Oregon State University TRIGA Reactor

Annual Report

To satisfy the requirements of:

A. U.S. Nuclear Regulatory Commision, License No. R-106 (Docket No. 50-243), Section 6.7(e) of the Technical Specifications, for the reporting period July 1, 1984 through June 30, 1985.

B. U.S. Department of Energy, University Reactor Fuel Assistance Contract No. DE-AC06-76-ER01953, for the reporting period July 1, 1984 through June 30, 1985.

C. Oregon Department of Energy, ODOE Rule No. 30-010, for the reporting period July 1, 1984 through June 30, 1985.

Written by:

T. V. Anderson, Reactor Supervisor
B. Dodd, Assistant Reactor Administrator
D. S. Pratt, Radiation Specialist
A. G. Johnson, Assistant Director, Radiation Center

Submitted by:

C. V. Smith, Jr.
Reactor Administrator & Director, Radiation Center

Radiation Center
Oregon State University
Corvallis, Oregon 97331
Telephone: 503-754-2341

September 15, 1985

IE25
1/1

Table of Contents

PART 1 -- INTRODUCTION AND SUMMARY

PART 2 -- GENERAL INFORMATION

PART 3 -- OSTR OPERATIONAL DATA

PART 4 -- OSTR UTILIZATION DATA

PART 5 -- OSTR ENVIRONMENTAL AND RADIATION PROTECTION DATA

List of Tables

List of Figures

PART 1

INTRODUCTION AND SUMMARY

PART 1

INTRODUCTION AND SUMMARY

A. INTRODUCTION TO THE OREGON STATE TRIGA REACTOR (OSTR) ANNUAL REPORT

The reporting period for this annual report is from July 1, 1984 through June 30, 1985. Because this report satisfies the requirements of more than one organization, all of the information included may not be of equal concern to all recipients, but a comprehensive table of contents has been included to aid in selecting and locating information of specific interest to the reader.

This report does not include detailed data for the OSTR's original (20% enriched) standard TRIGA fuel core; however, a summary of the important operational statistics for that core's use period (1967-76) is included in Table 3.2. For more information on the original fuel, the reader is referred to the 1976-77 OSTR annual report, which contains considerably more details on the history of the original (20% enriched) core.

This year's report uses statistics which reflect the operating history of the (70% enriched) FLIP fueled core. This core has been in use at the OSTR since August of 1976 and remains in use through the present time. For reporting purposes, the FLIP core has been established as the basis for operational history in all OSTR annual reports beginning with the one issued for the 1976-77 reporting period.

B. SUMMARY OF OSTR USE DURING THE REPORTING PERIOD

During the year July 1, 1984 through June 30, 1985:

1. The OSTR generated 39.4 MWD of energy.
2. The OSTR consumed 49.5 grams of ^{235}U.
3. The OSTR was pulsed 72 times.
4. No fuel elements were added to the core.

5. Eighteen academic courses were accommodated, which included seven OSU courses in nuclear engineering, seven OSU courses in chemistry, one OSU course in radiation protection, one OSU thesis project in physics, one OSU thesis project in geology, and one OSU thesis project in oceanography. Reactor use time for these programs totaled 81 hours.
6. One person, sponsored by a U.S. Department of Energy Grant administered by the University of Virginia, is being trained in reactor operations. A total of 14 hours of reactor time was used exclusively for this training. This time has been listed under teaching because of the nature of the training.
7. Forty-eight funded research projects (a 23% increase over the previous reporting period) and four unfunded research (teaching) projects were accommodated by the reactor. Reactor use time for funded research programs totaled 713 hours (a 27% increase over the previous reporting period), and totaled 18 hours for unfunded research (teaching) projects.
8. During the reporting period, 769 visitors (a 46% increase over the previous reporting period) viewed the reactor during scheduled tours and other university functions. Reactor use time for visitor open-house (demonstration) events totaled 11 hours, while the remainder of the visitors viewed the reactor during times when it was being operated for regularly scheduled research and teaching.
9. The reactor was in use an average of 36 hours during a typical 45-hour work week. Hence, the reactor was used approximately 81% of the available time.

C. SUMMARY OF RADIATION CENTER LABORATORY USE DURING THE REPORTING PERIOD

The Radiation Center laboratories provided much support for reactor related research and teaching, in addition to supporting numerous other academic projects not directly associated with the OSTR. In the former category, the Center's diversified radiation counting instrumentation accommodated work ranging from introductory undergraduate counting experiments through sophisticated analysis using gamma spectroscopy. Likewise, the many laboratory facilities supported a wide range of nuclear and radiochemical procedures for materials previously irradiated in the reactor.

Ranking high among the other research and teaching which the Center accommodated, were irradiations using the 300 kVp X-ray facility and irradiations using the Cobalt-60 gamma irradiator. Thirty different irradiations (a 58% increase over the previous reporting period) were conducted using the Cobalt-60 irradiator, and included such samples as fish eggs and fish cells, wheat seeds, blackberry plants, and other horticultural specimens.

D. SUMMARY OF OSTR ENVIRONMENTAL AND RADIATION PROTECTION DATA

1. Liquid Waste Data (See Table 5.1)	Year July 1, 1984 Through June 30, 1985
a. Total estimated quantity of radioactivity released (to the sanitary sewer)(in curies)(1)	6.50×10^{-5}
b. Detectable radionuclides in the liquid waste	^{3}H, ^{60}Co, ^{152}Eu
c. Estimated average concentration of released radioactive material at the point of release (in microcuries per cubic centimeter)	7.26×10^{-6}
d. Percent of applicable MPC for released liquid radioactive material at the point of release (%)	0.25%(2) 0.0075%(3)
e. Total volume of liquid effluent released, including diluent (in gallons)(4)	2366

---

(1) The OSU operational policy is to subtract only detector background from our water analysis data and not background radioactivity in the Corvallis city water.

(2) Based on values listed in 10 CFR 20, Appendix B, Table 2, Column 2.

(3) Based on values listed in 10 CFR 20, Appendix B, Table 1, Column 2, applicable to sewer disposal.

(4) Total volume of effluent plus diluent does not take into consideration the additional mixing with approximately 95,000 to 115,000 gallons per year of liquids and sewage normally discharged by the Radiation Center complex into the same sanitary sewer system.

2. Gaseous Waste Data (See Table 5.2)	Year July 1, 1984 Through June 30, 1985
a. Total estimated quantity of radioactivity released (in curies)(1)	9.41
b. Detectable radionuclides in the gaseous waste(2)	^{41}Ar ($T_{1/2}$= 1.83 hr)
c. Estimated average atmospheric diluted concentration of argon-41 at the point of release (in microcuries per cubic centimeter)	5.70×10^{-8}
d. Percent of applicable MPC for diluted concentration of argon-41 at the point of release (%)	1.43
e. Total estimated release of radioactivity in particulate form with half-lives greater than 8 days (in curies)(3)	none

3. Solid Waste Data (See Table 5.3)	Year July 1, 1984 Through June 30, 1985
a. Total amount of solid waste packaged and disposed of (in cubic feet)	11.0
b. Detectable radionuclides in the solid waste	^{24}Na, ^{65}Zn, ^{51}Cr, ^{75}Se, ^{58}Co, ^{124}Sb ^{59}Fe, ^{140}La ^{60}Co, ^{54}Mn
c. Total radioactivity in the solid waste (in curies)	9.46×10^{-4}

---

(1) The decrease in the total argon-41 released during the current reporting period is due to several modifications designed to reduce the argon-41 effluent to as-low-as-reasonably-achievable levels. The total argon-41 output for the 1984-85 reporting period is approximately 74.3% of last year's value (a drop of 25.7%).

(2) Routine gamma spectroscopy analysis of the gaseous radioactivity in the stack discharge indicated that it was virtually all argon-41.

(3) Evaluation of the detectable particulate radioactivity in the stack discharge confirmed its origin as naturally occurring radon daughter products, predominantly lead-214 and bismuth-214, which are not associated with reactor operations.

4. Radiation Exposure Received by Facility Personnel and Visitors (in mrem)(See Table 5.4)	Year July 1, 1984 Through June 30, 1985
a. Facility operating personnel (mrem)	
1) Average whole body	64.00
2) Average extremities	111.00
3) Maximum whole body	380.00
4) Maximum extremities	1090.00
b. Facility research personnel (mrem)	
1) Average whole body	1.00
2) Average extremities	2.00
3) Maximum whole body	25.00
4) Maximum extremities	70.00
c. Visitors (mrem)	
1) Average whole body	<1.00
2) Maximum whole body	10.00

5. Number of Area and Offsite Environmental Monitoring Samples Evaluated	Year July 1, 1984 Through June 30, 1985
a. Area composite dosimeters inside the TRIGA facility:	
1) Beta-gamma-sensitive component in the dosimeter	96
2) Neutron-sensitive component in the dosimeter	32
b. Vendor(1)- supplied TLD monitors on the reactor facility fence	72
c. OSU TLD monitors on the reactor facility fence	108
d. Readings from integrating ionization chambers on the reactor facility fence	216
e. μR/hr measurements around the perimeter of the reactor facility fence	108
f. Offsite environmental soil samples	16
g. Offsite environmental water samples	12

(1) Radiation Detection Company, Sunnyvale, California, was the vendor.

5. Number of Area and Offsite Environmental Monitoring Samples Evaluated (Continued)	Year July 1, 1984 Through June 30, 1985
h. Offsite environmental vegetation samples	56
i. Offsite vendor(1)-supplied TLD monitors	88
j. Offsite OSU TLD monitors	231
k. Readings from offsite integrating ionization chambers	264
l. µR/hr measurements at the offsite airborne gamma monitoring stations	228

(1) Radiation Detection Company, Sunnyvale, California, was the vendor.

PART 2

GENERAL INFORMATION

PART 2

GENERAL INFORMATION

A. RADIATION CENTER

The Oregon State TRIGA Reactor (OSTR) is located in the Radiation Center at Oregon State University. The Radiation Center was designed and established to:

1. Accommodate all types of internal and off-campus instructional programs involving nuclear science, nuclear engineering, radiation protection and related areas.
2. Support research, deve'opment and service programs involving nuclear science and engineering.
3. Provide a place especially designed for the use and handling of radioisotopes and other sources of ionizing radiation.
4. Provide a variety of ionizing radiation sources for fast and thermal neutrons, and X-ray and gamma radiation.

Construction of the Radiation Center was divided into two phases. The first phase was completed in June of 1964 and consisted of 32,397 square feet of office and laboratory space. The second phase was completed in March of 1967 and consisted of a nuclear research reactor housed in a 9,956 square foot building adjacent to the existing Radiation Center. In 1975 temporary space totaling 1,600 square feet was added for interim accommodation of the rapidly expanding nuclear engineering program. In 1977 additional temporary space equaling another 1,600 square feet was added. The Radiation Center complex at present totals 45,553 square feet.

The Radiation Center currently incorporates a variety of laboratories and equipment. In particular, these facilities are designed to accommodate:

1. Instructional programs in nuclear engineering, radiation biology, radiation protection, and nuclear and radiation chemistry.
2. Instrumental and radiochemical neutron activation analysis.
3. High-level and low-level radiochemical research.
4. Neutron radiography and neutron diffraction.
5. Irradiation experiments involving X-rays, gamma-rays, and neutrons.

6. Measurement of various types of ionizing radiation.
7. Calibration of a wide variety of nuclear instrumentation used in radiation protection.
8. Bioassay procedures for various radionuclides.
9. Radioactive waste management, packaging of radioactive materials for transportation and emergency response to accidents involving radionuclides.
10. Consultation in the application of radioisotopes and other radiation research.
11. Exploratory programs involving the novel uses of radioisotopes and other radiation sources.

B. FACULTY MEMBERS RESIDING AT THE RADIATION CENTER

*Smith, Clifford V., Jr.
Director, OSU Radiation Center
Director, OSU Institute of Nuclear Science and Engineering
OSTR Reactor Administrator
Head, Department of Nuclear Engineering

Anghaie, Samim
Assistant Professor of Nuclear Engineering

*Binney, Stephen E.
Associate Professor of Nuclear Engineering
Chairman, OSTR Reactor Operations Committee

Daniels, Malcolm
Professor of Chemistry

*Dodd, Brian
OSTR Assistant Reactor Administrator
Associate Professor of Nuclear Engineering

*Johnson, Arthur G.
Assistant Director for Radiation Protection and Regulatory Affairs, OSU Radiation Center
Senior Health Physicist, OSU Radiation Center
Professor of Nuclear Engineering and Radiation Health

*Loveland, Walter D.
Professor of Chemistry

Luo, Shyr-Tung
Instructor of Nuclear Engineering

---

*Reactor users for research and/or teaching.

Mac Vicar, Robert
President Emeritus, OSU

*Pastorek, Christine
Instructor of Chemistry

Popovich, Milosh
Vice President Emeritus, OSU

*Ringle, John C.
Professor of Nuclear Engineering
Associate Dean of the Graduate School, OSU

*Robinson, Alan H.
Professor of Nuclear Engineering

*Schmitt, Roman A.
Professor of Chemistry

Sugihara, Thomas T.
Dean of Science, OSU

Wang, Chih H.
Professor Emeritus, OSU

*Woods, W. Kelley
Professor of Nuclear Engineering

C. RESEARCH PERSONNEL USING RADIATION CENTER FACILITIES

1. Post-Doctorate Research Associates

Name	Field	Research Program Director
*Collier, Robert W.	Oceanography	J. Dymond
*Dudek, Nan	Oceanography	P. A. Wheeler
*Hughes, Scott S.	Chemistry	R. A. Schmitt
*LaTouche, Y. D.	General Science	D. L. Willis

2. Scientists and Trainees

Name	Field (Affiliation)	Advisor or Research Program Director
*Ahmed, Kamal	Reactor Operations Training (IAEA, Bangladesh Atomic Energy Commission)	B. Dodd

*Reactor users for research and/or teaching

Name	Field (Affiliation)	Advisor or Research Program Director
*Alamgir, Mohammad	Neutron Activation Analysis (Bangladesh Atomic Energy Commission)	R. A. Schmitt
Al Hassan, Layla	Nuclear Chemistry (Jeddah College for Girls, Saudi, Arabia)	M. Daniels
*Barry, Gene	Neutron Activation Analysis (University of Oregon)	R. A. Schmitt
*Brandon, Alan D.	Neutron Activation Analysis (University of Oregon)	R. A. Schmitt
*Brodman, Bruce	High Speed Neutron Radiography (U.S. Department of Defense)	A. H. Robinson
*Busamongkol, Yuthapong	Reactor Operations (IAEA Fellowship, Thailand)	B. Dodd
*Clingman, William	Neutron Activation Analysis (University of Oregon)	R. A. Schmitt
*Collier, Patricia	Neutron Activation Analysis (Oceanography/OSU)	R. W. Collier
*Conard, Roberta A.	Neutron Activation Analysis (Oceanography/OSU)	R. W. Collier
*Dawydiak, Orysia	Delayed Neutron Counting (General Science, OSU)	D. L. Willis
*Espenan, Gregory	Neutron Activation Analysis (Louisiana State University)	R. A. Schmitt
*Fritz, Raimund	High Speed Neutron Radiography (Federal Republic of Germany)	A. H. Robinson
Gallagher, Jennifer	Autoradiography (Vet. Medicine/OSU)	C. H. Wang
*Geist, Dennis	Neutron Activation Analysis (University of Oregon)	R. A. Schmitt

* Reactor users for research and/or teaching

Name	Field (Affiliation)	Advisor or Research Program Director
*Goles, Gordon	Neutron Activation Analysis (University of Oregon)	R. A. Schmitt
Golightly, Vivian	Autoradiography (Chemistry/OSU)	C. H. Wang
*Haines, Sara	Neutron Activation Analysis (University of Oregon)	R. A. Schmitt
*Hu, Ziuzhen	Neutron Activation Analysis (People's Republic of China)	R. A. Schmitt
*Knaus, Ronald	Neutron Activation Analysis (Louisiana State University)	R. A. Schmitt
*Laul, J. C.	Neutron Activation Analysis (Battelle Pacific Northwest Lab.)	R. A. Schmitt
Liu, Xueyu	Radiation Biology (Jiangsu Academy of Agricultural Science, People's Republic of China)	C. H. Wang
*Murali, Ahobila Vajjula	Neutron Activation Analysis (NASA/JSC Houston)	R. A. Schmitt
*Pham, Cecilia	Reactor Operator Trainee	B. Dodd
*Radosevich, Stephan	Neutron Activation Analysis (University of Oregon)	R. A. Schmitt
*Redecker, Klaus H.	High Speed Neutron Radiography (Federal Republic of Germany)	A. H. Robinson
*Reich, David	Neutron Activation Analysis (University of Oregon)	R. A. Schmitt
*Ritchie, Beatrice	Neutron Activation Analysis (University of Oregon)	R. A. Schmitt
*Schmitt, Joe	Neutron Activation Analysis (Battelle Pacific Northwest Lab.)	R. A. Schmitt
*Smith, Monty	Neutron Activation Analysis (Battelle Pacific Northwest Lab.)	R. A. Schmitt

* Reactor users for research and/or teaching.

Name	Field (Affiliation)	Advisor or Research Program Director
*Sonnenthal, Eric	Neutron Activation Analysis (University of Oregon)	R. A. Schmitt
*Taffet, Michael	Neutron Activation Analysis (University of Oregon)	R. A. Schmitt
*Ungerer, Carl	Neutron Activation Analysis (Oceanography/OSU)	J. Suess
*Vincenzi, Ed	Neutron Activation Analysis (University of Oregon)	R. A. Schmitt
*Vogel, Allan	Neutron Activation Analysis (Portland State University)	R. A. Schmitt
Wang, Guoli	Neutron Activation Analysis (People's Republic of China)	R. A. Schmitt
Wang, Zhifen	Neutron Activation Analysis (People's Republic of China)	R. A. Schmitt

3. Graduate Students

Name	Degree Program	Field	Advisor
*Anellis, Lawrence	MS	Rad Health	A. G. Johnson
Bostick, Kent	MS	Nuclear Engr	S. E. Binney
Bukar, Kyari A.	MS	Nuclear Engr	S. E. Binney
Butler, Hugh	PhD	Crop Science	A. P. Appleby
*Carter, Stephen	MS	Geology	R. D. Lawrence
*Casey, Coreen	PhD	Chemistry	W. D. Loveland
Cheng, Beato	PhD	Nuclear Engr	S. E. Binney
*Chidester, Stephen	MS	Nuclear Engr	A. H. Robinson
Cho, Byung-OH	MS	Nuclear Engr	A. H. Robinson
*Coe, Douglas H.	MS	Nuclear Engr	A. H. Robinson
*Greek, Kevin	MS	Nuclear Engr	A. H. Robinson
Ha, Jae-Joo	MS	Nuclear Engr	S. Anghaie
*Harris, Richard	MS	Nuclear Engr	S. E. Binney
Humphries, Larry	MS	Nuclear Engr	B. Dodd
Johnson, Susan	MS	Nuclear Engr	S. E. Binney
Kaya, Sadi	MS	Nuclear Engr	S. Anghaie

*Reactor users for research and/or teaching

Name	Program	Field	Advisor
*Keizer, Philip	MS	Rad Health	D. L. Willis
Lee, Chungchan	PhD	Nuclear Engr	A. H. Robinson
Lopez, Ricardo	PhD	Nuclear Engr	S. E. Binney
Luo, Shyr-tung	PhD	Nuclear Engr	S. E. Binney
*Newell, Daniel	MS	Nuclear Engr	A. H. Robinson
*Pyle, Douglas	MS	Geology	J. Dasch
Saamin, Shaharudin	PhD	Horticulture	M. Thompson
Salahuddin, Kamran	MS	Nuclear Engr	J. C. Ringle
Saleh, Hassan	PhD	Nuclear Engr	S. E. Binney
Samuels, Jeffrey	PhD	Nuclear Engr	S. Anghaie
Savidge, William	MS	Bio. Ocean	D. A. Carey
Sui, Yueh-Chun	MS	Nuclear Engr	A. H. Robinson
*Tenbrook, Warren	MS	General Sci	S. E. Binney
*Verplanck, Philip	MS	Geology	R. A. Schmitt
*Von Breymann, Marta	PhD	Oceanography	R. W. Collier
*Walker, Robert	PhD	Geology	R. A. Schmitt
*Walsh, Ian	MS	Oceanography	J. Dymond
Yilmaz, Tamer	PhD	Nuclear Engr	S. Anghaie
Youssefnia, Mohammad	PhD	Nuclear Engr	J. C. Ringle

D. CLASSIFIED STAFF AT THE RADIATION CENTER

Name	Title
Anderson, Terrance V.	Reactor Supervisor
Busby, Harold L.	Scientific Instrument Technician
Campbell, Ken	Custodian (through April 30, 1985)
Campbell, Shirley C.	Accounting Clerk I
Conrady, Michael R.	Chemist
Cramer, Dana L.	Secretary
Cunningham, Patricia A.	Clerical Assistant
Flickinger, Evelyn C.	Administrative Assistant
Hall, Arthur D.	Custodian
Johnson, Dean E.	Research Assistant
Latham, Jennie	Clerical Assistant
Liedtke, James D.	Research Assistant
Moeller, Wanda M.	Clerical Specialist
Neyhart, Shirley N.	Business Manager
Pratt, David S.	Radiation Specialist
Woodrow, Doyle K.	Scientific Instrument Technician

*Reactor users for research and/or teaching.

E. REACTOR OPERATIONS STAFF

Title	Name
Reactor Administrator	C. V. Smith, Jr.
Assistant Director, Radiation Center, and Principal Security Officer	A. G. Johnson
Assistant Reactor Administrator	B. Dodd
Reactor Supervisor	T. V. Anderson
Senior Reactor Operators	T. V. Anderson
	S. E. Binney
	B. Dodd
Reactor Operator	W. T. Carpenter
Senior Health Physicist	A. G. Johnson
Radiation Specialist	D. S. Pratt
Radiation Protection Student Workers	Lane, Vincent T.
	Lewis, Bryan R.
	Nelson, Lindsay A.
	Pauley, Keith A.
	Persinger, Richard W.
	Scarbrough, Allen L.

F. REACTOR OPERATIONS COMMITTEE

Name	Affiliation
S. E. Binney, Chairman	Nuclear Engineering
T. V. Anderson	Radiation Center
B. Dodd	Radiation Center and Nuclear Engineering
A. G. Johnson	Radiation Center and Nuclear Engineering
J. C. Ringle	Nuclear Engineering and Graduate School
A. H. Robinson	Nuclear Engineering
R. A. Schmitt	Chemistry
S. A. Stone	School of Engineering, Dean's Office
D. L. Willis	General Science (Radiation Biology & Radiation Health)

G. RADIATION SAFETY COMMITTEE

Name	Affiliation
S.E. Binney	Nuclear Engineering
A.G. Johnson	Radiation Center and Nuclear Engineering
N.I. Kerkvliet	Veterinary Medicine
G.E. Little	Radiation Safety Officer
R.G. Senechal (Current Chairman)	Geology
H.W. Shaup	Biochemistry
R. Morita	Microbiology
T. Beasley	Oceanography - Marine Science Center
G. Beaudreau	Agricultural Chemistry

PART 3

OSTR OPERATIONAL DATA

PART 3

OSTR OPERATIONAL DATA

A. BRIEF CHRONOLOGY OF THE OSTR OPERATING HISTORY

1. The OSTR has operated for 18 years.
2. From March of 1967 to August 1969 the maximum reactor power level was restricted to 250 kW.
3. In August 1969 the reactor was licensed to operate at a maximum reactor power level of 1 MW. From that date until June of 1971 the OSTR could operate at 1 MW for short periods of time only, due to the lack of sufficient cooling capacity.
4. In June of 1971 the cooling capacity was upgraded to allow continuous operation at 1 MW.
5. In July of 1976 the reactor was shut down for one month while a new FLIP fuel core (70% enriched fuel) was installed.(1)(2)(3)

This year's annual report will not attempt to review in detail the OSTR's operation over the past 18 years. Only the operating statistics for the FLIP fueled core will be presented in detail since we have established the (70% enriched) FLIP fuel as the basis for operational history for this and subsequent annual reports. More detailed information concerning the original (20% enriched) standard fuel core can be obtained from the 1976-77 OSTR annual report, dated August 31, 1977.

---

(1) See Table 3.1 for a tabular review of the OSTR's operating statistics with the FLIP core.

(2) See Table 3.2 for a summary of the OSTR's operating statistics with the original (20% enriched) standard TRIGA fuel core.

(3) See Figure 3.5 for a graphic review of the OSTR's energy production with the FLIP core.

B. OPERATING STATISTICS

The utilization of the OSTR for the current reporting period showed a very slight increase (∿0.2%) in the total number of operating hours when compared to the previous reporting period (see Table 3.1).

The thermal energy generated in the reactor during the reporting period was 39.4 MWD. The cumulative thermal energy generated by the FLIP core now totals 275 MWD from August 1, 1976 through June 30, 1985. Reactor use time averaged approximately 81% of the normal nine-hour, five-day week schedule. See Table 3.1 and Tables 3.3 through 3.5 for operating statistics applicable to this reporting period.

Excess reactivity showed an increase (∿30¢) during the current reporting period, which is mainly due to the rate of consumption of the burnable poison in the fuel. This is consistent with the fuel manufacturer's prediction that the FLIP fuel should initially decrease in reactivity until about 200 MWD, after which time it should begin a net gain in reactivity due to depletion of the burnable poison. It is expected that the OSTR with its burnup of 275 MWD will continue to show an increase in excess reactivity during the coming years. The excess reactivity is predicted to peak after about 4.5 MW years.

C. EXPERIMENTS PERFORMED

At the present time there are 12 approved reactor experiments on the active list which can be utilized in reactor related programs. These experiments are listed below:

NOTE: Missing numbers identify reactor experiments which are in the inactive file and are not currently being used.

A-1 Reactor Operation in Any of its Modes With No Sample Irradiation.

B-3 Irradiation of Materials in Assorted Matrices for Elements H to Bi Inclusive, Plus Natural Th and U, for Neutron Activation Analysis.

B-8 Isotope Production for Elements 1 Through 83 (H to Bi) Excluding Cd.

B-11 Nuclear Reaction Studies Using the Irradiation of Stable Elements to Produce Any Nuclide Formed During the Neutron Irradiation of Natural Uranium.

B-12 Exploratory Experiments to Investigate the TRIGA's Capability to Achieve Certain Experimental Goals. (If the TRIGA can achieve the desired goals, a regular experiment is established.)

B-21 Advanced Neutron Radiography Using Beam Port #3. (Radiography of all conventional items plus ordinance materials.)

B-23 Measurement of Neutron and Gamma Fluxes in the Thermal Column for Nuclear Engineering Laboratory Classes.

B-24 General Neutron Radiography Using Beam Port #1. (Ordinance items are excluded from radiography in this experiment.)

B-25 Measurement of Relative Neutron Fluxes in the Reactor Core and Experimental Facilities Using Fission Probes and Self-Powered Detectors.

B-29 Fuel Element Reactivity Worth Measurements.

B-30 Irradiation of Jet, Diesel, and Furnace Fuels. (Irradiation of various fuel oils for neutron activation analysis.)

B-31 Determination of Neutron Fluxes in the OSTR Core and Irradiation Facilities, Using Various Activation Foils or Wires.

Of the 12 currently approved experiments, six were used during the reporting period. A tabulation of information relating to reactor experiment use is given in Table 3.6, and includes a listing of the experiments which were used, how often each was used, and the general purpose of the use. Presently, 25 additional experiments are in the inactive file and could be reapproved for use as needed.

D. UNPLANNED SHUTDOWNS

There were no unplanned shutdowns (scrams) during the current reporting period. Table 3.7 will contain a tabulation of unplanned shutdowns when they occur.

E. CHANGES TO THE FACILITY, TO FACILITY PROCEDURES, AND TO REACTOR EXPERIMENTS PURSUANT TO 10 CFR 50.59

1. Introduction

The information contained in this section of the report provides a summary of OSTR changes made during the reporting period under the provisions of 10 CFR 50.59. As applicable, changes have been grouped into three categories: those dealing with the facility itself; those dealing with the facility's procedures; and those involving OSTR experiments. For each change identified, a brief description of the change and a summary of the safety evaluation is included.

The information presented in this section is also submitted annually to the USNRC in a separate report in order to meet the requirements of 10 CFR 50.59(b).

2. 10 CFR 50.59 Changes to the Facility

There were three changes to the facility itself which were reviewed and performed under the provisions of 10 CFR 50.59 during the reporting period. A summary of each change and its safety evaluation follows.

a. Installation of an Annunciator on the Ventilation Fan for the Argon Ventilation System

Description

The ventilation system for the OSTR's experimental facilities (e.g., the rotating rack, the beam ports, the thermal column, etc.) is commonly called the argon ventilation system. This system is equipped with a fan to enhance air flow through the various experimental facilities, and ultimately the system discharges its ventilation air into the intake plenum for the main reactor building ventilation fan. Although the fan for the argon ventilation system definitely creates a higher air flow through the system when it is operating, the negative pressure in the main reactor building fan intake plenum (relative to the argon ventilation system) induces a reasonable air flow through the argon ventilation system without the argon fan operating.

Operation of the ventilation fan for the argon ventilation system is indicated by an "on" light on a control panel located on the west wall of the reactor control room. However, this indicator light is not in a conspicuous location and is difficult to see from the reactor console. Because of this situation, it was considered unlikely that the reactor operator would be immediately alerted if the argon ventilation system fan stopped.

As a result of the above, the OSTR operations staff recommended the installation of a visible and audible annunciator for the argon vent fan circuit which would immediately alert the reactor operator in the event that the fan stopped operating.

Safety Evaluation

The installation of an annunciator on the argon vent system fan enhances safety and provides needed notification in the event the fan malfunctions. However, failure of the vent fan poses no safety problem, as an acceptable air flow through the argon ventilation system is still maintained due to the pressure-induced flow mentioned previously. Air flow through the system is also an optional matter controlled by the reactor operations staff using flow regulating valves installed as part of the original reactor equipment and upgraded and supplemented within the last 2 years under the provisions of 10 CFR 50.59.

The above facility change was reviewed by a Reactor Operations Staff Subcommittee and approved by the Reactor Operations Committee (ROC) prior to being made. It was concluded that this facility change did not require a change in the Technical Specifications, or constitute an unreviewed safety question as defined in 10 CFR 50.59(a)(2).

b. Electronic Timing Device For Measuring Control Rod Drop Times

Description

During an NRC inspection of the OSTR in March of 1985, the NRC representative suggested that consideration be given to an alternative method for measuring the drop times for the reactor control rods. At that time, the semiannual measurements of control rod drop times were being performed manually with a stop watch.

NOTE: The stop watch method has been very adequate considering the two-second drop time limit in the OSTR Technical Specifications.

In an effort to respond to the NRC's suggestion, the OSTR staff investigated possible ways of implementing an electronic timing device for the control rod drop time measurements. The results of the study demonstrated that the control rod drop time could be measured electronically using a commercially available counter-timer, a relay, and a five-position selector switch. The process utilized existing but non-used contacts on: 1) the control rod drive foot switches, and 2) the console CONT/ON switches. The CONT/ON switch starts the timer at the instant the control rod starts to drop, and the foot switch stops the timer when the rod reaches the bottom. The selector switch determines which control rod drop time is being measured.

Safety Evaluation

This addition does not compromise safety or affect any control circuitry. Spare contacts were used on existing control rod microswitches to make the rod drop measurements possible. The rod drop measurement apparatus is passive and responds to switch actuation only, and will not have an effect on the control rod drive circuits, even if the apparatus should malfunction.

The above addition was reviewed by a Reactor Operations Staff Subcommittee and approved by the ROC prior to being made. It was concluded that the addition did not require a change in the Technical Specifications or constitute an unreviewed safety question as defined in 10 CFR 50.59(a)(2).

c. Change in the Display Locations for Data from the Instrumented Fuel Element Thermocouples

Description

The display locations for fuel temperature data from the instrumented fuel element thermocouples (TCs) were changed so that the TCs measuring the maximum and minimum fuel temperature now display their readings on the more accurate OMEGA digital instrument, and the middle TC displays its reading on the analog meter in the left-hand console drawer.

Safety Evaluation

There are no adverse safety implications associated with this change. All of the TCs are routinely calibrated and were recalibrated after this change. It is known that the newer OMEGA digital instrumentation is more accurate, and therefore it increases safety to have the maximum and minimum fuel temperatures displayed on this new digital meter. (Note: Only one TC reading is actually required by the Technical Specifications.)

The above change was reviewed by a Reactor Operations Staff Subcommittee and approved by the ROC prior to being made. It was concluded that this change did not require a change in the Technical Specifications or constitute an unreviewed safety question as defined in 10 CFR 50.59(a)(2).

3. 10 CFR 50.59 Changes to Facility Procedures

There was one 10 CFR 50.59 change to facility procedures made during the reporting period. A description of this change follows.

a. Revisions to the OSTR and Radiation Center Emergency Response Plan

Description

Following an action drill and a number of training sessions held during the week of July 16-20, 1984, it was recognized that there were a number of minor corrections required to the Emergency Plan. These changes are listed below:

Change No.	Page	Change
1	Cover page	Change revision date to July 24, 1984
2	2-2	Add definition of ERIP
3	2-3	Retyped to include text carried over from page 2-2. No revision of content.
4	3-4	Change Student Health Service to Student Health Center. This is necessary due to a name change.
5	3-5	Change Student Health Service to Student Health Center. This is necessary due to a name change.
6	3-6	Correct the spelling of responsibilities
7	3-6	Reverse the order of the first two people in the line of succession for the Emergency Director. This order is more consistent with personnel responsibilities.
8	3-8	Correct the spelling of responsibilities in two places on this page.
9	3-11	Change Radiation Center Clerical Specialist to Radiation Center Clerical Assistant. The original job title was in error.
10	7-1	Change OSU Student Health Service to Center

Change No.	Page	Change
11	7-2	Change Trojan nuclear plant to Trojan Nuclear Plant
12	7-3	Add an 's' to air sampler
13	7-8	Change shutdown to shut down
14	7-14	Change shutdown to shut down
15	8-5	Revise paragraph b) completely to allow for the new 911 Emergency Dispatch system now in use in the Corvallis area
16	8-6	Change Student Health Service to Center
17	8-7	Change Student Health Service to Center

Safety Evaluation

Most of the changes are typographical and English corrections and therefore do not have any safety implications. One change reverses the order of two people who may fill the Emergency Director position. The new sequence is more consistent with personnel responsibilities and will enhance safety. The latter is true because the Assistant Director will now only fill the position of Senior Health Physicist, enabling him to concentrate on the radiological safety aspects of the emergency. Another change is required because, since the plan was written, the Corvallis area has instigated a 911 emergency dispatch system. This also enhances safety because there are now less telephone numbers to call if there is need for assistance.

The above changes were reviewed by a Reactor Operations Staff Subcommittee and approved by the ROC prior to being made. It was concluded that these changes did not require a change in the Technical Specifications or constitute an unreviewed safety question as defined in 10 CFR 50.59(a)(2).

4. 10 CFR 50.59 Changes to Reactor Experiments

There was one 10 CFR 50.59 change to a reactor experiment during this reporting period. A description of this change follows.

a. Change to Experiment B-24-Revised to Add Subpart B-24F

Description

A change to experiment B-24-Revised was made which added Subpart B-24F and thereby enabled zone plate encoded neutron holography to be performed in beamport #1. The experiment change provided for the radiography of small (less than 3.0 cm^3) objects and was designed so that only neutrons scattered approximately 90 degrees (toward the access plug for the beam port shield) would be recorded by the film.

The new subpart B-24F will allow the imaging of neutrons which are scattered from any object currently allowed to be radiographed by experiment B-24-Revised. Before imaging on the film, the scattered neutrons will pass through a gadolinium or cadmium encoding aperture which will encode target spatial information on the film. The film cassette will be placed at a 90° angle to the beam axis and will be surrounded by a lead "igloo" which will enclose the cassette except for one small imaging aperture which allows scattered neutrons to reach the film, and for one entire face (the surface facing the access plug) which allows the cassette to be removed easily. The cassette will be attached to a support that extends from the access plug, allowing it to be rolled into and out of the beam port with the access plug.

A barrier of aluminum, wood, and masonite will be placed above the lead igloo and will serve the personnel access barrier function usually provided by an aluminum plate connected to the end of the access plug's carriage. This carriage necessarily will be removed during this experiment. A cadmium sheet on the beam side of the igloo will help to reduce interfering neutron radiation.

When it is necessary to change the target and/or coding aperture during experimentation, all requirements of Experiment B-24-Revised will be met before entering the beam port blockhouse. This includes shutdown of the reactor and full coordination with health physics personnnel.

No modifications to the beam port #1 facility as stated above affect any existing shielding or alter any approved operating procedures, or reduce any safety margins now in use. All procedures, safety requirements, and facility descriptions of B-24-Revised, excluding the specific (not applicable) procedures and requirements for Sub-parts A through E, will apply to this experiment.

Safety Evaluation

None of the new experimental apparatus will be in the direct beam and so from a safety viewpoint this experiment is very similar to standard neutron radiography experiments. Other very minor modifications to the facility will provide radiation protection features which are equivalent to those for other approved neutron radiography experiments.

The above experiment change to B-24-Revised was reviewed by a Reactor Operations Staff Subcommittee and approved by the ROC prior to being made. It was concluded that the change did not require a change in the Technical Specifications or constitute an unreviewed safety question as defined in 10 CFR 50.59(a)(2).

5. Forthcoming Changes to Be Made Under 10 CFR 50.59

At some point in the future we expect to replace the rotating specimen rack in the TRIGA reactor under the provisions of 10 CFR 50.59.

F. SURVEILLANCE AND MAINTENANCE

1. Non-Routine Maintenance

15 AUG 84	The "Agastat" delay timer, which initiates the stack monitor filter failure alarm, was replaced.
24 AUG 84	The scram relay designated K-1 was replaced.
5 SEP 84	The rectifier in the control rod magnet circuit was replaced
21 NOV 84	New evacuation horn batteries were installed.
13 DEC 84	The calibration potentiometers on the stack monitor gas and particulate channels were replaced.

21 JAN 85	The primary coolant pump bearing was replaced.
25 JAN 85	A replacement blower motor was installed in the reactor top continuous air monitor and the old motor repaired.
27 MAR 85	Area Radiation Monitor #11 was repaired.
3 APR 85	A new DOWN switch for the regulating control rod was installed.
5 APR 85	A new UP switch for the shim control rod was installed.
24 APR 85	A new motor for cooling tower for #1 was installed.
21 JUN 85	The argon ventilation system fan was repaired.

2. Routine Surveillance and Maintenance

The OSTR has a routine surveillance and maintenance (S&M) program. Examples of typical S&M lists are presented in Figures 3.1 through 3.4. Those items marked with an asterisk (*) are required by the OSTR Technical Specifications.

G. REPORTABLE OCCURRENCES

There was one reportable occurrence during the reporting period. This involved an indication that the OSTR might have experienced a very small fuel cladding leak. In keeping with the requirements of the OSTR Technical Specifications, a written report of the event was sent to the Region 5 Office of the USNRC.

The existence of the initially suspected fuel cladding leak has not to this date been established despite extensive efforts to confirm its presence or absence. An additional surveillance program designed to detect any changes in the current situation has been added to the OSTR's operational routine.

Table 3.1

OSTR Operating Statistics (Using the FLIP Fuel Core)

Operational Date for FLIP Core	1 AUG 76 through 30 JUN 77(1)	1 JUL 77 through 30 JUN 78	1 JUL 78 through 30 JUN 79	1 JUL 79 through 30 JUN 80	1 JUL 80 through 30 JUN 81	1 JUL 81 through 30 JUN 82	1 JUL 82 through 30 JUN 83	1 JUL 83 through 30 JUN 84	1 JUL 84 through 30 JUN 85
Operating Hours (critical)	875	819	458	875	1255	1192	1095	1205(2)	1208
Megawatt Hours	451	496	255	571	1005	999	931	943(2)	946
Megawatt Days	19	20.6	10.6	23.8	41.9	41.6	38.8	39.3	39.4
Grams ^{235}U Used	24	25.9	13.4	29.8	52.5	52.4	48.6	49.3	49.5
Hours at Full Power (1MW)	401	481	218	552	998	973	890	929(2)	904
Number of Fuel Elements Added to Core	85	0	2	0	0	1	0	0	0
Number of Irradiation Requests	443	375	329	372	348	408	396	469	407

(1) Reactor shutdown July 26, 1976 for one month in order to completely refuel the reactor with a new FLIP fuel core.

(2) These values have been changed in this report to correct errors detected in the 1983-84 report. All subsequent reports will show these new values.

Table 3.2
OSTR Operating Statistics with the Original (20% Enriched) Standard TRIGA Fuel Core

Operational Data for 20% Enriched Core	8 MAR 67 through 30 JUN 68 (1)	1 JUL 68 through 30 JUN 69	1 JUL 69 through 31 MAR 70 (2)	1 APR 70 through 31 MAR 71 (3)	1 APR 71 through 31 MAR 72	1 APR 72 through 31 MAR 73	1 APR 73 through 31 MAR 74	1 APR 74 through 31 MAR 75	1 APR 75 through 31 MAR 76	1 APR 76 through 26 JUL 76 (4)	TOTAL: MAR 67 through JUL 76
Operating Hours (critical)	904	610	567	855	598	954	705	563	794	353	6903
Meagawatt Hours	117.24	102.47	138.05	223.77	195.11	497.82	335.94	321.45	408	213	2553
Megawatt Days	4.88	4.27	5.75	9.3	8.1	20.74	13.99	13.39	17	9	106.4
Grams ^{235}U Used	6.13	5.36	7.21	11.7	10.2	26.031	17.57	16.81	21.35	10.7	133
Hours at Full Power (250 KW)	429	369	58	--	--	--	--	--	--	--	856
Hours at Full Power (1 MW)	--	--	20	23	100	401	200	291	460	205	1700
Number of Fuel Elements Added to Core	70 (Initial)	2	13	1	1	1	2	2	2	0	1560
Number of Irradiation Requests	429	433	391	528	347	550	452	396	357	217	4100
Number of Pulses	202	236	299	102	98	249	109	183	43	39	1560

(1) Reactor went critical on March 8, 1967 (70 element core; 250 KW). Note: This period length is 1.33 years as initial criticality

(2) Reactor shutdown August 22, 1969 for one month for upgrading to 1 MW (did not upgrade cooling system). Note: This period length is only 0.75 years as there was a change in the reporting period from July-June to April-March.

(3) Reactor shutdown June 1, 1971 for one month for cooling system upgrading.

(4) Reactor shutdown July 26, 1976 for one month for refueling reactor with a new full FLIP fuel core. Note: This period length is 0.33 years.

Table 3.3

Present OSTR Operating Statistics

Operational Data for FLIP Core	Annual Values for 1 JUL 84 through 30 JUN 85	Cumulative Values for 1 AUG 76 through 30 JUN 85
1. MWH of energy produced	946	6597
2. MWD of energy produced	39.4	275
3. Grams ^{235}U used	49.5	345.40
4. Number of fuel elements added to core	0	85 + 3 FFCR(1)
5. Number of pulses	72	1004
6. Hours reactor critical	1208	8982
7. Hours at full power (1 MW)	904	6346
8. Number of startup and shutdown checks	253	2255
9. Number of irradiation requests processed(2)	407	3547
10. Number of samples irradiated	4583	43996

(1) Fuel Follower Control Rod.

(2) Each irradiation request could authorize from 1 to 120 samples. The number of samples per irradiation request averaged approximately 11 during the current reporting period.

Table 3.4A

OSTR Use Time in Terms of Operational Functions

OSTR Operational Function	Annual Values for 1 JUL 84 through 30 JUN 85 (hours)	Cumulative Values for 1 AUG 76 through 30 JUN 85 (hours)
1. Checkout, core excess and shutdown	380	3284
2. Load and unload samples	59	766
3. Reactor in operation(1)	1467	10817
4. Total reactor use time	1906	14867

(1) Includes preclude time. (Preclude time is the time the reactor is not available for use due to performance of surveillance and maintenance items, such as fuel element inspections, transient rod lubrication, control rod calibration, power calibration, etc.)

Table 3.4B

OSTR Use Time in Terms of Specific Use Categories

OSTR Use Category	Annual Values for 1 JUL 84 through 30 JUN 85 (hours)	Cumulative Values for 1 AUG 76 through 30 JUN 85 (hours)
1. Teaching (departmental and others)(1)(2)	95	1653
2. OSU Research(1)(3)	477	4073
3. Off-campus research(1)(3)	236	953
4. Forensic services(1)	0	101(4)
5. Reactor preclude time	614	5214
6. Facility time(5)	473	2723
7. Visitor demonstration(6)	11	150
8. Total reactor use time	1906	14867

(1) Includes sample loading and unloading.

(2) See Tables 4.1 and 4.2 for teaching statistics.

(3) See Table 4.5 for research statistics

(4) Prior to the 1981-1982 reporting period, forensic services were grouped under another use category. Since then and for subsequent reports, it will be a separate category and the cumulative hours will be compiled beginning with the 1981-1982 report.

(5) The time OSTR spent operating to meet NRC facility license requirements. Whenever possible, beneficial uses of the reactor were made, including irradiation of items requiring long irradiation times.

(6) Reactor use time for visitor open-house (demonstration) events was 11 hours. The remainder of the visitors viewed the reactor during times when the reactor was being operated for regularly scheduled research and teaching.

Table 3.5

OSTR Multiple Use Time(1)

Number of Users	Annual Values for 1 JUL 84 through 30 JUN 85 (hours)	Cumulative Values for 1 AUG 76 through 30 JUN 85 (hours)
1. Two users	75	653
2. Three users	26	89
3. Four users	7	12
4. Total multiple use time	108(2)	754(3)

(1) Multiple use time is that time when two or more irradiation requests are being concurrently fulfilled by operation of the reactor.

(2) Represents 9% of the total hours the reactor was critical during this reporting period.

(3) Represents 8% of the total hours the reactor was critical since startup with FLIP fuel in August of 1976.

Table 3.6

Use of OSTR Reactor Experiments(1)

Reactor Experiment Number(2)	Research		Teaching				
	Funded Thesis Work	Other Scholarly Research	Unfunded Thesis Work	Scheduled Classes	Unscheduled Classes	Facility Use	TOTAL
A-1	0	0	0	13	8	138	159
B-3	6	115	5	12	0	0	138
B-8	7	4	28	18	0	0	58
B-12	3	0	0	0	0	0	3
B-21	16	19	1	0	0	0	36
B-24	11	2	0	0	0	0	13
TOTAL	43	140	34	43	8	138	407

(1) This table displays the number of times each formal reactor experiment was used for a particular purpose.

(2) The following tabulation gives the reactor experiment number and its corresponding title:

A-1 Normal TRIGA Operation
B-3 Neutron Activation Analysis
B-8 Isotope Production
B-12 Exploratory Experiments
B-21 Beam Port #3 Advanced Neutron Radiography
B-24 Beam Port #1 Neutron Radiography

Table 3.7

Unplanned Shutdowns (Scrams)

Type of Scram	Number of Occurrences	Cause of Scram
None during this reporting period	0	Not Applicable

Figure 3.1

Monthly Surveillance and Maintenance (Sample Form)

OSTROP 13 SURVEILLANCE & MAINTENANCE FOR MONTH OF _____

	SURVEILLANCE & MAINTENANCE TO BE PERFORMED			LIMITS	AS FOUND	DATE COMPLETED	REMARKS & INITIALS
* 1	FUNCTIONAL CHECK OF REACTOR WATER LEVEL ALARMS			MAXIMUM MOVEMENT ±3 INCHES			
2	MEASUREMENT OF THE REACTOR PRIMARY WATER pH			MIN: 5 MAX: 7.5			
3	MEASUREMENT OF THE BULK SHIELD TANK WATER pH			MIN: 5 MAX: 7.5			
4	EMERGENCY POWER SYSTEM BATTERY CHECKS INVERTER* GENERATOR△			*LIQUID: < 1" DN *S.G. DISCS UP △S.G.: >1.250 △VOLTS 12VDC CORR: NONE			
5	EVACUATION HORN & P.A. EMERGENCY SYSTEM BATTERY CHECKS			LIQUID: FULL S.G.: >1.250 VOLTS 12VDC CORR: NONE			
6	INSPECTION OF THE BRUSHES ON THE PNEUMATIC TRANSFER SYSTEM BLOWER MOTOR			CHANGE WHEN 1/4" LEFT			
7	FUNCTIONAL CHECK OF EVACUATION ALARM			ALL WORKING			
8	*This section is not currently used.*						
9	AVERAGE MONTHLY PRIMARY WATER CONDUCTIVITY			<2 MICROMHOS/CM			
10	GREEN LIGHT BULB REPLACEMENT			MAX: 75 WATT			
11	CHANGE LAZY SUSAN FILTER						
12	LUBRICATE THE TRIGA TUBE LOADING TOOL (REEL)			USE GUN OIL			
13	REACTOR TOP CAM OIL LEVEL CHECK			OSTROP 13.13			
14	PROPANE TANK LIQUID LEVEL CHECK (% FULL)			>50%			
15	ARM SYSTEM L.E.D. TESTS:	#8	#1 BEAM PORT	$10^2 \equiv 100$ mR/hr			
		#9	FUEL STORAGE PIT	$10^2 \equiv 100$ mR/hr			
		#10	#3 BEAM PORT	$10^2 \equiv 100$ mR/hr			
		#11	RX BAY TABLE	$10^2 \equiv 100$ mR/hr			
		#12	TOP OF RX	$10^2 \equiv 100$ mR/hr			
		#13	DEMINERALIZER ROOM	$10^2 \equiv 150$ mR/hr			
		#14	RABBIT ROOM	$10^2 \equiv 90$ mR/hr			

*Techninal Specifications Requirement

Revised 6/85

Quarterly Surveillance and Maintenance (Sample Form)

OSTROP 14

SURVEILLANCE & MAINTENANCE FOR THE QUARTER OF ___/___/___ 19___

	SURVEILLANCE & MAINTENANCE TO BE PERFORMED	LIMITS	AS FOUND	DUE DATE	DATE COMPLETED	REMARKS & INITIALS
*1	REACTOR OPERATIONS COMMITTEE (ROC) AUDIT OF REACTOR OPERATIONS FOR / / QUARTER	QUARTERLY				
*2	QUARTERLY ROC MEETING	QUARTERLY				
*3	FUEL ELEMENT RADIATION LEVEL MEASUREMENTS IN WATER	≥ 23 R/hr @ 2' IN WATER				
4	INSPECTION OF THE SOLENOID VALVES IN THE PNEUMATIC TRANSFER SYSTEM	FUNCTIONAL				
5	PNEUMATIC TRANSFER SYSTEM INSERTION TIME CHECK	≤ 6 SECONDS				
6	ROTATING RACK CHECK FOR UNKNOWN SAMPLES	QUARTERLY				
7	FUNCTIONAL CHECK OF EMERGENCY LIGHTS	FUNCTIONAL				
8	WESTRONIC RECORDER SLIDE WIRE CLEANING	QUARTERLY				
9	STACK MONITOR CHECKS (OIL DRIVE MOTORS, H.V. SOURCE)	PART: 500 V ±50 GAS: 900 V ±50				
10	TRACERLAB AREA RADIATION MONITOR (ARM) VOLTAGE CHECKS	25 V SUPPLY±10% H.V. 560 V ±10%				
11	ARM SYSTEM ALARM CHECKS CHAN 1 2 3 4 5 6 7 8 9 10 11 12 13 14 AUD LIGHT PANEL ANN	FUNCTIONAL				
12	OPERATOR QUARTERLY CHECK NAME / DATE CHECKED / DATE LICENSE EXPIRES	QUARTERLY				
13	CHECK FILTER TAPE SPEED ON STACK MONITOR	1"/HR ± 0.2				
14	INCORPORATE OIB & FCB INTO DOCUMENTATION	QUARTERLY				
15	TRANSIENT ROD CALIBRATION	OSTROP 9.0				

*Technical Specifications Requirement

Revised 4/85

OSTROP 15

SEMI-ANNUAL SURVEILLANCE & MAINTENANCE FOR _____

	SURVEILLANCE & MAINTENANCE TO BE PERFORMED						LIMITS	AS FOUND	DUE DATE	DATE COMPLETED	REMARKS & INITIALS
*1	FUNCTIONAL CHECKS OF REACTOR INTERLOCKS	a) SOURCE INTERLOCK SETTING					≥2 cps	a			
		b) SIMULTANEOUS WITHDRAWAL OF 2 RODS					1 only	b			
		c) PULSE INITIATION ABOVE 1 kW					≤1 kW	c			
		d) PULSE INTERLOCK ON RANGE SWITCH					OSTROP 15-1.d	d			
		e) TRANSIENT ROD CYLINDER AIR INTERLOCK					OSTROP 15-1.e	e			
		f) PULSE MODE ROD MOVEMENT INTERLOCK					NO MOVE	f			
		g) PREVENTS PULSING ABOVE $2.55 p INSERTION					≤$2.55	g			
*2	SAFETY CIRCUITS TEST	a) PERCENT CHANNEL					≤110%	a			
		b) SAFETY SCRAM					≤110%	b			
		c) PERIOD SCRAM					≥3 sec	c			
*3	CONTROL ROD WITHDRAWAL, INSERTION & SCRAM TIMES		TRANS	SAFE	SHIM	REG					
		a) ROD DROP					<2 sec	a			
		b) WITHDRAWAL					≤50 sec	b			
		c) INSERTION					≤50 sec	c			
*4	PULSE COMPARISON (PREVIOUS PULSE: _____ MW, _____ °C)						≤20%				
*5	REACTOR BAY VENTILATION SYSTEM SHUT DOWN TEST						OSTROP 17				
*6	CALIBRATION OF THE FUEL ELEMENT TEMPERATURE CHANNEL						±2°C				
*7	MATERIALS BALANCE REPORT										
*8	CLEANING & LUBRICATION OF TRANSIENT ROD CARRIER INTERNAL BARREL						3 IN 1 or GUN OIL				
9	LUBRICATION OF BALL-NUT DRIVE ON TRANSIENT ROD CARRIER						3 IN 1 or GUN OIL				
10	LUBRICATION OF THE ROTATING RACK BEARINGS						10 W OIL				
11	CONSOLE CHECK LIST (OSTROP 15-11)						OSTROP 15-11				
12	CONSTANT AIR MONITOR RECORDER MAINTENANCE										
13	WESTRONICS RECORDER ZERO & CALIBRATION CHECKS										
14	STANDARD CONTROL ROD MOTOR CHECKS										
15	FLUKE FUEL TEMPERATURE INSTRUMENT "D" CELL CHANGE										

*Technical Specification Requirements

Revised 4/85

Figure 3.4

Annual Surveillance and Maintenance (Sample Form)

OSTROP 16

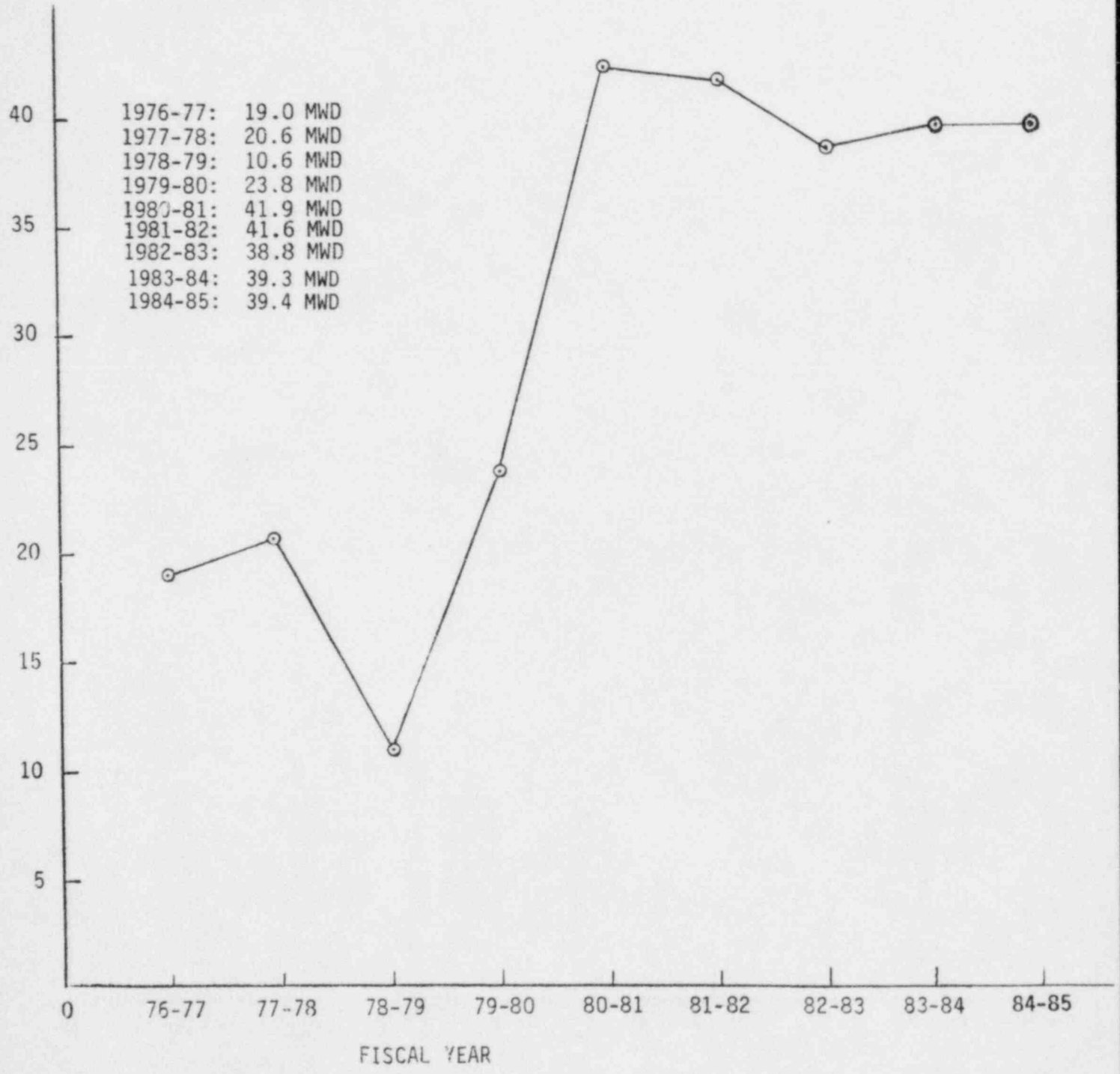
ANNUAL SURVEILLANCE & MAINTENANCE FOR THE YEAR _____

	SURVEILLANCE & MAINTENANCE TO BE PERFORMED		LIMITS	AS FOUND	DUE DATE	DATE COMPLETED	REMARKS & INITIALS
*1	BI-ANNUAL INSPECTION OF ALL CONTROL RODS		OSTROP 12.0				
*2	ANNUAL REPORT (DUE JUNE 30 + 75 DAYS)		SEPT 13				
*3	STANDARD CONTROL ROD CALIBRATION		OSTROP 9.0				
*4	REACTOR POWER CALIBRATION		OSTROP 8.0				
*5	CALIBRATION OF REACTOR TANK WATER TEMPERATURE METERS		OSTROP 16.5				
*6	CONSTANT AIR MONITOR CALIBRATION		RCHPP 18.0				
*7	STACK MONITOR:	a) CALIBRATE PARTICULATE MONITOR	RCHPP 18.0				
		b) CALIBRATE GAS MONITOR	RCHPP 18 & 26				
*8	AREA RADIATION MONITOR CALIBRATION		RCHPP 18.0				
*9	WATER MONITOR CALIBRATION		RCHPP 18.0				
*10	REACTOR OPERATOR REQUALIFICATION PROGRAM		GRADE OF ≥70%				
11	SNM PHYSICAL INVENTORY		OSTROP 20.0				
12	EMERGENCY RESPONSE PLAN DRILL						
13	STANDARD CONTROL ROD DRIVE INSPECTION						
14	SECURITY GUARD RETRAINING						
15	50.59 REPORT		SEPT				
16	INTRUSION ALARM RESPONSE DRILL (C.P.D. & OSU SECURITY)		RESPONSE ≤5 MIN				
17	EMERGENCY POWER INVERTER TEST		OSTROP 22.0				
18	REPLACE P.A. & EVAC SYSTEM LEAD-ACID BATTERIES		EVERY 4 YEARS				

*Technical Specification Requirements.

Figure 3.5

OSTR Annual Energy Production Vs. Time (Fiscal Year)



PART 4

OSTR UTILIZATION DATA

PART 4

OSTR UTILIZATION DATA

A. TEACHING PROGRAMS THAT UTILIZED THE OSTR

The OSTR accommodated 18 OSU academic courses and one special Radiation Center Reactor Operator Training Program. These programs utilized 95 hours of reactor time.

The academic courses are listed below:

Oregon State University Academic Courses

NE 101	Nuclear Engineering Orientation
NE 102	Nuclear Engineering Orientation
NE 203	Nuclear Radiation Detection and Measurement
NE 406	Projects (Nuclear Engineering)
NE 441	Nuclear Reactor Experiments
NE 461	Radiation Protection Engineering
NE 503	Thesis (Nuclear Engineering)
CH 107	General Chemistry Laboratory
CH 206H	General Chemistry Honors
CH 207	General Chemistry Laboratory
CH 316	Nuclear Reactor Chemistry
CH 419	Radioactive Tracer Methods
CH 505	Reading and Conference
CH 528	Activation Analysis
GS 405A/ 505A	Field Practices in Radiation Protection
G 503	Thesis (Geology)
OC 503	Thesis (Oceanography)
PH 503	Thesis (Physics)

Oregon State University Special Classes

One person sponsored by a U.S. Department of Energy grant administered by the University of Virginia participated in a special research reactor operator training program conducted at the Radiation Center. This training program is one of five in the U.S. for disadvantaged citizens, particularly minorities and females. The trainee has completed about 50% of the two year program, and will be taking the NRC reactor operator examination in September of 1985.

Part of the special reactor operator training program required exclusive use of the reactor. This time is reflected in Table 4.1 under the heading of "Special Classes." Most of the training program has been integrated into the regular operating schedule and, therefore, the special class hours do not represent the total training time but only that used exclusively for training purposes.

Additional data regarding the use of the OSTR for teaching and academic programs are shown in Tables 4.1 through 4.4.

B. RESEARCH PROJECTS THAT UTILIZED THE OSTR

During the reporting period, there were forty-eight funded research projects which utilized 713 hours of reactor time, and four unfunded research (teaching) projects which utilized 18 teaching hours of reactor time (see Table 4.1, footnote 1 and Table 4.5). Of these research projects, twenty-eight were conducted by Oregon State University, fifteen by the University of Oregon, three by Lousiana State University, one by Portland State University, one by Western Washington University, and four by Battelle Pacific Northwest Laboratories (under DOE prime contract number DE-AC06-76RLO-1830) in Richland, Washington. Table 4.5 gives statistics regarding OSTR-funded research hours, and Table 4.6 gives a summary of the different research projects.

C. PUBLICATIONS RESULTING FROM RESEARCH USING THE OSTR

1. Publications in Print

Bogard, D.D., G.J. Taylor, K. Keil, M.R. Smith and R.A. Schmitt "Impact melting of the Cachari eucrite 3 Gy ago." Geochim. Cosmochim. Acta, 49, 1985, 941-946.

Bogard, D., R.A. Schmitt, M.R. Smith and M. Rhodes (1984) "Petrology and shock age of the Palo Blanco Creek eucrite" Meteoritics 19, 219-220.

Budahn, J. R., and R. A. Schmitt, "Petrogenetic Modeling of Hawaiian Tholeiitic Basalts: A Geochemical Approach," Geochim. Cosmochim. Acta, 49, 1985, 67-88.

Davidson, G.R., G.P. Lahti, J.D. Ingle, Jr., J.C. Westall, C.W. Bennett, E.I. Jolma, and D.M. McDonald, "Post Accident Systems for Monitoring of Primary Coolant for Fission Products, Boron, pH, and Chloride," EPRI Report NP-3513, 1984.

Dickinson, T., G.J. Taylor, K. Keil, R.A. Schmitt, S.S. Hughes and M.R. Smith "Apollo 14 Aluminous Mare Basalts and Their Possible Relationship to KREEP." J. Geophys. Res., 90, 1985 C365-C374.

Hughes, S.S. and R.A. Schmitt "Whole-rock INAA Elemental Abundances and REE Variations in EH4, EH5 and EL6 Chondrites. Lunar and Planetary Science XVI, 1985, 372-373.

Hughes, S.S. and R.A. Schmitt "Zr/Hf Ratios in Lunar Mare Basalt Groups--Interim Report #2. Lunar and Planetary Science XVI, 1985, 374-375.

Laul, J.C., M.R. Smith, J.J. Papike and S.B. Simon "Agglutinates as Recorders of Regolith Evolution: Application of the Apollo 17 Drill Core." Proceedings of the 15th Lunar and Planetary Science Conference, 89, 1984, 161-170.

Laul, J.C., R.J. Walker, C.K. Shearer, J.J. Papike and S.B. Simon, "Chemical Migration by Contact Metamorphism Between Pegmatite/Country Rocks: Natural Analogs for Radionuclide Migration." Scientific Basis for Radioactive Waste Management VII, Material Research Proceedings, 26, 1984, 951-958.

Laul, J.C., M.R. Smith, S.B. Simon and J.J. Papike, "Chemistry and Petrology of Apollo 12 Drive Tube 12027." Proceedings of the 15th Lunar and Planetary Science Conference, 90, 1985, 507-516.

Laul, J.C., C.K. Shearer, J.J. Papike, "Chemistry of Potassium Feldspars from Three Zoned Pegmatites, Black Hills, South Dakota: Implications Concerning Pegmatite Evolution." Geochim. Cosmochim Acta, 49, 1985, 663-673.

Laul, J.C., T.D. Swindle, M.W. Caffee, C.M. Hohenberg, G.B. Hudson, S.B. Simon and J.J. Papike, "Noble Gas Organization in Apollo 14 Regolith Breccia 14318: ^{129}I and ^{244}Pu Regolith Chronology," Proceedings of the 15th Lunar and Planetary Sciences Conference, 90, 1985, 517-540.

Laul, J.C., C.K. Shearer, J.J. Papike, S.B. Simon and R. Christian, "Pegmatite/Wall Rock Interactions, Black Hills, South Dakota: Progressive Boron Metasomatism Adjacent to the Tip Top Pegmatite." Geochim. Cosmochim Acta, 48, 1984, 2563-2580.

Laul, J.C., M.R. Smith, M.-S. Ma, T. Huston, R.M. Verkouteren, M.E. Lipschutz and R.A. Schmitt, "Petrogenesis of the SNC (Shergottites, Nakhalites, Chassignites) Meteorites; Implications for Their Origin From a Large Dynamic Planet, Possibly Mars." Proceedings of the 14th Lunar and Planetary Science Conference, 1984, 612-630.

Laul, J.C., J.W. Shervais, L.A. Taylor and M.R. Smith, "Pristine Highland Clasts in Consortium Breccia 14305: Petrology and Geochemistry." Proceedings of the 15th Lunar and Planetary Science Conference, 89, 1984, 25-40.

Laul, J.C., N. Hubbard, R.W. Perkins, "The Use of Natural Radionuclides to Predict the Behavior of Radwaste Radionuclides in Far-Field Aquifers." Scientific Basis for Radioactive Waste Management VII, Material Research Society Proceedings, 26, 1984, 891-897.

2. Theses

Espenan, G.D., "Determination of Aluminum in Fish Tissue by Instrumental Neutron Activation Analysis." Master's Thesis, Louisiana State University, 1985.

Gastill, D. Kurt, "Peturbed Angular Correlation Measurements of Hyperfine Fields in Liquid and Amorphous Selenium-Tellurium Alloys." Ph.D. Thesis, OSU, 1984.

Greek, K.J., "Optimum Temporal and Contrast Consideration in Extremely High Speed Motion Neutron Radiography." Master's Thesis, Nuclear Engineering, OSU, 1985.

Newell, D., "Real Time Neutron Radiography." Master's Thesis, Nuclear Engineering, OSU, 1985.

3. Reports Submitted for Publication

Dickinson, T., K. Keil, L. Lapaz, D.D. Bogard, R.A. Schmitt, M.R. Smith and J.M. Rhodes, "Petrology and shock age of the Palo Blanco Creek Eucrite." Chemie der Erde, 1985.

Hughes, S.S., R. A. Schmitt, Y.L. Wang and G.J. Wasserburg, Trace Element Characteristics and Sr-Nd Isotopic Constraints on the Magmatic Sources of Serra Geral Continental Flood Basalts, Southern Brazil." Contributions to Mineralogy and Petrology. (submitted).

Jaeger, H., J.A. Gardner, J.C. Haygarth and P.L. Rasera, "Structural Characterization of Zirconia Ceramics by Perturbed Angular Correlation Spectroscopy." Submitted to J. American Ceramic Society.

Robinson, A.H., "High Speed Radiography of Solid Caseless Ammunition." Submitted to U.S. Army, 1985.

D. PRESENTATIONS AT PROFESSIONAL MEETINGS RESULTING FROM RESEARCH USING THE OSTR AND FROM OSTR OPERATIONS

Davidson, G.R., G.P. Lahti, J.D. Ingle, Jr., J.C. Westall, and M.D. Naughton, "Post Accident Systems for Monitoring Primary Coolant Radioactivity and Chemistry." Trans. Am. Nucl. Soc., 46, 1984 329.

Fukuoka, T., J. C. Laul, M.R. Smith, S.S. Hughes and R.A. Schmitt, "Chemistry of Yamato-791197 Meteorite: Evidence for Lunar Highland Origin." Proceedings of Tenth Symposium on Antartic Meteorites (2 page abstract), 1985.

Gardner, J.A., H. Jaeger, J. C. Haygarth, and R.L. Rasera, "Microscopic Structure and Phase Transformation of Zirconia-Based Ceramics Determined by PAC." American Ceramic Soc. Bulletin, August 1984, p. 991, paper presented at the 1984 Pacific Coast meeting of American Ceramic Society.

Jaeger, H., J.A. Gardner, J.C. Haygarth, and R.L. Rasera, "PAC Investigation of Oxygen-Defect Dynamics in Zirconia-Yttria." Bull. Am. Phys. Soc. 30, 319 (1985), paper presented at the 1985 March meeting of the American Physical Society.

Jaeger, H., J.A. Gardner, J. C. Haygarth, and R.L. Rasera, "PAC Investigation of Oxygen-Defect Dynamics in Zirconia-Yttria," American Ceramic Society Bulletin, March 1985, p. 464. Paper presented at the 1985 Annual Meeting of the American Ceramic Society.

Laul, J.C., "Rare Earth Elements Behavior in the Development of Energy Resources." Proceedings of the Fifth International Conference on Nuclear Methods in Environmental and Energy Research, CONF-840408, 425-437, 1985.

Schmitt, R.A., "The Relevance of Lunar Meteorites to Planetary Science" Suess Symposium (Dec 13-14, oral presentation at Univ. Calif., San Diego), 1984.

Willis, D.L., and Y.D. LaTouche, "Biokinetic Studies of Natural Uranium in Rats Following Oral Administration of Drinking Water." Oral report during the International Workshop on Gastrointestinal Absorption of Actinides and other Metals at Seattle, Washington, June 24, 1985.

E. PUBLIC RELATIONS

The continued interest of the general public in the TRIGA reactor is evident by the number of people who have toured the facility. In addition to many unscheduled visitors, and interested individuals who stopped in without appointments because they were in the vicinity,

a total of 769 people (a 46% increase over the previous reporting period) were given pre-planned and scheduled tours during this reporting period. See Table 4.7 for statistics on scheduled visitors.

F. PLANNED CHANGES IN OSTR UTILIZATION

A one-shift operation (45 hours per week) is the current OSTR utilization schedule. At the present time there are no planned changes to this mode of reactor utilization.

Table 4.1

OSTR Teaching Hours

Description	Annual Values for 1 JUL 84 through 30 JUN 85 (hours)	Cumulative Values for 1 AUG 76 through 30 JUN 85 (hours)
Departmental(1)	81	1463
Chemistry	25	
Nuclear Engineering	38	
Geology	8	
Oceanography	8	
Special Classes(2)	14	190
Total Teaching Time(3,4,5)	95	1653

(1) Approximately 22% of the departmental use hours (18 hours) was used for unfunded thesis and research work, which we consider to be teaching.

(2) Six special training classes were conducted for the Reactor Operator Training program.

(3) Includes sample loading and unloading

(4) See Table 4.3 for classes and student enrollment.

(5) See Table 3.4B.

Table 4.2

Other Educational Institutions Which Utilized the OSTR(1)

Institution	Number of Faculty Involved	Number of Students Involved	Number of Visits to Facility
Southern Oregon College	1	5	2
University of Oregon	6	8	2
Portland State University	1	1	2
Western Washington University	1	0	1
Louisiana State University	1	1	1

(1) Does not count community college, high school, and elementary school classes which came through for special tours. These are listed under the section on "Public Relations."

Table 4.3
Student Enrollment in Nuclear Engineering and Nuclear Science Courses

Course	Credit	Course Title	Number of Students Fall 1984	Winter 1985	Spring 1985
Nuclear Engineering Courses					
*NE 101	2	Nuclear Engineering Orientation	29	--	--
*NE 102	2	Nuclear Engineering Orientation	--	20	--
NE 103	3	Intro. Nuclear Engineering & Comp.	--	--	13
NE 201	3	Nuclear Energy Fundamentals	37	--	--
NE 202	3	Nuclear Radiation & Matter	--	28	--
*NE 203	3	Nuclear Radiation Detection & Measurement	--	--	22
NE 405	1-15	Reading & Conference			
*NE 406	1-15	Projects	5	2	2
NE 407	1	Seminar	--	17	--
NE 415	4	Principles Radiation Safety	26	--	--
NE 417	4	Principles Radiation Safety	--	14	--
NE 419	4	Principles Radiation Safety	--	--	14
NE 421	3	Nuclear Reactor Analysis & Computation	26	--	--
NE 422	3	Nuclear Reactor Analysis & Computation	--	21	--
NE 423	3	Nuclear Reactor Analysis & Computation	--	--	10
NE 430	3	Nuclear Fuel Cycle	--	16	--
NE 431	3	Reactor Thermal Hydraulics	22	--	--
NE 432	2	Reactor Design	--	14	--
NE 433	3	Reactor Design	--	--	4
NE 435	2	Nuclear Materials	--	--	13
**NE 441	3	Nuclear Reactor Experiments	--	14	--
*NE 461	3	Radiation Protection Engineering	--	22	--
NE 465	3	Nuclear Rules & Regulations	--	--	17
NE 501	1-15	Research	--	1	--
*NE 503	1-15	Thesis	14	11	11
NE 505	1-15	Reading & Conference	4	1	1
NE 505A	1-15	R&C/Neutron Transport	2	1	--
*NE 505B	1-15	R&C/Reactor Safety	1	--	--
NE 506	1-15	Projects	--	1	--
NE 507	1-15	Seminar	--	3	--
NE 511	2	Neutron Transport Theory	6	--	--
NE 512	2	Neutron Transport Theory			
NE 522	3	Reactor Environmental Problems	9	--	--
NE 523	2	Advanced Reactor Design	--	5	--
NE 524	3	Advanced Reactor Design	--	--	5
NE 531	3	Nuclear Reactor Kin.	--	--	8
NE 541	3	Advanced Nuclear Fuel Cycle	--	--	3
NE 542	3	Advanced Thermal Hydraulics	--	--	2
NE 552	3	Computational Methods for Nuclear Reactors	--	--	10
NE 553	3	Computational Methods for Nuclear Reactors			
NE 581	3	ST/Waste Management & Decommissioning	--	8	--
NE 583	3	Advanced Nuclear Waste Management	--	--	3
Chemistry Courses					
*CH 107	2	General Chemistry Lab	70	--	--
*CH 206 H	2	General Chemistry Honors	--	--	--
*CH 207	2	General Chemistry Lab	4	--	--
*CH 316	4	Nuclear Chemistry	28	--	--
**CH 419	4	Radioactive Tracer Methods	7	--	--
*CH 505	1-15	Reading & Conference (Chemistry)	--	--	--
*CH 528	4	Activation Analysis	--	7	--
Other Courses					
**G 503	1-15	Thesis (Geology)	1	1	--
**GS 405A/ 505A	2-3	Field Practices in Radiation Detection	--	1	1
**PH 503	1-5	Thesis (physics)	--	--	--
**OC 503	1-15	Thesis (Oceanography)	1	1	--

*OSTR used occasionally for demonstration experiments

**OSTR used heavily

Table 4.4

Graduate Student Thesis Research Which Utilized the OSTR

Student's Name	Program	Academic Department	Faculty Advisor	Thesis Topic
LOUSIANA STATE UNIVERSITY				
Espenan, G.	MS	Nuclear Science	Knaus	Aluminum Concentration in Fish Gills
OREGON STATE UNIVERSITY				
Coe, D.H.	MS	Nuclear Engineering	Robinson	Neutron Holography
Greek, K.	MS	Nuclear Engineering	Robinson	High Speed Neutron Radiography
Harris, R.	MS	Nuclear Engineering	Binney	Modeling of Gamma Pulse Height Distributions
Hoover, A.	MS	Geology	Snee	Idaho Batholiths
Jaeger, H.	PhD	Physics	Gardner	Petrubed Angular Correlations of Decay Gammas
Keizer, P.J.	MS	General Science (Rad. H)	Willis	Pharmacokinetics of Natural Uranium in Drinking Water
Newell, D.	MS	Nuclear Engineering	Robinson	Real Time Neutron Radiography
Verplanck, P.L.	MS	Geology	Snee	Pegmatite Geochemistry
Von Breymann, M.	PhD	Oceanography	Suess	Mg/Nh_4 Exchange
PORTLAND STATE UNIVERSITY				
Vogel, A.H.	PhD	Biology	Petersen	Oregon Lake Sediments
UNIVERSITY OF OREGON				
Brandon, A.	MS	Geology	Goles	Petrogenesis of Bear Creek Lavas
Connelly, T.	PhD	Geology/Anthropology	Aikens	Archaeological Horizon Markers
Desonte, D.L.	MS	Geology	McBirney	Geochemistry of the San Esteban Suite
Geist, D.	PhD	Geology	McBirney	Geology of San Cristobal
Radosevich, S.	PhD	Anthropology	Lukacs/Dumond	Trace Elements in Ancient Bones
Ritchie, B.	MS	Geology	McBirney	Geology of the Western Cascades
Sonnenthal, E.L.	MS	Geology	Goles	Pegmatoid Crystallization in Basalts of North-Central Oregon
Vicenzi, E.	MS	Geology	McBirney	Geology and Geochemistry of Marclena Island, Galapagas

Table 4.5

OSTR Funded Research Hours

Types of Research	Annual Values for 1 JUL 84 through 30 JUN 85 (hours)	Cumulative Values for 1 AUG 76 through 30 JUN 85 (hours)
OSU Research(1)	477	4073
Off-Campus Research(1)	236	953
Total Research(1)	713(2)(3)(4)	5026

(1) Includes sample loading and unloading time.

(2) 18% (130 hours) of the total research hours were funded thesis research.

(3) Research hours, OSU funded: 249

(4) Research hours, others funded: 464

Table 4.6

Summary of Research Projects Involving the OSTR, and the Funding Agencies

Listing Number	Name of Person(s) Using Reactor	Department and Institution	Project Title	Description	Funding Agency
1	R. A. Schmitt S. Hughes V. Golightly P. Verplanck	Chemistry, OSU	Tantalum/Ilmenites & Rutiles	INAA For Rutiles and Ilmenites From Worldwide Locations. (These Minerals are Used in Ta Production, as Well as Other Metals)	US Bureau of Mines
2	R. A. Schmitt P. Verplanck L. W. Snee	Geology OSU	Pegmatite Geochemistry	INAA for Trace Elements in Certain Geologic Samples to Determine Source Rock Signature	Radiation Center, OSU (Unfunded Research)
3	R. A. Schmitt S. Hughes	Chemistry, OSU	Lunar and Meteoritic Activation Analysis	Chemical and Petrological Characterization of Rock Clasts in Brecciated Meteorites	NASA
4	R. A. Schmitt A. Hoover L. W. Snee	Geology, OSU	Idaho Batholiths	INAA for Trace Element (REE) to Investigate Source Rock Affinities, Crustal Contamination and Mixing Relationships of Plutons of Idaho Batholith	Sigma Xi
5.	R. A. Schmitt S. S. Hughes Xiuzhen Hu	Radiation Center, OSU	INAA of Trace Elements in Deep Sea Drilling Projects	Study Major and Minor Trace Elements in Sedimentry Rock Samples Obtained in 1000 Meters Below Sea Floor at Site 530A, DSDP	Radiation Center, OSU
6	R. A. Schmitt M. Conrady	Radiation Center, OSU	Semi Conductor Analysis	INAA Determination of Mg Concentrations in Semi Conductor Samples	Union Carbide
7	R. A Schmitt R. Walker M. Conrady	Physics, OSU	Metal Purities	INAA of Purified and Unpurified Samples of Te & Se for Na Content	Physics

Table 4.6 (continued)

Listing Number	Name of Person(s)	Department and Institution	Project Title	Description	Funding Agency
8	R. A. Schmitt M. Conrady R. Walker	Radiation Center, OSU	Ash Peak Rhyolites	INAA of Rhyolitic Volcanics and Andesitic Volcanics to Determine Possible Genetic Relationship to Sn, Mn, Ag Mineralization	Chemistry, OSU
9	R. A. Schmitt S. Hughes	Chemistry, OSU	Miscellaneous Geological Material	On-going Investigation of Various Geological Materials to Coincide with Extra Terrestrial Studies	NASA
10	R. A. Schmitt M. Conrady	Radiation Center, OSU	Exxon Sediments	INAA of Ocean Sediments to Determine Barium Deposited During Drilling Operations	EXXON
11	R. A. Schmitt J. C. Laul A. V. Murali S. Hughes	Battelle N.W. Lab	Manganese Nodules	Study of the Meteoritic Fingerprints (That May Have Caused the Extinction of Dinosaurs) In Manganese Nodule Layers	BNWL
12	R. A. Schmitt M. Conrady	Radiation Center, OSU	La & Cr Analysis	INAA of Mineral Samples for La and Cr	U.S. Bureau of Mines
13	R. A. Schmitt S. Carter S. Hughes	Geology, Western Washington University	Geology & Petrology of Fifes Peak Formation (Central Washington)	INAA of Basaltic & Dacitic Dike & Lava Flow Rocks	USDOE
14	A. H. Robinson K. Greek D. Newell	Nuclear Engineering, OSU	Neutron Radiography	High Speed Neutron Radiography Of Brief Events (~2 Milliseconds)	USDOD
15	A. H. Robinson D. H. Coe	Nuclear Engineering, OSU	Neutron Holography	Formation of Psuedo Holograms Using Scattered Neutrons From an Irradiated Target Passing Through a Coded Aperture	Radiation Center, OSU

Table 4.6 (continued)

Listing Number	Name of Person(s) Using Reactor	Department and Institution	Project Title	Description	Funding Agency
16	A. H. Robinson D. Newell	Nuclear Engineering, OSU	Real-Time Neutron Radiography	Modification of High Speed N.R. by Replacing a 16mm Camera with a Video Camera For Real Time Events	Nuclear Engineering, OSU
17	S. E. Binney M. R. Conrady C. W. Bennett	Nuclear Engineering, OSU	U-Tn	Analysis of Solid Samples For U & Th as a Pilot Project for Future Research	Radiation Center, OSU
18	W. D. Loveland C. Casey	Chemistry, OSU	Log P Measurement	Measurement of M-Octanol/Water Partition Coefficient for Selected Metal Chelates	USDOE
19	S. E. Binney M. Conrady	Nuclear Engineering, OSU	U-Th	INAA of Selected Solid Samples for U & Th (Pilot Project)	Radiation Center, OSU
20	S. E. Binney R. Harris	Nuclear Engineering, OSU	Modeling of Gamma Pulse Height Distribution	Study of Modeling Gamma Pulse Height Distributions to Calculate the Feasibility of Monitoring Key Nuclides as Indicators of Fuel Element Damage	Radiation Center, OSU (Unfunded Research)
21	D. L. Willis D. LaTouche D. Bergman	General Science, OSU	Uranium in Drinking Water	Activation of Rat Blood, Kidney, Liver in Freeze Dry Form, Followed by Delayed Neutron Counting, to Determine Uranium Concentrations in Tissues (Rat Drinking Water Spiked with Uranium)	USEPA
22	D. L. Willis D. Bergman A. G. Johnson	General Science, OSU	U Concentration in Selected Metal Scrapings	Study of the Feasibility of Detecting Uranium in Scrapings From a Metal Pipe	General Science, OSU
23	R. Collier J. Dymond	Oceanography, OSU	JDF	Characterization of Particle Flux in the Juan De Fuca Hydrothermal Vent Area	NSF

Table 4.6 (continued)

Listing Number	Name of Person(s) Using Reactor	Department and Institution	Project Title	Description	Funding Agency
24	R. Collier J. Dymond R. Conard	Oceanography, OSU	BETTIS	INAA For Trace Elements of Sediment Trap Samples	Sandia
25	R. Collier A. Ungerer M. VonBreymann	Oceanography, OSU	Mg/Nh_4 Exchange	Investigation of the Role of Organic Matter During the Formation of Marine Dolmites. This Experiment Examines the Ammonium Exchangeable Mg Associated with Humic Materials	Radiation Center, OSU (Unfunded Research)
26	R. Collier J. Dymond R. Conard	Oceanography, OSU	MANOP	Analysis of Sediment Trap Samples to Determine Fluxes of Elements to the Ocean Floor in Mid-Pacific Ocean.	NSF
27	T. Beasley D. McCullough Chi-An Huh	Oceanography, OSU	Marine Transuranic Research	Study of Fate of Transuranic and Other Long-Lived Radionuclides in Columbia River Estuary and N.E. Pacific Ocean	USDOE
28	R. Buhler R. Reed	Agricultural Chemistry, OSU	Dimethylsulfone Metabolism	Explore Possibility of Using Reactor to Prepare Na_2S For Mammalian Metabolism Experiment	Radiation Center, OSU (Unfunded Research)
29	K. Krane J. Gardner H. Jaeger	Physics, OSU	Perturbed Angular Correlation ^{181}Hf	Study Structure of Hf Compounds Using Perturbed Angular Perturbation Method	Physics Dept., OSU (Wah Chang Grant)
30	J. C. Laul	Battelle Pacific Northwest Lab.	Lunar Chemical Characterization	INAA for Chemical Study of Lunar and Meteorite Samples and Some Terrestrial Rock Separates	DOE Prime Contract DE-AC06-76-RLO-1830, Special Agreement B-73282-A-U
31	J. C. Laul	Battelle Pacific Northwest Lab.	Geological Samples	Study of Chemical Migration of Various Elements in Geological Matrices Surrounding Nuclear Waste	DOE Prime Contract DE-AC06-76-RLO-1830, Special Agreement B-73282-A-U

Table 4.6 (con

Table 4.6 (continued)

Listing Number	Name of Person(s) Using Reactor	Department and Institution	Project Title	Description	Funding Agency
32	J. C. Laul E. Lepel	Battelle Pacific Northwest Lab.	Oil Shale	Analysis of Hg, As, Se in Samples Obtained From Oil Shale Retort Process	USDOE Prime Contract DE-AC06-76-RLO-1830, Special Agreement B-73282-A-U
33	J. C. Laul E. Lepel	Battelle Pacific Northwest Lab.	Polyethylene	Determine Trace Elements in Polyethylene	USDOE Prime Contract DE-AC06-76-RLO-1830, Special Agreement B-73282-A-U
34	J. Lyngdal M. Conrady	Tektronix Material Lab.	Glass Analysis	Characterization of Glass	Tektronix
35	R. Petersen A. H. Vogel	Biology, PSU	Oregon Lake Sediments	Measurement of the Distribution of 12 Physiological Important Metals in Lake Sediments	USDOE
36	R. Knaus G. Espenan	Nuclear Science, LSU	Aluminum in Fish Gills	Determination of Aluminum Concentrations in Gills of Fish Killed by Aluminum	USDOE
37	R. Knaus	Nuclear Science, LSU	Insect Migration	Characterization of Different Races of Moths	USDOE
38	R. Knaus	Nuclear Science, LSU	A Cretion Study of The Gulf Marsh	A Study in Marking Marsh-Mud Horizons with Rare Earths Equivalent to Soil Horizon Markers	USDOE
39	G. Goles R. A. Schmitt S. Radosevich J. R. Lukacs D. E. Dumond	Anthropology, U of O	Trace Elements in Ancient Bones	A Study of Trace Elements in Human Bone to Determine Diet Proportions in Ancient Populations of People From South Asia	USDOE

Table 4.6 (continued)

Listing Number	Name of Person(s) Using Reactor	Department and Institution	Project Title	Description	Funding Agency
40	G. Goles R. A. Schmitt D. Reich	Geology, U of O	Trace Elements in Coast Range Intrusive Rocks	INAA of Oregon Coast Intrusive Rock Samples From Cougar Mountain, Table Mountain, Blodget Peak, and Cannibal Mountain	USDOE
41	G. Goles R. A. Schmitt E. L. Sonnenthal	Geology, U of O	A Thermodynamic & Petrologic Study in the Picture Gorge Basalt of North-Central Oregon	INAA of Selected Rocks to Determine REE Yields Which Will Be Used to Model Processes That May Have Occurred During Cooling of These Flows	USDOE
42	G. Goles R. A. Schmitt T. Connelly C. M. Aikens	Geology/Anthropology, U of O	Archaeological Horizon Markers	Analysis of Pumice Collected From From Two Archaeological Sites in Western Oregon to Establish Tentative Identification as Mazama Tephra	USDOE
43	G. Goles R. A. Schmitt A. Brandon	Geology, U of O	Bear Creek lavas	INAA Determination of the Petrogenesis of Lavas in The Bear Creek Drainage in Central Oregon	USDOE
44	G. Goles R. A. Schmitt A. R. McBirney D. L. Desonte	Geology, U of O	Geochemical Features of The San Esteban Suite	A Study to Reconstruct the History of the Isla San Esteban and Its Relationship to the Tectonic Development of the Northern Gulf of California	USDOE
45	G. Goles A. R. McBirney D. Geist	Geology, U of O	San Cristobal Geology	INAA of Selected Geological Specimens from the Galapagos Islands in an Attempt to Identify the Mantle Sources for the Lavas & Determine Tectonic Origin of the Islands	USDOE

Table 4.6 (continued)

Listing Number	Name of Person(s) Using Reactor	Department and Institution	Project Title	Description	Funding Agency
46	G. Goles R. A. Schmitt W. W. Clingman	Geology, U of O	Geochemistry of Domes Near Crater Lake	INAA of Selected Geological Samples for Geochemical Analysis	USDOE
47	G. Goles R. A. Schmitt G. J. Retallack	Geology, U of O	Tuffaceous Palaeu Sediments	INAA of Tuffaceous Sediments From The Midland Valley of Scotland and From the Fort Ternan Area of Kenya to Establish Whether Carbonatitic Volcanism of Palaeozoic Age Occurred in The Northern British Isles.	USDOE
48	G. Goles R. A. Schmitt A. R. McBirney B. Ritchie	Geology, U of O	Calapooya Thesis	INAA of Volcanic Samples From The Western Cascades in Order to Document a Tholeiitic Suite of Tertiary Lavas and Determine heir Genetic Relationship to he Orogenic Rocks of the Cascade Volcanic Series	USDOE
49	G. Goles R. A. Schmitt A. R. McBirney E. Vicenzi	Geology, U of O	Geology & Geochemistry of Marchena Island, Galapagos	INAA of Geological Samples From . rchena Island to Analyze Each D.:ernable Stratigraphic Horizon in Addition to Xenoliths Transported to the Surface by Pyroclastic Eruptions.	USDOE
50	G. Goles R. A. Schmitt	Geology, U of O	Columbia River Basalt Stratigraphy and Petrogenesis	Continuation of Studies of Stratigraphic Succession and Petrogenetic Origins of Columbia River Basalt Flows	USDOE
51	G. Goles R. A. Schmitt E. Vicenzi	Geology, U of O	Geochemistry Of Paracutia Volcano (Mexico)	INAA of the Samples From the Paracutia Volcano in Mexico to Determine the Causes of the Abrupt Shifts in Lava Type and Composition	USDOE
52	G. Goles R. A. Schmitt R. Barry	Geology, U of O	Highwoods Analysis	INAA of Bulk Rocks & Mineral Separates From Pseudo-Leucite-Bearing Intrusive Rocks From The Highwood Mountains, MT	USDOE

Table 4.7
Summary of Visitors to the OSTR
July 1, 1984 through June 30, 1985

Date	No. of Visitors	Name
07-26-84	15	Adventures in Learning
07-31-84	10	English Language Institute
10-16-84	10	OSU Chemistry Dept. (Chem. 419)
11-01-84	40	OSU Chemistry Dept. (Chem 107/207)
11-08-84	40	OSU Chemistry Dept. (Chem 107/207)
11-26-84	7	Sichuan University, PRC
01-29-85	22	Tigard High School Science Students
01-30-85	12	English Language Institute
02-02-85	40	Beaver Open House
01-11-85	12	English Language Institute
02-16-85	82	Dad's Weekend Open House
02-20-85	15	Coquille High School Physics Class
03-01-85	6	Oregon Episcopal School
03-11-85	12	Problems in Safety Health 482
03-22-85	10	Community College Science Advisors
04-11-85	20	LBCC Physical Science Class
04-20-85	10	Albany High School Science Students
04-25-85	7	University of Oregon (Geol. 420)
04-25-85	~105	E-Spree Day High School Students
04-30-85	24	OSU Chemistry Dept. (Chem 206)
05-04-85	45	Moms' Weekend Open House
05-29-85	18	Westernview Intermediate School
06-04-85	12	Fujan Higher Ed. Delegation
06-13-85	25	Up With People
	769	

PART 5

OSTR ENVIRONMENTAL AND RADIATION PROTECTION DATA

Table of Contents

PART 5

OSTR ENVIRONMENTAL AND RADIATION PROTECTION DATA

(July 1, 1984 through June 30, 1985)

List of Tables

List of Figures

PART 5

OSTR ENVIRONMENTAL AND RADIATION PROTECTION DATA
(July 1, 1984 through June 30, 1985)

A. INTRODUCTION

The data contained in this section have been prepared to comply with the current requirements of Nuclear Regulatory Commission (NRC) Facility License No. R-106 (Docket No. 50-243) and the Technical Specifications contained in Appendix A to that license. The material has also been prepared in compliance with Oregon Department of Energy Rule No. 345-30-010, which requires an annual report of environmental effects due to research reactor operations.

Within the scope of Oregon State University's radiation protection program, it is standard operating policy to maintain all releases of radioactivity to the unrestricted environment and all occupational exposures to radiation and radioactive materials at levels which are consistently "as low as reasonably achievable" (ALARA).

B. A SUMMARY OF THE NATURE AND AMOUNT OF RADIOACTIVE EFFLUENTS RELEASED OR DISCHARGED TO THE ENVIRONS BEYOND THE EFFECTIVE CONTROL OF THE LICENSEE, AS MEASURED AT OR PRIOR TO THE POINT OF SUCH RELEASE OR DISCHARGE.

1. Liquid Waste (Summarized on a Monthly Basis)

   a. The radioactivity in liquid waste discharged during the applicable reporting period has been summarized according to the following items. All liquid waste data pertaining to these items are contained in Table 5.1.

      1) The total estimated quantity of radioactivity released (to the sanitary sewer) (in curies).

      2) The detectable radionuclides present in this waste.

3) An estimate of the specific activity for each detectable radionuclide present, if the specific activity of the released material after dilution was greater than 1 x 10^{-7} microcuries/cubic centimeter.

4) A summary of the total release (in curies) for each radionuclide identified in 2) above for the reporting period, based on representative isotopic analysis.

5) The estimated average concentration of the released radioactive material at the point of release for the reporting period (in terms of microcuries/cubic centimeter) and the fraction of the applicable MPC value.

b. The total volume in gallons of effluent water (including diluent) released during each period when liquid waste was released also has been summarized in Table 5.1.

2. Gaseous Waste (Summarized on a Monthly Basis)

a. The radioactivity in gaseous waste discharged during the applicable reporting period has been summarized according to the following items. All gaseous waste data pertaining to these items are contained in Table 5.2.

1) The total estimated quantity of radioactivity released (in curies) determined by an appropriate sampling and counting method.

2) The detectable radionuclides present in this waste.

3) The total estimated quantity of argon-41 released during the reporting period (in curies), based on data from an appropriate monitoring system.

4) The estimated average atmospheric diluted concentration of argon-41 released during the reporting period (in terms of microcuries/cubic centimeter) and the fraction of the applicable MPC value.

5) The total estimated quantity of radioactivity in particulate form with half-lives greater than eight days released during the reporting period (in curies), as determined by an appropriate particulate monitoring system.

6) The average concentration of radioactive particulates with half-lives greater than eight days released during the reporting period (in microcuries/cubic centimeter).

7) An estimate of the average concentration of other significant radionuclides present in the gaseous waste discharge (in terms of microcuries/cubic centimeter) and the fraction of the applicable MPC value for the reporting period, if the estimated release was greater than 20% of the applicable MPC.

3. Solid Waste (Summarized on an Annual Basis)

a. The radioactivity in solid waste discharged during the applicable reporting period has been summarized according to the following items. All solid waste data pertaining to these items are contained in Table 5.3.

1) The total amount of solid waste packaged (in cubic feet).

2) The detectable radionuclides present in this waste.

3) The total radioactivity in the solid waste (in curies).

b. The dates of shipment and the disposition of solid wastes (if shipped offsite) also have been listed in Table 5.3.

C. AN ANNUAL SUMMARY OF THE RADIATION EXPOSURE RECEIVED BY FACILITY PERSONNEL AND BY VISITORS, IN TERMS OF THE AVERAGE RADIATION EXPOSURE PER INDIVIDUAL AND THE GREATEST EXPOSURE PER INDIVIDUAL FOR EACH OF THE TWO GROUPS.

The annual summary of the radiation exposure received by facility personnel and by visitors for the applicable reporting period has been included in Table 5.4.

D. AN ANNUAL SUMMARY OF THE RADIATION LEVELS AND THE LEVELS OF CONTAMINATION OBSERVED DURING ROUTINE SURVEYS PERFORMED AT THE FACILITY, IN TERMS OF THE AVERAGE AND THE HIGHEST LEVELS.

The annual summary of radiation and contamination levels observed during routine facility surveys for the applicable reporting period has been included in Table 5.5.

E. THE LOCATION AND MAGNITUDE OF THE MAXIMUM MEASURED OR CALCULATED DIRECT RADIATION LEVEL IN UNRESTRICTED AREAS DUE TO DIRECT RADIATION FROM THE FACILITY AND DIRECT RADIATION FROM FACILITY EFFLUENTS

1. The Maximum Direct Radiation Level in Unrestricted Areas Due to Direct Radiation from the Facility

The location and magnitude of the maximum (measured and calculated) direct radiation level in an unrestricted area due to direct radiation from the facility can best be understood by referencing Figures 5.1 and 5.2, and Tables 5.6 and 5.7.

Early in the operating history of the OSU TRIGA reactor, two potential sources of direct radiation from the TRIGA facility were identified. These were the demineralizer tank for the reactor primary water system and the graphite-natural uranium subcritical pile located in the main reactor room (see Figure 5.1).

On January 3, 1972, the demineralizer tank was removed from its original position, shown in Figure 5.1, and was moved to location "A" in Figure 5.1. Henceforth, it ceased to be a major contributor to the direct radiation from the facility. On February 23, 1972, the east side (the exterior wall side) of the subcritical pile and the entire demineralizer tank were conservatively shielded with concrete and lead, further limiting any small direct radiation contribution from the demineralizer tank and effectively reducing the direct radiation to unrestricted areas from both the subcritical pile and the demineralizer tank to essentially zero millirem per year.

With the elimination of the preceding two sources of direct radiation from the facility, two additional sources of lesser magnitude became apparent. One of these was the particulate filter for the reactor primary water system, which is located on the demineralizer platform (see Figure 5.1), while the second is best collectively termed "normal use of reactor experimental facilities and operating areas for research and teaching."

The particulate filter was completely shielded by July 10, 1972, and the new shield eliminated any further radiation contribution in unrestricted areas from this source. The second source, relating to normal use of the OSU research reactor, takes into consideration the routine handling of radioactive materials within the entire facility, and the need for relatively frequent access into reactor experimental and irradiation facilities. Both of these latter activities create a small potential for very low level direct radiation exposure (of reactor facility origin) in immediately adjacent unrestricted areas.

Surveillance over direct radiation levels in unrestricted areas (which potentially may arise from inside the TRIGA facility) is maintained by utilizing four different types of routine area radiation measurements. These measurements include data from a continuously operating 14-channel area radiation monitoring system inside the facility, with remote detector stations located throughout the TRIGA operating area, plus results obtained by numerous (and in many cases, daily) on-the-spot direct radiation measurements made inside and outside the facility by members of the radiation protection staff during TRIGA operations. In addition, data from integrating area monitoring composite dosimeters installed at strategic locations within the TRIGA reactor operating area are routinely documented and are used to indicate locations where direct radiation from the facility might be entering unrestricted areas. The composite dosimeter data can be corrected, as needed, to reflect radiation attenuation in the reactor facility walls. Finally, assessment of direct radiation levels in unrestricted areas is conducted on the basis of area monitoring data collected through our thermoluminescent dosimetry (TLD) program. Most of the TLDs in this program are actually located in unrestricted areas, or are located on the TRIGA reactor area fence which surrounds the accessible sides of the TRIGA reactor building (see Figure 5.2). Therefore, these monitors are a valuable source of information for assessing the likelihood of radiation doses in unrestricted areas due to direct radiation from the TRIGA facility.

The specific locations of pertinent vendor supplied area monitoring composite dosimeters (consisting of a beta-gamma film plus a CR-39 polycarbonate plastic track-etch neutron dosimeter) inside the restricted operating area at the OSU TRIGA facility are shown in detail in Figure 5.1, and are again shown as part of an overall area diagram in Figure 5.2. For the period involved in this year's report, there was one vendor for the composite dosimeters, namely Radiation Detection Company (R.D. Co.), Sunnyvale, California.

Figure 5.2 also identifies the area monitoring stations on the reactor area fence. For the 1984-85 reporting period, each of these fence stations utilized OSU supplied and processed TLD area monitors (normally three Harshaw ^{7}LiF TLD-700 chips per monitor) and Radiation Detection Company (R.D. Co.) $CaSO_4$ TLD area monitors (2 $CaSO_4$ TLDs per monitoring packet). The R.D. Co. TLDs were started during the 1977-78 reporting period to replace R.D. Co. beta-gamma-neutron film packs, which were used previously at the reactor fence monitoring stations. R.S. Landauer Co. TLD monitors were started July 1, 1981 and were used throughout the 1981-82 reporting period along with the R.D. Co. TLD monitors. Because the results from the two vendors were very similar, in July of 1982 we returned to R.D. Co. as our only vendor of TLD monitors for the fence and other offsite environmental monitoring stations. OSU will also continue to supply and process TLD area monitors as we have in the past.

With the addition of the fence around the reactor area (in September of 1972), area monitoring data from inside the TRIGA facility (contained in Table 5.6) no longer have a high degree of correlation to direct radiation levels in surrounding unrestricted areas. Nevertheless, we believe the data from inside the facility reflect the general character of our operation and therefore we plan to continue including it in all reports of this type.

In Figures 5.1 and 5.2 composite dosimeter area monitoring locations within the TRIGA reactor facility are abbreviated to indicate their position on a north, south, east, or west wall of the main reactor bay, or their locations in the reactor's adjacent heat exchanger room. For example, MRCTSE is interpreted as Monitor

Radiation Center TRIGA, South dosimeter, East wall of the main reactor bay building. Similarly, MRCTHXS is the dosimeter for the adjacent Heat EXchanger room, South wall. Monitoring locations on the reactor area fence are simply designated MRCFE-1 through MRCFE-9, and imply Monitor Radiation Center Fence Environmental, followed by the monitoring station number.

After the addition of the previously described shielding inside the reactor facility and the addition of the reactor area fence, direct radiation levels in unrestricted areas due to the TRIGA facility dropped to essentially background radiation levels. Data presented in Table 5.6 show the generally low annual doses recorded inside the reactor facility's operating area. Likewise, Table 5.7 provides information regarding the low annual doses at the reactor area fence, based on three different groups of radiation measurements. Specifically, Table 5.7 includes results from the two different groups of area monitoring TLDs which were located on the reactor area fence, plus data from the direct micro-roentgen-per-hour exposure rate measurements collected at each fence area monitoring station. See footnote (6) of Table 5.7 for a further explanation of the μR/hr data and its application. Most importantly, however, use of the data in Tables 5.6 and 5.7 demonstrates that the total annual radiation dose in the unrestricted areas adjacent to the reactor area fence generally falls within the rather broad dose range typically expected for natural background radiation in Oregon.

As a final note on the fence monitoring stations, it should be pointed out that throughout the full year period covered by this report there was no continuous occupancy, and little or no occupancy of any duration, in the unrestricted area along the entire perimeter of the reactor facility fence. In addition, there was only very intermittant, short duration occupancy in the area between the reactor fence and the reactor building.

OSU has continued its efforts to achieve close agreement between R.D. Co. (our regular vendor for area monitoring and environmental TLDs) and OSU TLD data. Present control and QA procedures used by OSU for its outside dosimetry vendor will continue to be carefully reviewed, and will be modified as deemed necessary.

In line with our commitment to maintain surveillance over vendor-supplied TLD results, for the 1981-82 reporting year OSU tried an additional TLD vendor (R.S. Landauer, Jr. and Co.) for area and environmental monitoring. To facilitate comparison of TLD results between vendors and to prevent loss of data from the monitoring locations currently being reported, it was considered necessary to use TLDs from the new vendor along with those from R.D. Co.

As indicated earlier, TLD results from the two vendors appeared to be very similar and were normally within the statistical range of values obtained by OSU. Because of this, OSU continued to use only one outside TLD vendor (Radiation Detection Co.) for the 1984-85 reporting period.

2. The Maximum Direct Radiation Level in Unrestricted Areas Due to Direct Radiation from the Facility Effluents.

The location and magnitude of the maximum (measured and calculated) direct radiation level in unrestricted areas due to direct radiation from facility effluents will be reviewed in light of both liquid and gaseous releases.

As reported in Table 5.1, the total annual quantity of radioactivity released in liquid effluents has been quite small. The microcurie quantity for the reporting period in even a few hundred cubic centimeters of solution would not normally present a significant direct radiation potential, particularly when the radionuclide composition of the radioactivity is examined. In our particular operation, the majority of the liquid radioactive effluent is now normally associated with a single annual demineralizer resin change. When released from the reactor facility, potentially radioactive liquid is mixed in a holdup tank on a batch basis with up to 3,000 gallons of waste water from the Radiation Center laboratories before final discharge into the unrestricteed area (the sanitary sewer system).

The annual average concentration for total reactor facility radioactivity in liquid effluents entering the unrestricted area equaled 7.26×10^{-6} µCi/cc for the year of July 1, 1984 through

June 30, 1985. With respect to this value and the total radioactivity released in the liquid effluent, recall that no city water background radioactivity has been substracted. Also, note that the main contributor to the total microcuries released was tritium (65.0 μCi of tritium out of a total of 65.03 μCi released). Even though nearly all of the liquid effluent volume from the reactor facility originated during the annual changing of the demineralizer resins, it appears that little of the tritium was of reactor origin. Our routine analysis of Corvallis city water over the past four years indicates a normal variation in tritium background concentration within a range of 1.70×10^{-5} μCi/cc to 1.92×10^{-6} μCi/cc for the years July 1, 1981 through June 30, 1985. Our annual average concentration for tritium based on all liquids released to the unrestricted area from the reactor facility was within this range at 7.26×10^{-6} μCi/cc for the year July 1, 1984 through June 30, 1985. If the tritium is omitted from the total radioactivity released in the reactor's liquid effluent and a calculation performed using the remaining radioactivity ($\sim$0.03 μCi), some of which was also possibly city water background, the annual average concentration for reactor facility radioactivity entering the unrestricted area becomes 3.35×10^{-9} μCi/cc for the year July 1, 1984 through June 30, 1985.

In view of the radionuclides present, and the relative abundance of each, it can be easily determined (as shown in Table 5.1) that the annual average concentration of total reactor facility radioactivity in reactor related liquid effluents represents but a small fraction (0.25% or 0.0075%--see Table 5.1) of the applicable unrestricted area concentration limits. In addition, the average μCi/cc concentrations stated above DO NOT take into consideration the additional mixing of these liquid discharges with approximately 95,000 to 115,000 gallons per year of liquids and sewage normally discharged by the Radiation Center complex into the same sanitary sewer system. For these reasons, we have concluded that the direct radiation to unrestricted areas due to radioactivity in reactor liquid effluents has been negligible.

On pages 4-53 through 4-58 of the Safety Analysis Report (SAR) for the OSU TRIGA Research Reactor, dated August 1968, consideration is given to the routine discharge and atmospheric dilution of gaseous effluents from the reactor facility. In this particular analysis in the 1968 SAR, the evaluations were conducted using the original TRIGA facility stack height of 55 feet above ground level. Furthermore, on pages 4-53 and 4-57 of the 1968 SAR it is specified that the continuous radioactivity discharge rate assumed for the purpose of evaluating effluent dilution and for dose calculations was 100 MPC, meaning 100 times the normal 4×10^{-8} µCi/cc argon-41 unrestricted area maximum permissible concentration, or a continuously released concentration of argon-41 equaling 4×10^{-6} µCi/cc. On page 4-58 (Table 4.11 of the 1968 Safety Analysis Report), it is concluded that under the most unfavorable atmospheric conditions (with the 55-foot stack) a person standing on the ground for a full year at the point of maximum argon-41 concentration would be exposed to less than 9% (8.53%) of the normal unrestricted area MPC for argon-41. As a result, a person could stand on the ground at that point continuously for one year (year after year) under the most unfavorable atmospheric conditions, while the reactor operated continuously at 1000 kW, (and continuously discharged an assumed worst case concentration of 4×10^{-6} µCi/cc of argon-41) and receive a whole body gamma dose from argon-41 of less than 45 mrem (42.6 mrem) integrated over an entire year's occupancy.

Since the OSU TRIGA does not operate on a 24-hour per day basis, does not operate continuously at 1000 kW while it is operating, and does not discharge argon-41 at 4×10^{-6} µCi/cc while operating at 1000 kW, the annual average argon-41 concentration, as measured by the facility stack monitor, has always been much less than the assumed calculational value of 4×10^{-6} µCi/cc. Consequently, the maximum annual dose in the unrestricted area due to direct radiation from gaseous effluents has also been significantly less than the nominal 45 mrem per year value conservatively projected in the 1968 Safety Analysis Report.

As indicated in OSU's May 16, 1973 report of 10 CFR 50.59 items to the former USAEC Division of Reactor Licensing, (a copy of which also went to the former Oregon Nuclear and Thermal Energy

Council) on February 23, 1972, the TRIGA facility stack height was increased from its original 55 feet above the ground to 65 feet, 10 inches above ground level. As a result of the new stack height, new atmospheric dispersion calculations were necessary in order to evaluate the atmospheric dilution of gaseous effluents from the reactor facility and to predict the corresponding radiation doses in the unrestricted area from these diluted effluents. The results of the original 1968 atmospheric dilution and dose calculations and the first such evaluation following the stack height change were included in Table 2 of OSU's May 16, 1973 10 CFR 50.59 report to the USAEC. The data in this report indicated a slightly lower concentration at the point of maximum concentration on the ground because of the higher stack. Additional plume dispersion studies during 1973 and 1974, and again during 1978 using USNRC Regulatory Guide 1.111, evaluated the influence of the new stack height on gaseous effluent dispersion and corresponding dose, and essentially confirmed earlier data. Only a slight change is introduced if the most unfavorable values from the expanded 1973-74 study and the newer 1978 study are used.

Using the same basic assumptions employed for the shorter stack and, in particular, a continuous argon-41 discharge rate of 100 MPC, the 1973-74 results indicated that for atmospheric conditions giving the highest ground concentration of argon-41 (i.e., the worst atmospheric conditions) a person standing at the point of maximum concentration on the ground would encounter approximately 3.018% (as opposed to 2.85% in the 1972 report) of the unrestricted area MPC for argon-41. Furthermore, the 1978 study produced a nearly identical value of 3.005% of the unrestricted area argon-41 MPC. As a result, a person could stand at this point of maximum concentration on the ground (currently projected to be 130 meters from the stack as opposed to 150 meters in the 1973-74 calculation, and 135 meters in the 1972 report) continuously for one year under the worst atmospheric conditions, while the reactor continuously discharged 100 times the argon-41 MPC, (i.e., 4×10^{-6} µCi/cc) and receive a whole body gamma dose from argon-41 of 15 (15.03) mrem integrated over an entire year's occupancy.

As we have indicated, the OSU TRIGA does not operate on a 24-hour per day basis nor does it operate continuously at 1,000 kW. Also, the facility's annual average argon-41 concentration is always much lower than the 4×10^{-6} µCi/cc value used for purposes of calculation. As a result, the maximum annual dose in the unrestricted area due to direct radiation from gaseous effluents of reactor origin consistently remains much less than the nominal 15 mrem per year dose projected by using the new stack height and the 1978 plume dispersion data.

In order to arrive at the most conservative estimate of the maximum dose in the unrestricted area due to direct radiation from OSTR gaseous effluents during the reporting period, one should assume continuous annual occupancy at the point of maximum argon-41 concentration on the ground. Furthermore, it will be necessary to assume the existence of the most unfavorable meteorological conditions for a full year in order to achieve the maximum concentration at the specified point for one entire year. If these highly conservative (and, for all practical purposes, impossible) assumptions are applied in conjunction with the reported annual average argon-41 concentration (5.70×10^{-8} µCi/cc), as derived from actual measurements at the point of release with the facility's continuous stack monitor (see Table 5.2), then the maximum annual dose in the unrestricted area (at 130 meters from the stack under the most unfavorable atmospheric conditions) would be approximately 0.21 mrem for the year July 1, 1984 through June 30, 1985.

Even in the presence of an essentially negligible argon-41 dose contribution in the unrestricted area, which has been repeatedly verified by using the most conservative assumptions in dose calculations and by a longstanding environmental monitoring program in the unrestricted area, OSU chose to implement several actions during the 1981-1982 reporting period (and during later periods) designed to minimize argon-41 release even further, and to clearly demonstrate our continuing commitment to the as-low-as-reasonably-achievable (ALARA) concept relative to argon-41 discharge. As expected, these measures, and others implemented later, proved effective in reducing even further the barely detectable concen-

tration of argon-41 from the OSTR, but had no measurable impact on detectable dose in the environment since this was already well below detection thresholds. Nevertheless, during the current reporting period the argon-41 released and the corresponding theoretical dose commitment in the unrestricted area was lower. As noted in Table 5.2, Footnote 3, the total argon-41 released during this reporting period decreased by about 25.7% with respect to last year's value, although the megawatt-days of operation during both reporting periods were nearly the same (1984-85 showed a 0.3% increase in megawatt-days).

The observed reduction in argon-41 released during this reporting period was due to a full year's use of several 1981-82, 1982-83 and 1983-84 modifications in the reactor's experimental facilities ventilation system (usually termed the argon ventilation system). The two changes which had the greatest impact on minimizing argon-41 releases were the addition of a nitrogen purging system to the ventilation scheme for the rotating rack facilty and the reduction of room air flow through the thermal column. In the first situation, nitrogen gas from a liquid nitrogen tank flows through the rotating rack in place of normal room air; while in the case of the thermal column, ventilation flow valve adjustments were made which resulted in much less room air being passed through this facility. Also, during the 1983-84 reporting period, much more sensitive devices for indicating and adjusting air flow were installed on the ventilation lines for the thermal column and beam ports 1 and 3, which allowed much better adjustment of the (room) air flow through these facilities and a further reduction in argon-41 release. Since the rotating rack, the thermal column and the two beam ports currently in use represent the major potential sources of argon-41 generation due to neutron activation of argon-40 in normal atmospheric air in these facilities, the elimination or reduction of this target argon isotope significantly reduces the already vey low argon-41 production and release from the OSTR.

Another modification to the entire argon ventilation system (during the 1982-83 interval) added a positive exhaust-isolation valve into the system's discharge line. This valve was installed

at a point just before the argon vent line joins the main reactor building exhaust stream, and the valve is normally closed at the end of the operating day when the argon ventilation system is shut off. Closure of this valve eliminates any further escape of argon-41 from this system, and since argon-41 has a 1.83 hour half-life, it decays overnight while the argon ventilation system is shut off and secured.

A similar exhaust-isolation valve was also added to the pneumatic-transfer system discharge line (in the 1982-83 interval) just before it joins the main reactor building exhaust stream. This valve will also be kept closed, except during times when the pneumatic-transfer system is in use. Keeping the valve closed eliminates a small induced air flow through the system caused by a lower pressure in the reactor building exhaust stream. Elimination of this air flow through the pneumatic-transfer system's in-core region eliminates any possible argon-41 production during periods of non-use, and thereby contributes even further to the reduction in total argon-41 released.

F. A DESCRIPTION OF THE GENERAL METHODS USED AND AN ANNUAL SUMMARY OF THE RESULTS OF ENVIRONMENTAL SURVEYS PERFORMED OUTSIDE THE FACILTIY

The overall environmental radiation monitoring effort will be categorized according to "onsite" and "offsite" environmental radiation monitoring programs. A description of the two environmental monitoring program categories follows.

1. The Onsite Environmental Radiation Monitoring Program

The onsite environmental radiation monitoring program currently incorporates the TRIGA facility radioactivity stack monitor, the onsite area monitoring composite dosimeters (consisting of a film badge and a CR-39 plastic track-etch neutron detector), the onsite area monitoring TLDs, the onsite 0-200 mrem gamma-sensitive integrating ionization chambers (self-reading pocket dosimeter type), and the monitoring procedures associated with the analysis of radioactivity in liquid effluents from the reactor facility. Also, routine (daily, weekly, and monthly) radiation surveys con-

ducted by the OSU TRIGA radiation protection staff provide a wealth of essential information on existing radiation conditions throughout the various onsite areas.

The reactor facility dual channel radioactivity stack monitoring system consists of a continuously-moving-filter-paper particulate monitoring channel, followed by a separate chamber and detector which function together as the gas monitoring channel. The system is consistently placed in operation before the reactor is started up, remains in operation at all times while the reactor is in use, and is kept operable after reactor shutdown until both detection channels reach normal background or it is determined by other means that no significant radioactivity is being discharged from the stack. The system is equipped with an isokinetic sampling head which draws its sample near the point of discharge in the reactor building stack. The particulate channel is calibrated at least annually with standardized particulate samples containing NBS traceable radioactivity of an appropriate type and energy, and the gas channel is calibrated at the same frequency with known quantities of argon-41 gas. The system reads out continuously in both the particulate and gaseous channels, with each channel having its own count rate meter and recorder. A count integrating scaler is also attached to the gas channel to increase the accuracy in determining the argon-41 released. The system is equipped with alarm circuits which will automatically shut off the facility ventilation system fans and close dampers on the intake and exhaust ventilation lines in the event preset airborne radioactivity concentration limits are reached. One of the most valuable applications of this system from the standpoint of environmental monitoring is the data derived from its operation which can be applied to determining potential exposures in unrestricted areas from gaseous radioactive effluents, namely argon-41.

Onsite area-monitoring composite dosimeters each consist of a standard personnel-type beta-gamma film pack and a CR-39 plastic track-etch neutron detector, located at strategic positions inside the reactor facility operating area (see Figures 5.1 and 5.2). The composite dosimeters within the facility are changed once per month.

During the 1984-85 reporting period, onsite area monitoring using TLDs consisted of two different types of dosimeters, both located at identical positions on the reactor area fence (see Figure 5.2). One type of TLD monitor was supplied and interpreted by Radiation Detection Company (R.D. Co.), Sunnyvale, California. This system utilized $CaSO_4$ TLDs (2 $CaSO_4$ TLDs per monitoring pack) prepackaged by R.D. Co. and exchanged on a quarterly basis. The R.D. Co. TLDs were placed in thin sheet metal boxes located on the reactor area fence, and were accompanied at each location by the second TLD monitoring package which was prepared and interpreted by OSU. Each OSU TLD monitoring device normally consisted of three lithium fluoride chips, presently Harshaw TLD-700s, exchanged on a quarterly basis.

Prior to April 1976, each OSU supplied onsite TLD monitoring device was first packaged in a plastic mount which was then placed inside an outer container made out of thin-walled copper tubing. The copper tube was subsequently taped to the reactor area fence. The plastic mount and copper container were essentially identical to those being used at the time by the Oregon Radiation Control Section in their TLD monitoring program.

In April of 1976, the copper tube outer containers were discontinued for the OSU-supplied TLDs on the reactor area fence. As an alternative, the original inner plastic mounts were placed inside the thin sheet metal boxes (mentioned above) which are located at each of the reactor area fence monitoring stations. This was done to reduce data-loss due to increasing theft of the small copper tube TLD packs. OSU and R.D. Co. TLD packs were located at each of the nine reactor area fence monitoring positions (identified in Figure 5.2) during the current reporting period.

In addition to the above radiation monitoring devices, each of the nine reactor area fence monitoring positions is presently (and will continue to be) equipped with two 0-200 mrem gamma-sensitive integrating ionization chambers (self-reading pocket dosimeter type). These dosimeters are located inside the thin sheet metal box at each fence monitoring station which also contains the pre-

viously mentioned TLD monitoring packets. The ionization chamber dosimeters are read every month and are used as backup monitors for each station.

For the July 1, 1984 through June 30, 1985 reporting period, an additional onsite environmental radiation monitoring effort was continued. This effort involved the monthly measurement of the direct gamma radiation exposure rate in terms of microroentgens per hour (μR/hr) at each reactor fence monitoring station. Measurements were taken with an Eberline Instrument Company micro-R per hour rate meter containing a 1" x 1" NaI detector. The monthly readings (normally 12 annually) were then averaged and ultimately converted to an expected (calculated) annual gamma dose equivalent (in mrem) for each location. (See footnote 6 to Table 5.7).

In terms of environmental radiation monitoring, the onsite area monitoring composite dosimeters, the reactor fence (onsite) TLDs and integrating ionization chambers, and the direct radiation exposure rate measurements at appropriate locations around the reactor fence can be used to estimate maximum potential radiation doses in nearby unrestricted areas. Normally, these estimates are made to reflect the annual radiation dose equivalent which could be delivered in the unrestricted area assuming continuous occupancy, although occupancy of unrestricted areas adjacent to the reactor facility is virtually zero throughout the year.

The routine onsite determination of the total radioactivity in liquid effluents (with isotopic identification as appropriate) prior to discharging the liquids into the unrestricted area (the sanitary sewer system) allows a conservative estimate to be made of the magnitude of the reactor facility contribution to potential environmental radiation exposures to the general public from this source.

2. The Offsite Environmental Radiation Monitoring Program

The offsite environmental radiation monitoring program collectively includes a routine soil, water, and vegetation monitoring program, and an airborne gamma radiation monitoring program.

The soil, water, and vegetation monitoring effort centers around the collection of a limited number of samples in each category on a quarterly basis. It is conducted to monitor both the TRIGA reactor facility and the OSU Radiation Center, and is considered useful for indicating the general trend of the radioactivity concentration in each of the various substances sampled. See Figure 5.3 for the locations of the sampling stations for grass (G), soil (S), water (W), and rainwater (RW) samples.

The airborne gamma radiation environmental monitoring program is generally described on pages 4-59 and 4-60 of the August 1968 Safety Analysis Report for the OSU TRIGA Reactor. As of January 1, 1975, nine additional offsite airborne gamma monitoring stations were put into service, to increase the total number of these stations now in use to 19. See Figure 5.3 for the locations of the 19 airborne gamma radiation monitoring stations, each designated as a TE location on the figure.

As of January 1, 1975, the coding technique used to designate each specific offsite airborne gamma radiation monitoring station was modified slightly to indicate the radiation monitoring devices present at a particular station. Under the new coding system, stations which contain only a standard OSU TLD monitoring pack (described previously in this report) will have an "L" after the station number. For example, MRCTE-2L is interpreted as Monitor Radiation Center TRIGA Environmental Station Number 2, with a standard OSU TLD pack in a copper tube being the only monitoring device at this station. (NOTE: The copper tube outer container is still used for all OSU TLD packs employed in the offsite environmental monitoring program. They were discontinued only for the OSU TLDs used on the reactor area fence). At offsite stations where only an OSU TLD monitor is used, the copper tube containing the TLDs is taped directly onto a mounting post or other permanent object used to identify the monitoring station. Stations which have no "L" after the station number consist of a thin weather-tight aluminum box mounted on a post about four feet off the ground. For the 1984-85 reporting period, each of these stations contained one R.D. Co. TLD pack, one standard OSU copper tube TLD monitoring

pack identical to those previously described, and two 0-200 mrem gamma-sensitive integrating ionization chambers (self-reading pocket dosimeter type) as backup monitors. At these stations, the OSU TLDs were not enclosed inside the aluminum box, but instead the copper tube was taped directly onto the box mounting post at the station. All TLD monitors in the offsite airborne gamma radiation environmental monitoring program were exchanged on a quarterly interval. The ionization chambers (dosimeters) were read once every month throughout the year.

For the July 1, 1984 through June 30, 1985 reporting period, the previously described effort involving monthly onsite measurements of the direct gamma radiation exposure rate in μR/hr at each reactor fence monitoring station was extended for another year to include each of the 19 offsite airborne gamma radiation monitoring stations. The data were handled as described in Footnote (7) to Table 5.9, and the objective was to derive an expected (calculated) annual gamma dose equivalent (in mrem) for each monitoring location, based on an annual average μR/hr exposure rate.

3. An Annual Summary of Onsite and Offsite Environmental Radiation Monitoring Results

A summary of the environmental radiation monitoring results for the year July 1, 1984 through June 30, 1985 is given below. As appropriate for the measurement under consideration, the results include:

a. The number of sampling locations;
b. The total number of samples per year;
c. The annual average concentration of total radioactivity, and in some cases, concentrations of specific radionuclides in the substance being assayed; and
d. The total annual millirem of external radiation dose for a particular location, as well as a general description of that location.

The data from the onsite and offsite environmental radiation monitoring programs will be presented under headings which correspond to the specific individual program components. These components were identified previously in conjunction with a description of the onsite and offsite environmental radiation monitoring programs.

a. Reactor Facility Stack Monitor (Onsite)

1) The system has one sampling location as indicated previously.

2) Samples are continuous (i.e., prior to, during, and after reactor operation). It is normal for the stack monitor to begin operation as one of the first systems in the morning and to cease operation as one of the last systems at the end of a normal operating day.

3) The annual average concentration of gross radioactivity based on the facility stack monitor is given in Table 5.2. As indicated in this table, the only gaseous component identified has been argon-41, and only naturally occurring particulate radioactivity (radon daughter products) has been detected by the particulate channel. During the reporting period, the normal concentration for the naturally occurring particulate daughters of radon remained about the same as in previous years, and was within a range of 1.38×10^{-9} μCi/cc to 3.22×10^{-11} μCi/cc.

b. Reactor Facility Area Monitoring Composite Dosimeters, Reactor Fence TLDs and Integrating Ionization Chambers, and Direct Radiation Measurements (Onsite)

1) There were eight applicable area monitoring composite dosimeters within the TRIGA reactor facility operating area. There were also nine R.D. Co. supplied $CaSO_4$ TLD monitoring packs, nine standard OSU TLD monitoring packs, and 18 (two per station) 0-200 mrem gamma-sensitive integrating ionization chambers on the reactor area fence. There were also nine specific locations (the fence monitoring stations) where routine monthly μR/hr measurements were made. All of these have application as onsite environmental radiation monitors.

2) Since the beta-gamma sensitive component (i.e., film) in each composite dosimeter within the TRIGA facility was changed once per month, there was a total of 96 different samples of this type during the reporting period. Since the neutron sensitive component in this dosimeter package was changed quarterly, there was a total of 32 different

samples of this type. Quarterly changes of the fence TLD monitors resulted in another 72 R.D. Co. supplied TLD samples (9 stations x 2 $CaSO_4$ TLD monitors per station x 4 changes per year = 72 samples), and 108 OSU TLD samples (9 stations x 3 TLD chips per station x 4 changes per year = 108 samples) for these locations for the reporting period. The 18 integrating ionization chambers were read once every month and thus resulted in approximately 216 samples (readings) for the reporting period. There are normally a total of 12 μR/hr measurements made at each of the nine fence monitoring stations during the year for a total of approximately 108 such measurements in the reporting period.

3) TRIGA operating area sampling locations are identified in Figure 5.1, with composite dosimeters being located on the inside of the indicated walls at approximately head height above the floor. Locations of the dosimeter packets are coded Monitor Radiation Center TRIGA North badge, East wall (MRCTNE) and so on. Locations for the TRIGA operating area composite dosimeters plus locations of the radiation monitors on the reactor area fence are shown in Figure 5.2. Fence monitoring locations are coded Monitor Radiation Center Fence Environmental, location 1 (MRCFE-1) and so on through MRCFE-9. TLD monitors on the fence are in sealed moisture-resistant packages inside thin sheet metal mailboxes about four feet off the ground. The integrating ionization chambers are also contained in the metal boxes. Annual radiation dose equivalent totals for the onsite area monitoring locations are given in Tables 5.6 and 5.7.

c. Analysis of Reactor Contributed Radioactivity in Liquid Effluents (Onsite)

1) TRIGA liquid effluent is analyzed at the time it is released to a collection point (a holdup tank), and is analyzed again in conjunction with other radioactivity from the

Radiation Center, which also may be present in the holdup tank prior to discharge from this collection point into the unrestricted area (the sanitary sewer system).

2) The total number of samples were as follows: two reactor liquid effluent samples for the period July 1, 1984 through June 30, 1985, in conjunction with two separate releases of reactor liquid effluent into the holdup tank collection point; and one liquid effluent sample from the holdup tank for the period July 1, 1984 through June 30, 1985 before the single release of the liquid effluent in the holdup tank collection point to the unrestricted area (the sanitary sewer).

3) The liquid effluent data for environmental assessment have been summarized for the reporting period in Table 5.1. Section 5.E.2 of this report also addresses the virtually undetectable level of direct radiation from this liquid and the negligible potential dose commitment in the unrestricted area from the radioactivity in this liquid effluent.

d. Soil, Water, and Vegetation Monitoring (Offsite)

1) For this environmental monitoring effort there are now a total of 22 sampling locations; four soil locations, four water locations (when water is available), and 14 vegetation locations.

2) Samples (as available) are taken at each location on a quarterly basis. Samples have been collected as follows for the period July 1, 1984 through June 30, 1985:

Total number of samples	84
Total number of soil samples	16
Total number of water samples	12*
Total number of vegetation samples	56

*(During the reporting period there were four instances when a collection point was dry).

3) The annual average concentration of total net beta radioactivity (minus ^{3}H) for samples collected at each offsite environmental soil, water, and vegetation sampling location (sampling

station) is given in Table 5.8. Identification of specific radionuclides is not routinely carried out as part of this program, but would be conducted if unusual radioactivity levels above natural background were evident. Locations of sampling points relative to the reactor facility are given in Figure 5.3, and as shown in this figure, most locations are within a 1,000-foot radius of the reactor building and the Radiation Center. In general, samples are collected over a local area having a radius of about ten feet at the positions indicated in Figure 5.3.

e. Airborne Gamma Radiation Monitoring (Offsite)

1) The offsite airborne gamma environmental radiation monitoring program currently utilizes 19 stations, and each station is considered a sampling location. For the 1984-85 reporting period, 11 stations had an R.D. Co. supplied $CaSO_4$ TLD monitoring packet, a standard OSU TLD monitoring pack, and two 0-200 mrem gamma pocket dosimeters. Eight stations had only a standard OSU TLD monitoring pack. In addition, each of the 19 monitoring stations were included in the ongoing program for measurement of the μR/hr exposure rate.

2) The TLDs at each airborne gamma monitoring station were changed once every calendar quarter resulting in a total of 88 R.D. Co. TLD samples during the 1984-85 reporting year (11 stations x 2 $CaSO_4$ monitors per station x 4 changes per year = 88 samples), and a total of 231 OSU TLD samples during the same period (18 stations x 3 TLD chips per station x 4 changes per year = 216 samples [minus 1 TLD dosimeter containing 3 chips for a one quarter interval, due to theft during the current reporting period], plus 1 station with 6 TLD chips x 4 changes per year = 24 samples [minus 1 TLD dosimeter containing 6 chips for a one quarter interval due to theft during the current reporting period] for a final total of 231 OSU samples for the

current year). The two backup monitors (integrating ionization chamber dosimeters) were read every month, which resulted in approximately 264 individual dosimeter readings for the reporting period. There are normally a total of 12 μR/hr measurements made at each of the 19 airborne gamma radiation monitoring stations each year for a total of approximately 228 individual measurements of this type annually.

3) Locations of the 19 airborne gamma radiation monitoring stations are shown in Figure 5.3. Like the soil, water, and vegetation sampling locations, most of the airborne gamma monitoring stations are within a 1,000-foot radius of the reactor building. These locations generally correspond to distances of interest with respect to the atmospheric (plume) dispersion results mentioned earlier in this report.

The dosimetry data from the airborne gamma radiation monitoring stations are summarized in Table 5.9, and are based on the vendor-supplied TLD data, the OSU TLD data, and results obtained from the μR/hr measurements. See Footnote (7) of Table 5.9 for a further explanation of the μR/hr data and its application.

The OSU in-house TLD program was started in 1974, and we believe a number of improvements have been made in the program since that time. There are, however, a few aspects which we continue to work on and some which may still require added refinement. In particular, we are continuing to study our background for the airborne gamma monitoring stations, because we feel that our reported average natural background value is occasionally lower than the natural background some stations are actually experiencing. We increased the number of background stations during 1976, and between July 1, 1984 and June 30, 1985 we continued to make a series of direct background measurements with our μR/hr monitoring equipment (started July 1, 1977) in order to obtain a better profile of the background variation. We plan to continue our surveillance of this variable.

From our viewpoint, the major purpose of the airborne gamma radiation monitoring stations is to give an indication of general increases or trends in unrestricted area radiation levels which might be linked to argon-41 released from the OSU TRIGA reactor. Past experience (over the last 15 years) has shown that annual results per location vary slightly from year to year. Although the data have not been included in this report, by following the mrem per year history for any single station and comparing the annual mrem total for that station to the curies of argon-41 emitted for the corresponding year, it becomes evident that there is no consistent correlation or pattern to the results and that other factors must be responsible for the minor mrem per year variations at the stations. For example, such variations may be the result of small, annual differences in cosmic or terrestrial background radiation, fallout, etc. In any event, the amount of argon-41 released annually (particularly in view of the very low concentrations involved and the significant decreases registered over the past three to four years) does not seem to be a significant factor which effects the total mrem per year measured (or predicted from μR/hr readings) for any particular monitoring station. A comparison of the data contained in Table 5.9 to past results from these monitoring stations, and a comparison to the values in Footnote (7) of Table 5.9, leads us to the firm conclusion that there has been no detectable or meaningful increase in the unrestricted area gamma radiation levels due to the very low concentrations of short-lived argon-41 released by the OSU TRIGA reactor during the defined reporting period.

6. REFERENCES

1. Eisenbud, Merril, Environmental Radioactivity, Second Edition, p. 190, Academic Press, New York, NY (1973).

2. U.S. Environmental Protection Agency, "Estimates of Ionizing Radiation Doses in the United States, 1960-2000," ORP/CSD 72-1, Office of Radiation Programs, Rockville, Maryland (1972).

3. U.S. Environmental Protection Agency "Radiological Quality of the Environment in the United States, 1977," EPA 520/1-77-009, Office of Radiation Programs; Washington, D.C. 20460 (1977).

Table 5.1

Monthly Summary of Liquid Waste Discharges
For the Year July 1, 1984 through June 30, 1985(1)

Date of Discharge (Month & Year)	Total Quantity of Radioactivity Released (To Sanitary Sewer) (Curies)	Detectable Radionuclides in the Waste	Specific Activity for Each Detectable Radionuclide in the Waste Discharged, Where the Released Concentration After Dilution Was $>1.0 \times 10^{-7}$ µCi/cc (µCi/cc)	Total Curies of Each Detectable Radionuclide Released in the Waste (Curies)	Average Concentration of Released Radioactive Material at the Point of Release (To the Sanitary Sewer) (µCi/cc)	Percent of Applicable MPC for Released Radioactive Material (%)	Total Volume of Liquid Effluent, Including Diluent, Released (To Sanitary Sewer) (Gallons)
JUL 84	None	None	Not Applicable	None	Not Applicable	Not Applicable	None
AUG 84	None	None	Not Applicable	None	Not Applicable	Not Applicable	None
SEP 84	None	None	Not Applicable	None	Not Applicable	Not Applicable	None
OCT 84	None	None	Not Applicable	None	Not Applicable	Not Applicable	None
NOV 84	None	None	Not Applicable	None	Not Applicable	Not Applicable	None
DEC 84	None	None	Not Applicable	None	Not Applicable	Not Applicable	None
JAN 85	None	None	Not Applicable	None	Not Applicable	Not Applicable	None
FEB 85	6.50×10^{-5}	^{3}H ^{152}Eu ^{60}Co	7.26×10^{-6} --- ---	6.50×10^{-5} 1.76×10^{-8} 1.42×10^{-8}	7.26×10^{-6}	0.25% (2) 0.0075%(3)	2366
MAR 85	None	None	Not Applicable	None	Not Applicable	Not Applicable	None
APR 85	None	None	Not Applicable	None	Not Applicable	Not Applicable	None
MAY 85	None	None	Not Applicable	None	Not Applicable	Not Applicable	None
JUN 85	None	None	Not Applicable	None	Not Applicable	Not Applicable	None
ANNUAL TOTAL	6.50×10^{-5}	See Above	Not Applicable	6.50×10^{-5}	7.26×10^{-6}	0.25% (2) 0.0075%(3)	2366(4)

(1) The OSU operational policy is to subtract only detector background from our water analysis data and not background radioactivity in the Corvallis city water.

(2) Based on values listed in 10 CFR 20, Appendix B, Table 2, Column 2.

(3) Based on values listed in 10 CFR 20, Appendix B, Table 1, Column 2, which are applicable to sewer disposal.

(4) The total volume of liquid effluent plus diluent does not take into consideration the additional mixing with approximately 95,000 to 115,000 gallons per year of liquids and sewage normally discharged by the Radiation Center complex into the same sanitary sewer system.

Table 5.2

Monthly Summary of Gaseous Waste Discharges for the Year

July 1, 1984 through June 30, 1985

Date of Discharge (Month & Year)	Total Estimated Radioactivity Released (Curies)	Total Estimated Quantity of Argon-41 Released(1) (Curies)	Estimated Average Atmospheric Diluted Concentration of Argon-41 at Point of Release (Reactor Stack) (μCi/cc)	Percent of the Applicable MPC for Diluted Concentration of Argon-41 at Point of Release (Reactor Stack) (%)	Total Estimated Quantity of Radioactivity in Particulate Form with Half-Lives >8 Days(2) (Curies)	Average Concentration of Radioactive Particulates Released With Half-Lives >8 Days (Curies)	Estimated Average Concentration of Other Significant Radionuclides in Discharge if >20% of the Applicable MPC (μCi/cc)	Percent of MPC if the Estimated Release was >20% of the Applicable MPC
JUL 84	0.68	0.68	4.85×10^{-8}	1.21	None	Not Applicable	Not Applicable	Not Applicable
AUG 84	0.83	0.83	5.92×10^{-8}	1.48	None	Not Applicable	Not Applicable	Not Applicable
SEP 84	0.98	0.98	7.20×10^{-8}	1.80	None	Not Applicable	Not Applicable	Not Applicable
OCT 84	0.94	0.94	6.68×10^{-8}	1.67	None	Not Applicable	Not Applicable	Not Applicable
NOV 84	0.96	0.96	7.09×10^{-8}	1.77	None	Not Applicable	Not Applicable	Not Applicable
DEC 84	0.87	0.87	6.22×10^{-8}	1.56	None	Not Applicable	Not Applicable	Not Applicable
JAN 85	1.06	1.06	7.57×10^{-8}	1.89	None	Not Applicable	Not Applicable	Not Applicable
FEB 85	0.57	0.57	4.52×10^{-8}	1.13	None	Not Applicable	Not Applicable	Not Applicable
MAR 85	0.58	0.58	4.16×10^{-8}	1.04	None	Not Applicable	Not Applicable	Not Applicable
APR 85	0.63	0.63	4.66×10^{-8}	1.17	None	Not Applicable	Not Applicable	Not Applicable
MAY 85	0.53	0.53	3.86×10^{-8}	0.97	None	Not Applicable	Not Applicable	Not Applicable
JUN 85	0.78	0.78	5.71×10^{-8}	1.43	None	Not Applicable	Not Applicable	Not Applicable
ANNUAL VALUE	9.41(3)	9.41(3)	5.70×10^{-8}	1.43	None	Not Applicable	Not Applicable	Not Applicable

(1) Routine gamma spectroscopy analysis of the gaseous radioactivity in the stack discharge indicated that it was virtually all argon-41.

(2) Evaluation of the detectable particulate radioactivity in the stack discharge confirmed its origin as naturally-occurring radon daughter products, predominantly lead-214 and bismuth-214, which are not associated with reactor operations.

(3) The 25.7% decrease in total argon-41 released during the current reporting period is due to several modifications designed to reduce the argon-41 effluent to as-low-as-reasonably-achievable levels. We anticipate that our argon-41 releases will remain at about the 1984-85 level.

Table 5.3

Annual Summary of Solid Waste Discharges

For the Year

July 1, 1984 through June 30, 1985

Total Amount of Solid Waste Packaged (Cubic Feet)	Detectable Radionuclides in the Waste	Total Quantity of Radioactivity in Solid Waste (Curies)	Dates of Shipment and Disposition(1)(2)
11.0	^{24}Sodium ^{51}Chromium ^{54}Manganese ^{58}Cobalt ^{59}Iron ^{60}Cobalt ^{65}Zinc ^{75}Selenium ^{124}Antimony ^{140}Lanthanum	9.46×10^{-4}	December 5, 1984

(1) OSTR solid radioactive waste is routinely transferred onsite (within the Radiation Center building) to the OSU Radiation Safety Committee, where it is held on the University's State of Oregon radioactive materials license, along with other campus waste, prior to shipment to U.S. Ecology.

(2) All solid radioactive waste is transferred by the University Radiation Safety Committee to our radioactive waste disposal service vendor, the U.S. Ecology Company, for burial at their installation located near Richland, Washington.

Table 5.4

Annual Summary of Radiation Exposure Received

By Facility Personnel and Visitors for the Year

July 1, 1984 through June 30, 1985

Personnel Group	Average Annual Exposure for Each Personnel Group		Greatest Individual Exposure per Personnel Group	
	Whole Body (mrem)	Extremities (mrem)	Whole Body (mrem)	Extremities (mrem)
Facility Operating Personnel	64.00(1)	111.00	380.00(1)	1090.00
Key Facility Research Personnel	1.00	2.00	25.00	70.00
Facility Visitors:				
Film Badges/TLDs	(2)	(2)	(2)	(2)
Pocket Dosimeters	<1.00	N/A(3)	10.00	N/A(3)

(1) The increased whole body exposure for the facility operating personnel during the 1984-85 reporting period can be attributed to the one-time search for a possible leaking TRIGA fuel element in January and February of 1985.

(2) OSU TRIGA reactor policy does not normally allow people in the visitor category to become actively involved in the use or handling of radioactive materials. Therefore, visitor whole-body film badges and extremity TLDs are not normally necessary, and no visitor data are available in these categories.

(3) Not applicable.

Table 5.5

Annual Summary of Radiation Levels and Contamination Levels Observed Within the Reactor Facility During Routine Radiation Surveys For the Year July 1, 1984 through June 30, 1985

Accessible Locations Within the Reactor Facility	Whole Body Radiation Levels (mrem/hr)($\beta\gamma$+neutrons)		Contamination Levels (dpm/100 cm^2)($\beta\gamma$)(1)	
	Average	Maximum	Average	Maximum
Reactor Top	5.70	160.00	<500(2)	3600(2)
General Reactor Room Area	<1.00	10.00	<500	<500
Beam Port Facilities (Reactor Room, First Floor)	<1.00	22.00	<500	<500
Sample Handling Area (Reactor Room, First Floor)	<1.00	12.00	<500	<500
Demineralizer Tank-Outside Shield (Reactor Room, First Floor)	<1.00	8.00	<500	<500
Class Experiments	1.50	37.00	<500	<500

(1) No contamination equal to or above the specified 500 dpm/100 cm^2 reporting limit was found during the entire reporting period at locations designated <500. The <500 dpm/100 cm^2 value used in this table was based on a consideration of the radionuclides likely to be present in any contamination detected, on the normal beta counting efficiency for such radionuclides when using the radiation survey instruments routinely employed at the OSTR to measure contamination in the field, and on a radiation survey result which showed the net counting rate to be less than the background counting rate, provided the background counting rate was ≤100 cpm (i.e., field contamination measurements would normally have to show a gross counting rate equal to at least twice the background counting rate, but generally never greater than 200 gross cpm, before contamination would be considered present). However, in addition to normal field screening for contamination by direct surveys and smear samples, smears of particular importance or interest and any smears otherwise suspected of containing removable radioactive contamination were routinely counted in a more sensitive radiation detection system. Based on usual counting times, a normal instrument counting efficiency, and a typical background counting rate, such a detection system generally showed a lower limit of detection (LLD) at 95% confidence of approximately 15 dpm for the radionuclides normally expected to be on smears if contamination was present. As a matter of initial conservatism, smearing efficiency for radioactivity removal is routinely assumed to be <100%. If warranted, after assessing each situation, positive smear results are multiplied by a factor to correct for smearing efficiency before final conversion to dpm/100 cm^2.

(2) The contamination shown for this location was a unique one-time situation and was removed immediately. As a result, during the reporting period the average contamination level on the reactor top was, for all practical purposes, <500 dpm per 100 cm^2.

Table 5.6

Total Dose Equivalent Recorded on Operating Area Composite Dosimeters Located Inside the TRIGA Reactor Facility for the Year July 1, 1984 through June 30, 1985

Location(1)	Total Recorded mrem for the Year July 1, 1984 through June 30, 1985 (2)(3)(4)
MRCTNE	15(7)
MRCTSE	<10(5)
MRCTSW	15
MRCTNW	<10
MRCTWN	<10
MRCTEN	30(7)
MRCTHXS	<10(6)
MRCTHXW	<10

(1) These locations do not represent radiation exposure through an exterior wall directly into an unrestricted area.

(2) Totals do not include natural background contribution.

(3) The beta-gamma dosimeter in each of the above composite dosimeters was exchanged on a monthly basis, while the neutron dosimetry packet was exchanged on a quarterly frequency.

(4) The total millirem values listed above are entirely due to gamma radiation, and reflect the summation of results from 12 different beta-gamma dosimeters for each location. Those listed at <10 mrem showed twelve separate readings below the vendor's gamma dose reporting threshold of 10 mrem. All July 1984 through June 1985 quarterly fast neutron doses were below R.D. Co.'s fast neutron detection threshold of 30 mrem.

(5) This is the composite dosimeter opposite the subcritical pile. This dosimeter primarily monitors the subcritical pile and the effectiveness of the subcritical pile shield. It is, therefore, somewhat shielded from other reactor room activities. The shield for the subcritical pile was completed February 23, 1972. The total mrem reported for this location reflects an annual total below the neutron and gamma dose detection thresholds listed in footnote (4).

Table 5.6 (continued)

(6) This is the composite dosimeter opposite the particulate filter for the reactor primary water cleanup system. A shield around this filter was completed July 10, 1972, and the annual dose equivalent totals from this operating area composite dosimeter are below the neutron and gamma dose detection thresholds listed in footnote (4).

(7) These dose equivalent totals are attributed to the short term use of small gamma-emitting sealed sources inside the reactor bay for laboratory classes in nuclear engineering and radiation protection.

Table 5.7
Total Dose Equivalent at the TRIGA Reactor Area Fence
For the Year July 1, 1984 through June 30, 1985

Fence Location (1)	Total Recorded mrem for the Year July 1, 1984 through June 30, 1985 Based on R.D. Co. TLDs (2)(3)	Total Recorded mrem for the Year July 1, 1984 through June 30, 1985 Based on OSU TLDs (3)(4)(5)	Total Calculated mrem for the Year July 1, 1984 through June 30, 1985 Based on the Annual Average R/hr Exposure Rate Measured at Each Location (5)(6)
MRCFE-1	113.8	75.8 ± 10.6	71.8 ± 35.3
MRCFE-2	118.5	64.0 ± 31.3	79.5 ± 28.3
MRCFE-3	118.5	62.9 ± 8.6	71.9 ± 31.3
MRCFE-4	128.1(7)	100.4 ± 30.4(7)	79.5 ± 40.4
MRCFE-5	112.8	76.3 ± 10.9	69.0 ± 33.1
MRCFE-6	113.8	76.3 ± 13.8	71.5 ± 44.0
MRCFE-7	113.8	69.3 ± 15.7	68.0 ± 24.7
MRCFE-8	110.9	69.8 ± 8.6	64.2 ± 25.7
MRCFE-9	112.8	70.8 ± 9.2	63.5 ± 31.6

(1) The TRIGA reactor area fence was installed September 15, 1972.

(2) Radiation Detection Company (R.D. Co.), Sunnyvale, California, TLD totals include their annual natural background contribution of 91.8 mrem for the reporting period. Average Corvallis area natural background using R.D. Co. TLDs totals 100.4 mrem for the same period.

(3) TLD monitoring packets are exchanged on a quarterly interval.

(4) OSU fence TLD totals include a measured annual natural background contribution of 63.4 ± 6.0 mrem for the reporting period.(5)

(5) ± values represent the standard deviation of the total value at the 95% confidence level.

(6) The annual average microroentgen (μR) per hour exposure rate for each location is normally determined by averaging 12 separate μR/hr measurements taken at monthly intervals throughout the year. The total mrem for the period is then calculated by multiplying this average μR/hr value by 8760 hours per year and then by converting microroentgens to millirem. Normal μR/hr values for the U.S. (terrestrial plus cosmic radiation) range between about 7.0 and 11.0 μR/hr (Ref. 1) (excluding areas of unusually high natural radioactivity). These exposure rates correspond to annual dose equivalent totals of about 59 to 93 mrem per year. The U.S. EPA (Ref. 2,3) estimates the total annual terrestrial plus cosmic dose equivalent for Oregon to be about 110 mrem per year.

(7) A possible slight increase in the dose equivalent recorded by this station is not believed to be related to reactor operations, but instead is attributed to the same short term use of small gamma-emitting sealed sources inside the reactor bay for laboratory classes in nuclear engineering and radiation protection as mentioned in footnote (7) of Table 5.6. Monitoring station FE-4 views that portion of the reactor building which includes the steel equipment access doors, and as expected, the attenuation provided by these doors is not quite equal to that provided by the building's concrete walls.

Table 5.8

Annual Average Concentration of the Total Net Beta Radioactivity (Minus ^{3}H)
For Offsite Environmental Soil, Water, and Vegetation Samples
For the Year July 1, 1984 through June 30, 1985

Sample Location Identification Number, Type & Reporting Units		Annual Average Concentration of Total Net Beta Radioactivity (Minus ^{3}H)(1)(2)(3)(4)
1-water	(μCi/cc)	$3.40 \times 10^{-8} \pm 5.19 \times 10^{-10}$ (5)
4-water	(μCi/cc)	$3.41 \times 10^{-8} \pm 5.54 \times 10^{-10}$ (6)
11-water	(μCi/cc)	$3.39 \times 10^{-8} \pm 1.85 \times 10^{-9}$
19-rainwater	(μCi/cc)	$3.42 \times 10^{-8} \pm 1.74 \times 10^{-9}$ (5)
3-soil	$(\frac{\mu Ci}{\text{gram of dry soil}})$	$4.98 \times 10^{-5} \pm 3.60 \times 10^{-5}$
5-soil	$(\frac{\mu Ci}{\text{gram of dry soil}})$	$4.55 \times 10^{-5} \pm 3.07 \times 10^{-5}$
20-soil	$(\frac{\mu Ci}{\text{gram of dry soil}})$	$7.31 \times 10^{-5} \pm 1.30 \times 10^{-5}$
21-soil	$(\frac{\mu Ci}{\text{gram of dry soil}})$	$7.18 \times 10^{-5} \pm 1.52 \times 10^{-5}$
2-grass	$(\frac{\mu Ci}{\text{gram of dry ash}})$	$3.09 \times 10^{-4} \pm 2.71 \times 10^{-4}$
6-grass	$(\frac{\mu Ci}{\text{gram of dry ash}})$	$2.73 \times 10^{-4} \pm 2.33 \times 10^{-4}$
7-grass	$(\frac{\mu Ci}{\text{gram of dry ash}})$	$3.21 \times 10^{-4} \pm 5.21 \times 10^{-5}$
8-grass	$(\frac{\mu Ci}{\text{gram of dry ash}})$	$2.81 \times 10^{-4} \pm 2.30 \times 10^{-4}$
9-grass	$(\frac{\mu Ci}{\text{gram of dry ash}})$	$2.76 \times 10^{-4} \pm 3.64 \times 10^{-4}$
10-grass	$(\frac{\mu Ci}{\text{gram of dry ash}})$	$2.71 \times 10^{-4} \pm 1.94 \times 10^{-4}$
12-grass	$(\frac{\mu Ci}{\text{gram of dry ash}})$	$3.37 \times 10^{-4} \pm 2.27 \times 10^{-4}$
13-grass	$(\frac{\mu Ci}{\text{gram of dry ash}})$	$3.78 \times 10^{-4} \pm 9.46 \times 10^{-5}$

Table 5.8 (continued)

Sample Location Identification Number, Type & Reporting Units	Annual Average Concentration of Total Net Beta Radioactivity (Minus ^{3}H)(1)(2)(3)(4)
14-grass ($\frac{\mu Ci}{gram\ of\ dry\ ash}$)	$2.33 \times 10^{-4} \pm 2.50 \times 10^{-4}$
15-grass ($\frac{\mu Ci}{gram\ of\ dry\ ash}$)	$2.43 \times 10^{-4} \pm 1.52 \times 10^{-4}$
16-grass ($\frac{\mu Ci}{gram\ of\ dry\ ash}$)	$3.09 \times 10^{-4} \pm 1.90 \times 10^{-4}$
17-grass ($\frac{\mu Ci}{gram\ of\ dry\ ash}$)	$3.25 \times 10^{-4} \pm 2.34 \times 10^{-4}$
18-grass ($\frac{\mu Ci}{gram\ of\ dry\ ash}$)	$3.47 \times 10^{-4} \pm 2.47 \times 10^{-4}$
22-grass ($\frac{\mu Ci}{gram\ of\ dry\ ash}$)	$3.67 \times 10^{-4} \pm 2.29 \times 10^{-4}$

(1) "±" values represent the standard deviation of the average value at the 95% confidence level.

(2) Annual average concentrations were calculated using sample results which exceeded the lower limit of detection (LLD), except that sample results which were $\leq$ the LLD were averaged in at the corresponding LLD concentration.

(3) For this report, the lower limit of detection (LLD) has been defined as the smallest amount or concentration of radioactive material (in terms of µCi per unit volume or unit mass) in a representative sample, which has a 95% probability of being detected. It is equivalent to 4.66 times the standard deviation of the detection system's background counting rate obtained with a blank sample (provided the relative standard deviation of the background rate, the coefficient of variation, is less than 25%) and is expressed in µCi divided by the typical volume or mass of the sample type involved. For the year July 1, 1984 through June 30, 1985, the LLD for total net β in water samples averaged 3.39×10^{-8} µCi/cc and ranged between 3.52×10^{-8} µCi/cc and 3.29×10^{-8} µCi/cc. For total net β in vegetation samples, the LLD averaged 2.00×10^{-5} µCi/gm of dry ash, and ranged between 2.39×10^{-5} µCi/gm of dry ash and 1.88×10^{-5} µCi/gm of dry ash. For total net β in soil samples, the LLD averaged 1.34×10^{-5} µCi/gm of dry soil, and ranged between 1.54×10^{-5} µCi/gm of dry soil and 1.26×10^{-5} µCi/gm of dry soil. The preceding LLD values for net β radioactivity in water, vegetation, and soil samples exclude ^{3}H.

Table 5.8 (continued)

(4) The total net beta radioactivity is based on the total net beta disintegration rate per sample, excluding ^{3}H. Calculation of the total net beta disintegration rate incorporates subtraction of only the counting system background from the gross beta counting rate, followed by application of an appropriate beta counting system yield (or efficiency). This value (the total net beta radioactivity) is also sometimes implied (but not in this report) by use of the term gross beta radioactivity, usually when the beta background is a small fraction of the gross beta counting rate.

(5) Results are based on three quarterly samples. The indicated collection point was dry during one collection period.

(6) Results are based on two quarterly samples. The indicated collection point was dry during two collection periods.

Table 5.9

Total Dose Equivalent at the Offsite Airborne Gamma Monitoring Stations
For the Year July 1, 1984 through June 30, 1985

Monitoring Station (1)(2)	Total Recorded mrem for the Year July 1, 1984 through June 30, 1985 Based on R.D. Co. TLDs (3) (4)	Total Recorded mrem for the Year July 1, 1984 through June 30, 1985 Based on Standard OSU TLDs (4)(5)(6)(8)	Total Calculated mrem for the Year July 1, 1984 through June 30, 1985 Based on the Annual Average µR/hr Exposure Rate Measured at Each Location (6) (7)
MRCTE-2L	-----	57.0 ± 14.4	51.6 ± 19.8
MRCTE-3	120.5	68.6 ± 6.8	77.5 ± 33.3
MRCTE-4	116.6	61.2 ± 5.3	71.1 ± 27.7
MRCTE-5L	-----	60.6 ± 5.7	71.1 ± 24.5
MRCTE-6	119.5	75.3 ± 36.2	81.3 ± 25.5
MRCTE-7L	-----	59.6 ± 13.5	72.9 ± 23.9
MRCTE-8	131.0	66.5 ± 18.2	83.3 ± 35.7
MRCTE-9	125.2	68.7 ± 17.4	81.3 ± 38.4
MRCTE-10	110.9	65.2 ± 12.1	62.8 ± 28.4
MRCTE-11	105.2	68.4 ± 8.3 (8)	60.0 ± 16.9
MRCTE-12	126.2	69.6 ± 8.3	80.9 ± 32.3
MRCTE-13L	-----	59.5 ± 6.4	74.6 ± 23.4
MRCTE-14L	-----	62.8 ± 14.7	57.9 ± 22.6
MRCTE-15	111.9	57.6 ± 12.1	77.5 ± 54.5
MRCTE-16L	-----	58.6 ± 9.0 (8)	73.9 ± 34.8
MRCTE-17	113.8	63.9 ± 20.6	66.9 ± 31.6
MRCTE-18L	-----	67.7 ± 14.9	71.1 ± 27.5
MRCTE-19	124.3	62.0 ± 21.0	81.3 ± 21.4
MRCTE-20L	-----	71.7 ± 10.2	66.9 ± 30.7

(1) Monitoring stations coded with an "L" contained one standard OSU TLD monitoring pack only (no R.D. Co. TLD pack).

(2) Monitoring stations not coded with an "L" contained one R.D. Co. TLD monitoring pack, two 0-200 mrem gamma pocket dosimeters, and one standard OSU TLD monitoring pack.

(3) Radiation Detection Company (R.D. Co.), Sunnyvale, California, TLD totals include their annual natural background contribution of 96.6 mrem for the reporting period. Average Corvallis area natural background using R.D. Co. TLDs totals 100.4 mrem for the same period.

(4) TLD monitoring packets are exchanged on a quarterly interval.

(5) OSU offsite airborne gamma TLD totals include a measured annual natural background contribution of 64.3 ± 8.3 mrem for the reporting period.(6)

(6) "±" values represent the standard deviation of the total value at the 95% confidence level.

(7) The annual average microroentgen (µR) per hour exposure rate for each location is normally determined by averaging 12 separate µR/hr measurements taken at monthly intervals throughout the year. The total mrem for the period is then calculated by multiplying this average µR/hr value by 8760 hours per year and then by converting microroentgens to millirem. Normal µR/hr values for the U.S. (terrestrial plus cosmic radiation) range between about 7.0 and 11.0 µR/hr (Ref. 1) (excluding areas of unusually high natural radioactivity). These exposure rates correspond to annual dose equivalent totals of about 59 to 93 mrem per year. The U.S. EPA (Ref. 2,3) estimates the total annual terrestrial plus cosmic dose equivalent for Oregon to be about 110 mrem per year.

(8) Total mrem for the reporting period is based on three calendar quarters of measured data, plus a fourth quarter of data derived from averaging the three quarters of measured data. These calculations to supplement the measured results were necessary due to lost or stolen TLD monitors in one quarter.

Figure 5.1

Operating Area Composite Dosimeter Locations for the TRIGA Reactor Building External Walls

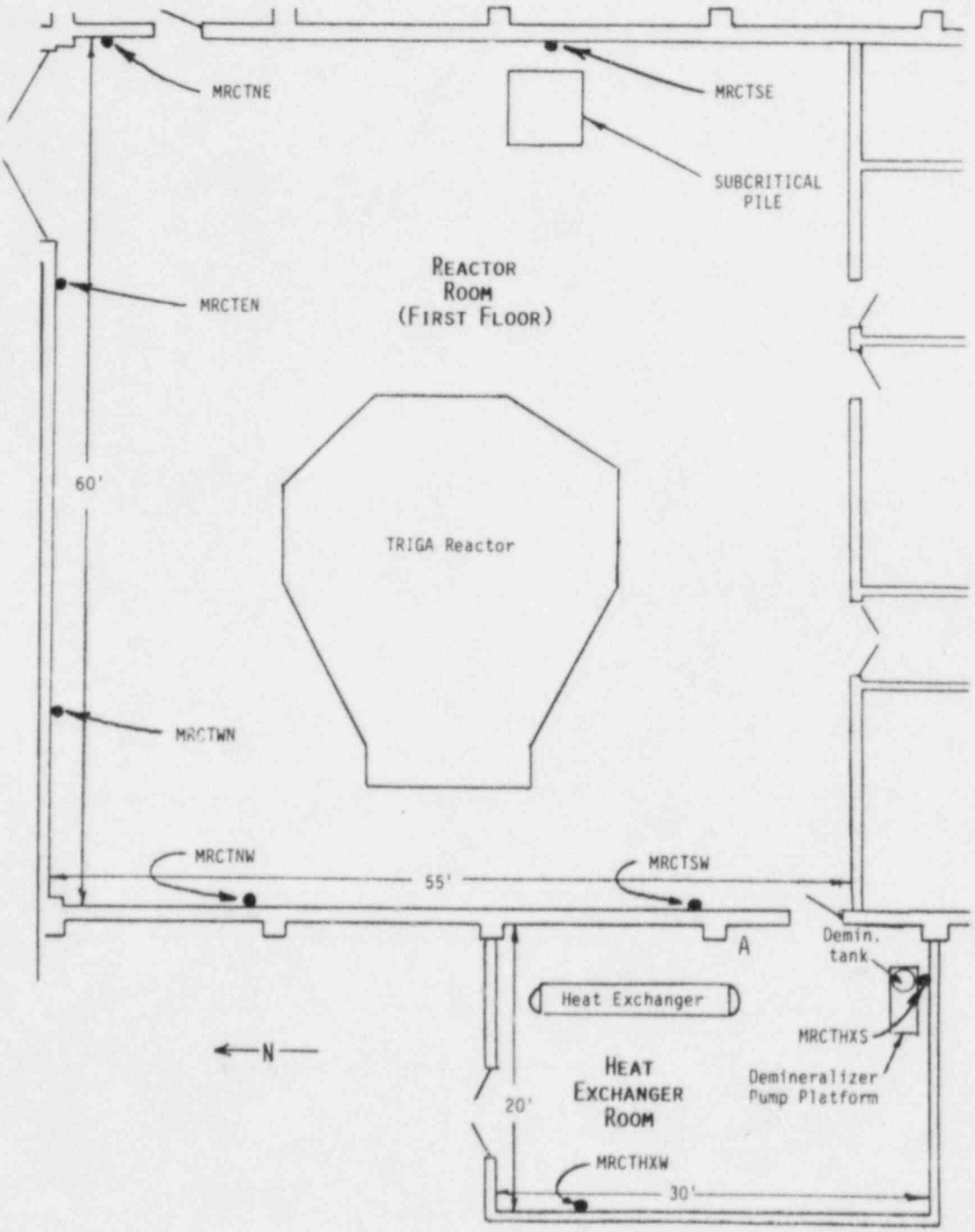


Figure 5.2

Area Radiation Monitor Locations for the

TRIGA Reactor, and on the TRIGA Reactor Area Fence

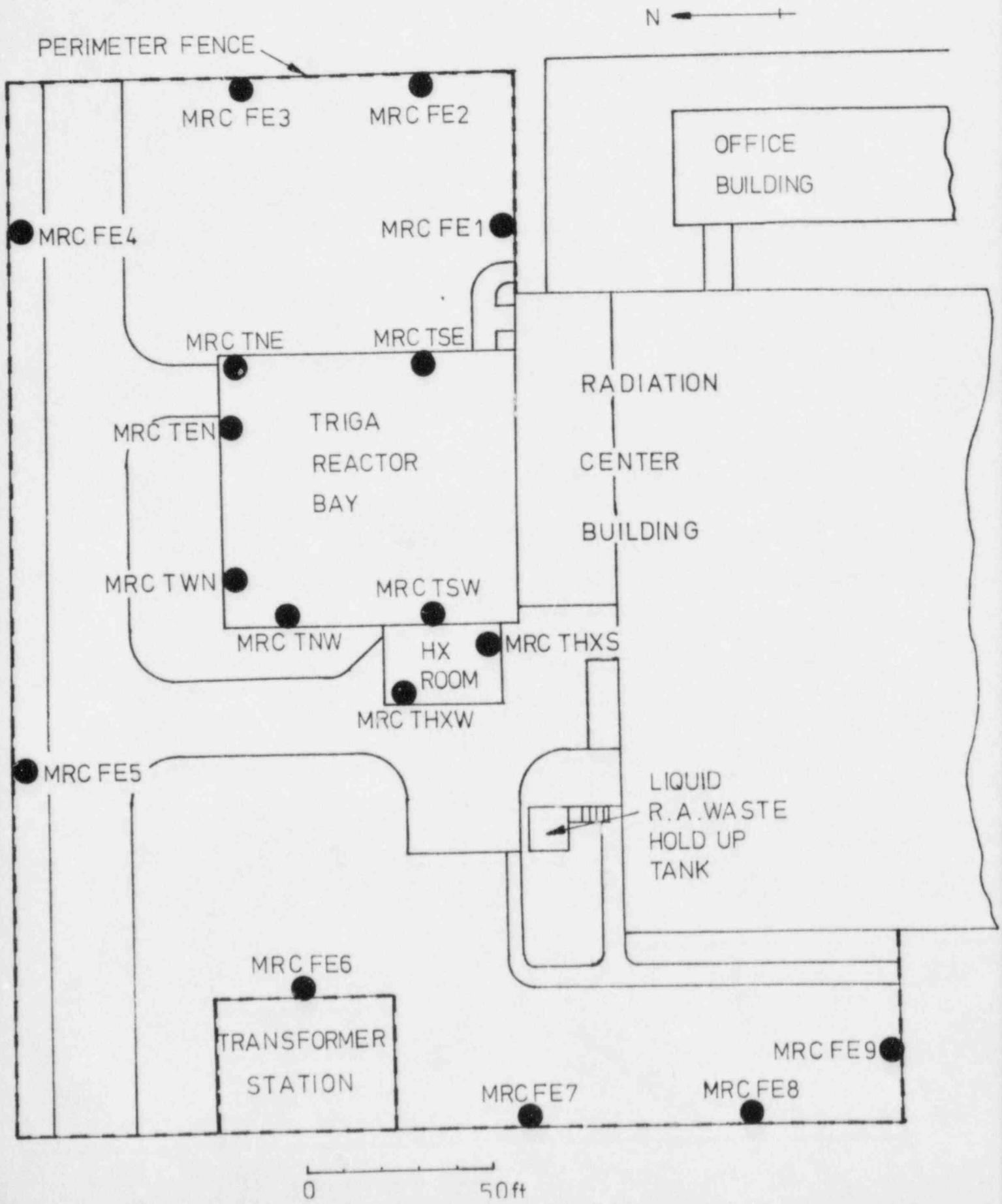
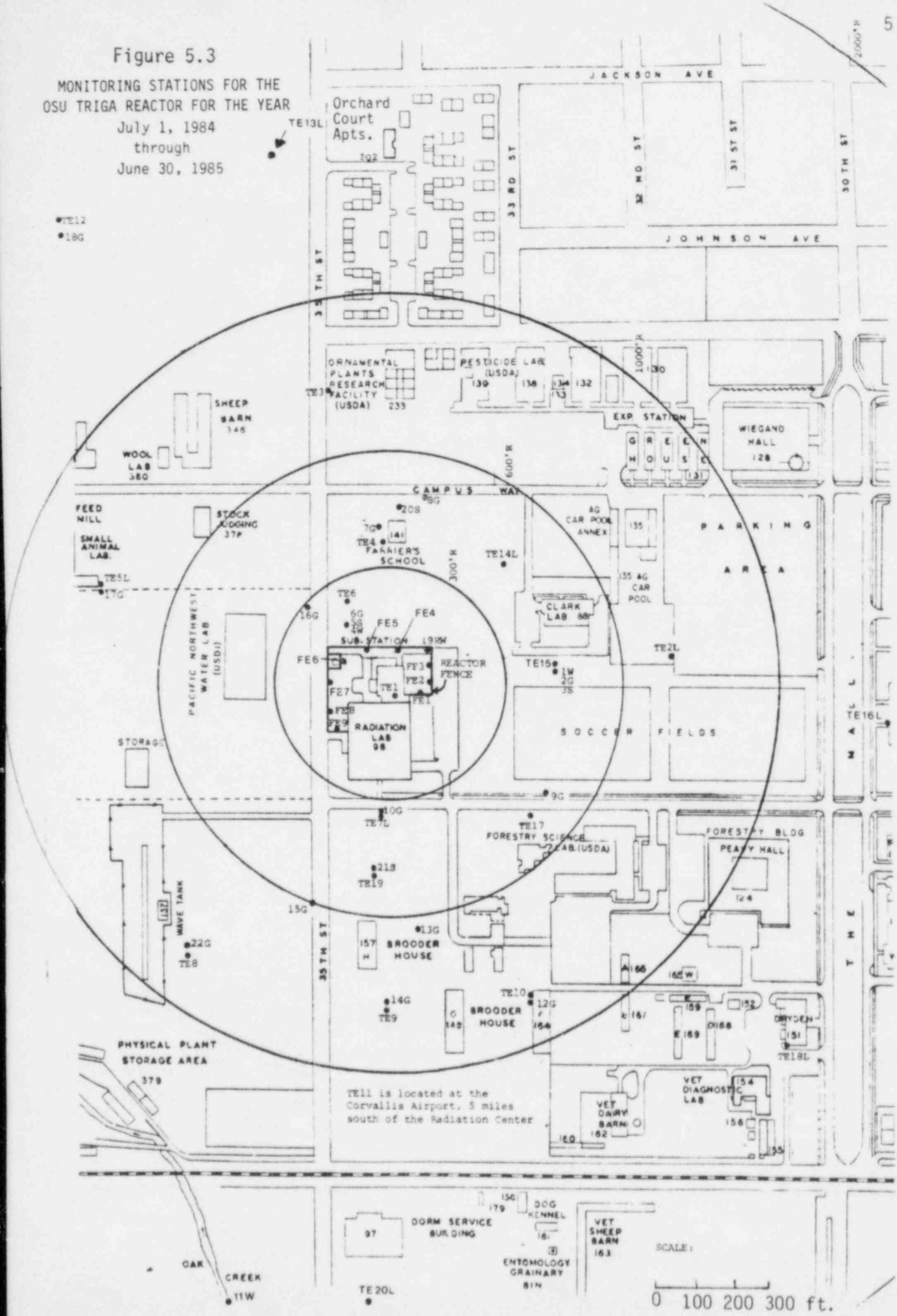


Figure 5.3

MONITORING STATIONS FOR THE OSU TRIGA REACTOR FOR THE YEAR July 1, 1984 through June 30, 1985



Radiation Center

Oregon State University

Corvallis, Oregon 97331 (503) 754-2341

September 15, 1985

U.S. Nuclear Regulatory Commission
Region V
1450 Maria Lane, Suite 210
Walnut Creek, CA 94596-5368

Attention: Regional Administrator

Gentlemen:

Reference: Oregon State University TRIGA Reactor (OSTR), Docket No. 50-243, License No. R-106

In accordance with the requirements specified in the Technical Specifications for the Oregon State University TRIGA Reactor, we are submitting to you five copies of the OREGON STATE UNIVERSITY TRIGA REACTOR ANNUAL REPORT for the reporting period July 1, 1984 through June 30, 1985. In addition, please note that we are using this letter of transmittal to send two copies of the report to the Director of the Office of Inspection and Enforcement in Washington, D.C., plus one copy of the report to Mr. Robert Carter, Division of Licensing, Washington, D.C., and one copy to the USNRC Document Control Desk, Washington, D.C.

As you will observe, the OSTR annual report contains information required by the U.S. Nuclear Regulatory Commission, the U.S. Department of Energy, and the Oregon Department of Energy. However, if there is any additional information that you desire, we will be most happy to provide it.

Sincerely,

Clifford V. Smith Jr.

Clifford V. Smith, Jr.
Reactor Administrator
Director, Radiation Center

CVS/ef

Enclosure

cc w/enclosure:

Document Control Desk, USNRC, Washington, D.C. (1 copy)
Standardization and Special Projects Branch, Division of Licensing, USNRC, Washington, D.C., ATTN: Mr. Robert Carter (1 copy)
Director, Office of Inspection & Enforcement, USNRC, Washington, D.C. (2 copies)

IE25

cc w/out enclosure:

Director, Oregon Department of Energy
T.D. Parsons, Vice President for Administration, OSU
A.G. Johnson, Assistant Director, Radiation Center, OSU
B. Dodd, OSTR Assistant Reactor Administrator, OSU
T.V. Anderson, OSTR Reactor Supervisor, OSUfor the reporting period