are made of its radioactivity, which runs about 6.3×10^{-9} μc/ml (gross β).

The Nuclear Reactor Laboratory has no sewer connections, either sanitary or storm. Surface water run-off outside the Laboratory is directed into a University-owned storm sewer system. This portion of the system empties into Waller Creek about two blocks east of the Laboratory.

On a few occasions in the past small amounts of storm water have entered the Laboratory room. For this and other reasons, the top of the reactor pit was raised about ten inches above the room floor to preclude any possibility of contaminating the reactor pool water with storm water, and vice versa.

2.5. Seismology

According to "Earthquake History of the United States, Part I" (Coast and Geodetic Survey), there have been nine earthquakes in Texas "from the earliest times" to the close of 1956:

Date	Location	Intensity	Felt Distance, Miles	Distance from Austin, Miles
1891	Rusk	7	200 (est.)	200
1917	Panhandle	6	Local	430
1925	N.W. Texas	5	250	430
1931	W. Texas	8	380	415
1932	Mexia	5-6	30	130
1934	N. Texas	5	55	290
1936	N.W. Texas	5	140	450
1948	N.W. Texas	6	125	510
1951	N.W. Texas	6	125 (est.)	510

The intensity is on the Modified Mercalli scale of 1931, on which intensity 7 indicates negligible damage in buildings of good design at the epicenter of the shock. The felt distance has been computed as the radius of a circle containing the recorded felt area, which is approximately that area included within isoseismal 3. Intensity 3 is "vibration like passing of a truck."

Of the listed earthquakes, only the 1891 one might have been felt at Austin. Its felt area was not recorded, and the estimated felt distance in this case was determined from typical areas of other intensity 7 earthquakes. It is highly probable that no detectable damage of any kind occurred in Austin due to this shock.

Study of a map of earthquakes in the above publication shows that Central Texas is one of the most earthquake-free areas in the entire country.

3. DESCRIPTION OF BUILDING AND REACTOR

3.1. Reactor Laboratory

The TRIGA-I reactor is located in Room 131 of Taylor Hall, the mechanical engineering building (see Figures 5 and 6). This room is of brick and reinforced-concrete construction, including the floor. The roof of the room is of steel girder and gypsum construction. Partitions are of metal lath and cement plaster fireproof construction. Entrance to the reactor laboratory is normally through a pair of sequential doors leading to the corridor. Access to the laboratory is restricted as described in section 4.4.1.

An area between the two access doors serves as a health-physics control point where personnel monitoring equipment is issued and information is obtained for the preparation of records in accordance with 10 CFR 20 and any other pertinent codes in effect. The outside double doors on the south wall of the laboratory will serve as emergency exit doors. Only with the reactor shut down and under the direct supervision of authorized Laboratory personnel, are these doors used for any other purpose.

The Laboratory is air-conditioned and sealed off from the adjoining parts of Taylor Hall by tight fireproof walls and weatherstripped doors. Heat, when needed, is supplied by steam-fed unit heaters in the Laboratory.

Figure 6 shows the arrangement of the major pieces of equipment in the Laboratory, and Figure 9 shows the arrangement of the pit relative to the existing floor, ground, and outside wall of the building. The floor in most of the room is of 3" and 3.5" suspended concrete slab construction with strengthening concrete beams underneath. A small portion of the floor along the east side of the room is a 6" concrete slab on grade. Around the periphery of this portion of Taylor Hall, including the south and west sides of the

Laboratory, is a service trench for various utility lines. This trench provides more space under the floor than would otherwise be available, as the floor slab is only about 3 or 4 feet above grade here and even less at other locations.

The ceiling height in the main part of the Laboratory is about 16 feet to the bottom of the steel girders and about 17 feet between the girders. Over the east side of the room the ceiling is of saw-tooth shape and varies from about 12 to 20 feet in height. The ceiling (roof) is supported by steel girders and consists of 4.5-inch gypsum board and fill, one inch of rigid insulation, waterproofing, and built-up roofing. The floor area of the room is approximately 1700 ft^2 and the room volume is about 24,000 ft^3.

The exterior (south and west) walls are solid brick 17" thick. The north wall and the present east wall are solid brick 13" thick. The walls of the storage room and the adjacent office are of structural clay tile 6" thick. The interior surfaces of all of these walls are completely covered with approximately 3/4" of gypsum plaster.

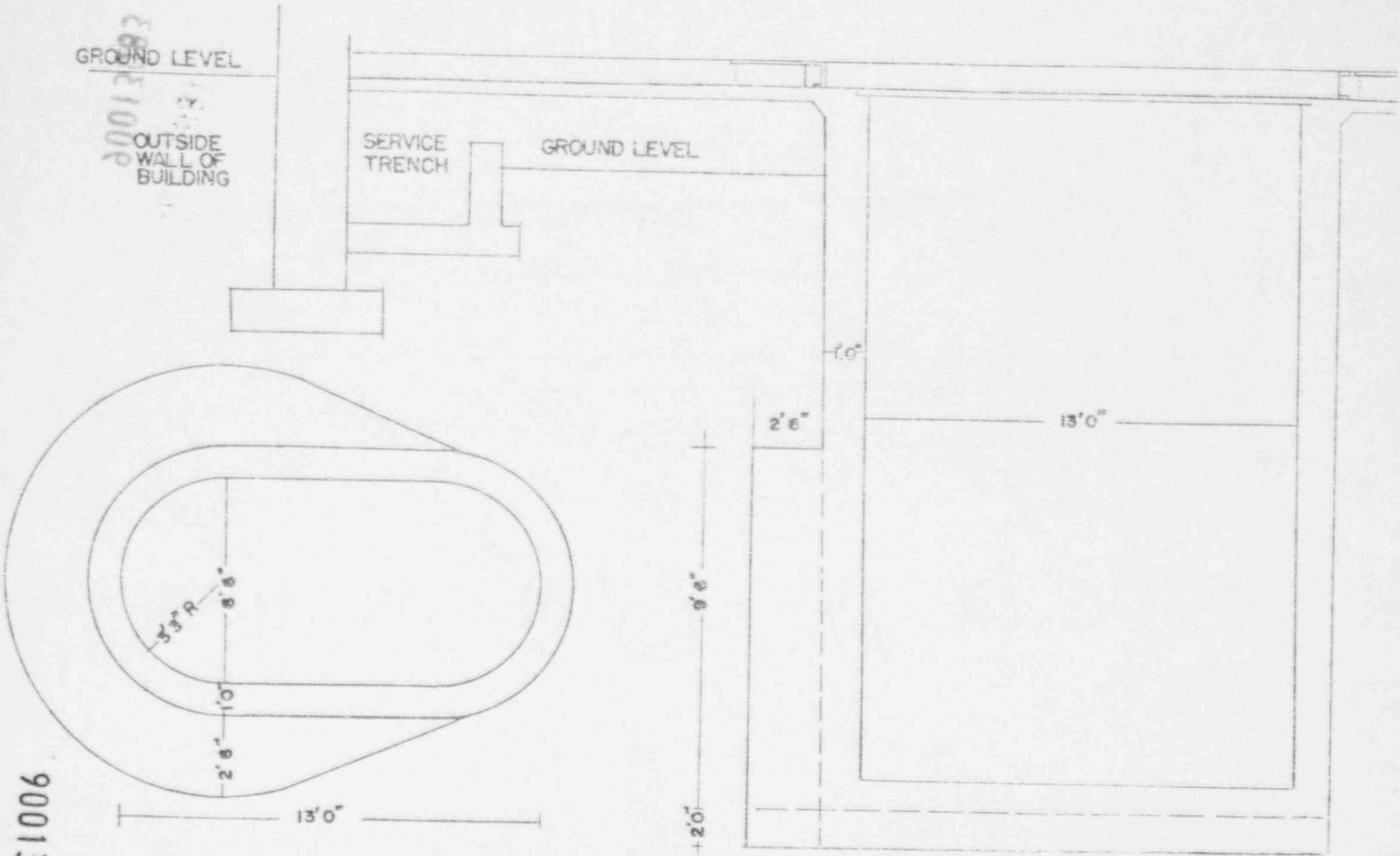
The south and west walls have projected steel windows with double-strength clear glass. They are sealed with plastic caulking compound and welded shut to prevent any future opening. The doors to the room are sealed by means of commercial weatherstripping on the top and sides and on the bottom.

An analysis by General Atomic based on experimental data has shown that no special building requirements are necessary for housing a TRIGA reactor. A normal building, without any special provisions for air confinement,

is satisfactory. In case of a fuel-element-cladding failure, the concentration of the airborne radioactive fission products at a short distance away from the Laboratory would be very low due to fresh-air dilution. The analysis indicates that the maximum integrated dose which a person might receive in the area of maximum concentration outside the building would be approximately 40 mr over a 1-hr period, based on infinite reactor operation at 250-kw power.

The only possible condition which might create airborne radioactive particulate matter would be a fuel-element-cladding failure. Should an accident occur, it is possible that small amounts of radioactive noble gases would be dispersed from the reactor pool into the air of the reactor building, and these would decay into particulate matter.

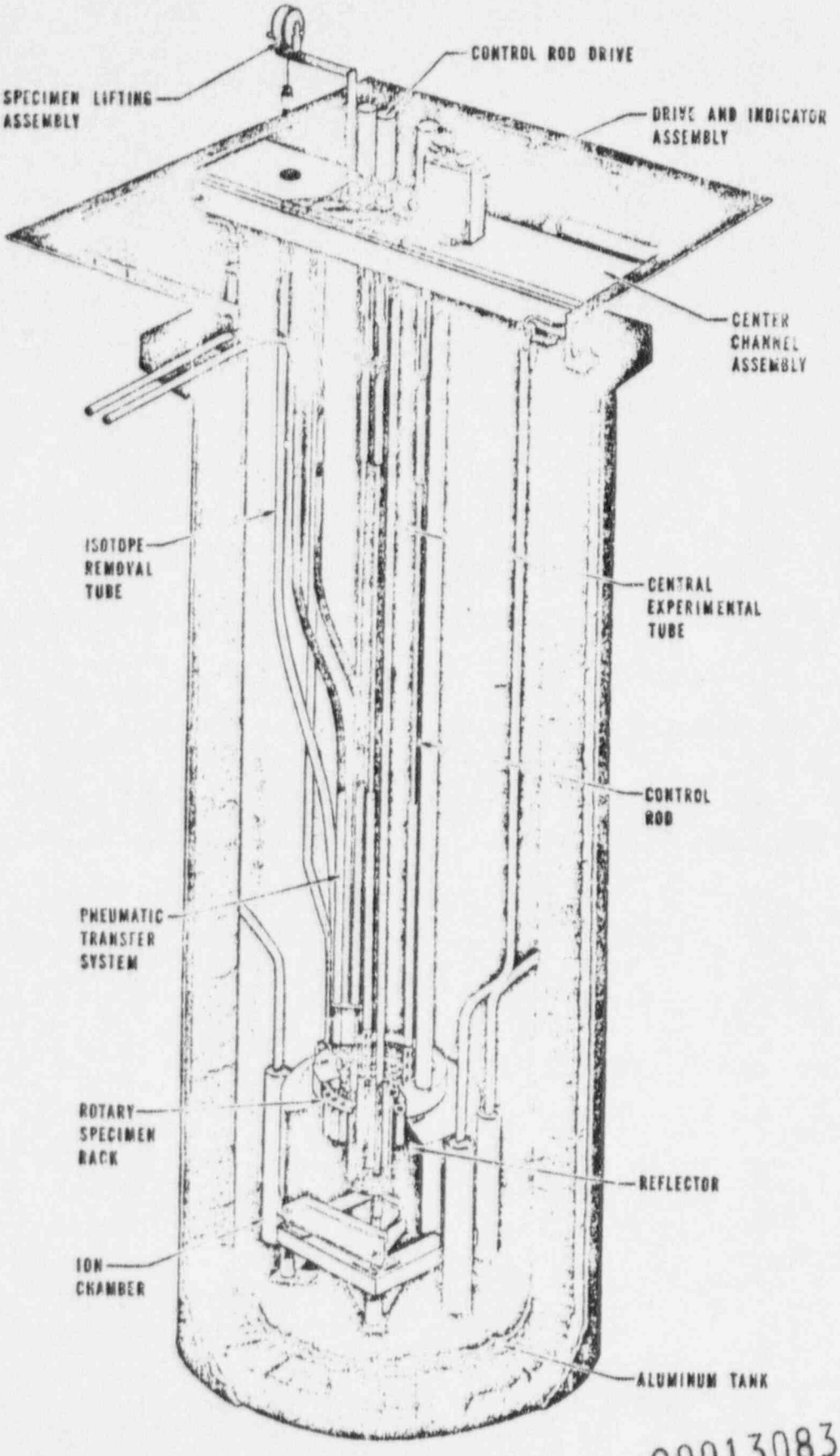
According to the handbook Heating, Ventilating, and Air-Conditioning, the number of air changes in a building without a ventilation system varies from 0.5 to 2 air changes per hour under average conditions. Based on this, it can be assumed that most of the fission-product gases leak out of the reactor building within 1 hr. The fission-product gases are mainly xenon and krypton, with an average total activity in the building of 2.45 curies. The average concentration of xenon and krypton in the reactor-building air would be 1.21×10^{-3} $\mu c/cm^3$. If it is assumed that the fission-product gases escape from the building at a lower rate than previously postulated, the integrated dose received by a person would be approximately the same as before, due to the longer decay period for the fission-product gases before leaving the building. If the Laboratory ventilation system is operating during the release of fission-product gases, the integrated dose received by a person near the building would be less than 16 mr for a 1-hr period, due to air dilution.



scale 1/4" = 1' 0"

Figure 9

NUCLEAR REACTOR PIT
RM. 131
TAYLOR HALL



REACTOR AND PIT.

Figure 10

The TRIGA fuel elements especially developed for pulsing have safely received more than 1000 pulses of between 1.5% and 2.25% δ k/k in the prototype TRIGA Mark F reactor at San Diego. Assuming that a fuel-element failure occurred once each year, the concentration of the fission-product gases leaving the building, averaged over a 1-yr. period, would be 1.38×10^{-7} μc/cm^3, and this concentration would be very much decreased by fresh-air dilution after traveling a short distance away from the building.

3.2. Reactor Pit

The reactor is located near the bottom of a pit below ground level (see Figures 9 and 10). The pit is lined with an aluminum tank 0.25 inches thick. Outside the aluminum is poured concrete shielding varying in thickness from 48 inches in the vicinity of the core to 14 inches around the upper portion of the tank. The assembly rests upon a 24-inch poured concrete slab as shown.

The reactor is located near one end of the elongated pit shown in Figure 9; the remainder of the pit is available for experimental purposes. Figure 10 represents the standard TRIGA-I arrangement in a circular pit and closely approximates its appearance at one end of the proposed elongated tank.

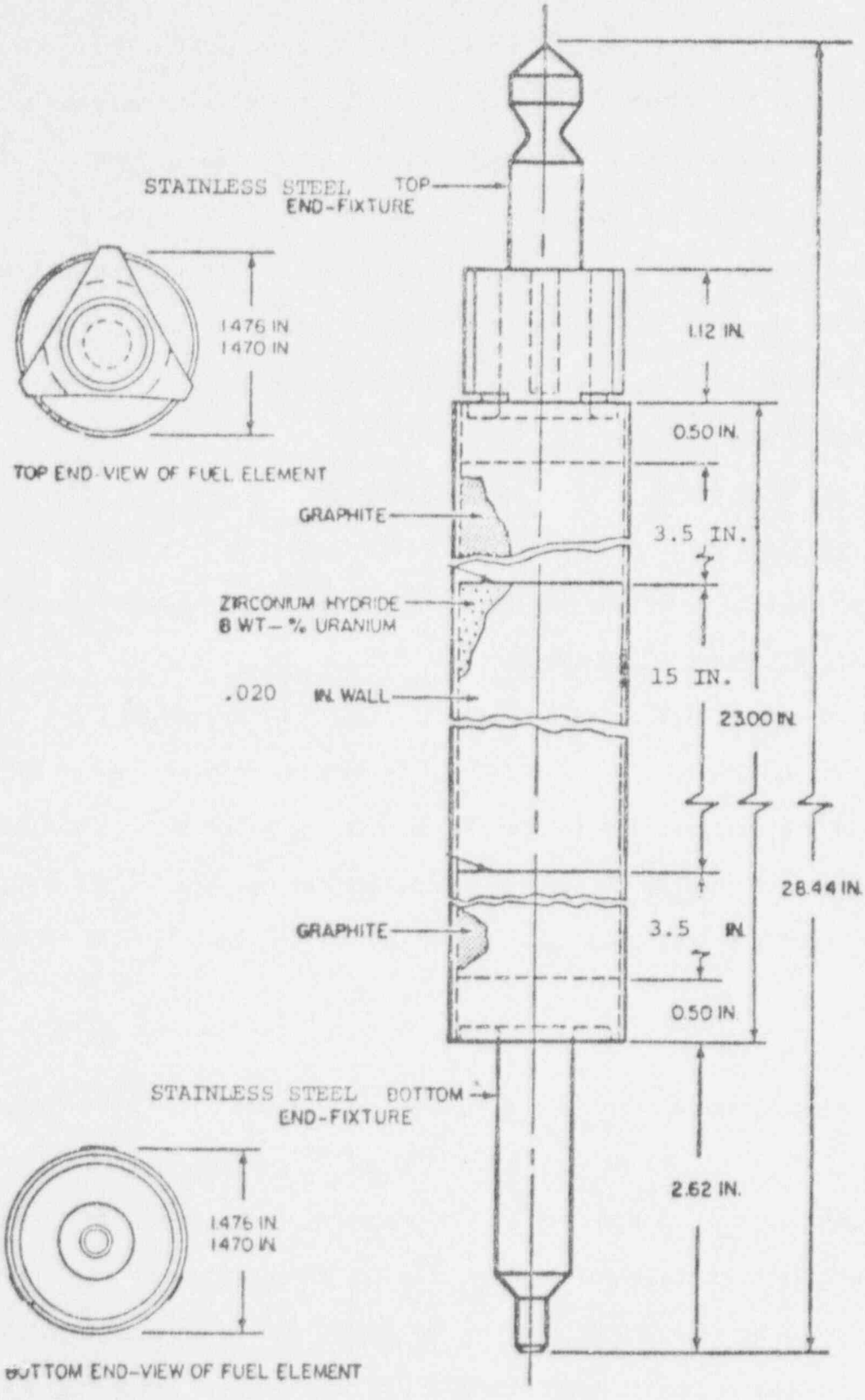
The aluminum tank was completely shop-fabricated except for one Heli-arc field weld of an aluminum pressure relief line. All shop welds were 100% X-rayed and tested for leaktightness by dye penetrant tests and helium-sniffing. The one field weld was helium sniffed with six times the normal helium pressure to compensate for the impossibility of X-raying at this point. The tank was coated on the exterior surface for corrosion protection and grouted in place on the previously poured bottom slab. The radial concrete shielding was then poured around it. Shielding above the core is provided by approximately 16 feet of water.

Three storage pits are located immediately adjacent to the reactor tank. These pits are made of vertical steel pipes 10 inches in diameter and 10 feet long and lined with an organic coating. They may be filled with water and used for temporary storage of irradiated specimens or used fuel elements prior to their ultimate disposal.

3.3. Reactor Core

As shown in Figure 10, the core forms a right circular cylinder and consists of a lattice of cylindrical fuel-moderator elements, graphite dummy elements, and control rods all immersed in water. The active portion of each fuel element (Figure 11) is 1.42 inches in diameter by 15 inches long and is a solid, homogeneous mixture of hydrided uranium-zirconium alloy containing 8 wt-% uranium enriched to 20% in U^{235}. The H-Zr atomic ratio is approximately 1.0. Each element is jacketed with a .020 inch thick stainless steel can. Three inch sections of graphite are inserted in the can above and below the fuel to serve as top and bottom reflectors for the core. Stainless steel end fixtures are attached to both ends of the can to support and position the elements in the core. The overall length of the element is 28.44 inches.

The elements are spaced so that about 33% of the core volume is occupied by water. This fuel-to-water ratio in the core was selected because calculations show that it gives very nearly the minimum critical mass. A total of 69 fuel elements in the core are required for 250 kw operation. Eighty-five fuel-element positions are available in the lattice; the unused positions are occupied by graphite dummy elements, i.e., elements in which the uranium-zirconium-hydride fuel is replaced by graphite.



FUEL MODERATOR ELEMENT

Figure 11

The elements are supported and spaced by top and bottom grid plates of 6061 aluminum. The bottom grid plate is 3/4 in. thick, with holes drilled in it to receive the lower end-fixtures of the elements. These lower end-fixtures have 1/4-in.-diameter projections on the bottom. A 5/8-in. shoulder is provided on the end-fixture, and the hole in the bottom grid plate is countersunk by a corresponding amount. The weight of the element rests on this shoulder, not on the bottom of the end-fixture, which is used only to position the element as it is being put into place.

The top grid plate is also 3/4 in. thick and has 1-1/2 in.-diam holes. The top grid plate does not support any of the weight of the elements. The holes serve only to determine the lateral position of the elements and to permit their withdrawal from the core.

The core is cooled by natural circulation of water, which flows through the core from bottom to top. In addition to flowing in over the top of the bottom grid plate, space for the passage of cooling water through the bottom grid plate is provided by 36 special holes, and through the top grid plate by the gap between the triangular section of the fuel element and the round grid hole.

3.4. Reactor Reflector

The core is surrounded by a cylindrical reflector 12 in. thick, 17 in. ID, 42 in. OD, and 22 in. high. This reflector is completely encased in a welded aluminum can, and flooding of the graphite, in the event that the can should leak, will decrease reactivity. The top and bottom reflectors are the 3-in. graphite sections encased in the fuel-element cans, so that the reflector in this region is approximately 67% graphite and 33% water, by volume. The reflector assembly is supported at the bottom by an aluminum structure, as indicated in Fig. 10.

3.5. Irradiation Facilities

Special irradiation facilities are provided for the production of radioisotopes. These include a rotary specimen rack located in the well in the reflector can, a pneumatic transfer tube, and a central thimble. In addition, odd-shaped specimens may be irradiated in the water outside the reflector (see Figure 10).

The rotary specimen rack consists basically of an aluminum ring which can be rotated around the core. Forty aluminum cups, evenly spaced, are hung from the ring and serve as irradiation specimen holders. The ring can be rotated manually from the top of the reactor pit, so that any one of these cups can be aligned with the single isotope-removal tube which runs up to the top of the reactor pit. This tube is used for removing and replacing irradiation specimens. An indexing and keying device is provided to ensure positive positioning of the cups.

The rotary specimen rack is completely enclosed in a welded aluminum box. The aluminum ring is located at approximately the level of the top grid plate, with the specimen cups extending from the ring down to about 4 in. below the top of the active lattice. In the radial direction, the centers of the cups are about 4 in. from the inner edge of the reflector assembly. The box enclosing the rotary specimen rack has been designed to ensure that it will remain watertight. Flooding of this box will decrease the reactivity of the reactor.

The pneumatic transfer tube is provided for the production of isotopes with short half lives. It consists, in essence, of two tubes leading down through the water tank to any position in the core, where the tubes are joined. The specimen is fed in and out through one of the tubes, and a blower connected

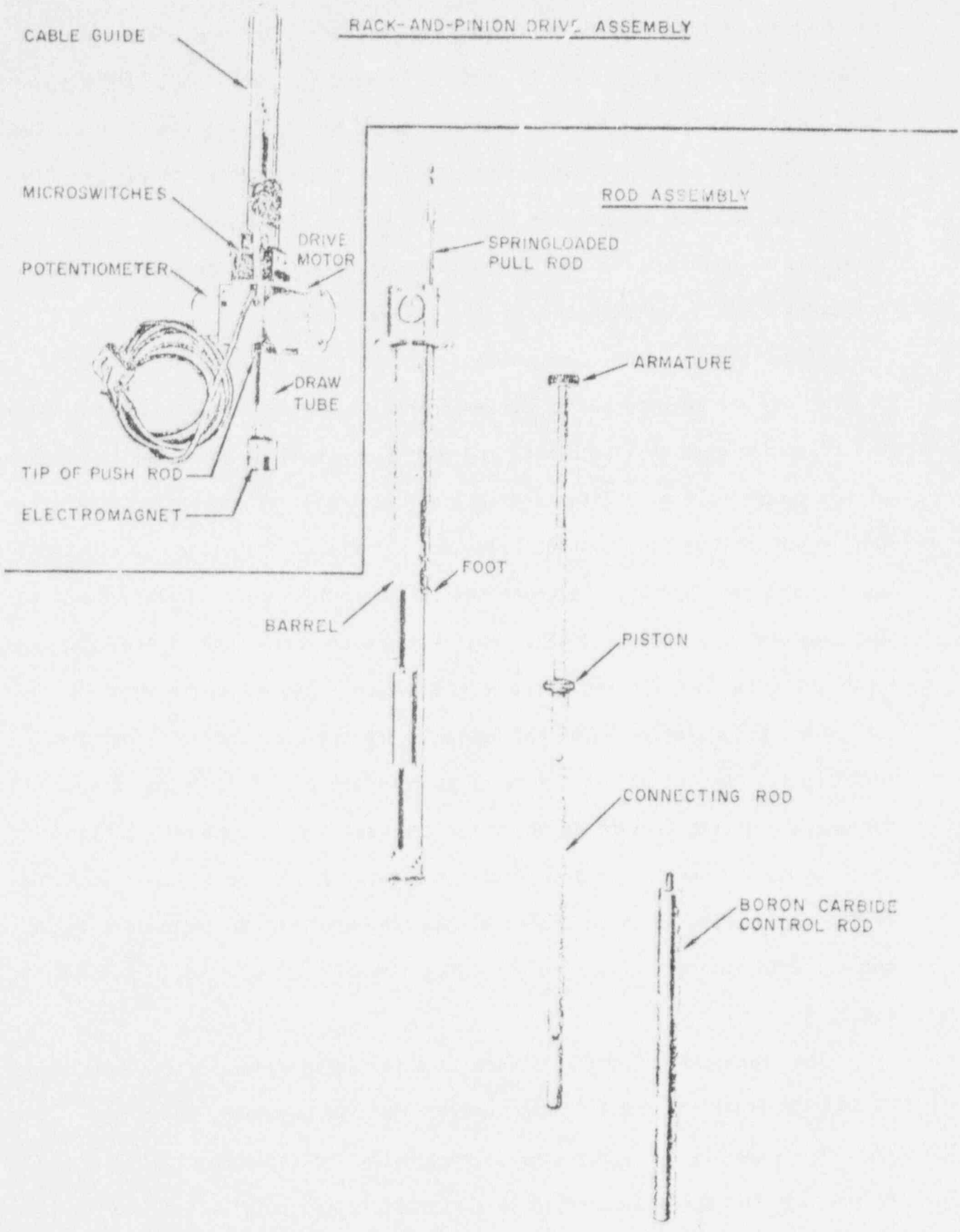
to the tubes provides the pressure difference required to inject or eject the specimen. Specimens are inserted into and removed from the pneumatic system on the north wall of the radio-chem room.

A single thimble is provided to permit irradiations or experiments in any region of the core. It consists of a vertical aluminum tube 1.33 in. ID leading from the top of the reactor pit to any position in the reactor core and terminating below the bottom of the core. The bottom of the tube is capped, but holes drilled in the wall of the tube near the upper grid plate ensure that the portion in the active lattice will always be filled with water during reactor operation. The portion of the tube above the core is normally filled with water but can be emptied to serve as a beam tube by applying compressed air or other gas to expel the water above the holes. A dry tube is also available for experimentation and can be placed in any region of the core.

3.6. Control Rods and Drives

Three control rods (safety-transient, shim, and regulating) operate in perforated aluminum guide tubes. The control rods for the shim and regulating rods are in sealed aluminum tubes containing powdered boron carbide as a neutron poison. The safety-transient rod is similar except that a solid bar of borated graphite is used in place of the boron carbide. The upper end of the control rod screws into the control rod drive assembly extension tube. Each control rod is approximately 20 inches long. The outside diameter of the regulating rod is 7/8 inch, of the safety-transient rod 1 inch, and the shim rod is 1-1/4 inches in diameter. The vertical travel of the control rods is approximately 15 inches.

The safety-transient rod, located in the D-ring, is completely out of the core during normal steady-state operation, and its stroke is adjusted such that it will not be worth more than 1.5% δ k/k during pulsing. The



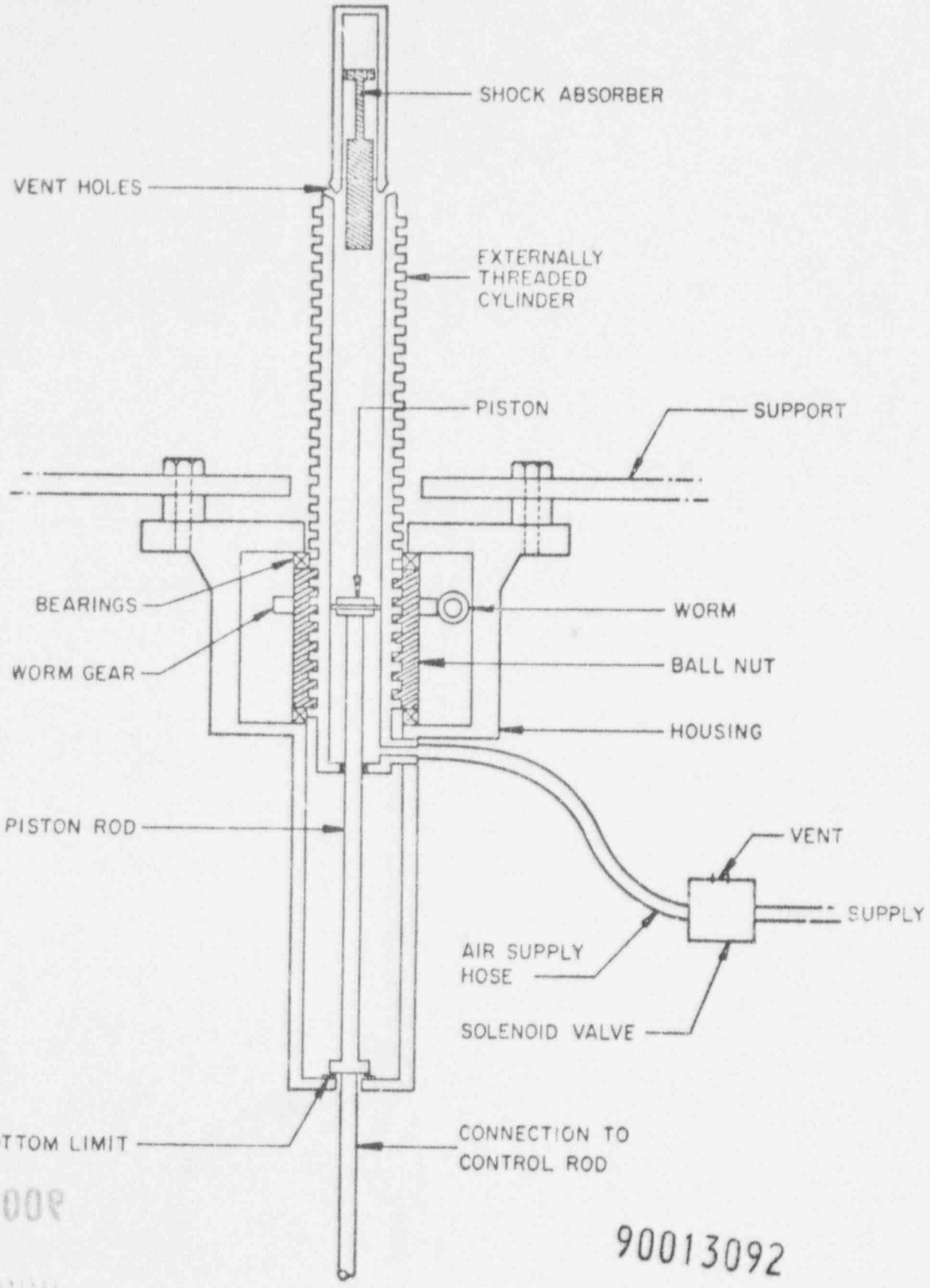
SHIM AND REGULATING CONTROL RODS AND DRIVES.

Figure 12

total worth of this rod, however, will not exceed 1.7% δ k/k. The shim rod is worth approximately 2.6% δ k/k and is located in the C-ring. The regulating rod, located in the E-ring, is worth approximately 0.96 % δ k/k. This makes the usable total rod worth approximately 5.06% δ k/k. Total excess reactivity in the core will be not greater than 2.25% δ k/k. The maximum reactivity insertion rate associated with the withdrawal of any rod, except the safety-transient rod, is approximately 0.2% δ k/k per second.

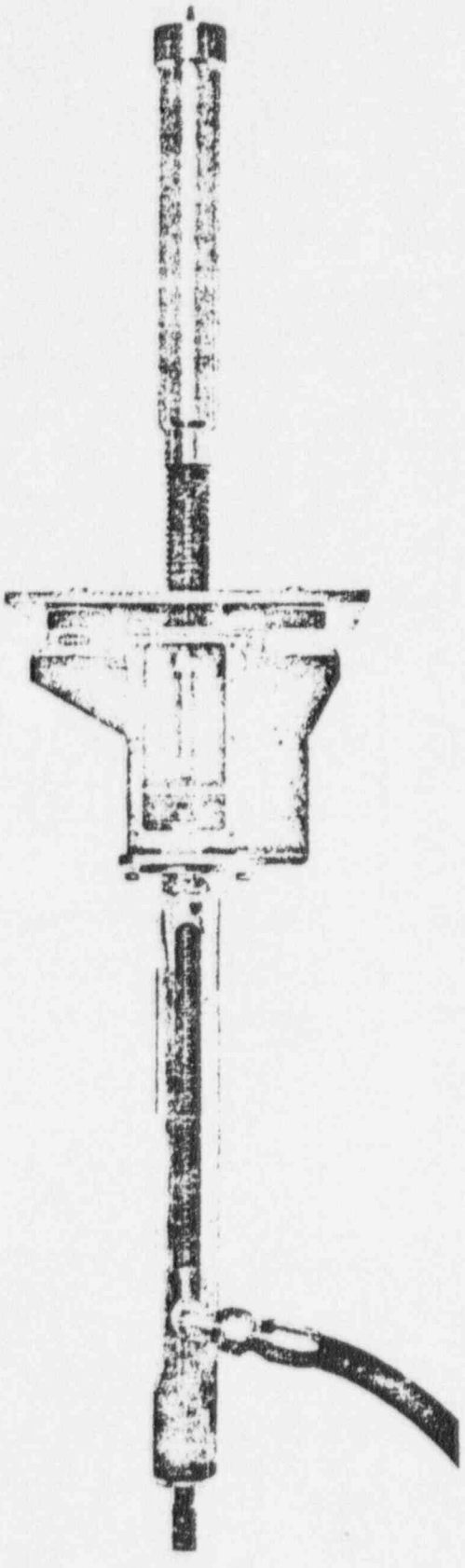
The control-rod-drive assemblies for the shim and regulating rods (Figure 12) are mounted on the center-channel assembly and consist of a motor and reduction gear driving a rack and pinion. A helipot connected to the pinion generates the position indication. Each control rod has an extension tube which extends to a dashpot below the surface of the water. The dashpot and control-rod assembly are connected to the rack through an electro-magnet and armature. In the event of a power failure or scram, the control-rod magnet is de-energized and the rod falls into the core. The rod-drive motor is nonsynchronous, single-phase, and instantly reversible, and will insert or withdraw the control rod at a rate of approximately 11.4 in./min. Electrical dynamic and static braking on the motor are used for fast stops. Limit switches mounted on the drive assembly indicate the "up" and "down" positions of the magnet, the "down" position of the rod, and whether the magnet is in contact with the rod. The complete drive assembly is enclosed in an aluminum can.

The transient-rod drive is used to expel instantaneously (in less than 0.1 second) the transient rod from the core by means of pneumatic pressure. The drive assembly is illustrated schematically and photographically in Figures 13 and 14. For pulsing operations, air pressure (75 psig) is applied to the



Schematic of Transient-Rod Drive Assembly

Figure 13



PHOTOGRAPH OF TRANSIENT ROD DRIVE

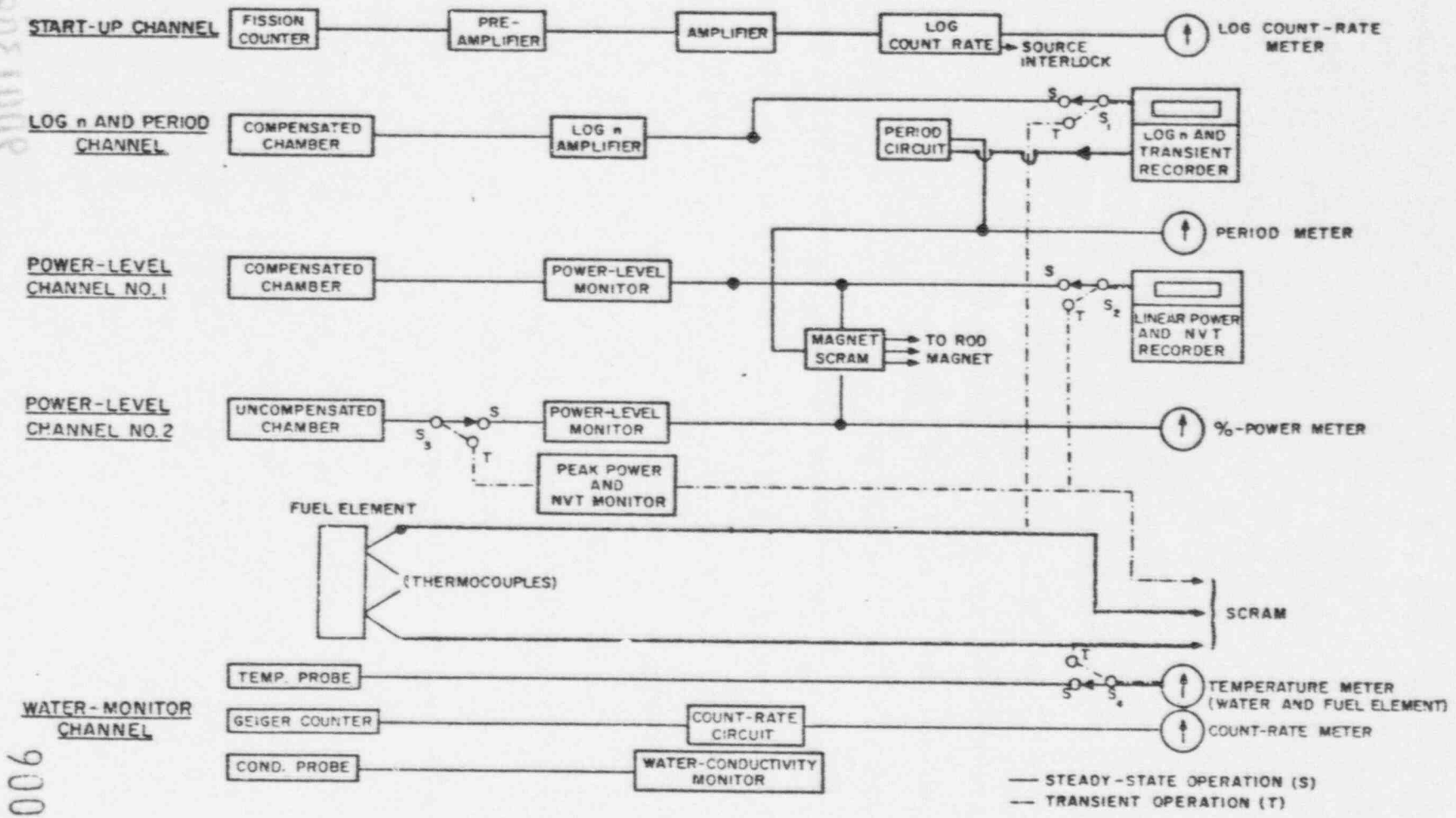
Figure 14

bottom face of the piston to drive it upward until stopped by a hydraulic shock absorber. When air pressure is released by a fast-acting solenoid control valve, the piston will return by gravity action to its bottom limit to fully insert the transient rod in the core. The lower limit of rod travel is fixed by the housing and remains unchanged. The upper limit of rod travel is determined by the upper end of the cylinder which may be moved by turning the ball nut within the rod drive housing. This ball nut engages the external threads on the cylinder. A revolution counter connected to the ball nut worm drive indicates the position of the cylinder. Full-up and full-down indications for cylinder position are also provided.

3.7. Instrumentation

The reactor instrumentation for steady-state operation includes four neutron channels; water-radioactivity, water-temperature, and water-conductivity monitors; an area-radiation monitor; and a control console. The neutron channels are fed by four neutron-sensitive ion chambers that are sealed in aluminum cans and mounted around the reflector in the ion-chamber tube-mounting assembly. The type of chamber used and its location determine the range of the neutron flux measured. The reactor instrumentation described in this section is shown in block-diagram form in Fig. 15.

1. The Start-up Channel consists of a fission counter, a pulse amplifier, and a log count-rate circuit. The channel gives useful power indication from source level (about 1 milliwatt) to greater than 1 watt. The log count rate is indicated on a meter at the upper left corner of the console panel (see Figure 16). The circuit also includes a minimum source-strength interlock signal which will not allow the rods to be raised unless the source strength is adequate.



Block diagram of instrumentation

Figure 15

2. The Log n and Period Channel consists of a compensated ion chamber and a log n and period circuit. The compensated ion chamber feeds a logging circuit which produces a voltage logarithmically dependent on the chamber current. This voltage is displayed by the log n recording unit of the dual-pen recorder at the center of the console panel and gives log power from less than 1/10 watt to full power. Reactor period information is also derived from the log n recording unit and is displayed from -40 sec to infinity to +7 sec on the period meter at the lower left side of the console panel. The operator may adjust this meter to cause a scram for any positive reactor period less than a selected value; this period scram is adjustable between 7 and ∞ sec.

3. Power-level Channel No. 1 consists of a compensated ion chamber and a recording micromicroammeter with a range switch to give accurate power information from source level to full power on the linear recording unit of the dual-pen recorder. This circuit also provides one of the power-level scrams.

4. Power-level Channel No. 2 consists of an uncompensated ion chamber, a power-level-monitor circuit, and a meter calibrated in percent of full power. The meter is located at the upper right side of the console panel. This channel furnishes the second power-level scram, which can be adjusted by the operator from 0% to 110% of full power.

5. The Dual-pen Recorder on the control console displays both the linear power level and the log n data. It is composed of two complete and separate recorders, each with its own recording pen and ink color. Pen cross-over is accomplished by one pen writing slightly forward of the other. The chart recording width is 11 in.

The log n scale is continuous from .3 to $3x10^5$. The scale of the linear power level is controlled by the range switch on the operator's panel.

6. The Automatic Power-control System has been removed from the system.

7. The Water-radioactivity Monitor consists of a Geiger tube, count-rate circuit, and meter. The Geiger tube is located in the water-monitor vessel and is connected to a meter and an alarm in the console which will ring when the radioactivity exceeds a preset level.

8. The Water-conductivity Monitor consists of two conductivity probes and a Wheatstone bridge circuit. One probe is located upstream of the water demineralizer and the second probe is located downstream. Each conductivity probe consists of a platinum-electrode conductivity cell shielded with glass. Each cell is connected by two conductors, through a selector switch, to a Wheatstone bridge circuit containing a one-stage amplifier, an electronic "tuning-eye," and a rectified power supply. This Wheatstone bridge, which reads water conductivity at either point, is located on the console. The selector switch permits the operator to connect the desired conductivity cell to the bridge circuit.

9. The Water-temperature Monitor consists of a temperature probe in the water line leaving the demineralizer and a meter, on the console. The temperature probe is a resistance thermometer of the tip-sensitive type.

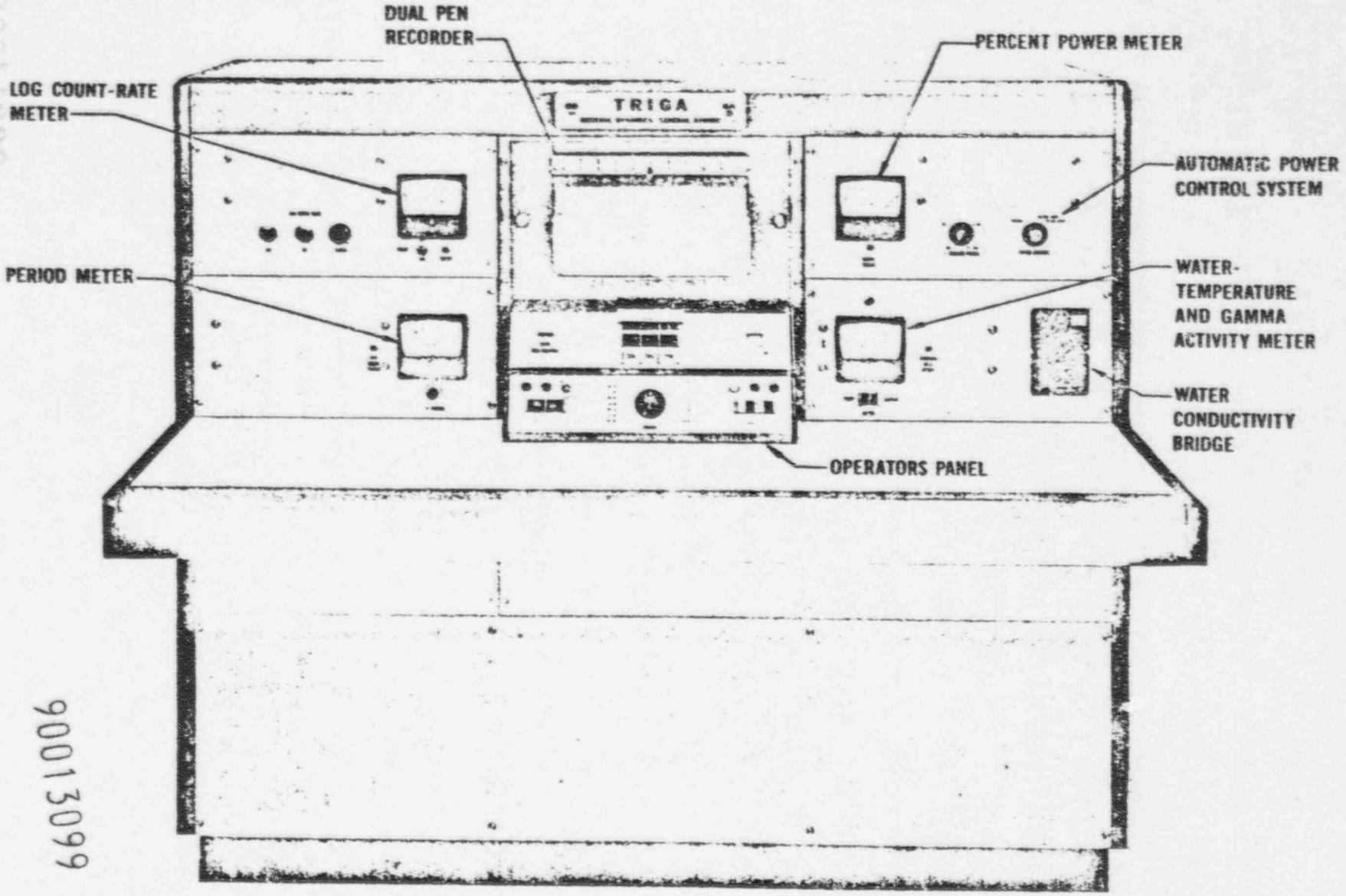
The reactor control console contains the control, indicating, and recording instrumentation required for operation of the reactor. The instrumentation described above is located, as indicated, on either side of the dual-pen recorder and operator's panel.

On the operator's panel are (1) rod-position indicators to show the position of the shim and regulating rods to within 0.2%; (2) control-rod

buttons to control the position of each rod, and annunciator lights to indicate the up and down positions of each rod and rod-magnet control; (3) monitor alarm lights; and (4) additional pilot lights to indicate power on, cooling system on, and adequate source strength. Other annunciator lights on the console indicate the source of a scram signal.

Scram is initiated by (1) excessive power level on either of the power-level channels, (2) too short a positive period, (3) ion-chamber power-supply failure, (4) operating-power failure, (5) manual scram or (6) excessive fuel temperature. Following a scram or the dropping of any rod, and after the rod reaches the "full-in" position, the drive mechanism automatically follows the rod down to re-establish contact. A key-operated switch breaks the rod-magnet circuit so that the console may be operated without rod withdrawal if the switch is off.

A core loading which includes a special fuel element and a pneumatic rod are employed to permit routine pulsed operation of the reactor. These special elements, equipment, and instrumentation permit the rapid insertion of 1.5% δ k/k of excess reactivity, resulting in a pulsed peak power level of approximately 250,000 kw. A special fuel element is equipped with three thermocouples each for temperature measurement and scram following pulsed operation. The additional equipment includes a high-speed, pneumatic rod-drive mechanism that is substituted for one of the standard drives. This mechanism will withdraw the transient rod in about 0.1 sec. The transient rod will also serve as a safety rod if in the full "out" position during steady-state operation. In the steady state mode an interlock prevents application of air to the transient rod unless the shim and regulating rods are in the full "in" position.



Control console (without pulsing instrumentation)

Figure 16

Transient operation is achieved by (1) bringing the reactor to criticality with the transient rod inserted and then to a reasonable power level in the steady-state operating mode; (2) turning the mode switch (S_1, S_2, S_3, and S_4 ganged; see Fig. 16) to pulsing mode; and (3) actuating the pulse button. Changing the switch from the steady-state to the transient mode removes the air interlock mentioned above, thus allowing pulsing to take place. Furthermore, it is not possible to move any rod except the transient rod during transient-mode operation. This fact, coupled with the period scram during operation in the steady-state mode, will prevent pulsing while on a fast period. Since normal instrumentation is disconnected during transient operation, the usual scrams for steady-state operation are not used. Scram is obtained during transient operation either by two fuel-temperature scram circuits or by excessive nv (neutron flux). The fuel-temperature scram will normally be set at 400°C, or less. Turning the mode switch eliminates all neutron channels except one (the uncompensated chamber). This chamber is connected to a peak-power and integrating circuit which will measure the peak power and the integrated nvt in the pulse, store the information in a memory, and then record this information sequentially on the linear recorder within a few seconds. Fuel temperature is indicated accurately only on the two fuel temperature meters although the fuel temperature signals are also fed to the log n recorder.

A thermocouple selector switch is provided which will allow any thermocouple to be momentarily connected to the meters during steady-state operation. Thus, the various thermocouples can be checked against each other at a fairly high power level. The 7-sec period scram during steady-state operation would produce a scram if the rod withdrawal were made fast enough

to produce an excess reactivity of $0.50 or greater. A 2 Ci Americium-beryllium start-up source, with a source strength of about 10^7 neutrons/sec, is located in a special source holder on the edge of the reactor core. The source can be raised from its normal position by means of either the fuel-element handling tool or a wire that is attached to the top of the source element and tied to a cross member at the top of the reactor pit.

3.8. Cooling and Water-treatment System

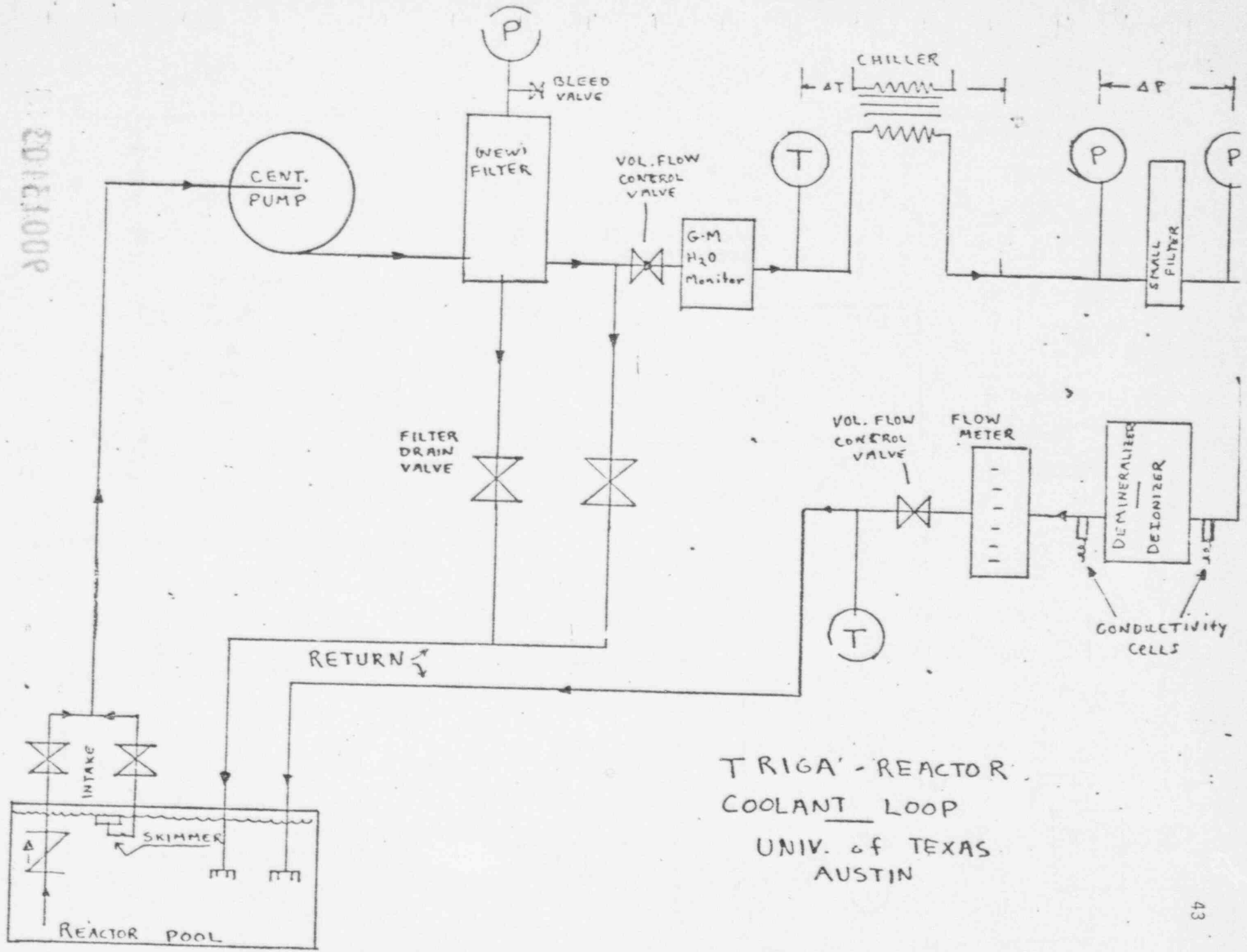
The reactor is cooled by natural convection of the pool water. A 5-ton Freon vapor-compression chiller with an air-cooled condenser is used as the heat sink. Water from the reactor tank goes to a pump, a large filter tank and then to the water monitor, where the gamma activity is measured. It then goes back to the reactor tank and/or to the chiller unit. From the chiller it goes through a filter, a demineralizer, a conductivity cell and then another conductivity cell, a temperature probe to the inlet of a rotameter, where flow rate is measured. The water is then returned to the tank. Figure 17 shows a schematic of the cooling and water-treatment system.

3.8.1. Secondary Cooling System

A second cooling system, as shown in Figure 18, has been added to facilitate removal of the extra heat generated by the increase in power to 250 kw. A detailed description of this system is found in the Manual for Operation of the 250 kw Heat Exchanger.

3.9. Core Physics

The physics of the reactor has been studied in considerable detail on a critical-assembly mock-up as well as on the operating prototype. In the succeeding sections, the more important features of the core physics are discussed.



TRIGA - REACTOR
COOLANT LOOP
UNIV. of TEXAS
AUSTIN

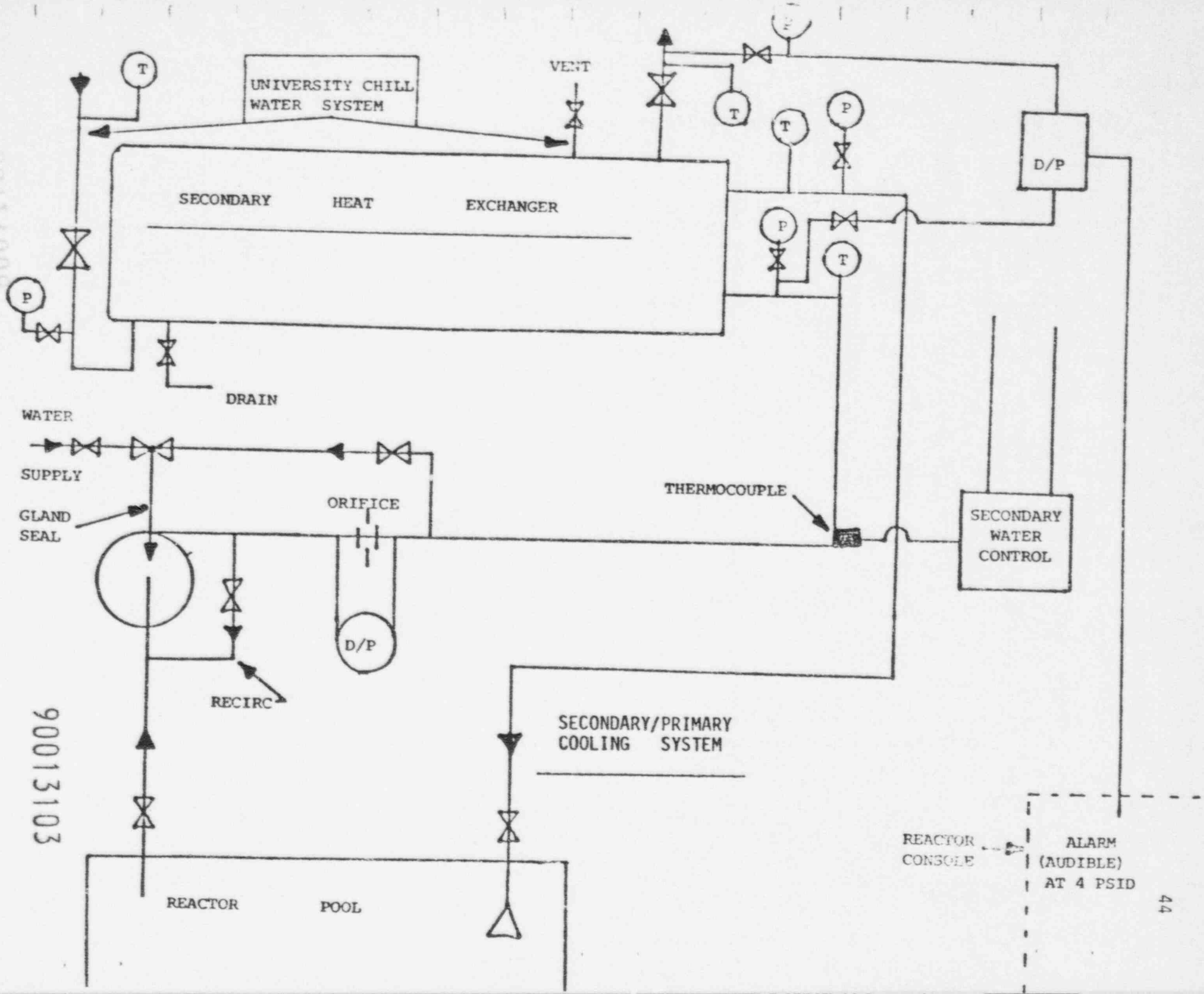


Figure 18

3.9.1. Critical Mass

The prototype TRIGA reactor attained criticality with 54 fuel elements, or about 1.9 kg of U^{235}.

3.9.2. Void Coefficient

The void coefficient should be very similar to the values obtained on the critical-assembly, which were measured to be -0.15% δ k per 1% water void at 23°C in the central region of the core and +0.04% δ k per 1% water void at 23°C at the core-reflection interface, where there is a region of graphite-loaded dummy elements. The core average value was approximately -0.10% δ k per 1% water void, which agrees with calculations.

Calculations indicate that the total reactivity value of the water in the core is about 20%.

3.9.3. Moderating Properties of Zirconium Hydride

Experiments performed by General Atomic personnel at the Brookhaven National Laboratory have shown that zirconium hydride has very unusual moderating properties for slow neutrons. The results of these experiments can be explained by assuming that the hydrogen-atom lattice vibrations can be described by an Einstein model with a characteristic energy $h\nu = 0.130$ ev. This description is consistent with the theory that the hydrogen atom occupies a lattice site at the center of a regular tetrahedron of zirconium atoms. The basic consequences of this model, which have been experimentally verified, are that

1. Neutrons with energies of less than $h\nu$ cannot lose energy in collisions with zirconium hydride.
2. A slow neutron can gain an energy $h\nu$ in a collision with zirconium hydride with a probability proportional to $\exp(-h\nu/kT)$, which increases very rapidly with temperature.

Since $h\nu >> kT$, it has been found that zirconium hydride is not effective in thermalizing neutrons but that it can speed up neutrons already thermalized by water by transferring to them a quantum of energy $h\nu$.

3.9.4. Temperature Coefficient

The fuel temperature coefficient of the TRIGA reactor has been experimentally demonstrated to be -0.013% δ k/k per °C rise in average fuel temperature. The temperature coefficient associated with heating the water and the fuel in the TRIGA reactor core is extremely small. The total reactivity contribution due to this latter coefficient over the range of 10° to 60°C is less than 0.08% δ k/k. The operational characteristics of the reactor are therefore primarily determined by the extremely large prompt negative temperature coefficient. The experiments performed to determine this temperature coefficient demonstrate that it is a prompt coefficient negative temperature coefficient. The experiments performed to determine this temperature coefficient demonstrate that it is a prompt coefficient and that it is nearly constant over the power range from 0 to 1.4 Mw. This corresponds to maximum fuel operating temperatures from ambient to greater than 500°C. At a power level of 250 kw, the maximum fuel temperature is approximately 240°C.

3.9.5. Reactivity Perturbations

Sources of reactivity associated with experimental facilities are:

1. The rotary specimen rack fully "poisoned" is worth -0.15% δ k/k.
2. The pneumatic tube system, located in the sixth (or outside) ring of the core, when fully "poisoned" is worth -0.14% δ k/k.

3. A fuel element replacing the central thimble adds about 1.5% δ k/k. A control rod in the same position is worth about -4% δ k/k.
4. The reactivity effect associated with replacing a void in the central thimble with water is approximately +0.04% δ k/k.
5. All other locations (excluding fuel positions) are outside the reflector and contribute only a trivial amount of reactivity.
6. Extra elements added to the periphery of the core replacing graphite dummy elements would be worth about +0.4% δ k/k each.

4. REACTOR OPERATIONS

The TRIGA-I reactor for The University of Texas in Austin is designed to operate continuously at a power level of 250 kw. The maximum available excess reactivity, for any temperature conditions which could conceivably be encountered, will be limited to 2.25% δ k/k. Experiments conducted with General Atomic's prototype TRIGA show that the reactor power level will be limited to safe values even if all of this available excess reactivity should be suddenly introduced into the reactor (see Section 5.1.4, "Reactor Power Transients").

4.1. Legal Requirements

All operations shall be in strict accordance with The Code of Federal Regulations, Title 10- Energy and with any pertinent State, County and City codes then in effect. The reactor shall be operated only in the manner and for the purposes authorized by the Nuclear Regulatory Commission. All reactor operations shall be performed by, or under the direction and in the presence of, a N.R.C.-licensed reactor operator. All use of "byproduct material" (10 CFR 30), "source material" (10 CFR 40), and "special nuclear material" (10 CFR 70) shall be in accordance with licenses issued by the N.R.C.

4.2. Initial Loading and Start-up

4.2.1. Operational Limitations

During the initial loading and start-up operation the reactor will be protected by at least two level trip channels at all times. In addition, the period trip, which will be operable at all power levels, will be set to cause scram on a 7-sec period.

4.2.2. Operational 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 were performed.

1. Pre-Critical tests:
   a. Functional tests of mechanical equipment.
   b. Check-out of instrumentation and control system.
2. Initial loading of fuel elements to criticality using inverse multiplication data.
3. Tests to be performed upon reaching criticality:
   a. Approximate calibration of control rods by rod drop method.
   b. Adjustment of excess reactivity by addition of fuel elements and preliminary control rod calibration by the period method.
   c. Final calibration of control rods by the period method and determination of final excess reactivity.
   d. Calibration of period meter.
4. Tests to be performed at higher powers:
   a. Check of instrument linearity by going to full power in decade steps.
   b. Power calibration by the method of rate of rise of water temperature.
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.

c. Demonstration of pulsing performance of reactor system.

4.3. Routine Operational Procedures

A routine checkout procedure has been established and is performed prior to start-up each day. The reactor operating procedures will prohibit reactor operation under the following conditions:

1. If the fission counter or more than one of the other neutron-measuring channels is inoperative.
2. If the count rate prior to start-up is below the specified minimum.
3. If any other condition exists which in the judgment of the Reactor Operator, Reactor Supervisor, or Laboratory Director might lead to a potential hazard.

Conditions which will cause the reactor to scram automatically are listed in Section 3.7. The reactor will be shut down routinely by pressing the manual scram button on the operator's console and by driving all the control rods in.

A "Proposed Experiment" description must be prepared in full detail by the experimenter and submitted to the Reactor Supervisor. When and if approved by the Reactor Supervisor, Laboratory Director, Chairman of the Reactor Committee, and Chairman of the Radiation Safety Committee, it will become a "Special Experiment" and may be performed once. Then, if desired, the experiment and its results may be resubmitted for review, and if approved for the purpose by the same persons listed above, it will become a "Routine Experiment" and may be performed at any suitable time within the limitations of law, licenses, or other restrictions in force.

Significant alterations of the above procedure may be made only with the unanimous written approval of all of the above listed persons.

4.4. Health-Safety Procedures

4.4.1. Access to Laboratory

Access to the Nuclear Reactor Laboratory will be restricted. Access will be governed by regulations approved by all of the following:

1. Director, Nuclear Reactor Laboratory
2. Chairman, Department of Mechanical Engineering
3. Chairman, Reactor Committee
4. Chairman, Radiation Safety Committee
5. Radiation Safety Officer

Modifications of the regulations may be made only with the approval of all of the above persons. The Director of the Nuclear Reactor Laboratory will have ultimate control over access to the Laboratory. He may at any time admit, refuse to admit, or expel any person within the scope of the above regulations.

Persons entering the Laboratory will be logged and monitored as described in Section 4.4.3.

4.4.2. Materials Controls

Storage. The fuel elements will be installed in the reactor by licensed reactor laboratory personnel and, in normal operation, will remain in place in the reactor. In the event of fuel-element failure, the element will be removed from the reactor by the use of a shielded cask and will be placed in one of the emergency storage pits temporarily. The U.S.N.R.C. will then be contacted for permanent disposal of the element.

Accountability. The Director of the Nuclear Reactor Laboratory is responsible for fuel elements until they are returned to the NRC or its authorized representative.

Radioactive Wastes. Radioactive wastes will be disposed of in accordance with Title 10, Code of Federal Regulations, Part 20. It is anticipated that spent-fuel-element disposal will not be required for a number of years.

4.4.3. Health Physics

Equipment. The following items or their equivalents will be included in the health-physics equipment to be utilized in the Reactor Laboratory:

Victoreen Thyac III alpha-beta-gamma survey meter,

Technical Associates alpha-beta-gamma neutron survey meter,

Nuclear Chicago Model 2673N fast-slow neutron survey meter,

Film badges,

Landsverk thermal-neutron and gamma-sensitive pocket dosimeters,

Nuclear Measurements air sampler.

Surveys. The reactor room will be constantly monitored by either a portable or installed G-M Area Radiation Monitor and a constant air monitor. Reactor pool water is monitored via the purification system and by quarterly samples taken by the Radiation Safety Officer. Surface contamination wipe surveys will be performed quarterly by the Radiation Safety Office and at other times as deemed necessary by the Reactor Supervisor.

Unusual Events. Any person within the Reactor Laboratory is required to report unusual events to the Reactor Supervisor, who will investigate the event and recommend corrective measures to the Laboratory Director.

Personnel Safety Practices. Entrance to the reactor laboratory will be controlled as described in Section 4.4.1., "Access to Laboratory." Laboratory personnel will be issued film badges, and occasional visitors will be required to wear pocket dosimeters. A log will be kept on all personnel entering, including film-badge number and ion-chamber reading.

Personnel in the laboratory will wear protective clothing as required and will handle radioactive materials with gloves, with tongs, or with suitable remote-handling equipment.

4.5. Organization

An organizational flow chart showing lines of responsibility for the Reactor facility is shown in Figure 19.

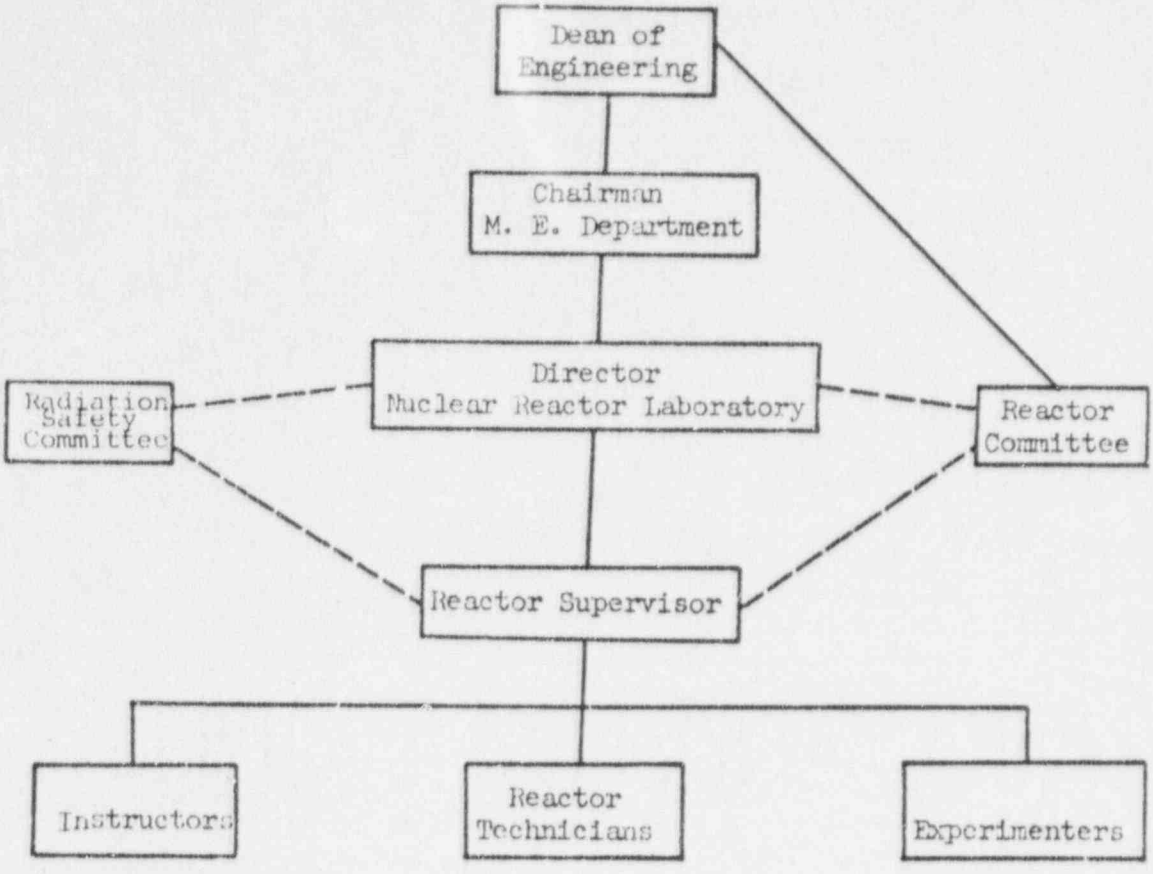
4.5.1. Reactor Committee

Composition of this committee shall be a chairman plus at least three other members.

Appointment to or removal from this committee shall be by the Dean of Engineering on the basis of qualifications and interests, and the members are responsible to him. A maximum of one person may belong to both the Reactor Committee and the Isotopes Committee, provided that he not be chairman of both committees.

Functions of this committee are:

1. To assist the Nuclear Reactor Laboratory Director in establishing operational procedures.
2. To meet and receive a report at least quarterly from a designated committee member concerning his inspection of the Nuclear Reactor Laboratory and the operation thereof.
3. To meet as needed to approve or disapprove reactor experiments as submitted by the Nuclear Reactor Laboratory Director.
4. To meet as needed to approve or disapprove proposed changes in Nuclear Reactor Laboratory procedures and



——————— LINE OF RESPONSIBILITY

– – – – – – – – – – CONSULTATION AND VETO POWER

ORGANIZATIONAL CHART
FIG. 19

proposed significant alterations to reactor instrumentation and proposed changes in the N.R.C. utilization facility license.

5. To order a reactor experiment to be stopped and/or the reactor not to be operated if in their opinion any significant hazard exists.
6. To transmit to the Nuclear Reactor Laboratory Director or Supervisor whatever suggestions or comments they consider desirable.
7. The chairman shall cause written records to be kept of all committee actions.
8. At his discretion the chairman may declare any number of committee members to constitute a quorum.

Modifications of the above may be made with the written approval of all of the following:

Dean of Engineering
Chairman, Mechanical Engineering Dept.
Chairman, Reactor Committee
Director, Nuclear Reactor Laboratory

4.5.2. Radiation Safety Committee

Composition of and appointment to or removal from this committee shall be determined by the Vice President and Provost, and the members are responsible to him.

Functions of this committee are determined by the Vice President and Provost. The specific functions of this committee or their designated Radiological Protection Officer relative to the Nuclear Reactor Laboratory are:

1. To furnish advice and assistance concerning radiological health upon request from Nuclear Reactor Laboratory personnel.
2. To meet and receive a report at least semi-annually from the designated Radiological Protection Officer or committee member concerning his inspection of the radiological health aspects of the Nuclear Reactor Laboratory.
3. To meet as needed to approve or disapprove reactor experiments from the radiological health standpoint as submitted by the Nuclear Reactor Laboratory Director.
4. To meet as needed to approve or disapprove from the radiological health standpoint proposed changes in Nuclear Reactor Laboratory procedures as submitted by the Nuclear Reactor Laboratory Director.
5. To order an experiment or other operation to be stopped if in their opinion a significant radiological health hazard exists.
6. To transmit to the Nuclear Reactor Laboratory Director or Supervisor whatever suggestions or comments they consider desirable.
7. The chairman shall cause records to be kept of all committee actions pertaining to the Nuclear Reactor Laboratory.
8. At his discretion, the chairman may declare any number of committee members to constitute a quorum in regard to their Nuclear Reactor Laboratory functions.

Modifications of the above which affect the Nuclear Reactor Laboratory may be made with the written approval of all of the following:

Vice President and Provost
Chairman, Radiation Safety Committee
Director, Nuclear Reactor Laboratory

5. POTENTIAL HAZARDS

5.1. Hazards Associated with the Operation of the Reactor

Certain potential hazards associated with the operation of the reactor system have been studied and have been found to cause no serious environmental hazard, nor any hazard to operating personnel. However, in the handling of all radioactive material, because of its very nature, it is necessary to observe accepted safety precautions. In the normal use of TRIGA it will be necessary to handle irradiated samples, and standard health-physics procedures will be followed. On the rare occasions when it is necessary to remove highly radioactive fuel elements from the water shield, special equipment and safety procedures will be utilized.

Specifically, the following problems were investigated: activation of the soil surrounding the reactor pit; radioactive contamination of the shielding water; and the production of radioactive gases in the reactor. A description of the behavior of the reactor during a reactivity accident is presented. The possibility of loss of shielding water is discussed, as are the hazards of handling radioactive materials.

In order to evaluate the possible hazards associated with the operation of this reactor the following assumptions have been made:

1. This reactor normally will have two standard modes of operation, as follows:
   a. Non-pulsed steady-state operation at power levels up to 250 kw.
   b. Pulsed operation produced by rapid transient-rod withdrawal, which results in insertion of a step increase of reactivity into the system. The pulsing rod for the reactor system is designed to insert as a maximum 1.5% δ k/k (2.00 dollar)

above critical, which is twice the amount required to attain prompt criticality.

2. The reactor operations will be supervised by individuals who are qualified operators trained in the detection and evaluation of radiological hazards.

5.1.1. Soil Activation

The soil adjacent to the reactor pit will capture fast and thermal neutrons which escape from the pit. The magnitude of the radioactivity induced has been approximated for typical soils in order to determine whether leaching of activity in the soil by ground water might constitute a potential environmental hazard. The activities were below the permissible NRC tolerances for burial of radioactive materials. These calculations were based upon 250 kw steady-state operation with 48 inches of concrete shielding around the core and 24 inches of concrete below it.

5.1.2. Radioactive Contamination of Shielding Water

Contaminant material susceptible to neutron irradiation in the shield water is maintained at low levels by the demineralizing system. The hazards associated with a failure of the fuel-element cladding and consequent fission-product contamination of the water have been calculated and studied experimentally as described in section 5.1.11. The results show that in the improbable event of a cladding failure, the water activity may reach a maximum level of 0.8 $\mu c/cm^3$. The water may be decontaminated by using the demineralizer or by dilution. Manufacturing inspection and quality control assure that the possibility of a cladding failure is very small.

Three mechanisms (recoil, diffusion, and dissolution) govern the release of fission products from a fuel element. The most important of these mechanisms

is the fission-product recoil from near the surface of the fuel. This mechanism, therefore, is considered in the calculation of the fission-product release after a fuel-element failure covered in section 5.1.11. The dissolution rate of unclad fuel in water has also been studied experimentally at General Atomic and was found to be small.

5.1.3. Production of Radioactive Gases by the Reactor

Calculations and experiments nave shown that in regions of the reactor containing air, in which the neutron flux is assumed to be the same as that in the rotary specimen rack, an equilibrium concentration of 7.5 $\mu c/cm^3$ of 109-minute argon-41 is established after the reactor has operated at 250-kw for several hours. Under these conditions, the equilibrium argon-41 content in the rotary specimen rack is calculated to be approximately 250 millicuries (mc). After the argon-41 is purged from this cavity, it takes several hours to re-establish an equilibrium concentration.

The maximum permissible concentration (MPC) for radioactive argon is 2×10^{-6} $\mu c/cm^3$ for a 40-hr week.* This MPC would just be maintained in the reactor building which has an air volume of 1000 m^3 if approximately 8 mc of argon-41 were liberated in the room every 8 hours with no ventilation system in operation.

The transfer of argon from the rotary specimen rack to the reactor room is inhibited by the long, slender access tube, and the tube cover plate.

The pneumatic transfer system has been modified to use carbon dioxide gas in a closed loop system. Calculations indicate saturated activities of 7.7×10^{-2} $\mu c/cm^3$ of 7-second N-16 and 0.2×10^{-2} $\mu c/cm^3$ fo 29-second O-19 may be produced in the CO_2 exposed to the neutron flux in the outer ring of the core matrix. The total activity that could be produced in the pneumatic tube during steady state operation is, therefore, approximately 12.6 μc.

*Title 10, Chapter 1, Part 20, Code of Federal Regulations

Similar calculations show that the total activity induced in the CO_2 for a \$2.00 pulse is approximately 60 μc. If this activity were released instantaneously into the reactor room the maximum resulting concentration would be 6×10^{-8} μc/cm^3. This concentration is well below 1×10^{-6} μc/cm^3 which represents the MPC for CO_2 in an unrestricted area.* Further, it should be noted that these radioactive gases decay rapidly and that normally the CO_2 gas is contained in a closed loop system with a small leakage rate.

Significant quantities of 7-sec nitrogen-16 and small quantities of argon-41 (refer to section 5.1.13) are produced in the water of the reactor core. However, the transport time from the reactor core to the surface of the shielding water when the reactor is operating at 250-kw is sufficient to allow substantial decay of the nitrogen-16. Experiments on the General Atomic TRIGA reactor show the following:

1. Small quantities of nitrogen-16 and argon-41 are transferred from the reactor shielding water to the air in the reactor building.
2. Although the nitrogen-16 and argon-41 activities in the shielding water contribute somewhat to the radiation dose measured at the top of the reactor during operation, the dosage is well within acceptable limits for personnel exposure.

5.1.4. Reactor Power Transients

Calculations show that the rapid insertion of \$2.00 excess reactivity results in a pulse with a peak power of ~250 Mw, as shown in Fig. 1, and a reactor period of approximately 10 msec. Experiments performed on the General Atomic prototype TRIGA reactor in San Diego confirm this analysis. Upon rapid insertion of \$2.00 excess reactivity, the reactor power level increases on

*Title 10, Chapter 1, Part 20, Code of Federal Regulations

the predicted period and attains the predicted peak level. Within a few seconds after the initiation of the transient, the reactor power level returns to an equilibrium power level of about 200 kw. The rapid reduction in reactor power level following peak power is brought about through the prompt negative temperature coefficient, a characteristic of the reactor fuel-moderator elements. This temperature coefficient has been measured to be an average of about \$0.017 reactivity loss /C°rise in fuel temperature.

The reactor loading will be a maximum of 2.25% δ k/k (\$3.00) excess reactivity above a cold, critical, and compact condition. A "compact condition" means that the core is loaded so that all of the innermost fuel positions contain fuel, starting from the B-ring, which consists of six elements. Thus, the maximum reactivity transient that could possibly occur would be that produced by the rapid insertion of the entire available amount of excess reactivity. During the metallurgical development of the pulsing TRIGA fuel, the prototype TRIGA reactor at General Atomic was pulsed safely over 8000 times with reactivity insertions up to 2.25% δ k/k (\$3.00). The resulting maximum excursion attains a peak power of 1500 Mw; the reactor period is 4.0 msec; and the total energy release during the burst is approximately 20 Mw-sec. The maximum measured fuel temperature during this pulse is less than 500°C. Curves of the transient power level and of the fuel temperature during the \$2.00 pulse are shown in Fig. 22.

On the basis of the operating experience with the TRIGA prototype, it is concluded that there is no hazard associated with the standard mode of pulsed operation, i.e., routine rapid insertion of up to and including 1.5% δ k/k excess reactivity in this reactor; nor would there be any hazard associated with the sudden accidental insertion of the 2.25% δ k/k total available excess reactivity in this reactor.

5.1.5. Loss of Shielding and Cooling Water

Loss of water can occur by only two means: the tank may be pumped dry, or a tank failure may allow the water to drain.

The tank outlet water lines each have a 1/2-in.-diameter hole drilled a foot below the normal water level. The purpose of these holes is to break pump suction or accidental siphoning if the tank water level drops below this hole. Also the tank outlet pipe extends only about 3 ft below normal water level. Therefore, even if the water system were operated carelessly, for example, when the pump discharge line was disconnected for repairs, the tank could not be pumped dry accidentally. The tank can be pumped dry only by deliberate action of the operating crew. In the unlikely event that the tank must be drained for repairs, the fuel will be removed from the reactor and stored in shielded casks.

The TRIGA Mark I concrete shield has been designed by a firm experienced in the design of earthquake-proof structures. There is no record of a facility of this type sustaining damage through an earthquake. The aluminum reactor tank and welded aluminum experimental facilities are designed and thoroughly tested to provide tightness against leakage.

Even though the possibility of the loss of shielding water is believed to be exceedingly remote, a calculation has been performed to evaluate the radiological hazards associated with this type of accident under the condition that the reactor has been operating for a long period of time at 250 kw prior to losing all of the shielding water. The radiation dose rates were determined for the two different locations given in Table 1. The first location, 18 ft. above the unshielded reactor core at the top of the reactor tank, receives direct radiation. The second location, at the top of the reactor shield, is shielded from the direct radiation by the concrete reactor structure, but is subject to scattered radiation. The assumption is made that a thick concrete

ceiling 9 ft above the top of the reactor shield will maximize the reflected radiation dose. Normal roof structures would give considerably less back-scattering. Time is measured from the conclusion of 250 kw operation. Dose rates assume no water in the tank.

Table 1

RADIATION DOSE RATES AFTER 250-KW OPERATION
AND LOSS OF ALL SHIELDING WATER

Time	Direct Radiation (r/hr)	Scattered Radiation (r/hr)
10 sec	2.7×10^4	15
1 day	1.4×10^3	0.8
1 week	9.3×10^2	0.5
1 month	4.5×10^2	0.2

The above data show that if an individual does not expose himself to the core directly, he could work for approximately 90 min at the top of the shield tank after 1 day without being exposed to radiation in excess of the normal quarterly occupational dose of approximately 1.25 rem allowed by 10 CFR 20. A 90-min. period would probably be sufficient time in which to view the interior of the shield tank with a mirror and to make emergency repairs.

Outside the building the radiation from the unshielded core would be collimated upward by the shield structure and, therefore, would not be a hazard to the public.

Calculations have been made by General Atomic (GA-7860) to determine the temperature rise in a central (B-ring) fuel element if the cooling water

is lost instantaneously. These calculations shown in section 5.1.12 were made in a very conservative manner in order to give an upper-limit estimate of the fuel-cladding temperature.

The calculations show that the maximum fuel-element temperature following an instantaneous and complete loss of water would be less than 150°C.

On the basis of the considerations given above, the following conclusions were reached:

1. The probability of the loss of all of the shielding water is extremely small.
2. The concrete reactor shield provides adequate protection for nearby personnel in the event of the loss of all of the shielding water.
3. The reactor shielding is sufficient to allow emergency repair of the leak approximately 1 day following shutdown, even after prolonged operation at maximum power.
4. The gas pressure within the hottest fuel element would not rupture the cladding; consequently, the loss of water would not damage the core.

5.1.6. Production and Handling of Radioactive Material

Because radiation from radioactive isotopes produced by this reactor can be intense, reactor operation must be supervised by persons who are trained in the detection and evaluation of radiological hazards. These hazards, however, are the same as those encountered with any reactor operating at comparable power levels.

Calculations based on cobalt-60^m activation indicate that at 250-kw the reactor can produce a radioisotope equilibrium concentration of approximately 2000 curies in the rotary specimen rack. A second source of radioactivity is the activity produced in the fuel-moderator elements. Calculations indicate that if the reactor is operated at 250 kw, the equilibrium activity associated with one of these elements is approximatley 2×10^4 curies at the time of shutdown.

The maximum production of 2000 curies is distributed in 40 sample positions, each containing two sample containers; the maximum amount of activity that can be withdrawn at one time, therefore, is approximately 25 curies. Though intense, such radiation sources are handled routinely by competent personnel.

The radioactivity hazard associated with fuel elements is of the same nature as that associated with isotope production. Because of the high radiation level associated with the reactor fuel-moderator elements during and after use, the elements will normally be kept under water for shielding purposes. If an element is to be removed from the reactor tank, a conventional fuel-element transfer cask can be used to reduce the radiation level to within tolerable limits. The operation will be performed under strict supervision by safety cognizant staff members.

5.1.7. Radiation Dose Rates Around Reactor

Complete gamma dose measurements have been taken around typical TRIGA Mark I reactors during 250-kw steady-state operation. Typical measurements are listed in Table 2. No neutron leakage has been detected from operating TRIGA reactors except for a thermal neutron dose of 0.03 mrem/hr (15 neutrons/cm^2-sec) measured above the rotary specimen rack drive shaft tube during 100-kw operation.

As indicated in Table 2, the measured radiation dose rates are low enough to allow operating personnel to perform experiments at the edge of the reactor tank during full power operation.

Table 2

GAMMA DOSE RATES AROUND TRIGA MARK I REACTOR
DURING 250-KW STEADY-STATE OPERATION

Location of Instrument	Dose Rate (mr/hr)[a]
	Measured 250-kw Steady-State Operation
1 ft above surface of reactor tank water	15 to 30
At handrail adjacent to reactor tank	3 to 5
Maximum dose in the passage way beneath the floor of the reactor room[b]	~50

[a] Dose rates apply when the reactor cooling system is in operation. These doses are given in milliroentgens (mr).

[b] This dose rate is estimated from measurements taken during operation of the University of Texas TRIGA at 10 kw. A locked gate with instructions to contact the Reactor Supervisor, Nuclear Reactor Laboratory, for admission controls access to this area.

The maximum permissible dose in restricted areas established by 10 CFR 20, for persons whose previous radiation history is unknown, is 1.25 roentgens per quarter calendar year (approximately 100 mrem per week).

The International Commission on Radiation Protection (ICRP) recommends that the average external occupational exposure should not exceed 5 rem per year.

5.1.8. Reactivity Hazards Associated with Experimental Facilities

Sources of reactivity associated with experimental facilities are:

1. The rotary specimen rack fully poisoned is worth -0.15% δ k/k.
2. The pneumatic tube system, located in the F-ring of the core, fully poisoned is worth -0.14% δ k/k.
3. A fuel element replacing the central thimble adds about 1.5% δ k/k. A control rod in the same position is worth about -4% δ k/k.
4. All other locations (excluding fuel positions) are outside the reflector and contribute only a trivial amount of reactivity.
5. Extra elements replacing graphite dummy elements at the periphery of the core would be worth about +0.4% δ k/k each.

5.1.9. Fuel Element Failure

A cladding failure, or even the failure of the cladding of several fuel elements, would not constitute an undue hazard to the operating crew, or the general public. Should an accident occur, small amounts of radioactive noble gases could be dispersed from the reactor pool into the air the reactor building, and these would decay into particulate matter.

According to the handbook Heating, Ventilating, and Air-Conditioning Guide,* the number of air changes in the building without a ventilation system varies from 0.5 to 2 air changes per hour under average conditions. Based on this, it can be assumed that most of the fission-product gases leak out of

*Heating, Ventilating, and Air-Conditioning Guide, American Society of Heating and Ventilating Engineers, New York, New York, 1953, p. 223.

the reactor building within 1 hr. The fission-product gases are mainly xenon and krypton, with an average total activity in the building of 13.5 curies, as calculated and discussed in section 5.1.11. The average concentration of xenon and krypton in the reactor building air will be 1.35×10^{-2} $\mu c/cm^3$. If it is assumed that the fission-product gases leak out of one side of the building only, a person present for 1 hr near that side of the building could receive an integrated dose of 164 mr, resulting from the decay of the fission-product gases. If the fission-product gases are assumed to escape from the building at a lower rate than previously postulated, the integrated dose received by a person would be approximately the same as before, because of the longer decay period for the fission-product gases before leaving the building. This dose would be very much decreased by fresh-air dilution of the gases a short distance from the building. Assuming that a fuel-element failure occurred once each year, the concentration of the fission-product gases leaving the building, averaged over a 1-yr period, would be 1.54×10^{-6} $\mu c/cm^3$.

An example of the activity caused by a cladding failure is provided by an incident that occurred during operation of the prototype TRIGA reactor at San Diego. During transient tests performed to verify the operational safety of the TRIGA reactors, an early model non-pulsing-type fuel element suffered a cladding failure following the step insertion of 2.85% δ k/k ($3.90). The following summarizes the results:

1. The activity in the pool water reached a maximum of 0.2 $\mu c/ml$. It decayed very rapidly and was measured as 5×10^{-5} $\mu c/ml$, 24 hr after the cladding failure.
2. The activity in the air of the reactor room reached about 10 times the MPC for fission products and decayed very rapidly. The maximum exposure to operating personnel was less than 1 mr.

3. The noble gases were not collected on the filter samples used, but it may be inferred from the nature of the particulates observed that only the noble-gas fission products escape from the TRIGA reactor pool in significant proportions when cladding fails.

The preceding discussion shows that no significant hazard to the public results from such an accident.

5.1.10. Adequate Steady State Cooling

The bulk water temperature will be maintained below 120°F at all times by administrative control. The pool water is continuously monitored while the reactor is in operation. An alarm will be set to annunciate and corrective action will be taken if the maximum bulk pool temperature reaches 120°F.

The rate of increase in water temperature with no cooling at 250 kw is calculated as 8.58°F/hr and a secondary cooling system heat exchanger has been installed which will reduce bulk pool temperature at the rate of 9°F/hr. Thus, the reactor can be operated continuously at 250 kw with no increase in pool water temperature.

5.1.11. Calculated Maximum Fission-Product Release After a Fuel Element Failure

Calculations and a related experiment have been made to determine theoretically the maximum concentration of fission products that might be present in the reactor-room air following a fuel-element-cladding failure.

The calculations are based on the fact that as the reactor operates, fission products will be built up in the uranium-zirconium hydride fuel mixture until an equilibrium concentration is reached for each nuclide, dependent on

(1) the total energy release in the reactor, (2) the decay process for each nuclide, and (3) the yield of the species from fission. Of the various fission products produced in the fuel material, only certain nuclides will migrate into the gap between the fuel material and the fuel element cladding. These nuclides are the iodines, the xenons, the kryptons, and the decay products of all of these elements.

Calculation of Gaseous Fission Products. In the event of a rupture of the fuel element cladding, the fission products released to the water in the reactor tank will be limited to all or a portion of those fission products that have collected in the gap between the fuel material and the aluminum cladding. A portion of these fission products will then go into the air above the water, this fraction depending on the solubility of the species in water.

The quantity of gaseous fission products produced in the fuel element was determined by Blomeke and Todd.* The amounts of krypton, xenon, and iodine produced in a typical element after infinite operation at 250 kw are given in Table 3. These data are based on a loading of 36.8 grams of U^{235} per element where $N°_{25}$ is the initial number of U^{235} nuclei (9.4×10^{22}) and N_s is the number of nuclei of the isotopes in the fuel element, and on an average flux of 4×10^{12} thermal neutrons/cm^2-sec.

Experimental Determination of Fission Products in Gap. In order to determine the actual percentage of fission-product gases that escape from the fuel material and collect in the air gap between the cladding and the fuel material, the following experiment was conducted in the TRIGA reactor at General Atomic. A fuel element was fabricated with a sealed tube that vented the gap to a charcoal-filled cold trap at the surface of the reactor tank.

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*J. O. Blomeke and Mary F. Todd, "Uranium-235 Fission-Product Production as a Function of Thermal Neutron Flux, Irradiation Time, and Decay Time," Pt. I, Vol. 1, Oak Ridge National Laboratory, Report ORNL-2127, 1957.

Table 3

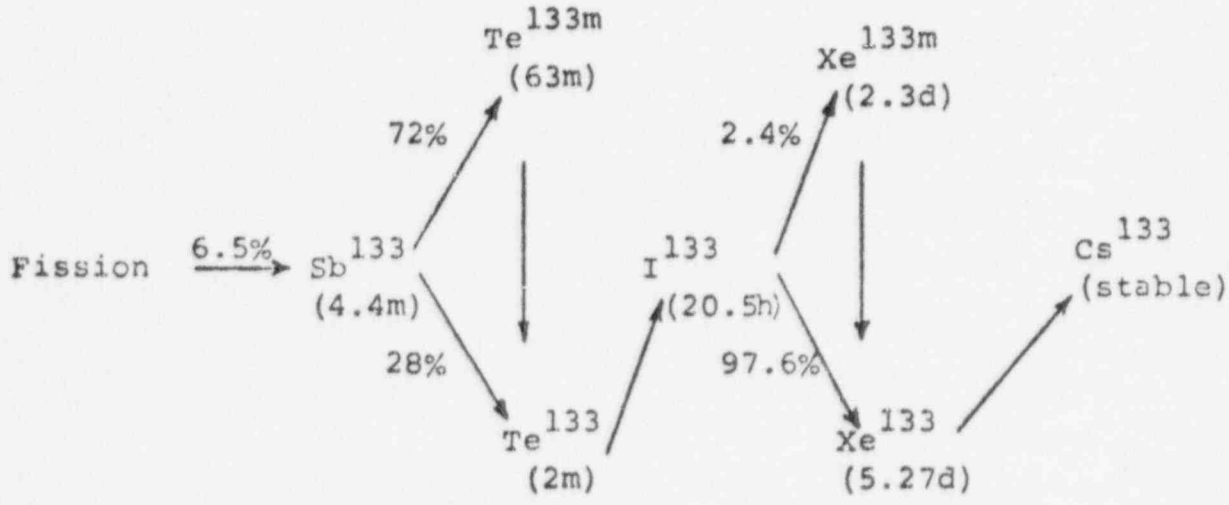
GASEOUS FISSION PRODUCTS PRODUCED IN FUEL ELEMENT

Isotope	N_s/N_{25}^o	N_s	Decay Constant, sec^{-1}	Activity, curies
Kr^{83m}	1.1×10^{-7}	1.0×10^{16}	1.0×10^{-4}	28
Kr^{85m}	7.8×10^{-7}	7.2×10^{16}	4.4×10^{-5}	85
Kr^{85}	3.0×10^{-3}	2.8×10^{20}	2.2×10^{-9}	16
Kr^{87}	3.2×10^{-7}	2.9×10^{16}	1.5×10^{-4}	119
Kr^{88}	1.2×10^{-6}	1.1×10^{17}	7.0×10^{-5}	208
Kr^{89}	2.0×10^{-8}	1.8×10^{15}	3.6×10^{-3}	179
Kr^{90}	5.8×10^{-9}	5.3×10^{14}	2.1×10^{-2}	303
Kr^{92}	2.7×10^{-10}	2.5×10^{13}	2.3×10^{-1}	155
I^{130}	1.6×10^{-6}	1.5×10^{17}	1.5×10^{-5}	59.6
I^{131}	7.0×10^{-5}	6.4×10^{18}	1.0×10^{-6}	174
Xe^{131m}	1.0×10^{-6}	7.2×10^{16}	6.7×10^{-7}	1.7
I^{132}	1.3×10^{-6}	1.2×10^{17}	8.0×10^{-5}	259
I^{133}	1.6×10^{-5}	1.5×10^{18}	9.3×10^{-6}	371
Xe^{133m}	1.0×10^{-6}	9.2×10^{16}	3.5×10^{-6}	8.7
Xe^{133}	1.0×10^{-4}	9.2×10^{18}	1.5×10^{-6}	373
I^{134}	8.0×10^{-7}	7.4×10^{16}	2.2×10^{-4}	438
I^{135}	4.7×10^{-6}	4.3×10^{17}	2.9×10^{-5}	339
Xe^{135m}	5.7×10^{-8}	5.2×10^{15}	7.4×10^{-4}	105
Xe^{135}	4.8×10^{-6}	4.4×10^{17}	2.1×10^{-5}	251
I^{136}	8.8×10^{-9}	8.1×10^{14}	8.1×10^{-3}	178
Xe^{137}	4.6×10^{-8}	4.2×10^{15}	3.0×10^{-3}	351
I^{138}	6.7×10^{-10}	6.2×10^{13}	1.2×10^{-1}	200
Xe^{138}	1.9×10^{-7}	1.8×10^{16}	6.8×10^{-4}	327
I^{139}	1.6×10^{-10}	1.5×10^{13}	2.6×10^{-1}	104
Xe^{139}	6.5×10^{-9}	6.0×10^{14}	1.7×10^{-2}	274
Xe^{140}	2.0×10^{-9}	1.8×10^{14}	4.3×10^{-2}	214
			Total	5120

All of the fission-product gases that accumulated in the gap were collected in the liquid-air-cooled charcoal trap by purging the system with helium, and the trap was analyzed. This measured amount of radioactive noble gases enabled the determination of the fraction of the fission products that diffused through the uranium-zirconium hydride material into the gap.

In the experiment, a total of 210 kw-hr of reactor operation took place in 1 hour. The test element was exposed to an average thermal flux of 2.1×10^{12} neutrons/cm^2-sec during this time, and the Xe^{133} that finds its way into the gap during the exposure of the element to a known amount of reactor flux, the percentage of Xe^{133} and consequently of all other fission-product gases that collect in the gap can be determined.

During the experiment, Xe^{133} was produced in the fuel element from the chain:



The activity of Xe^{133} present in the element at the end of 1 hour at 2.1×10^{12} neutrons/cm^2-sec was determined from the work of Bolles and Ballou.* In this work

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*R. C. Bolles and N. E. Ballou, "Calculated Activities and Abundances of U^{235} Fission Products," USNRDL-456, August 30, 1956.

the number of Xe^{133} nuclei present after the fission of 10,000 U^{235} nuclei is given as a function of time. Integrating these data over 1 hour gives 0.49×10^{-3} Xe^{133} nuclei present per fission. In the 1-hour experiment, the number of fissions is:

$$\text{No. of Fissions} = 9.4 \times 10^{22} \frac{U^{235} \text{ atoms}}{\text{element}} \times 3.8 \times 10^{-22} \frac{\text{cm}^2}{U^{235} \text{ atom}}$$

$$\times 2.1 \times 10^{12} \frac{\text{neutrons}}{\text{cm}^2\text{-sec}} \times 3.6 \times 10^{3} \text{ sec}$$

$$= 2.7 \times 10^{17}$$

Thus, the number of Xe^{133} nuclei is $0.49 \times 10^{-3} \times 2.7 \times 10^{17}$, or 1.3×10^{14} nuclei. This gives a total available Xe^{133} activity of

$$A = \frac{1.52 \times 10^{-6} \times 1.3 \times 10^{14}}{3.7 \times 10^{4}} \mu c = 5.4 \times 10^{3} \mu c.$$

As was indicated previously, the amount of Xe^{133} that was collected in the gap at the end of 1 hour was 78 microcuries. The percentage of the total inventory of Xe^{133} that escaped to the gap between fuel and cladding, therefore, can be calculated as follows:

$$\frac{Xe^{133} \text{ measured in gap}}{Xe^{133} \text{ total available}} = \frac{78 \mu c}{5.4 \times 10^{3} \mu c} = 1.4\%$$

As shown in Table 3, at 250 kw the total quantity of all fission-product gases in a TRIGA reactor fuel element at equilibrium is 5,120 curies. For the purposes of this calculation, it is assumed that the fractions of the iodine, krypton, and xenon isotopes produced that collect in the gap between fuel and cladding are the same as that determined for Xe^{133}. Thus, the total gaseous activity in the gap is calculated to be 71.8 curies.

Therefore, the maximum amount of fission products that could be released in the event of a cladding failure is:

Iodines	33.7 c
Xenons	24.2 c
Kryptons	13.9 c
Total Fission Products	71.8 c

The volume of water in the TRIGA reactor tank is approximately 4.5×10^7 cm^3. For purposes of this calculation, it is assumed that the total 71.8 curies of gaseous fission products in the gap escapes into the water. However, when these fission products (the iodines, kryptons, and xenons) are released from the fuel element into the water, the iodines will be dissolved in the water. Since the kryptons and xenons are quite insoluble in water, it has been assumed for purposes of this analysi, that a major portion of these isotopes will escape into the reactor-room atmosphere. Based on the assumption that xenon and krypton follow Henry's law, it is estimated that 95% of the krypton will escape into the air and that all of the iodines will remain in the water. Therefore, the activity in the water will be:

$$A_{water}\ \frac{\mu c}{cm^3} = \frac{33.7 + 0.05 \times 24.2 + 0.02 \times 13.9}{4.5 \times 10^7\ cm^3} \times 10^6\ \mu c$$

$$= \frac{35.2 \times 10^6\ \mu c}{4.5 \times 10^7\ cm^3} = 0.80\ \mu c/cm^3$$

and the activity in the air will be

$$A_{air} = \frac{0.95A\ \text{xenon}\ (\mu c) + 0.98A\ \text{krypton}\ (\mu c)}{\text{volume of air}\ (cm^3)}$$

$$= \frac{0.95 \times 24.2 + 0.98 \times 13.9}{1.0 \times 10^9\ cm^3} \times 10^6\ \mu c$$

$$+ \frac{36.6 \times 10^6 \mu c}{1.0 \times 10^9\ cm^3} = 3.66 \times 10^{-2}\ \mu c/cm^3$$

Since krypton and xenon are inert gases, the hazard presented by their being in air is from the dose a person in the room would receive from their decay. To estimate this dose, it is assumed that each decay of a krypton or xenon nucleus results in the emission of a 0.5-Mev gamma photon. Then the dose rate at the center of a hemispherical volume of 29,000 ft^3 (equivalent radius = 24.0 ft) is

$$D = \frac{(3.66 \times 10^{-8})(3.7 \times 10^{10})(1 - e^{-8 \times 10^{-5} \times 24.0 \times 30.5})}{(2 \times 8 \times 10^{-5})(1.085 \times 10^{6})}$$

$$= 0.445 \text{ rad/hr} = 445 \text{ mr/hr}$$

where the attenuation coefficient for air is taken as 8×10^{-5} cm^{-1} and the flux-to-dose conversion factor is 1.085×10^{6}.

To estimate an integrated dose to a person remaining in the reactor room for 1 hour after a rupture, the xenon and krypton activities in the room were averaged over 1 hour. This average is 13.5 curies, or an average concentration of 1.35×10^{-2} $\mu c/cm^3$.

If we assume that each disintegration is accompanied by a 0.5-Mev photon and that the room can be approximated by a hemisphere of equivalent volume, then by proportion, using the above initial dose value, the average dose rate will be

$$D = \frac{1.35 \times 10^{-8} \text{ c/cm}^3}{3.66 \times 10^{-8} \text{ c/cm}^3} \times 445 \text{ mr/hr} = 164 \text{ mr/hr}$$

It can thus be observed that even on the basis of the conservative assumptions made herein, a person could remain in the reactor room for about one-half hour after a fuel-element-cladding failure without exceeding the permissible radiation dose limit of 100 mr/week. Standard operating procedures for the facility will require prompt evacuation of the reactor room by the operating personnel on indication of airborne radioactivity in the reactor room.

According to the Heating, Ventilating, and Air-Conditioning Guide,* the number of air changes in a building without a ventilation system varies from 0.5 to 2 air changes per hour under average conditions. Based on this, it can be assumed that most of the fission-product gases leak out of the reactor building within 1 hour . With an average total activity in the building of 13.5 curies, as calculated above, the average concentration of xenon and krypton in the reactor-building air will be 1.35×10^{-2} $\mu c/cm^3$. If it is assumed that the fission-product gases leak out of one side of the building only, a person present for 1 hour near that side of the building would receive a maximum dose rate of 164 mr because of the decay of the fission-product gases. This dose is equal to that received by a person in the building for 1 hour.

If it is assumed that the fission-product gases escaped from the building at a lower rate than previously postulated, the integrated dose received by a person would be approximately the same as before, owing to the longer decay period for the fission-product gases before leaving the building.

Assuming that a fuel-element failure occurred once each year, the concentration of the fission-product gases leaving the building, averaged over a 1-year period, would be

$$A_{air} = \frac{1.35 \times 10^{-2} \ \mu c/cm^3 \times 1 \ hr}{8765.8 \ hr/yr} = 1.54 \times 10^{-6} \ \mu c/cm^2\text{-yr},$$

and this concentration would be very much decreased by fresh air dilution after traveling a short distance away from the building.

*Heating, Ventilating, and Air-Conditioning Guide, American Society of Heating and Ventilating Engineers, New York, 1953, p. 223.

5.1.12. Calculations for 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 after operation for infinite time at 250 kw. Results indicate that if the water loss in the core occurs immediately after the reactor has been shut down, the maximum temperature of the fuel, and consequently the aluminum cladding, is less than 150°C. This temperature is such that the pressure exerted by trapped air and fission product gases is less than 30 psi. This pressure produces a stress of about 660 psi, whereas, the yield stress for the aluminum cladding is 5000 psi at 150°C. 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 some will be removed by radiation.

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 fuel element considered in this calculation was equal to the temperature distribution in the hottest (B-ring) fuel element during steady-state operation of the reactor at 250 kw.

It was also assumed that the reactor had been operating for an infinite time at 250 kw with only 62 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 1700 watt-sec total in the B-ring 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:

$$\frac{q}{V} = 0.1\ p\ \frac{P}{V_f} \cos\left[0.78\ \frac{\pi}{L}\ (x - \frac{L}{2})\right]$$

$$\left\{\left[t + t_o + 10\right]^{-0.2} \quad -0.87\ \left[t + t_o + 2x10^7\right]^{-0.2} \quad -0.5\right\} \qquad (1)$$

where

p = Peak-to-average power density in the core = 1.72

P = Operating reactor power = 8.525×10^5 Btu/hr (250 kw)

V_f = Volume of the fuel in the core = 0.81 ft^3

L = Length of the fuel = 1.1666 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 (1) is a modification of the Untermeyer-Weill formula that matches the work of Stehn and Clancy*(1) to about 5×10^4 sec after shutdown. It is also conservatively assumed that all the energy produced by fission product decay in the element 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, tending to lower the fuel temperature. The air velocity through the channel can be determined by setting the frictional pressure loss equal to the buoyancy. Entrance and exit losses will be about 2-5% of the friction losses and have been ignored.

Stehan, 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.

$$\delta P \text{ (buoyancy)} = \delta P \text{ (friction)} \tag{2}$$

The term on the left is given by

$$\delta P \text{ (buoyancy)} = (\rho_0 - \rho_1)\frac{L}{2}, \tag{3}$$

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

$$\delta P \text{ (Friction)} = \frac{32\mu vL}{g\left[D_2^2 + D_1^2 - \dfrac{D_2^2 - D_1^2}{\ln (D_2/D_1)}\right]} \tag{4}$$

(2)

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 effective separation distance between the B-ring fuel 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, that is

$$b = \frac{1}{2} (0.01130 + .04930) = 0.0303 \text{ ft} \tag{5}$$

For the channel between the B-ring and the C-ring, the equation for the pressure balance (2) becomes

$$(\rho_0 - \rho_1) = 6.88 \times 10^{-5} \bar{\mu} \bar{v}, \tag{6}$$

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/ft^3).

The mass flow rate of air in the channel is

$$w = \bar{v} \bar{\rho} A_c, \tag{7}$$

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 (6) and (7), one obtains the mass flow rate

$$w = 405. (\rho_0 - \rho_1) \bar{\rho}/\bar{\mu} \tag{8}$$

Assuming that the average properties in equation (8) are the average of the properties at the entrance and exit, equation (8) becomes

$$w = 405. \frac{(\rho_0^2 - \rho_1^2)}{(\mu_0 + \mu_1)} \tag{9}$$

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

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

and (10)

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

where T is the temperature in °Rankine.

Using these expressions in equation (9) one obtains

$$w = \frac{3.24 \times 10^5 \ (T_1^2 - T_0^2)}{(T_1^2 \ T_0^2) \left[0.01135 + 0.30085 \times 10^{-4} \ (T_1 + T_0)\right]} \text{ lb/hr} \qquad (11)$$

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

$$h = 0.531 \frac{k_f}{L} \left[GrPr\right]^{0.25} \qquad (12)$$

where k_f is the thermal conductivity of the air film at temperature T_f (Btu/hr-ft-°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

$$k_a = (0.009 + 0.26 \times 10^{-4} T) \text{ Btu/hr-ft-°F}$$

and (13)

$$C_{pa} = 0.240 \text{ Btu/lb-°F}$$

with temperature, T, in °R. Using these values and those in equation (10) in equation (12) one obtains

$$h = 50.16 \left[\left\{0.0009 + 0.13 \times 10^{-4} \ (T_w + T_a)\right\}^3 \right.$$
$$\left\{\left[1.25 \times 10^{-2} \ (T_w + T_a)\right]^{-2} \left[0.5 \ (T_w + T_a)\right] \right. \qquad (14)$$
$$\left.\left.\left[0.01135 + 0.30085 \times 10^{-4} \ (T_w + T_a)\right]\right\}^{-1} (T_w - T_a)\right]^{0.25}$$

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 °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 1.1666-ft-long, 0.1225-ft-diameter cylinder of U-$ZrH_{1.0}$ with specific heat and thermal conductivity given by:

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

and (15)

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

where T is the local temperature in °R. The temperature drop in the clad was ignored because, at the time of peak temperature, it has been calculated to be insignificant (1-5°C). 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 Figure 20 the maximum fuel temperature calculated is plotted as a function of time after the loss of cooling water. The loss of water in the core was assumed to take place immediately at shutdown, and the maximum initial fuel temperature is 113°C.

To determine the pressure exerted on the cladding by released hydrogen, fission products, and air trapped in the fuel can, the conservative assumption will be made that the entire system is at the peak fuel temperature, i.e., about 150°C.

The total number of fission product nuclei released to the gap between the fuel and clad was determined from Blomeke and Todd,[3] and the results of the experiment described in section 5.1.11. The total quantity of Br, I, Kr, and Xe released to the gap in the B-ring fuel element after infinite operation at 250 kw will be

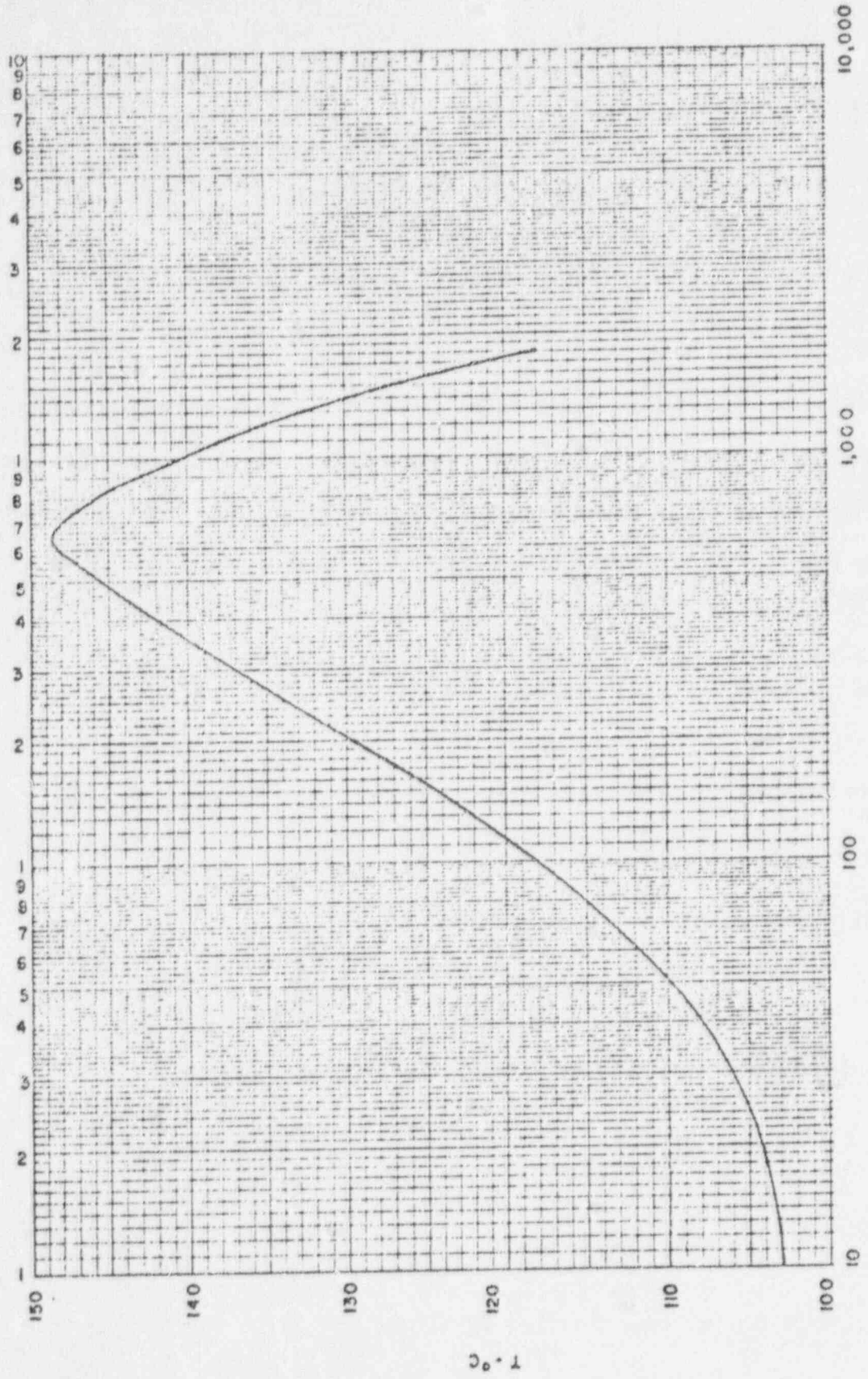

Fig. 20 --Maximum fuel temperature versus time after complete water loss (250 kW operating power)

$$N_i = 0.028 \times 6.886 \times 10^{20} = 1.933 \times 10^{19} \text{ atoms,} \quad (16)$$

The number of gram-atoms in the gap is

$$n_{fp} = \frac{1.933 \times 10^{19}}{6.02 \times 10^{23}} = 3.216 \times 10^{-5} \text{ gram-atoms,} \quad (17)$$

The partial pressure exerted by the fission products gases is

$$P_{fp} = n_{fp} \frac{RT}{V} \quad (18)$$

where initially, the volume, V, is taken as the 1/8-inch space between the end of the fuel and the 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

$$V_0 = \pi r^2 h = \pi(1.80)^2\ 0.317 \text{ cm}^3 = 3.23 \text{ cm}^3 \quad (19)$$

where the radius of the fuel material, r, is about 1.80 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

$$P_{fp} = \frac{3.216 \times 10^{-5}}{3.23} RT = 0.997 \times 10^{-5}\ RT \quad (20)$$

The partial pressure of the air in the fuel element is

$$P_{air} = \frac{RT}{22.4 \times 10^{-3}} = 4.46 \times 10^{-5}\ RT, \quad (21)$$

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

$$P_r = \left(1 + \frac{P_{fp}}{P_{air}}\right) P_{air} = \left(1 + \frac{0.007 \times 10^{-5}}{4.46 \times 10^{-5}}\right) P_{air} = 1.223 P_{air} \quad (22)$$

Also $P_{air} = 14.7 \frac{T}{273} \frac{V_0}{V}$ (23)

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

The equilibrium hydrogen pressure over the U-$ZrH_{1.0}$ fuel material is simply a function of the fuel temperature. At a temperature of 150°C, the hydrogen pressure is negligible (see Figure 21).

The total gas pressure at the maximum fuel temperature of 150°C is

$$P = P_H + P_r = 0 + 1.223\ (14.7)\ \left(\frac{150+273}{273}\right)\frac{3.23}{V} = \frac{90.0}{V}\ \text{psi}. \qquad (24)$$

Ignoring the negligible increase in volume caused by the expansion of the clad the total pressure then, is

$$P = \frac{90.0}{V} = \frac{90.0}{V_0} = \frac{90.0}{3.23} = 27.8\ \text{psi} \qquad (25)$$

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

$$S = Pr/t, \qquad (26)$$

where r is the radius of the fuel element can (1.80 cm) and t is the wall thickness (0.076 cm) or

$$S = 23.7P. \qquad (27)$$

The stress in the cladding is

$$S = 23.7(27.8) = 660\ \text{psi} \qquad (28)$$

From the Aluminum Company of American handbook [4] the yield stress for type 1100-0 aluminum at 150°C is > 5000 psi.

Consequently, it is concluded that, subsequent to the loss of cooling water after infinite operation at 250-kw, the release of hydrogen from the fuel and the expansion of air and fission product gases in the space between fuel and graphite end pieces will not result in the rupture of the fuel element cladding.

5.1.13. Argon Activation in Reactor Water

The argon activity in the reactor pool water results from the argon dissolved in water. In the calculation to determine the amount of argon

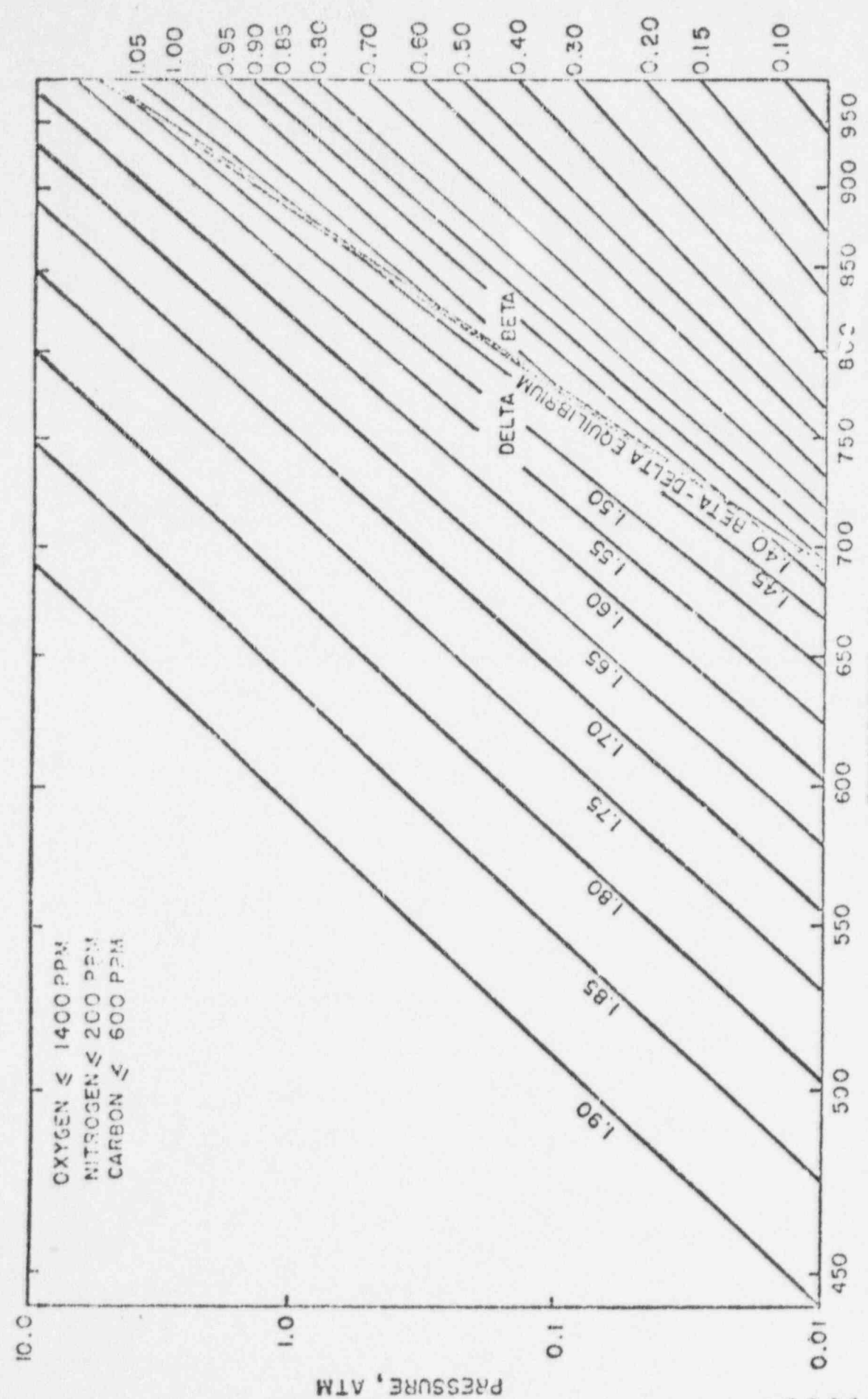

Fig. 21 --Dissociation pressure isochores of zirconium hydride (expressed as H/Zr atom ratios)

dissolved in the pool water, the assumption was made that argon follows Henry's law. If the water temperature is taken to be 70°F, the corresponding water vapor pressure is 26 mm Hg. The partial pressure of air is then 760 - 26 = 734 mm Hg. The argon content of air is 0.94% by volume; hence, the partial pressure of argon is $734 \times (9.4 \times 10^{-3}) = 7$ mm Hg.

The saturated concentration of argon in water, according to Henry's law, is

$$X = \frac{P}{K}, \tag{29}$$

where

X = Mole fraction of argon in water

P = Partial pressure of argon above water

K = Henry's constant, 2.84×10^{7} at 70°F.

Thus, $X = 2.46 \times 10^{-7}$ mole A^{40} per mole of $(H_2) + A^{40})$, or $X = 1.367 \times 10^{-8}$ mole A^{40} per 1 $cm^3 H_2O$.

Argon-41 production, assuming saturated conditions and an irradiation time t, is given by

$$N_1 = \frac{\bar{\Phi}_n \Sigma_a (1 - e^{-\lambda t})}{\lambda} \tag{30}$$

where

N_1 = Atomic density of A^{41}, atoms A^{41}/cm^3

$\bar{\Phi}_n$ = Average neutron flux, n/cm^2-sec

Σ_a = Macroscopic absorption cross section of A^{40}, cm^{-1}

λ = Decay constant for A^{41}, sec^{-1}

A corresponsing activity for small values of λt is

$$A_1 = \bar{\Phi}_n \lambda t \Sigma_a. \tag{31}$$

The average thermal flux in the reactor core is estimated to be 4×10^{12} n/cm^2-sec. The water circulates in the core mainly through natural convection and is estimated to change completely in 4 seconds. Since the core holds 2.4×10^4 cm^3 of water, the rate of flow of water through the core is 0.6×10^4 cm^3/sec. Substituting appropriate values for Σ_a and taking irradiation time as 4 seconds results in

$$A_1 = 7.4 \text{ disintegrations/cm}^3\text{-sec} \tag{32}$$

The total activity from the core is thus

$$Q_0 = 1.2\mu c/sec. \tag{33}$$

The travel time of argon-41 from the core to the water surface, a distance of 16 feet, was measured as 42 seconds. Applying the decay law, argon-41 activity reaching the surface is

$$Q = 1.9\mu c/sec. \tag{34}$$

Under saturated, steady-state conditions, the maximum rate at which argon-41 can escape from the water surface will be 1.19μc/sec, and an equivalent amount of argon-40 will dissolve in water in place of argon-41. In reality, a much smaller fraction of argon-41 will escape from the water. At increased water temperature, the partial pressure of water vapor will increase and the amount of dissolved argon-40 will decrease.

The radioactive argon escaping from the reactor pool will dissipate in the air of the reactor room. The volume of air, estimated to be 29,000 ft^3, is recirculated through an air-conditioning system. Because of the high air-circulation rate, it may be assumed that argon-41 is evenly distributed throughout the entire room. The concentration of argon-41 in the air may be obtained by considering the material balance of argon-41 (in μc/sec).

Letting Q be the production rate, P the exhaust rate and D the decay rate, the accumulation rate will be

$$A = Q - P - D \tag{35}$$

If the reactor is operated for a long time, equilibrium conditions in the air may be assumed. The accumulation rate, then, is equal to zero. Hence,

$$Q - D = P.$$

From Equation (30), $Q = 1.19 \mu c/sec$.

The average decay rate may be expressed as

$$S = Q - \frac{1}{\theta} \int_0^{\theta} Q_e^{-\lambda t} dt \mu c/sec, \tag{36}$$

where θ is the time in which decay occurs and is the average time or residence of an atom of argon-41 in the reactor room before it is vented to the atmosphere, i.e.,

$$\theta = \frac{29{,}000 \text{ ft}^3}{25 \text{ ft}^3/\text{sec}} = 1.16 \times 10^3 \text{ sec} \tag{37}$$

Integrating Equation (36) and substituting in Equation (37) yields

$$P = Q \frac{1 - e^{-\lambda\theta}}{\lambda\theta} \ \mu c/sec \tag{39}$$

Substituting for λ and θ yields

$$P = 1.12 \mu c/sec.$$

Since equilibrium conditions are assumed, the same concentration A^{41} will exist in the air of the reactor room as in the exhausted air. Concentration in the exhausted air is

$$1.12/(25)(2.832 \times 10^4) = 1.59 \times 10^{-6} \ \mu c/cm^3.$$

This is less than the maximum permissible occupational tolerance level of

2×10^{-6} $\mu c/cm^3$, which is recommended for a 40-hour week by the U.S. Federal Register. Moreover, it is a upper limit since actually a much smaller amount of argon-41 than 1.19 μc/sec will escape into the air.

5.2. Accidents Not Associated with Operation of the Reactor

5.2.1. Mechanical Damage to the Reactor

It is conceivable that a heavy weight, such as a lead transfer cask, could be dropped on the reactor core from above and could smash the core in such a way as to change the fuel-to-water ratio. The designed fuel-to-water ratio in the core was selected because this ratio was calculated to give very nearly the minimum critical mass. Consequently, smashing the core is likely to decrease the reactivity and at worst cannot increase it appreciably.

5.2.2. Failure of Utilities

The reactor control system is fail-safe in the event of power failure; i.e., loss of power will de-energize the magnets and release the control rods. Loss of water supply or other utilities will have no effect, as the reactor will not depend upon them.

5.2.3. Fire

The building is constructed almost entirely of fireproof materials. The load-bearing walls and the floors are of brick and reinforced concrete. Very little inflammable material is stored in the building. Carbon dioxide fire extinguishers are located in laboratories and at other strategic locations. Aid from the Austin Fire Department is available in less than five minutes. The possibility of fire does not contribute significantly to any radiological hazard.

5.3. Design Basis Accident

The design basis accident for this reactor system is the occurrence of a defect in the cladding of a fuel element either prior to or simultaneously with a 1.5% δ k/k transient.

It is concluded that a cladding failure or even the failure of the cladding of several fuel elements would not constitute an undue hazard to the operating crew or the general public.

In the event that a defect exists in the cladding of a fuel element at the time of pulsed operation, there may be some release of fission products to the atmosphere of the reactor room. Operating data have been obtained from the General Atomic TRIGA reactor to indicate the magnitude of such release.

During a higher transient than the 1.5% δ k/k mentioned above, namely a 2.9% δ k/k (3.90-dollar) transient conducted as part of an extensive program of reactor transient experiments at General Atomic, a cladding failure occurred which resulted in the release of a minor quantity of fission products.

The following summarizes the measurement made:

1. The activity in the cooling water tank reached a maximum of 0.2 μc/ml. It decayed very rapidly and was measured as 5×10^{-5} μc/ml 24 hr after the cladding failure.
2. The activity in the air of the reactor room reached about ten times the MPC for fission products and decayed very rapidly. Experiments were resumed 2 hr after the activity release. The maximum exposure to operating personnel was less than 1 mr.
3. The noble gases were not collected on the filter samples used, but it may be inferred from the nature of the particulates observed that only the noble-gas fission products escape

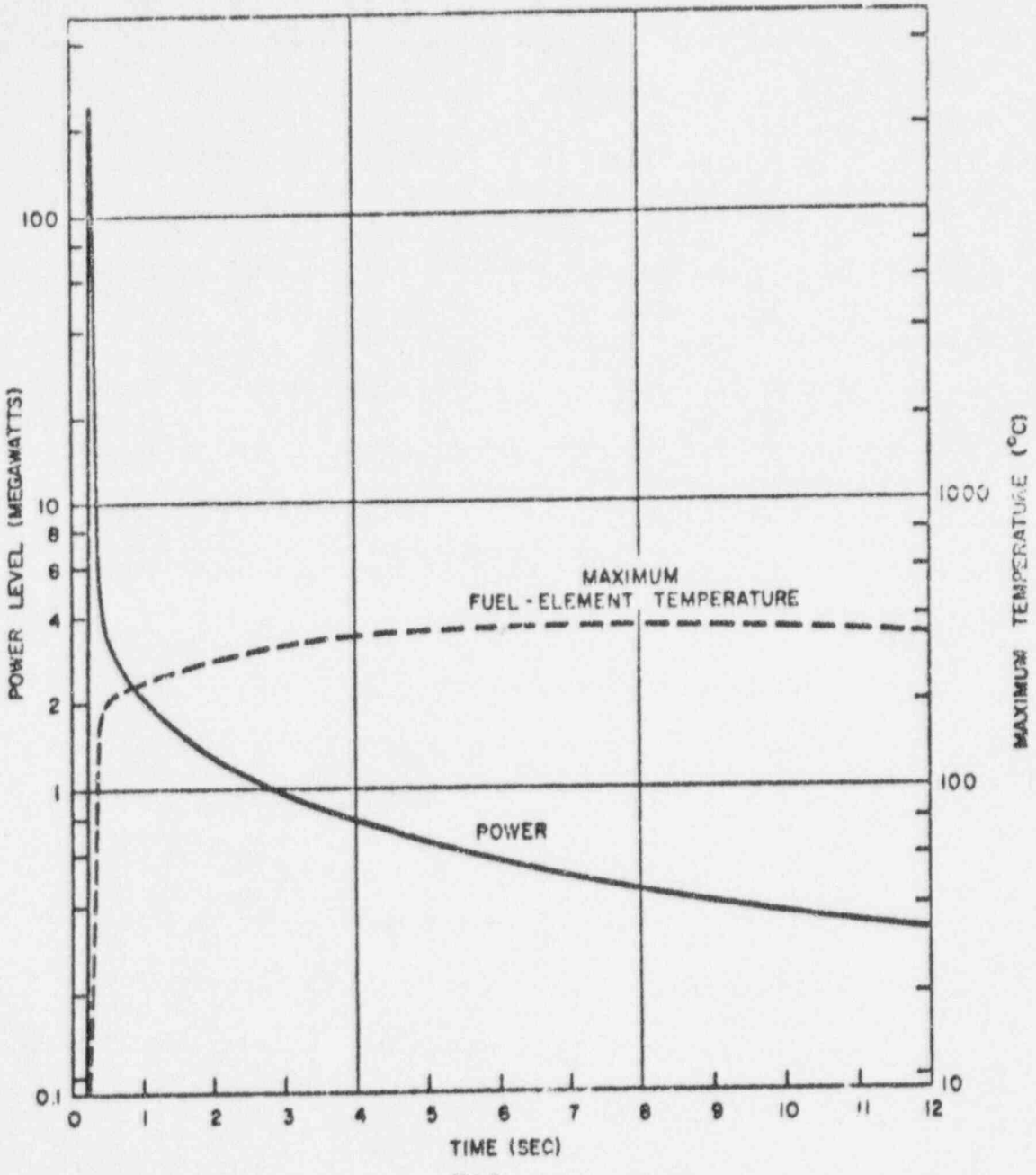

FIG. 22

TWO-DOLLAR REACTIVITY TRANSIENT.

from the TRIGA pool in significant proportions when a fuel cladding fails.

The analysis in Section 3.1, "Reactor Laboratory," shows that no significant hazard to the public results from such an accident.

5.4. Emergency Procedures

1. Radioactivity

The basic philosophy in case of emergency is, first, to remove all persons to safe locations and, secondly, for qualified individuals to return cautiously for evaluation of the situation.

Possible hazards are:
(1) Isotope spill
(2) Radiolytic gas release

Remote hazards are:
(1) Uncontrolled power excursion
(2) Chemical reactions

The amount of emergency action taken shall be based upon the most pessimistic view of the potential hazard. The Supervisor or his designated agent shall be in complete command of all persons in the Laboratory and all actions taken in the Laboratory during an emergency. During the emergency he shall follow orders from the Laboratory Director or from the Chairman of the Radiation Safety Committee only if in his opinion such orders lead to a more conservative or more cautious approach to the correction of the overall situation.

There shall be three grades (or classifications) of emergency situations ranging from the most minor to the maximum credible. The required action in each grade shall include all pertinent provisions of all lower grades.

a. Grade A:

This includes the most minor situations, such as a small spill of a non-volatile low-level radioactive liquid. Whoever first notices the situation is responsible for alerting all

others in the Laboratory. The reactor operator shall modify his plans for operation as he may deem necessary and shall be prepared to scram the reactor until the situation is remedied. The Supervisor is responsible for the evaluation and correction of the situation and for supervising all necessary monitoring, remote handling, use of protective clothing, and other health and safety measures.

b. Grade B:

Up to 100 mr/hr — Known cause.

This limiting level has been established as being near the upper limits permitted by the Nuclear Regulatory Commission and Texas Radiation Council for restricted areas if a person were to remain in the area for a period of about twelve hours per calendar quarter. This grade has been established for cautionary purposes. Occupancy of the Laboratory is restricted to employees with film badges.

c. Grade C:

Over 100 mr/hr, or
Over 5000 cpm on CAM } Unknown cause.

All persons shall immediately evacuate the Laboratory into TAY 125, picking up all readily available portable survey equipment and closing the doors on their way out. If there is any damage to the walls or roof of the Laboratory, the air-conditioning blower in room 131B shall be turned off (switch on right upon entering 131B If the situation warrants, the Supervisor or his designated agent shall

evacuate all adjacent rooms (104, 125, 129, 131B, 133, 135, 204). One person shall warn others not to enter the hall near the Laboratory from the north. Another will similarly guard at the corridor intersection adjacent to room 135. The Supervisor or his agent will station himself, with monitoring equipment, in the driveway south of the Laboratory. If radiation levels in the corridors adjacent to the Laboratory approach the legal maxima for unrestricted areas, the evacuation of all persons shall be continued as far as necessary to ensure compliance with these regulations. The following persons shall be notified:

(1) Laboratory Supervisor:

Joseph A. Burack CTX 5136

(2) Laboratory Director

Dale E. Klein CTX 5136 (459-0075)

(3) Radiation Safety Officer:

Bill Bryant CTX 4601 and 3511 (452-6689)

(4) Chief, Traffic & Security Officer (Chief, University Police)

Donald R. Cannon CTX 4441 PAX 1031 (836-4089)

(5) Radiation Control Division of State Health Department:

Martin C. Wukasch 454-3781, ext. 241 (465-5437)

(6) Laboratory Technician:

Bob B. Tomlin CTX 5136 (321-3560)

At his discretion, the Chief Traffic and Security Officer may request assistance from the City of Austin Police and Fire Departments.

If Laboratory personnel are unable to obtain sufficient survey equipment and/or protective clothing during the evacuation, such items shall be requested from the Radiation Safety Officer or the Radiation Control Division. At least once each calendar quarter the emergency procedures as outlined above shall be reviewed by the employees involved.

2. Fire and Other Hazards

In case of fire or other hazards not directly associated with radiation, the general procedure will be to scram the reactor and notify the Fire Marshal (CTX 3511) or the Chief, University Police (CTX 3511 and 4441). The Supervisor or his designated agent shall have the same authority and responsibility as he has in the case of a radiation hazard. He shall decide the type and extent of action to be taken based upon the location and characteristics of the hazard.

At least two carbon dioxide fire extinguishers shall be maintained in readiness in the Laboratory area.