Annual Report
of the
Oregon State University
Radiation Center
and
TRIGA Reactor

July 1, 1988 - June 30, 1989

To satisfy the requirements of:

A. U.S. Nuclear Regulatory Commission, License No. R-106 (Docket No. 50-243), Technical Specification 6.7(e).

B. Task Order No. 3, under Subcontract No. C84-110499 (DE-AC07-76ER01953) for University Reactor Fuel Assistance-AR-67-88, issued by EG&G Idaho, Inc.

C. Oregon Department of Energy, ODOE Rule No. 30-010.

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E. Summary of OSTR Environmental and Radiation Protection Data

1. Liquid Effluents Released (See Table V.B.1)	Year July 1, 1988 Through June 30, 1989
a. Total estimated quantity of radioactivity released (to the sanitary sewer)(in curies)(1)	8.53×10^{-4}
b. Detectable radionuclides in the liquid waste	^{3}H, ^{60}Co
c. Estimated average concentration of released radioactive material at the point of release (in microcuries per cubic centimeter)	3.00×10^{-5}
d. Percent of applicable MPC for released liquid radioactive material at the point of release (%)	1.0%(3) 0.03%(4)
e. Total volume of liquid effluent released, including diluent, which contained an OSTR contribution (in gallons)(4)	7542

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(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 the over 7,500,000 gallons per year of liquids and sewage normally discharged by the Radiation Center complex into the same sanitary sewer system.

2. Airborne Effluents Released (See Table V.B.2)	Year July 1, 1988 Through June 30, 1989
a. Total estimated quantity of radioactivity released (in curies)	6.3
b. Detectable radionuclides in the gaseous waste(1)	^{41}Ar ($T_{½}$ = 1.83 hr)
c. Estimated average atmospheric diluted concentration of argon-41 at the point of release (in microcuries per cubic centimeter)	4.0×10^{-8}
d. Percent of applicable MPC for diluted concentration of argon-41 at the point of release (%)	1.0
e. Total estimated release of radioactivity in particulate form with half-lives greater than 8 days (in curies)(2)	None

3. Solid Waste Released (See Table V.B.3)	Year July 1, 1988 Through June 30, 1989
a. Total amount of solid waste packaged and disposed of (in cubic feet)	21.0
b. Detectable radionuclides in the solid waste	^{3}H, ^{46}Sc, ^{51}Cr, ^{54}Mn, ^{58}Co, ^{59}Fe ^{60}Co, ^{75}Se, ^{82}Br, ^{124}Sb, ^{132}Te, ^{141}Ce, ^{144}Ce, ^{152}Eu, ^{154}Eu
c. Total radioactivity in the solid waste (in curies)	5.2×10^{-5}

(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.

4. Radiation Exposure Received by Personnel (in mrem) (See Table V.C.1)[1]

	Year July 1, 1988 Through June 30, 1989
a. Facility Operating Personnel	
1) Average whole body	12
2) Average extremities	72
3) Maximum whole body	80
4) Maximum extremities	500
b. Key Facility Research Personnel	
1) Average whole body	0
2) Average extremities	5
3) Maximum whole body	0
4) Maximum extremities	40
c. Physical Plant Maintenance Personnel	
1) Average whole body	<1
2) Maximum whole body	8
d. Laboratory Class Students	
1) Average whole body	0
2) Average extremities	7
3) Maximum whole body	0
4) Maximum extremities	100
e. Campus Police and Security Personnel	
1) Average whole body	0
2) Maximum extremities	0
f. Visitors	
1) Average whole body	<1
2) Maximum whole body	5

(1) "0" indicates that each of the beta-gamma dosimeters during the reporting period was less than the vendor's gamma dose reporting threshold of 10 mrem or that each of the neutron dosimeters was less than the vendor's threshold of 30 mrem, as applicable.

5. <u>Number of Routine Onsite and Offsite Monitoring Measurements and Samples</u> — <u>Year July 1, 1988 Through June 30, 1989</u>

a. Facility Survey Data

1) Area Radiation Dosimeters (See Table V.D.1)

a) Beta-gamma dosimeter measurements	136
b) Neutron dosimeter measurements	48

2) Radiation and Contamination Survey Measurements (See Table V.D.3)	~6000

b. Environmental Survey Data

1) Gamma Radiation Monitoring (See Tables V.E.1 and V.E.2)

a) Onsite monitoring	
-- OSU TLD monitors	108
-- Radiation Detection Co. TLD monitors	72
-- Monthly μR/hr measurements	108
b) Offsite monitoring	
-- OSU TLD monitors	264
-- Radiation Detection Co. TLD monitors	104
-- Monthly μR/hr measurements	252

2) Soil, Water and Vegetation Surveys (See Table V.E.3)

a) Soil samples	16
b) Water samples	16
c) Vegetation samples	56

PART IV
REACTOR

A. Operating Statistics

For the current reporting period, the operating statistics for the OSTR showed modest increases in most of the major categories when compared to the previous period. Operating data by individual category are given in Table IV.A.1 and in Figure IV.A.1. Table IV.A.2 is included for reference and summarizes the operating statistics for the original 20% enriched fuel.

The thermal energy generated in the reactor during this reporting period was 42.7 megawatt days (MWD). The cumulative thermal energy generated by the FLIP core now totals 444.2 MWD from August 1, 1976 through June 30, 1989. Reactor use time averaged approximately 90% of the normal ninehour, five-day per week schedule. Tables IV.A.3 through IV.A.6 detail the operating statistics applicable to this reporting period.

Excess reactivity increased approximately 18¢ during the current reporting period. This change was caused by three factors:

1. Consumption of the erbium burnable poison in the fuel (increased reactivity).
2. Fuel element shuffles to even out core burnup (increased reactivity).
3. Fuel burnup (decreased reactivity).

Table IV.A.1

OSTR Operating Statistics (Using the FLIP Fuel Core)
for the 10-Year Period August 1976 - June 1986

Operational Data 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	1 JUL 85 Through 30 JUN 86
Operating Hours (critical)	875	819	458	875	1255	1192	1095	1205	1208	1470
Megawatt Hours	451	496	255	571	1005	999	931	943	946	1042
Megawatt Days	19	20.5	10.6	23.8	41.9	41.6	38.8	39.3	39.4	43.4
Grams ^{235}U Used	24	25.9	13.4	29.8	52.5	52.4	48.6	49.3	49.5	54.5
Hours at Full Power (1 MW)	401	481	218	552	998	973	890	929	904	1024
Number of Fuel Elements Added or Removed (-)	85	0	2	0	0	1	0	0	0	0
Number of Irradiation Requests	44	375	329	372	348	408	396	469	407	403

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

Table IV.A.1 (continued)

OSTR Operating Statistics (Using the FLIP Fuel Core)
for the Period July 1986 - June 1989

Operational Data for FLIP Core	1 JUL 86 Through 30 JUN 87	1 JUL 87 Through 30 JUN 88	1 JUL 88 Through 30 JUN 89	1 JUL 89 Through 30 JUN 90	1 JUL 90 Through 30 JUN 91	1 JUL 91 Through 30 JUN 92	1 JUL 92 Through 30 JUN 93	1 JUL 93 Through 30 JUN 94	1 JUL 94 Through 30 JUN 95	1 JUL 95 Through 30 JUN 96
Operating Hours (critical)	1172	1352	1170							
Megawatt Hours	993	1001	1025							
Megawatt Days	41.4	41.7	42.7							
Grams ^{235}U Used	51.9	52.3	53.6							
Hours at Full Power (1 MW)	980	987	1021							
Number of Fuel Elements Added or Removed (-)	0(1)	-2(2)	0							
Number of Irradiation Requests	387	373	290							

(1) No fuel elements were added, but one fueled follower control rod was replaced.

(2) Two fuel elements were removed due to cladding deformation.

Table IV.A.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
Megawatt 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	94
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 occurred in March of 1967.

(2) Reactor shut down 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 shut down June 1, 1971 for one month for cooling system upgrading.

(4) Reactor shut down 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 IV.A.3

Present OSTR Operating Statistics

Operational Data for FLIP Core	Annual Values for 1 JUL 88 Through 30 JUN 89	Cumulative Values for 1 AUG 76 Through 30 JUN 89
1. MWH of energy produced	1,025	10,658
2. MWD of energy produced	42.7	444.2
3. Grams ^{235}U used	53.6	557.7
4. Number of fuel elements added to or removed from (-) the core	0	85 + 3 FFCR[1]
5. Number of pulses	18	1,148
6. Hours reactor critical	1,170	14,146
7. Hours at full power (1 MW)	1,021	10,358
8. Number of startup and shutdown checks	252	3,265
9. Number of irradiation requests processed[3]	290	5,000
10. Number of samples irradiated	3,177	59,115

(1) Fuel Follower Control Rod.

(2) Each irradiation request could authorize from 1 to 120 samples. The number of samples per irradiation request averaged 11.0 during the current reporting period.

Table IV.A.4

OSTR Use Time in Terms of Operational Functions

OSTR Operational Function	Annual Values for 1 JUL 88 Through 30 JUN 89 (hours)	Cumulative Values for 1 AUG 76 Through 30 JUN 89 (hours)
Checkout, core excess and shutdown	377	4,802
Reactor in use(1)	2,353	20,050
Total reactor use time	2,730	24,852

(1) This function includes preclude time, multirle reactor experiment time, and the time the reactor is in use for teaching but not necessarily operating. (Preclude time is the time the reactor is not available for regular use due to performance of surveillance and maintenance items, such as fuel element inspections, transient rod lubrication, control rod calibration, power calibration, as well as sample loading and unloading time.)

Table IV.A.5

OSTR Use Time in Terms of Specific Use Categories

OSTR Use Category	Annual Values for 1 JUL 88 Through 30 JUN 89 (hours)	Cumulative Values for 1 AUG 76 Through 30 JUN 89 (hours)
Teaching (departmental and others)(1)	260	2,994
OSU research(2)	386	5,782
Off-campus research(2)	791	2,909
Forensic services	11	154(3)
Reactor preclude time	938	8,346
Facility time(4)	328	4,472
Visitor demonstration(5)	16	195(6)
Total reactor use time	2,730	24,852

(1) See Tables III.A.2 and III.E.1 for teaching statistics.

(2) See Table III.A.3 for research statistics.

(3) Prior to the 1981-1982 reporting period, forensic services were grouped under another use category. Since then, this service has been a separate category and the cumulative hours have been compiled beginning with the 1981-1982 report.

(4) The time OSTR spent operating to meet NRC facility license requirements.

(5) This is the time that the reactor was used specifically for visitor open-house (demonstration) events. The remainder of the visitors viewed the reactor during times when the reactor was being operated for regularly scheduled research and teaching.

(6) An error in the preparation of the 1984-1985 report resulted in the reporting of 101 hours of OSTR operations for visitor demonstration while the actual value was 159 hours. The difference of 58 hours is added to this year's report to correct the total cumulative reactor use time value.

Table IV.A.6

OSTR Multiple Use Time(1)

Number of Users	Annual Values for 1 JUL 88 Through 30 JUN 89 (hours)	Cumulative Values for 1 AUG 76 Through 30 JUN 89 (hours)
Two	155	1,155
Three	78	278
Four	25	102
Five	6	10
Six	15	23
Seven	4	4
Total multiple use time	283(2)	1,572(3)

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

(2) This represents 24% of the total hours the reactor was critical during this reporting period.

(3) This represents 11% of the total hours the reactor was critical since startup with FLIP fuel in August of 1976.

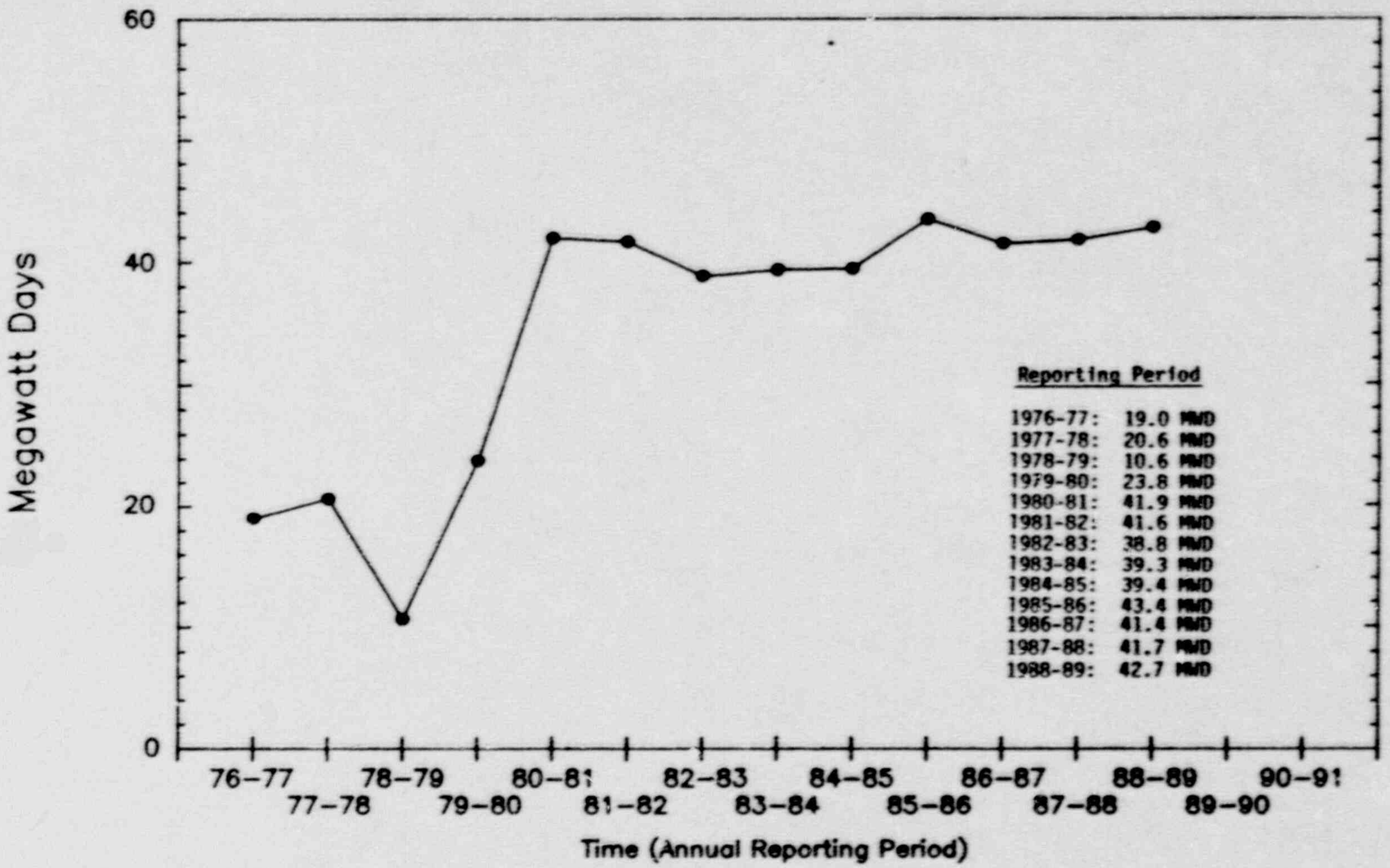


Figure IV.A.1 OSTR Annual Energy Production Vs. Time (Annual Reporting Period)

B. Experiments Performed

During the current reporting period there were 11 approved reactor experiments available for use in reactor related programs. The following list of reactor experiments identifies the 11 approved experiments. Missing numbers signify reactor experiments which are in the inactive file and are not currently being used.

A-1 Normal TRIGA Operation (No Sample Irradiation).

B-3 Irradiation of Materials in the Standard OSTR Irradiation Facilities.

B-11 Irradiation of Materials Involving Specific Quantities of Uranium and Thorium in the Standard OSTR Irradiation Facilities.

B-12 Exploratory Experiments.

B-21 Beam Port No. 3 Neutron Radiography Facility; Amendment No. 1 to B-21; Neutron Holography.

B-23 Studies Using TRIGA Thermal Column.

B-24 General Neutron Radiography.

B-25 Neutron Flux Monitors.

B-29 Reactivity Worth of Fuel.

B-30 NAA of Jet, Diesel, and Furnace Fuels.

B-31 TRIGA Flux Mapping.

Of the approved experiments on the active list, five were used during the reporting period. A tabulation of information relating to reactor experiment use is given in Table IV.B.1, and includes a listing of the experiments which were used, how often each was used, and the general purpose of the use. Presently, 25 experiments are in the inactive file and could be reapproved for use if needed.

Table IV.B.1

Use of OSTR Reactor Experiments[1]

Reactor Experiment Number[2]	Research	Teaching	Forensic	Facility Time[3]	TOTAL
A-1	0	33	0	81	114
B-3	132	17	2	0	151
B-11	20	0	0	0	20
B-23	0	2	0	0	2
B-31	2	0	0	0	2
TOTAL	154	52	2	81	289

(1) This table displays the number of times reactor experiments were used for a particular purpose.

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

A-1 Normal TRIGA Operation
B-3 Irradiation of Materials in the Standard OSTR Irradiation Facilities
B-11 Irradiation of Materials Involving Specific Quantities of Uranium and Thorium in the Standard OSTR Irradiation Facilities
B-23 Studies Using TRIGA Thermal Column
B-31 TRIGA Flux Mapping

(3) The time OSTR spent operating to meet NRC facility license requirements.

C. Unplanned Shutdowns

There were ten unplanned reactor shutdowns (scrams) during the current reporting period. Table IV.C.1 contains a summary of the unplanned shutdowns including a brief description of the cause of each.

Table IV.C.1

Unplanned Reactor Shutdowns (Scrams)

Type of Scram	Number of Occurrences	Cause of Shutdown
Safety Channel	6	Spurious scram signals. No cause or reason could be determined at the time. (These did not involve actual overpower situations.) Even though the safety channel ion chamber checked out good, it was decided to change the ion chamber. This action appears to have solved the problem.
% Power Channel	1	Spurious scram signal. Apparent noise in the mode switch caused this scram. Reactor was at 100 watts preparing for pulsing operation. When the mode switch was turned, the scram occurred.
Manual	1	The linear channel (blue pen) on the console recorder was behaving abnormally. It was observed that the drive line from the bull wheel to the pen was frayed. The reactor was scrammed and the pen drive repaired.
Manual	1	The reactor top Continuous Air Monitor (CAM) ceased operation (low flow alarm) and the reactor was scrammed manually. It was determined that the CAM blew a fuse. The fuse was replaced and reactor operation was resumed.
Manual	1	This scram was unintentional. The reactor operator reached for a writing pen and accidentally hit the manual scram button. Reactor operation was resumed.

D. Changes to the OSTR Facility, to Reactor Procedures, and to Reactor Experiments, and Tests Performed Pursuant to 10 CFR 50.59

The information contained in this section of the report provides a summary of changes and tests performed during the reporting period under the provisions of 10 CFR 50.59. For each item listed, we have included a brief description of the action taken and a summary of the applicable safety evaluation. Although it may not be specifically stated in each of the following safety evaluations, all actions taken under 10 CFR 50.59 were implemented only after it was established by the OSTR Reactor Operations Committee (ROC) that the proposed activity did not require a change in the facility's Technical Specifications and did not introduce or create an unreviewed safety question as defined in 10 CFR 50.59(a)(2).

1. 10 CFR 50.59 Changes to the Reactor Facility

There were eight changes to the reactor facility which were reviewed, approved, and performed under the provisions of 10 CFR 50.59 during the reporting period.

a. INSTALLATION OF A CADMIUM-LINED, IN-CORE, IRRADIATION TUBE

(1) Description

The reactor operations staff built and installed a cadmium-lined irradiation tube which can be permanently positioned in the reactor core. As shown in Figure IV.D.1, the facility consists of an air-filled aluminum tube with an offset bend, which is inserted into a convenient B-ring core grid position. The cadmium is approximately 0.025 inches thick and is permanently encased in aluminum inside and out. The tube is positively secured near the top to the center channel which extends across the reactor tank, and has a cap on top to seal it during reactor operation. To eliminate any small pressure increases due to radiolytic gas production, a suction is drawn on the irradiation tube by connecting a line from the rotating rack vent system to a tee-section on the irradiation tube.

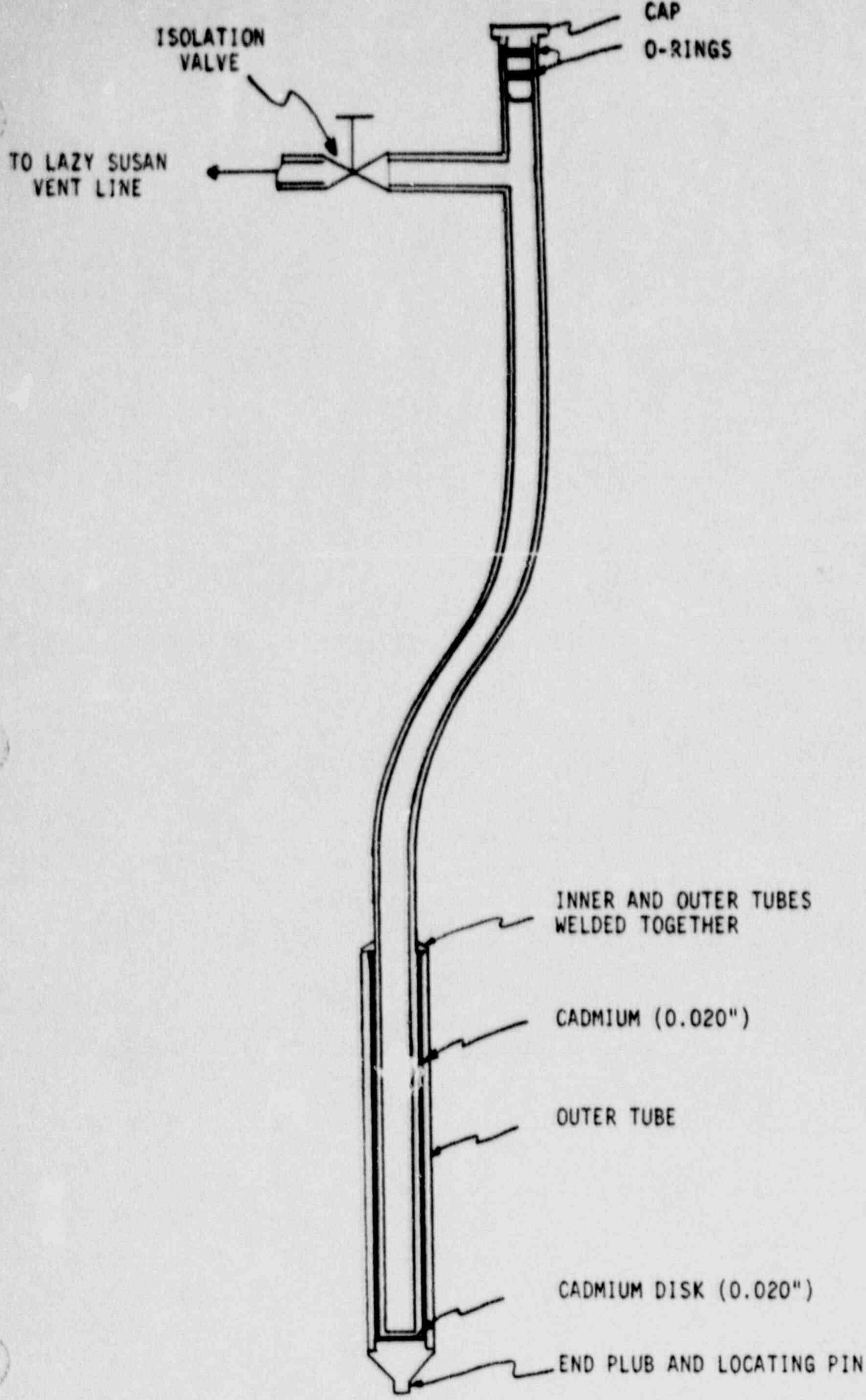


Figure IV.D.1 Cadmium-Lined, In-Core Irradiation Tube

NOTE: OSTROP 10 was revised to detail the procedures for using the new irradiation facility, and experiment B-3 was modified to allow irradiations in the cadmium-lined tube to be performed under this experiment. These changes, which were also made under 10 CFR 50.59, are discussed later in section D.

(2) Safety Evaluation

The safety considerations for this facility were very similar to those evaluated for the cadmium-lined pneumatic transfer tube which was previously installed under ROC approval and then recently removed from the reactor core. The reactivity changes were expected to be about the same, or slightly more negative because a similar amount of cadmium was used and this facility was placed in a core location with a slightly higher flux. The cadmium-lined tube was therefore estimated to be worth about -\$2.20 and was expected to reduce the core excess from about \$6.50 to about \$4.30. As indicated earlier, the tube was secured (bolted) to the reactor facility in such a manner that it could not be easily or unintentionally removed. This prevents any sudden, unplanned addition of reactivity to the reactor.

The tube was constructed in such a manner that the cadmium was completely sealed within an inner and outer aluminum tube. Therefore, there will be no potential for cadmium contamination of the samples or the reactor.

The offset bend in the tube is similar to that of the other in-core facilities and effectively precludes radiation streaming from the tube.

The procedures for the facility stipulate that the tube's cap is to be on the top of the tube except when samples are being inserted or removed and whenever the reactor is in operation. Additionally, under normal circumstances, samples will be removed from the tube only after the reactor has been shut down for a time period adequate to ensure that the bulk of the argon-41 activity has decayed (usually overnight). This prevents any unnecessary release of argon-41 to the reactor bay. The tube could be unloaded after a shorter decay period with specific health physics approval and appropriate precautions, but this option is no different than that currently employed with the rotating rack and therefore introduces no new safety considerations.

The procedures and limitations for encapsulation and irradiation of samples using the in-core cadmium-lined facility follow current requirements, particularly those in OSTROP 18. Therefore, no new or untried practices were introduced relative to the actual use of the new facility.

In order to position the cadmium-lined end of the tube into the core's B-ring, a fuel element must first be moved from grid location B1 to G6. The cadmium-lined tube can then be inserted into the vacant B-ring position. The estimated reactivity effect at each stage of the move was calculated and is given below:

i. Removal of the element from B1 = -35¢
ii. Insertion of this element into G6 = +30¢
iii. Insertion of the cadmium-lined tube into B1 = -$2.20
iv. Overall reactivity change = -$2.25

Core excess measurements were made at each step of the tube insertion procedure outlined above. The control rods were recalibrated after the tube was inserted and the core excess was remeasured.

b. CHANGES TO THE CADMIUM-LINED, IN-CORE, IRRADIATION TUBE

(1) Description

Following Reactor Operations Committee approval of the facility change regarding the cadmium-lined, in-core irradiation tube, the staff recommended two changes related to the irradiation tube. The first change involved the method to be used for removal and insertion of the sample support seat and the outer container holding the encapsulated sample(s) to be irradiated. This change was based on the fact that after the in-core irradiation facility was constructed it was discovered that the standard TRIGA tube handling device used to insert and remove TRIGA tubes for the rotating rack would also easily go down the new irradiation tube provided the handling device's outside borated shield was removed. Therefore, instead of a previously proposed, less desirable method for sample handling it was now proposed that samples be placed in the cadmium-lined facility in aluminum TRIGA tubes modified to have internal threads so that the containers could be inserted and removed using the standard TRIGA tube handling device (fishing pole and grapple) and standard procedures developed for the rotating rack. Similarly, the top of the sample support seat was modified by adding an aluminum TRIGA tube cap so that this too could be easily inserted and removed by this same method.

The second change involved moving the location of the facility's air suction tube to the irradiator tube cap, rather than having a "T" in the tube itself. It was determined that this change would allow much more flexibility in the angular positioning of the air tube.

(2) Safety Evaluation

It was judged that the first change improved the operational safety of the facility. The basis for this was related to the fact that the previously planned use of wire and cord to position and support samples was deleted, and with this deletion went any potential activation of wire, risk of wire breakage, and the risks associated with handling wire or cord when samples are pulled out of the tube. Conversely, the new method for sample insertion and removal employs currently approved procedures, which have been in practice at the OSTR for many years. The new design of the sample support seat also enables the staff to remove the seat easily when not in use, thus preventing any unnecessary activation.

There are no unfavorable safety implications involved with the second change. A slight air suction will still be pulled on the tube when it is in the core, in the same manner as before. Repositioning the suction tube on the cap simply allows this tube to be oriented in different directions and thus limits the inconvenience of having the tube in the way of other work which could be going on in the immediate area.

c. REMOVAL OF AN EXPERIMENTAL WATER RADIOACTIVITY MONITOR

(1) Description

As part of a graduate research project conducted approximately 8 years ago, an independent water monitoring loop was installed in the demineralizer circuit of the primary water system. The water monitor did not perform satisfactorily, and once the project was finished, the monitor was never used again. The reactor staff removed the water monitoring loop from the demineralizer circuit. Specifically, all of the piping and equipment between

valve DV 17 and the tee-section where valve DV 22 was attached to the main water line for the demineralizer circuit was removed (see Figure IV.D.2). The tee was plugged and valve DV 17 remained in the system as a sample collection valve. This change resulted in a revision of OSTROP 7 to remove all mention of this water monitoring loop.

(2) Safety Evaluation

There were no unfavorable safety implications associated with this facility change as this loop was never used to support reactor operations and the original water monitoring system for the reactor has remained fully functional. Appropriate health physics precautions were taken during the removal of the equipment. No significant contamination was found due in part to the long elapsed time since the loop was used, and due to the low radioactivity concentration of the reactor primary water.

d. ADDITION OF A PARTICULATE FILTER DOWNSTREAM OF THE DEMINERALIZER

(1) Description

The reactor staff installed a particulate filter downstream of the demineralizer tank in the reactor primary cooling water cleanup system. The new filter prevents resin fines from being introduced into the primary water system.

The filter assembly is mounted next to the east wall of the heat exchanger room adjacent to the demineralizer pump skid. The pipe and valving system (see Figure IV.D.3) is primarily 1" plastic pipe with three valves (two shutoff valves and a bypass valve), two pressure gauges (to measure the pressure drop across the filter) and a drainable housing for the 25 micron filter. The filter housing can be shielded by concrete blocks, if needed.

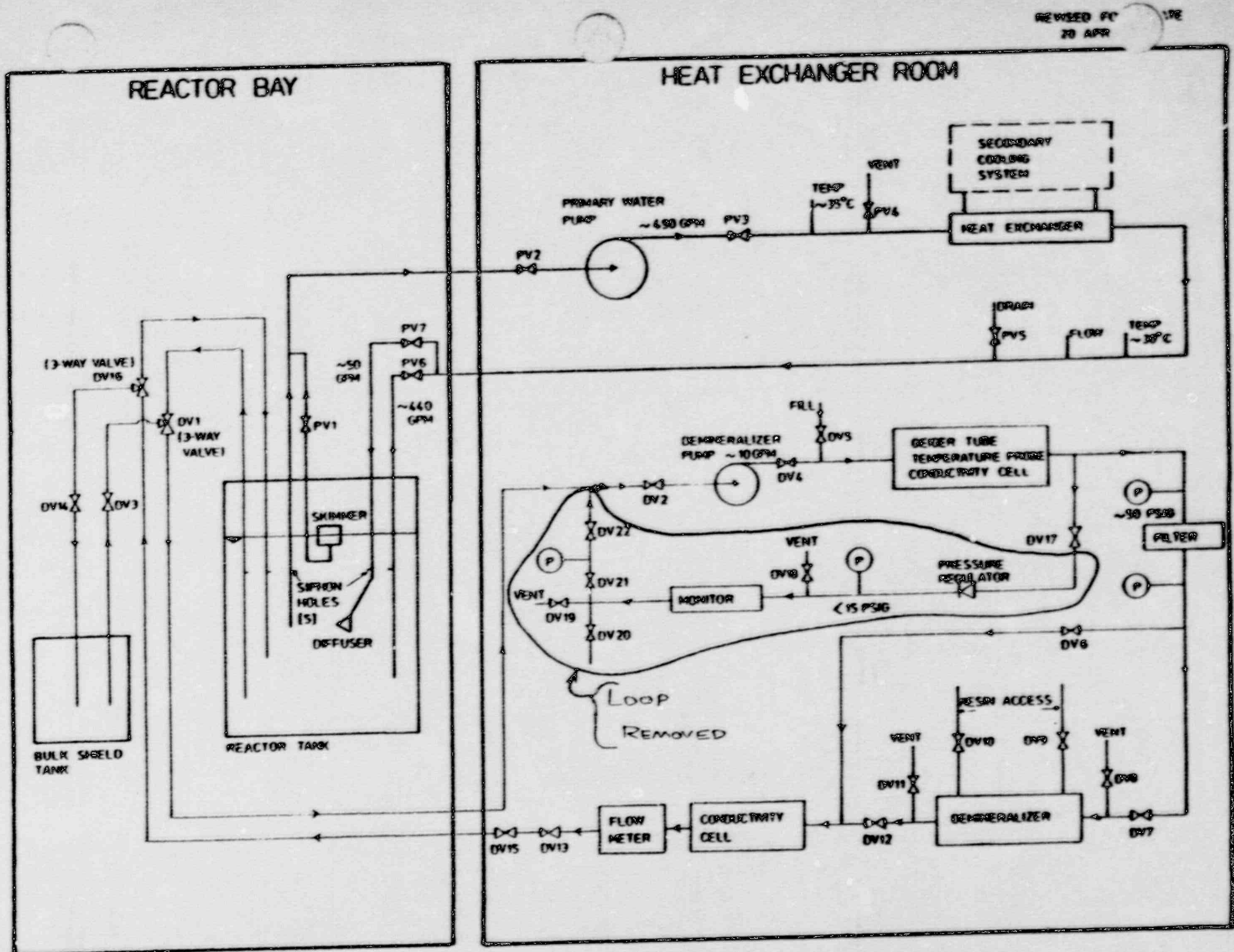


Fig. IV.D.2 Removal of an Experimental Water Radioactivity Monitor from the Primary Water Purification System

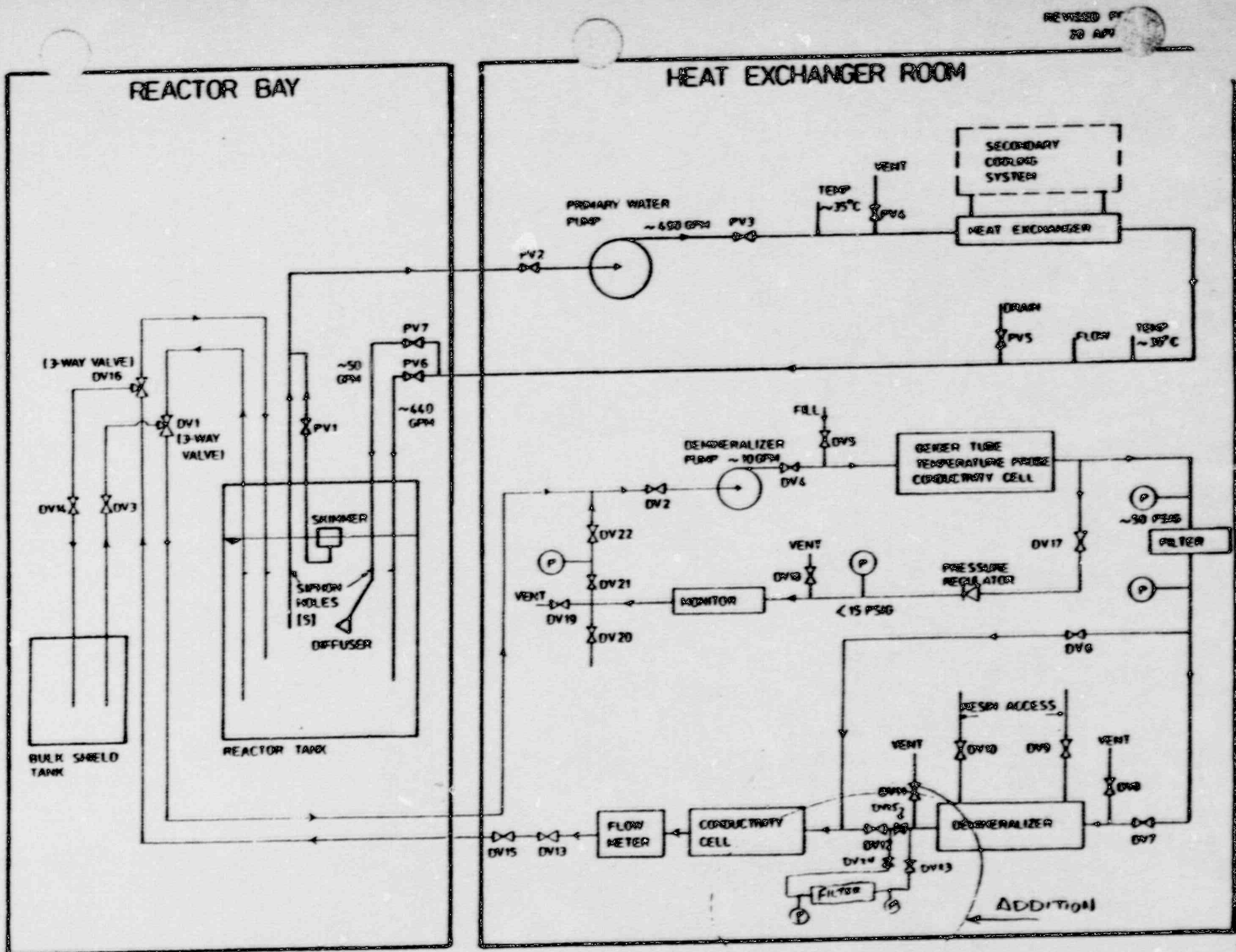


Fig. IV.D.3 Addition of a Particulate Filter Downstream of the Demineralizer in the Primary Water Purification System

(2) Safety Evaluation

There are no unfavorable safety implications related to the addition of this new filter. All of the materials used are of good quality, and the system passed all pre-operational tests. The new filter is essentially no different than the existing particulate filter for the primary water which has been in successful operation since the OSTR was built. Procedures used to change the new filter cartridge are the same as those used for the current filter. The new filter contributes to increased safety by removing any resin fines which might pass from the demineralizer system.

e. REPLACEMENT OF THE WATER CONDUCTIVITY MONITORING SYSTEM

(1) Description

The reactor staff upgraded the reactor water conductivity monitoring system to a digital readout instrument with automatic temperature compensation.

While the original conductivity monitoring system was still quite reliable, it had one disadvantage related to the fact that the "cat eye" indicating tube in the conductivity system had to be changed periodically. These old-style "cat eye" electronic tubes were becoming increasingly expensive and could possibly become unavailable in a few years.

In order to install the new system the following changes were made:

i. Two new signal cables were pulled from the heat exchanger room to the reactor console. The original cable did not have enough wire pairs to accommodate the temperature compensating feature of the new instrument.

ii. New temperature compensating conductivity probes were installed.

iii. Adapters were installed so that the new probes would fit into the water system piping.

iv. Modifications to the console side cabinet were made to accommodate the installation of the new digital readout instrument.

(2) Safety Evaulation

There is a great deal of historical data documenting the normal range of conductivity for the OSTR primary water. Any significant discrepancy observed between established values and results with the new system could and would be immediately investigated using other conductivity instruments.

Failure of the new system also does not create any immediate safety implications, since the conductivity of the reactor water changes very slowly with time, and thus allows plenty of time for detection and repair of conductivity equipment. In addition, the OSTR reactor water is kept at an extremely low level of conductivity, which gives an even greater margin of protection.

Because this is new equipment it is expected that the potential for failure will actually decrease and that this installation will provide more accurate conductivity readings. Hence, the new device will actually increase reliability and safety.

f. MONITORING OF REACTOR POWER AND FUEL TEMPERATURE WITH A CDAS DURING NON-PULSING OPERATIONS

(1) Description

On an as-needed basis, a computer-based data acquisition system (CDAS) can be connected to test terminals on the OSTR console to passively measure reactor power and fuel temperature signals during non-pulsing operations. Terminals TP2 and TP3 would be used to measure fuel temperature from the fuel thermocouple amplifier board (card XA16) in the left-hand console drawer. These terminals were extended to the rear of the console for easier and safer access. The LINEAR and LOG terminals already on the rear of the console would be used to measure the linear and log reactor power, respectively. No active circuitry will exist between the test terminals and the CDAS.

(2) Safety Evaluation

The referenced terminals are designed to be used as indicated above; however, additional measures will be taken to ensure the safe use of the CDAS while measuring reactor power and fuel temperature during non-pulsing operations. First, a 100-ohm resistor was permanently installed between console terminals TP3 and TP4 for the Nvt circuit, thus ensuring proper connection of the CDAS or other external recording devices to these terminals. Second, the CDAS system will be operationally tested to assure its proper functioning <u>before</u> it is connected to the reactor console. This will eliminate the possibility that the CDAS will be connected to the console with an improperly functioning or failed component. In addition, cables used to connect the CDAS to console terminals were labeled and color-coded to minimize the possibility of an incorrect connection. It is important to note, however, that an incorrect connection would create no safety problems and would not

affect console electronics. The impact would be an incorrect result on the external recording device. Furthermore, extending terminals TP2 and TP3 from the thermocouple amplifier board (card XA16) to the rear of the reactor console will increase safety by making these terminals more accessible, by eliminating the need to directly access electronics in the left-hand drawer of the console, and by making it easier to confirm that the connections are correct.

An evaluation of the worst-case failure of the CDAS while connected to the reactor console to record reactor power and fuel temperature indicated that the consequences are no worse than those created by the failure of an existing console component in the same circuit, and such consequences would be immediately obvious to the reactor operator so that appropriate action could be taken.

A new OSTROP 26 was written and approved. It details operating procedures for the CDAS when it is being used to measure reactor power level and fuel temperature.

g. MONITORING OF REACTOR PEAK POWER AND FUEL TEMPERATURE WITH A CDAS DURING PULSING OPERATIONS

(1) Description

The OSTR staff installed an operational amplifer inside the reactor console cabinet. The amplifier provides a gain of about 100 to amplify the peak power (Nvt) signal taken from console terminals TP3 and TP4. Signal amplification at these terminals is from about 60 mV to about 6 V. Although the amplifier circuit has been permanently installed inside the reactor console cabinet, it will be connected to Nvt circuit terminals TP3 and TP4 only as needed during pulsing. Also, as needed during pulsing operations, a computer-based data acquisition system

(CDAS) will be connected to the output from the operational amplifier to measure the peak reactor power during a pulse.

To measure fuel temperature during pulsing, the CDAS will also be connected to fuel temperature output terminals TP2 and TP3 on the back of the OSTR console in order to passively measure fuel temperature signals from the thermocouple amplifier board (card XA16) in the left-hand console drawer. No active circuitry will exist between the fuel temperature terminals and the CDAS.

(2) Safety Evaluation

The referenced terminals are designed to be used as indicated above; however, additional measures have been taken to ensure the safe use of the CDAS and amplifier during pulsing operations. First, a 100-ohm resistor has been permanently installed between console Nvt terminals TP3 and TP4, thus ensuring proper connection of the CDAS, the amplifier, or other external recording devices to these terminals. Permanently installing a 100-ohm resistor across terminals TP3 and TP4 increases the system's reliability by eliminating the need for a jumper cable across TP3 and TP4. Second, the amplifier and the CDAS system will be operationally tested to assure proper functioning <u>before</u> they are connected to the reactor console. This will eliminate the possibility that these devices will be connected to the console with an improperly functioning or failed component, which will thereby eliminate the chance that a signal from the amplifier or CDAS will affect console electronics. In addition, cables used to connect the amplifier to the console and the CDAS to the amplifier were labeled and color-coded to minimize the possibility of an incorrect connection. It is important to note, however, that an incorrect connection would

create no safety problems and would not affect console electronics. The impact would be an incorrect result on the external recording device. Furthermore, only cables from input channels of the covered distribution board for the CDAS will be present, and thus no output signal cables will be available for connection between the CDAS and the reactor console. As an added feature, protective diodes were added to the amplifier circuit to isolate the CDAS and the amplifier from the reactor console. This action will prevent the CDAS or amplifier from introducing a measurable charge to the Nvt circuit capacitor.

An evaluation of the worst-case failure of the CDAS and amplifier while connected to the reactor console to record peak pulse power and fuel temperature indicates that the consequences are no worse than those created by the failure of an existing console component in the same circuit, and such consequences would be immediately obvious to the reactor operator so that appropriate action could be taken.

In addition to the above safety considerations, fuel temperature monitoring during pulsing is described and fully evaluated by the 10 CFR 50.59 evaluation entitled "Monitoring of Reactor Power and Fuel Temperature with a CDAS During Non-Pulsing Operations," (see section IV.D.1.f).

Operating procedures for the CDAS and amplifier when used during pulsing are included in the new OSTROP 26 "Procedures for the Use of External Monitoring and Recording Devices."

h. INSTALLATION OF AN EXTERIOR LIGHT ON THE EAST WALL OF THE REACTOR BUILDING

(1) Description

In order to provide additional lighting for the northern half of the Radiation Center's parking lot, and to simultaneously enhance exterior lighting on the east side of the reactor building, an exterior light was installed at the north end of the east wall of the reactor building. Electrical power was supplied by extending a conduit from an interior east wall electrical outlet located adjacent to the east fire exit door.

(2) Safety Evaluation

The conduit penetration in the east reactor bay wall, as described above, was filled with an electrical conduit which was appropriately caulked inside and out to prevent air leakage. Therefore, this facility change does not affect the ability of the reactor building to maintain the originally designed containment integrity, and does not alter the probability of minimal radiological releases as described in the facility SAR. Consequently, the change introduces no increase in the probability or consequences of occurrences evaluated in the facility SAR. Furthermore, no new types of occurrences are introduced and no margin of safety is reduced by the proposed change.

2. <u>10 CFR 50.59 Changes to Reactor Procedures</u>

There were four changes to reactor procedures which were reviewed, approved, and performed under the provisions of 10 CFR 50.59 during the reporting period.

a. REVISION OF OSTROP 6

(1) Description

OSTROP 6.0 was modified to incorporate revisions necessitated by recent organizational and procedural changes. In particular, job descriptions for the Radiation Center Director, the Reactor Administrator, and most of the health physics group were modified. The revised ROC charter was also incorporated into the procedure. In addition, the access control procedure for the reactor bay was changed to incorporate the daytime use of signs when the security alarms are set during the day.

(2) Safety Evaluation

There are no unfavorable safety implications associated with the job description changes. All of the necessary responsibilities are well-covered, and the organizational changes have already been approved by the NRC and incorporated into the OSTR Technical Specifications. The changes to the ROC charter were also previously approved by the ROC, and these changes were merely being incorporated into OSTROP 6.0. The slight change to the reactor bay access procedure will not affect the security plan for the reactor. The change only affects those people with reactor bay keys and these people are the most responsible members of the Radiation Center security staff. Even if the procedure is not followed, security is not compromised, only an alarm is sounded unnecessarily.

b. REVISION OF THE EMERGENCY RESPONSE PLAN

(1) Description

A number of changes to the OSTR emergency response plan were made as a result of the annual review of the plan by the standing subcommittee of the ROC. This review was conducted on October 26, 1988. Most of the changes

were required as a result of the revised Radiation Center organization recently approved by the NRC. Remaining changes were merely updates of such things as telephone numbers, first aid qualifications, etc.

(2) Safety Evaluation

None of the changes involve revisions of the actual emergency response described in the plan. Instead, the changes simply update the plan to incorporate the current titles and organizational structure of the Radiation Center, and involve appropriate modifications to designated lines of succession in the emergency plan. As a result, none of the changes have any impact on safety. Changes to the plan were also reported to the USNRC under the provisions of 10 CFR 50.54(q).

c. REVISION OF OSTROP 10

(1) Description

The reactor staff amended section 10.7 of OSTROP 10 to specify the procedures for moving the cadmium-lined in-core irradiation tube from its storage location in the S-rack to the in-core position, and for returning the tube to the storage location.

In addition to amending section 10.7, the staff changed the title of OSTROP 10 to "Operating Procedures for OSTR Irradiation Facilities."

(2) Safety Evaluation

The reactivity effects of inserting and removing the cadmium-lined in-core irradiation tube have already been addressed in a previous 10 CFR 50.59 safety evaluation. The requirement relating to use of a specific set of control rod calibrations corresponding to whether the

tube is in or out of the core will ensure that accurate measurements of core excess and shutdown margin are made. A 2¢ limit for comparing control rod worths is also included in the revised section 10.7 because this is the estimated error for a rod calibration.

With respect to movement of the tube in the reactor tank, aluminum TRIGA tubes filled with lead shot will be placed inside the cadmium-lined in-core irradiation tube and the top cap will be sealed to ensure that the tube is neutrally bouyant. Therefore, the impact of accidentally releasing the tube will be minimal. In addition, the tube will not be over the core when it is passed under the center channel, and as a result there will be no possibility of dropping the tube on the core.

d. (ADDITIONAL) REVISIONS OF OSTROP 6.0

(1) Description

Three (additional) amendments were made to OSTROP 6.0, "Administrative and Personnel Procedures." The first amendment states that no external measuring or recording device will be connected to reactor measuring channels or safety channels without ROC approval of a 10 CFR 50.59 safety evaluation and any needed operating procedures. However, it was not intended that ROC approval be required for normal use of standard diagnostic equipment by the Scientific Instrument Technician or a designated replacement. A second amendment to OSTROP 6.0 states that when classes are in the reactor control room, the operator of record will not be the instructor of the class. Finally, a third amendment was added to ensure that any connection of an external system to reactor measuring or safety channels will be checked by the Reactor Supervisor.

Again, this requirement was not intended to apply to the normal use of standard diagnostic equipment by the Scientific Instrument Technician or a designated replacement.

(2) Safety Evaluation

All of the three revisions were made to increase safety, and therefore the safety implications are all positive. The first amendment noted above will prevent the addition of systems which might affect the reactor measuring or safety channels. The second amendment will decrease the possibility of the reactor operator being distracted by also having to instruct a class. The third amendment will help to ensure that any connections made to important console systems are made correctly.

3. 10 CFR 50.59 Changes to Reactor Experiments

There were three changes to reactor experiments which were reviewed, approved, and performed under the provisions of 10 CFR 50.59 during the reporting period.

a. REVISION OF OSTR EXPERIMENTS B-3 AND B-11

(1) Description

Experiments B-3 and B-11 were revised to add the new cadmium-lined in-core irradiation tube as one of the standard OSTR irradiation facilities.

(2) Safety Evaluation

The installation of the cadmium-lined tube in the core provides a new standard irradiation facility. From an operational and health physics standpoint, irradiation of samples in this facility is no different than irradiating samples in cadmium cups in the (dummy) sample-holding fuel element, in the rotating rack or in the pneumatic

transfer facility. In fact, using the new tube will enhance safety by reducing radiation doses associated with the use and handling of cadmium cups, which will not now be needed for many experiments. All irradiations will be controlled following the usual procedures associated with irradiation requests.

b. REVISION OF OSTR EXPERIMENT B-31

(1) Description

Experiment B-31 was revised to expand the types of materials that can be activated in OSTR facilities for flux-mapping purposes. All OSTR irradiation facilities can be used including the reactor tank and associated in-core locations, since this was the original intent of Experiment B-31.

(2) Safety Evaluation

No reduction in safety effectiveness results from these minor revisions. Reactivity values and radioactivity limits have not changed.

c. REVISION OF OSTR EXPERIMENTS B-3, B-11 AND B-12

(1) Description

Experiments B-3, B-11 and B-12 were revised to prohibit the irradiation of elemental mercury or substances where mercury is a major constituent.

(2) Safety Evaluation

The change increases safety by preventing the introduction of mercury into the reactor. Mercury reacts with aluminum and could therefore cause undesirable corrosion of reactor components.

E. Surveillance and Maintenance

1. Non-Routine Maintenance

18 AUG 88	Replaced a pneumatic damper motor in the reactor building ventilation system.
29 AUG 88	Replaced the carbon vanes in the stack radioactivity monitor pump.
16 SEP 88	Changed the ion chamber in the reactor safety channel.
12 DEC 88	Installed a 20 micron filter downstream of the reactor primaty water demineralizer tank.
21 DEC 88	Repaired the low flow alarm on the continuous air monitor.
3 JAN 89	Sent the fuel element handling tool back to the manufacturer for repair.
6 JAN 89	Installed a new cranking battery for the emergency generator.
25 JAN 89	Installed a new test potentiometer on the console for the safety channel.
30 JAN 89	Removed an unused water radioactivity monitor from the reactor's primary water purification system.
6 FEB 89	Performed a shuffle of fuel elements to even out burnup.
16 FEB 89	Replaced a diode in the reactor servo system for better servo response.
27 FEB 89	Repaired a frozen pre-heat coil in the reactor building ventilation system.
17 MAR 89	Replaced two bearings in the reactor building ventilation supply fan.
3 APR 89	Replaced bearings in the reactor primary water pump.
4 APR 89	Replaced bearings in the reactor primary water pump motor.
10 APR 89	Replaced bearings (again) in the reactor primary water pump.
10 APR 89	Replaced the GM tube in the stack monitor's particulate channel.
25 APR 89	Replaced the pump motor on the continuous air monitor.

27 APR 89	Replaced the time delay relay in the stack monitor.
4 MAY 89	Permanently installed a 100 ohm resistor across console terminals TP3 and TP4 for the Nvt circuit.
12 JUN 89	Installed a new outside light on the northeast side of the reactor building.
13 JUN 89	Replaced the control room access door closed circuit TV monitor.

2. Routine Surveillance and Maintenance

The OSTR has an extensive routine surveillance and maintenance (S&M) program. Examples of typical S&M checklists are presented in Figures IV.E.1 through IV.E.4. Items marked with an asterisk (*) are required by the OSTR Technical Specifications.

F. Reportable Occurrences

In a letter to the USNRC dated April 17, 1989, the OSU Radiation Center reported an event involving OSTR procedures for hooking up external monitoring equipment to the OSTR console. Based on the nature of the event, more stringent procedures were implemented for the connection of any external measuring device to the reactor control console. No citations were issued as a result of the report and a routine onsite inspection by the USNRC.

Figure IV.E.1

Monthly Surveillance and Maintenance (Sample Form)

OSTROP 13

SURVEILLANCE & MAINTENANCE FOR THE MONTH OF ______________________

	SURVEILLANCE & MAINTENANCE TO BE PERFORMED		LIMITS	AS FOUND	TARGET DATE	DATE NOT TO BE EXCEEDED	DATE COMPLETED	REMARKS & INITIALS
* 1	FUNCTIONAL CHECK OF REACTOR WATER LEVEL ALARMS		MAXIMUM MOVEMENT ±3 INCHES	UP: _____ inches DN: _____ inches ANN: _________				
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: 8.5					
4	EMERGENCY POWER SYSTEM BATTERY CHECKS	INVERTER	LIQUID: ∿ 1" DN					
			S.G. DISCS UP					
		GENERATOR	S.G.: >1.250					
			VOLTS $\geq$ 12V DC					
			CORR: NONE					
5	EVACUATION HORN & P.A. EMERGENCY SYSTEM BATTERY CHECKS		LIQUID: FULL					
			S.G.: >1.250					
			VOLTS $\geq$ 12V DC					
			CORR: NONE					
6	INSPECTION OF THE BRUSHES ON THE PNEUMATIC TRANSFER SYSTEM BLOWER MOTOR		CHANGE WHEN 1/4" LEFT					
7	GREEN LIGHT BULB REPLACEMENT		75 WATT					
8	CHANGE LAZY SUSAN FILTER		FILTER CHANGED					
9	LUBRICATE THE TRIGA TUBE LOADING TOOL (REEL)		USE GUN OIL	NEED OIL? _____				
10	REACTOR TOP CAM OIL LEVEL CHECK		OSTROP 13.10	NEED OIL? _____				
11	PROPANE TANK LIQUID LEVEL CHECK (% FULL)		>50%					
*12	BULK WATER TEMPERATURE ALARM CHECK		FUNCTIONAL					
13	PRIMARY PUMP BEARINGS OIL LEVEL CHECK		OSTROP 13.13	NEED OIL? _____				

*License Requirement
**Date not to be exceeded for license requirements is equal to the date completed last month plus six weeks.

Figure IV.E.2

Quarterly Surveillance and Maintenance (Sample Form)

OSTROP 14 SURVEILLANCE & MAINTENANCE FOR THE QUARTER OF ____/____/____ 19____

	SURVEILLANCE & MAINTENANCE TO BE PERFORMED	LIMITS	AS FOUND	TARGET DATE	DATE NOT TO BE EXCEEDED**	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	RACK SHOULD BE EMPTY					
7	FUNCTIONAL CHECK OF EMERGENCY LIGHTS (SEE CHECKSHEET)	FUNCTIONAL					
8	WESTRONIC RECORDER SLIDE WIRE CLEANING	SLIDE WIRE CLEANED					
9	STACK MONITOR CHECKS (OIL DRIVE MOTORS, H.V. READINGS)	MOTORS OILED					
		PART: 500 V ±50	VOLTS				
		GAS: 900 V ±50	VOLTS				
10	TRACERLAB AREA RADIATION MONITOR (ARM) VOLTAGE CHECKS	25 V SUPPLY ±10%	VOLTS				
		H.V. 560 V ±10%	VOLTS				
11	ARM SYSTEM ALARM CHECKS	FUNCTIONAL					
12	OPERATOR LOG	a) >4 hours: at console (RO) or as Rx.Sup.(SRO) b) Complete Operating Exercise	a) TIME	b) OPERATING EXERCISE			
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					
16	FUNCTIONAL CHECK OF EVACUATION ALARMS	ALL FUNCTIONAL					

Item 11 — ARM SYSTEM ALARM CHECKS:

CHAN	1	2	3	4	5	6	7	8	9	10	11	12	13	14
AUD														
LIGHT														
PANEL														
ANN														

Item 12 — OPERATOR LOG:

NAME

* License Requirement ‡ Physical Security Plan Requirement

** Date not to be exceeded for license requirements is equal to the date completed last quarter plus four months.

Revised March 88

Figure IV.E.3

Semi-Annual Surveillance and Maintenance (Sample Form)

OSTROP 15 — SEMI-ANNUAL SURVEILLANCE & MAINTENANCE FOR ______________________

	SURVEILLANCE & MAINTENANCE TO BE PERFORMED		LIMITS	AS FOUND	TARGET DATE	DATE NOT TO BE EXCEEDED**	DATE COMPLETED	REMARKS & INITIALS
*1	FUNCTIONAL CHECKS OF REACTOR INTERLOCKS	a) NEUTRON SOURCE COUNT RATE INTERLOCK	NO WITHDRAW	a1				
			≥5 cps	a2				
		b) TRANSIENT ROD AIR INTERLOCK	NO PULSE	b				
		c) PULSE PROHIBIT ABOVE 1 kW	≥1 kW	c				
		d) TWO ROD WITHDRAWAL PROHIBIT	1 only	d				
		e) PULSE MODE ROD MOVEMENT INTERLOCK	NO MOVEMENT	e				
		f) MAXIMUM PULSE REACTIVITY INSERTION LIMIT	≤ $2.50	f				
		g) PULSE INTERLOCK ON RANGE SWITCH	NO PULSE	g				
*2	SAFETY CIRCUIT TEST	PERIOD SCRAM	≥3 sec	c				
*3	CONTROL ROD WITHDRAWAL, INSERTION & SCRAM TIMES	TRANS / SAFE / SHIM / REG						
		a) SCRAM	≤2 sec	a				
		b) WITHDRAWAL	≤50 sec	b				
		c) INSERTION	≤50 sec	c				
*4	PULSE COMPARISON (PREVIOUS PULSE): PULSE # ____ $ ____ ____ MW ____ °C		≤20% CHANGE	PULSE # ____ $ ____ ____ MW ____ °C				
*5	REACTOR BAY VENTILATION SYSTEM SHUT DOWN TEST		DAMPERS CLOSE IN ≤5 SECONDS	4TH FLOOR ____ 1ST FLOOR ____				
*6	CALIBRATION OF THE FUEL ELEMENT TEMPERATURE CHANNEL		±2°C					
*7	MATERIALS BALANCE REPORT/FUEL MANAGEMENT		REPORTS DONE/ ~ EVEN BURNUP					
*8	CLEANING & LUBRICATION OF TRANSIENT ROD CARRIER INTERNAL BARREL		3-IN-1 or GUN OIL	CLEANED ____ OILED ____				
*9	LUBRICATION OF BALL-NUT DRIVE ON TRANSIENT ROD CARRIER		3-IN-1 or GUN OIL	MOLY KOTE ____ OILED ____				
10	LUBRICATION OF THE ROTATING RACK BEARINGS		10 W OIL	OILED ____				
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			OILED ____				
15	FLUKE FUEL TEMPERATURE INSTRUMENT "D" CELL CHECK		TEST POSITION READ >800°C					

*License Requirements

**Date not to be exceeded for license requirements is equal to the date last time plus 7½ months.

Revised 8/88

Figure IV.E.3 (Continued)

Semi-Annual Surveillance and Maintenance (Sample Form)

OSTROP 15 (continued)

SEMI-ANNUAL SURVEILLANCE & MAINTENANCE FOR ______________

SURVEILLANCE & MAINTENANCE TO BE PERFORMED			LIMITS	AS FOUND	TARGET DATE	DATE NOT TO BE EXCEEDED**	DATE COMPLETED	REMARKS & INITIALS
16	ION CHAMBER RESISTANCE MEASUREMENTS WITH MEGGAR INDUCED VOLTAGE	A. SAFETY CHANNEL	NONE (Info Only)					
		B. % POWER CHANNEL	NONE (Info only)					
17	FISSION CHAMBER RESISTANCE CALCULATION $R = \frac{900V}{\Delta I}$)	@ 100 V. I = ________ AMPS @ 900 V. I = ________ AMPS ΔI = ________ AMPS R = ________ Ω	NONE (Info only)					

Figure IV.E.4

Annual Surveillance and Maintenance (Sample Form)

OSTROP 16.0

ANNUAL Surveillance and Maintenance for the Year ______________

	SURVEILLANCE AND MAINTENANCE TO BE PERFORMED		LIMITS	AS FOUND	TARGET DATE	DATE NOT TO BE EXCEEDED**	DATE COMPLETED	REMARKS & INITIALS
*1	BIENNIAL INSPECTION OF CONTROL RODS:	a) FFCRS	OSTROP 12.0					
		b) TRANS						
*2	ANNUAL REPORT (DUE JUNE 30 + 75 DAYS)		SEPT 13					
*3	STANDARD CONTROL ROD CALIBRATION:	a) SAFE	OSTROP 9.0					
		b) SHIM						
		c) REG						
*4	REACTOR POWER CALIBRATION		OSTROP 8.0					
*5	CALIBRATION OF REACTOR TANK WATER TEMPERATURE METERS		OSTROP 16.5					
*6	CONTINUOUS AIR MONITOR CALIBRATION:	a) Particulate Monitor	RCHPP 18.0					
		b) Gas Monitor						
*7	STACK MONITOR CALIBRATION:	a) Particulate Monitor	RCHPP 18 & 26					
		b) Gas Monitor						
*8	AREA RADIATION MONITOR CALIBRATION		RCHPP 18.0					
*9	WATER MONITOR CALIBRATION		RCHPP 18.0					
10	REACTOR TANK AND CORE COMPONENT INSPECTION		NO POWDERY WHITE SPOTS					
11	SNM PHYSICAL INVENTORY		OSTROP 20.0					
12	EMERGENCY RESPONSE PLAN DRILL							
13	STANDARD CONTROL ROD DRIVE INSPECTION		OSTROP 16.13					
14	OSU POLICE AND SECURITY RETRAINING							
15	50.59 REPORT		SEPT					
16	INTRUSION ALARM RESPONSE DRILL (OSU POLICE AND SECURITY)		RESPONSE ≤5 MIN					
17	EMERGENCY POWER INVERTER TEST		OSTROP 22.0					
18	REPLACE P.A. & EVAC SYSTEM LEAD-ACID BATTERIES		EVERY 4 YEARS					

*License Requirements

**Date not to be exceeded for annual license requirements is equal to the date completed last year plus 15 months. For biennial license requirements, it is equal to the date completed last time plus 2½ years.

Revised 3/88

Figure IV.E.4 (Continued)

Annual Surveillance and Maintenance (Sample Form)

OSTROP 16.0 (continued) | ANNUAL Surveillance and Maintenance for the Year ____________ | Page 2

SURVEILLANCE AND MAINTENANCE TO BE PERFORMED

*19 REACTOR OPERATOR LICENSE CONDITIONS

	ANNUAL				BIENNIAL		EVERY 6 YEARS				
	REQUALIFICATION				MEDICAL		NRC REQUAL EXAM		LICENSE		
	WRITTEN EXAMINATIONS		OPERATING TEST						APPLICATION		EXPIRATION DATE
NAME	DUE DATE	DATE PASSED	DUE DATE	DATE PASSED	DUE DATE	DATE COMPLETED	DUE DATE	DATE PASSED	DUE DATE	DATE MAILED	

	SURVEILLANCE AND MAINTENANCE TO BE PERFORMED	LIMITS	AS FOUND	TARGET DATE	DATE NOT TO BE EXCEEDED**	DATE COMPLETED	REMARKS & INITIALS
20	FUEL ELEMENT INSPECTION FOR SELECTED ELEMENTS (B1, B2, B3, B4, B5, B6, C3, C5, D5, D6	PASS GO/NO GO TEST		Pulse # ______ Date ______	Pulse # ______ Date ______		

PART V
PROTECTION

A. Introduction

This section of the report deals with the **radiation protection program** at the OSU Radiation Center. The purpose of this program is to ensure the safe use of radiation and radioactive materials in the Center's teaching, research, and service activities, and in a similar manner to ensure the fulfillment of all regulatory requirements of the state of Oregon, the U.S. Nuclear Regulatory Commission, and other regulatory agencies. The comprehensive nature of the program is shown in Table V.A.1, which lists the program's major radiation protection requirements and the performance frequency for each item.

The radiation protection program is implemented by a staff consisting of a Senior Health Physicist, a Health Physicist, a Radiation Protection Technologist, and one to five part-time Radiation Protection Technicians (see Part II.F). Assistance is also provided by the reactor operations group, the neutron activation analysis group, the Scientific Instrument Technician, and the Radiation Center Director.

The data contained in the following sections 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 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. A summary of required data for the OSTR is provided in Part I.E for quick reference.

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 exposures to radiation and radioactive materials at levels which are consistently "as low as reasonably achievable" (ALARA).

Table V.A.1

Radiation Protection Requirements and Frequencies

FREQUENCY	RADIATION PROTECTION REQUIREMENT
Daily/Weekly/Monthly	Routine area radiation/contamination monitoring.
Weekly	Gamma spectroscopy of the (OSTR) continuous air monitor particulate filter.
Monthly	Routine response checks of radiation monitoring instruments. Monitor radiation levels (μR/hr) at the environmental monitoring stations. Collect and analyze TRIGA primary, secondary, and make-up water. Exchange personnel dosimeters and inside area monitoring dosimeters and review exposure reports. Laboratory inspections. Emergency and safety equipment checks. Neutron generator and tritium assembly contamination survey. Calculate previous month's gaseous waste discharge.
As Required	Process and record solid and liquid waste discharges. Prepare and record radioactive material shipments. Survey and record incoming radioactive material receipts. Monitor and record special radiation surveys. Perform thyroid and urinalysis bioassays. Conduct orientation and training. Issue radiation work permits and provide health physics coverage for maintenance operations.
Quarterly	Prepare, exchange and process environmental TLD packs. Collect and process environmental soil, water and vegetation samples. Orientation for classes using radioactive materials. Collect and analyze sample from reactor ventilation effluent line. Exchange personnel dosimeters and inside area monitoring dosimeters and review exposure reports.
Semi-Annual	Leak test and inventory sealed sources. Floor survey of corridors and the reactor bay. Calibrate portable radiation monitoring instruments and personnel pocket ion chambers. Inventory and inspect Radiation Center equipment located at the Student Health Center, Corvallis Fire Department Haz/Mat van, and Good Samaritan Hospital.
Annual	Calibrate reactor stack effluent monitor, continuous air monitors, remote area radiation monitors, water monitor, and air samplers. Measure face air velocity in laboratory hoods and exchange dust-stop filters and HEPA filters as necessary. Inventory and inspect Radiation Center emergency equipment. Facility radiation survey of the cobalt-60 irradiator and X-ray machine. Personnel dosimeter training.

Table V.B.1

Monthly Summary of Liquid Effluent Releases to the Sanitary Sewer
for the year July 1, 1988 through June 30, 1989(1)
(OSTR Contribution Shown in () and Bold Print)

Date of Discharge (Month & Year)	Total Quantity of Radioactivity Released (Curies)	Detectable Radionuclides in the Waste	Specific Activity For Each Detectable Radionuclide in the Waste, Where the Release Concentration Was >1 x 10^-7 μCi/cc (μCi/cc)	Total Quantity of Each Detectable Radionuclide Released in the Waste (Curies)	**Average Concentration of Released Radioactive Material at the Point of Release (μCi/cc)**	**Percent of Applicable MPC for Released Radioactive Material (%)**	**Total Volume of Liquid Effluent Released, Including Diluent(2)**
NOV 88 Radiation Center Plus OSTR	4.12×10^{-4}	^{3}H ^{60}Co ^{75}Se	3.76×10^{-5} --- ---	4.10×10^{-4} 6.59×10^{-7} 1.02×10^{-6}	3.78×10^{-5}	**1.4%(3)** **0.04%(4)**	**2872**
OSTR Contribution to Above	**(4.08×10^{-4})**	**(^{3}H)** **(^{60}Co)**	**(3.74×10^{-5})** **(---)**	**(4.08×10^{-4})** **(3.05×10^{-7})**	**(3.75×10^{-5})**	**(1.3%)(3)** **(0.04%)(4)**	
APR 89 Radiation Center Plus OSTR	5.90×10^{-4}	^{3}H ^{60}Co ^{65}Zn ^{75}Se ^{137}Cs	3.07×10^{-5} 1.86×10^{-7} --- 5.93×10^{-7} ---	5.44×10^{-4} 3.29×10^{-6} 1.14×10^{-6} 1.05×10^{-5} 2.76×10^{-7}	3.33×10^{-5}	**1.7%(3)** **0.06%(4)**	**4670**
OSTR Contribution to Above	**(4.45×10^{-4})**	**(^{3}H)** **(^{60}Co)**	**(2.51×10^{-5})** **(---)**	**(4.45×10^{-4})** **(8.33×10^{-8})**	**(2.51×10^{-5})**	**(0.8%)(3)** **(0.03%)(4)**	
Annual Total for Radiation Center Plus OSTR	1.00×10^{-3}	See Above	Not Applicable	1.00×10^{-3}	3.50×10^{-5}	**1.6%(3)** **0.06%(4)**	**7542**
OSTR Contribution to Above	**(8.53×10^{-4})**	**(^{3}H)** **(^{60}Co)**		**(8.53×10^{-4})** **(3.88×10^{-7})**	**(3.00×10^{-5})**	**(1.0%)(3)** **(0.03%)(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. There were no liquid effluent releases during months not listed.

(2) The total volume of liquid effluent plus diluent does not take into consideration the additional mixing with the over 7,500,000 gallons per year of liquids and sewage normally discharged by the Radiation Center complex into the same sanitary sewer system.

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

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

Table V.B.2

Monthly Summary of Gaseous Effluent Releases for the Year July 1, 1988 through June 30, 1989 (1)

Date of Discharge (Month & year)	Total Estimated Radioactivity Released (Curies)	Total Estimated Quantity of Argon-41 Released(2) (Curies)	Estimated Average Atmospheric Diluted Concentration of Argon-41 at Point of Release (Reactor Stack) (μCi/ml)	Percent of the Applicable MPC for Diluted Concentration of Argon-41 at Point of Release (Reactor Stack) (%)
JUL 88	0.42	0.42	3.1×10^{-8}	0.8%
AUG 88	0.42	0.42	3.2×10^{-8}	0.8%
SEP 88	0.63	0.63	5.0×10^{-8}	1.2%
OCT 88	0.49	0.49	3.7×10^{-8}	0.9%
NOV 88	0.41	0.41	3.2×10^{-8}	0.8%
DEC 88	0.56	0.56	4.2×10^{-8}	1.1%
JAN 89	0.41	0.41	3.0×10^{-8}	0.8%
FEB 89	0.41	0.41	3.4×10^{-8}	0.9%
MAR 89	0.96	0.96	7.2×10^{-8}	1.8%
APR 89	0.30	0.30	2.3×10^{-8}	0.6%
MAY 89	0.76	0.76	5.6×10^{-8}	1.4%
JUN 89	0.57	0.57	4.4×10^{-8}	1.1%
ANNUAL VALUE	6.3	6.3	4.0×10^{-8}	1.0%

(1) Airborne effluents from the OSTR contained no detectable particulate radioactivity resulting from reactor operations, and there were no releases of <u>any</u> radioisotopes in airborne effluents in concentrations greater than 20% of the applicable MPC value. (20% is a value taken from the OSTR Technical Specifications.)

(2) Routine gamma spectroscopy analysis of the gaseous radioactivity in the OSTR stack discharge indicated the only detectable radionuclide was argon-41.

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

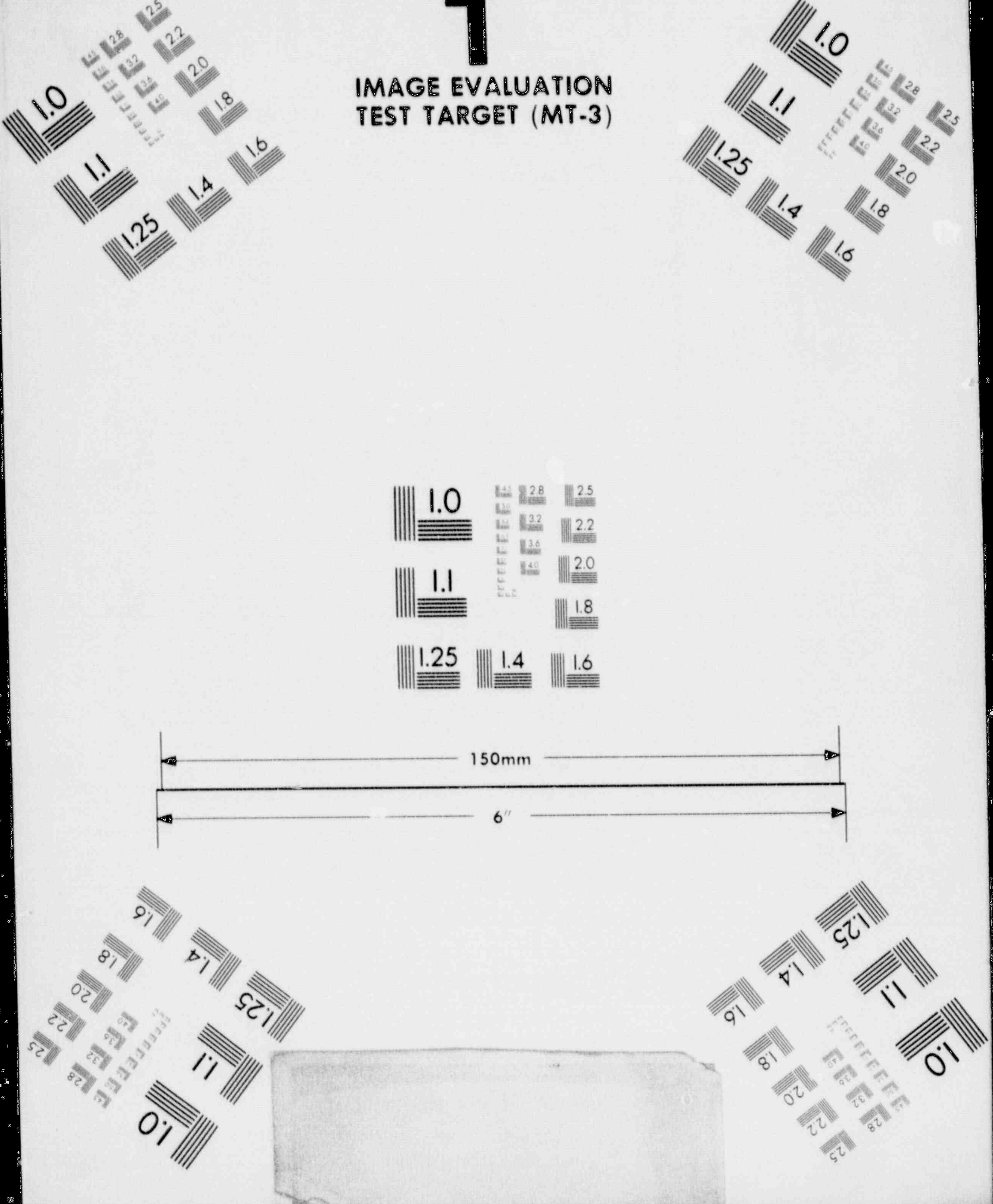


Table V.B.3

Annual Summary of Solid Waste Generated and Transferred for the Year July 1, 1988 through June 30, 1989

Origin of Solid Waste	Volume of Solid Waste Packaged (Cubic Feet)	Detectable Radionuclides in the Waste	Total Quantity of Radioactivity in Solid Waste (Curies)	Dates of Shipment to U.S. Ecology Company(1)
TRIGA Reactor Facility	21	3-Hydrogen 46-Scandium 51-Chromium 54-Manganese 58-Cobalt 59-Iron 60-Cobalt 75-Selenium 82-Bromine 124-Antimony 132-Tellurium 141-Cerium 144-Cerium 152-Europium 154-Europium	5.2×10^{-5}	8/24/88 6/15/89
Radiation Center Laboratories	12	46-Scandium 59-Iron 60-Cobalt 154-Europium 75-Selenium	6.25×10^{-6}	8/24/88 6/15/89

(1) All Radiation Center and OSTR solid radioactive waste is routinely transferred onsite (within the Radiation Center building) to the OSU Radiation Safety Office, 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 by the Radiation Safety Office.

Table V.C.1

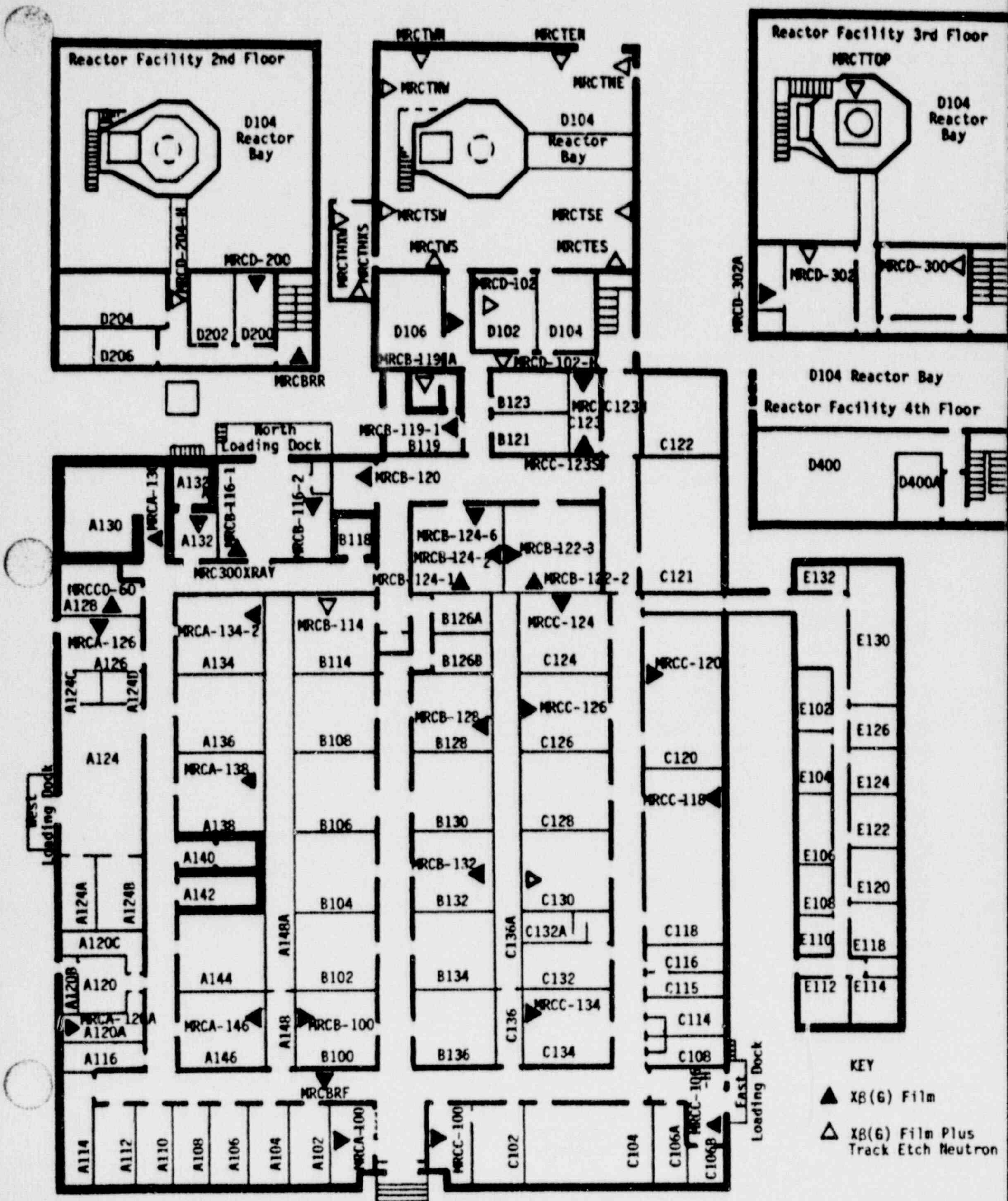
Annual Summary of Personnel Radiation Doses Received
For the Year July 1, 1988 through June 30, 1989

Personnel Group	Average Annual Dose (1)		Greatest Individual Dose (1)		Total Person-mrem For the Group (1)	
	Whole Body (mrem)	Extremities (mrem)	Whole Body (mrem)	Extremities (mrem)	Whole Body (mrem)	Extremities (mrem)
Facility Operating Personnel	12	72	80	500	245	1520
Key Facility Research Personnel	0	5	0	40	0	110
Physical Plant Maintenance Personnel	<1	N/A	8	N/A	42	N/A
Laboratory Class Students	0	7	0	100	0	320
Campus Security and Police Personnel	0	N/A	0	N/A	0	N/A
Visitors	<1	N/A	5	N/A	135	N/A

(1) "0" indicates that each of the beta-gamma dosimeters during the reporting period was less than the vendor's gamma dose reporting threshold of 10 mrem or that each of the neutron dosimeters was less than the vendor's threshold of 30 mrem, as applicable. "N/A" indicates that there was no extremity monitoring conducted or required for the group.

Figure V.D.1

TRIGA Facility and Radiation Center Area Dosimeter Locations



KEY

▲ Xβ(G) Film

△ Xβ(G) Film Plus Track Etch Neutron

Table V.D.1

Total Dose Equivalent Recorded on Area Dosimeters Located Within the TRIGA Reactor Facility for the Year July 1, 1988 through June 30, 1989

Monitor I.D.	TRIGA Reactor Facility Location (See Figure V.D.1)	Total Recorded Dose Equivalent (1)(2)	
		Xβ(G) (mrem)(3)	Neutron (mrem)
MRCTNE	D104 North Badge East Wall	65	0
MRCTSE	D104 South Badge East Wall	0	0
MRCTSW	D104 South Badge West Wall	65	0
MRCTNW	D104 North Badge West Wall	90	0
MRCTWN	D104 West Badge North Wall	0	0
MRCTEN	D104 East Badge North Wall	50	0
MRCTES	D104 East Badge South Wall	250	0
MRCTWS	D104 West Badge South Wall	340	0
MRCTTOP	D104 Reactor Top Badge	405	0
MRCTHXS	D104A South Badge HX Room	370	0
MRCTHXW	D104A West Badge HX Room	20	0
MRCD-302	D302 Reactor Control Room	145	0
MRCD-302A	D302A Reactor Supervisor's Office	15 (4)	N/A

(1) The total recorded dose equivalent values do not include natural background contribution and, except as noted, reflect the summation of the results of 12 monthly beta-gamma dosimeters or four quarterly fast neutron dosimeters for each location. A total dose equivalent of "0" indicates that each of the beta-gamma dosimeters during the reporting period was less than the vendor's gamma dose reporting threshold of 10 mrem or that each of the fast neutron dosimeters was less than the vendor's threshold of 50 to 100 mrem, as applicable. "N/A" indicates that there was no neutron monitor at that location.

(2) These dose equivalent values do not represent radiation exposure through an exterior wall directly into an unrestricted area.

(3) The total recorded dose equivalent values reflect the summation of eleven monthly beta-gamma dosimeters. September's dosimeters were lost at Radiation Detection Company during processing.

(4) The total dose equivalent reflects the summation of four quarterly beta-gamma dosimeters.

Table V.D.2

Total Dose Equivalent Recorded on Area Dosimeters
Located Within the Radiation Center for the Year
July 1, 1988 through June 30, 1989

Monitor I.D.	Radiation Center Facility Location (See Figure V.D.1)	Total Recorded Dose Equivalent (1)	
		X β (G) (mrem)	Neutron (mrem)
MRCA-100	Receptionist's Office	0 (2)	N/A
MRCA-120A	NAA Temporary Sample Storage, A120A	1780	N/A
MRCA-126	Campus RSO's Radioisotope Receiving Lab	15 (2)	N/A
MRCCO-60	^{60}Co Irradiator Room	40 (2)	N/A
MRCA-130	Shielded Exposure Room, A130	0	N/A
MRC300XRAY	X-Ray Console Room	0 (2)	N/A
MRCA-134-2	NAA Research	165 (2)	N/A
MRCA-138	Health Physics Laboratory, A138	0	N/A
MRCA-146	Gamma Analyzer Room (Storage Cave)	15	N/A
MRCB-100	Gamma Analyzer Room (Storage Cave)	0	N/A
MRCB-114	α Lab (^{226}Ra Storage Facility)	1530 (3)	0
MRCB-116-1	RSO's RAM Waste Processing Facility	15 (2)	N/A
MRCB-116-2	RSO's RAM Waste Facility Compactor Room	15 (2)	N/A
MRCB-119	Source Storage Room	15 (2)	N/A
MRCB-119A	Sealed Source Storage Room	3740 (3)	2060
MRCB-120	Instrument Calibration Facility	0 (2)	N/A
MRCB-122-2	Radioisotope Storage Hood	1040 (2)	N/A
MRCB-122-3	Radioisotope Research Laboratory	25 (2)	N/A
MRCB-124-1	Radioisotope Research Laboratory (Hood)	0 (2)	N/A
MRCB-124-2	Radioisotope Research Laboratory	70 (2)	N/A
MRCB-124-6	Radioisotope Research Laboratory	0 (2)	N/A
MRCB-128	Instrument Repair Shop	0 (2)	N/A
MRCB-132	Radioisotope Research Laboratory	200 (2)	N/A
MRCC-100	Director's Office	0 (2)	N/A
MRCC-106-H	East Loading Dock, C106H	0	N/A
MRCC-118	Radio-Chemistry Laboratory	0 (2)	N/A
MRCC-120	Student Counting Laboratory	0 (2)	N/A
MRCC-123N	Gamma Analyzer Room (Storage Cave), C123	135 (2)	N/A
MRCC-123S	Gamma Analyzer Room	0 (2)	N/A
MRCC-124	Student Computer Laboratory	0 (2)	N/A
MRCC-126	Student Counting Laboratory	0 (2)	N/A
MRCC-130	Radioisotope Laboratory	0 (2)	0 (4)
MRCC-134	Gamma Analyzer Room (Storage Cave), C134	145	N/A
MRCD-102	Pneumatic Transfer Terminal Laboratory	95 (3)	0
MRCD-102-H	1st Floor Corridor @ D102	0 (3)	0
MRCD-106-H	1st Floor Corridor @ D106	150 (2)	N/A
MRCD-200	Senior Health Physicist's Office	115 (2)	N/A
MRCD-204-H	2nd Floor Corridor @ D204	0 (3)	0
MRCD-300	3rd Floor Conference Room	15 (3)	0
MRCBRF	Front Personnel Dosimetry Storage Rack	0 (2)	N/A
MRCBRR	Rear Personnel Dosimetry Storage Rack	0 (2)	N/A

(1) The total recorded dose equivalent values do not include natural background contribution and, except as noted, reflect the summation of the results of 12 monthly beta-gamma dosimeters or four quarterly fast neutron dosimeters for each location. A total dose equivalent of "0" indicates that each of the beta-gamma dosimeters during the reporting period was less than the vendor's gamma dose reporting threshold of 10 mrem or that each of the fast neutron dosimeters was less than the vendor's threshold of 50 to 100 mrem, as applicable. "N/A" indicates that there was no neutron monitor at that location.

(2) The total dose equivalent reflects the summation of four quarterly beta-gamma dosimeters.

(3) The total dose equivalent reflects the summation of eleven monthly beta-gamma dosimeters. September's dosimeters were lost at Radiation Detection Company during processing.

(4) The total dose equivalent reflects the exposure for June 1989 only.

Table V.D.3

Annual Summary of Radiation Levels and Contamination Levels Observed Within the Reactor Facility and Radiation Center During Routine Radiation Surveys for the Year July 1, 1988 through June 30, 1989

Accessible Location (See Figure V.D.1)	Whole Body Radiation Levels (mrem/hr)		Contamination Levels (1) (dpm/100 cm^2)	
	Average	Maximum	Average	Maximum
TRIGA Reactor Facility:				
Reactor Top (D104)	1	101	<500	<500
Reactor 2nd Deck Area (D104)	6	42	<500	<500
Reactor Bay SW (D104)	<1	39	<500	<500
Reactor Bay NW (D104)	<1	15	<500	<500
Reactor Bay NE (D104)	<1	7	<500	<500
Reactor Bay SW (D104)	<1	22	<500	<500
Class Experiments (D104,D302)	<1	4	<500	<500
Demineralizer Tank-- Outside Shielding (D104A)	<1	7	<500	<500
Particulate Filter-- Outside Shielding (D104A)	<1	3	<500	<500
Radiation Center:				
NAA Counting Rooms (A146,B100,C134)	<1	3	<500	<500
Health Physics Laboratory (A138)	<1	<1	<500	<500
^{60}Co Irradiator Room (A128)	<1	7	<500	<500
Radiation Research Labs (B114,B122,B124,B132,C130)	<1	7	<500	<500
Radioactive Source Storage (B119A)	<1	29	<500	<500
Student Chemistry Laboratory (C118)	<1	<1	<500	<500
Student Counting Laboratories (C120,C126)	<1	3	<500	<500
Operations Counting Room (C123)	<1	<1	<500	<500
Pneumatic Transfer Laboratory (D102)	<1	3	<500	15000(2)

(1) <500 dpm/100 cm^2 = Less than the lower limit of detection for the portable survey instrument used.

(2) The contamination shown for this location assumes 100% smearing efficiency and was immediately removed. As a result, the average contamination level at this location during the reporting period was, for all practical purposes, <500 dpm per 100 cm^2.

Figure V.E.1

Area Radiation Monitor Locations for the TRIGA Reactor, and on the TRIGA Reactor Area Fence

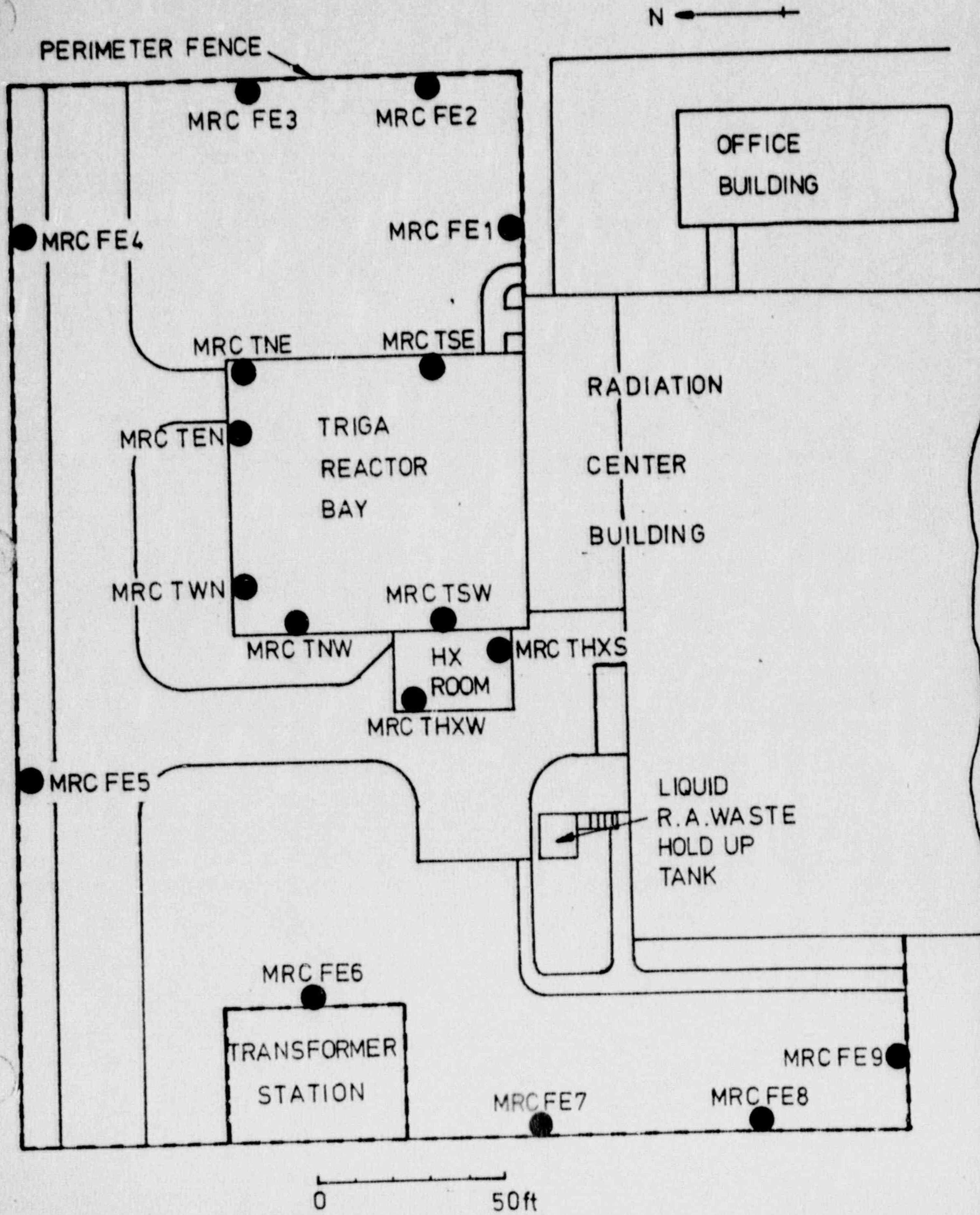


Table V.E.1

Total Dose Equivalent at the
TRIGA Reactor Facility Fence
for the Year July 1, 1988 through June 30, 1989

Fence Environmental Monitoring Station (See Figure V.E.1)	Total Recorded Dose Equivalent Based on R.D. Co. TLDs (1) (mrem)	Total Recorded Dose Equivalent Based on OSU TLDs (2)(3) (mrem)	Total Calculated Dose Equivalent Based on the Annual Average μR/hr Exposure Rate (3) (mrem)
MRCFE-1	100	81 ± 17	77 ± 20
MRCFE-2	106	92 ± 14	80 ± 18
MRCFE-3	106	87 ± 3	84 ± 19
MRCFE-4	112	85 ± 2	92 ± 22
MRCFE-5	98	78 ± 10	76 ± 20
MRCFE-6	98	85 ± 6	81 ± 37
MRCFE-7	101	84 ± 7	75 ± 24
MRCFE-8	103	82 ± 7	73 ± 19
MRCFE-9	96	79 ± 4	64 ± 15

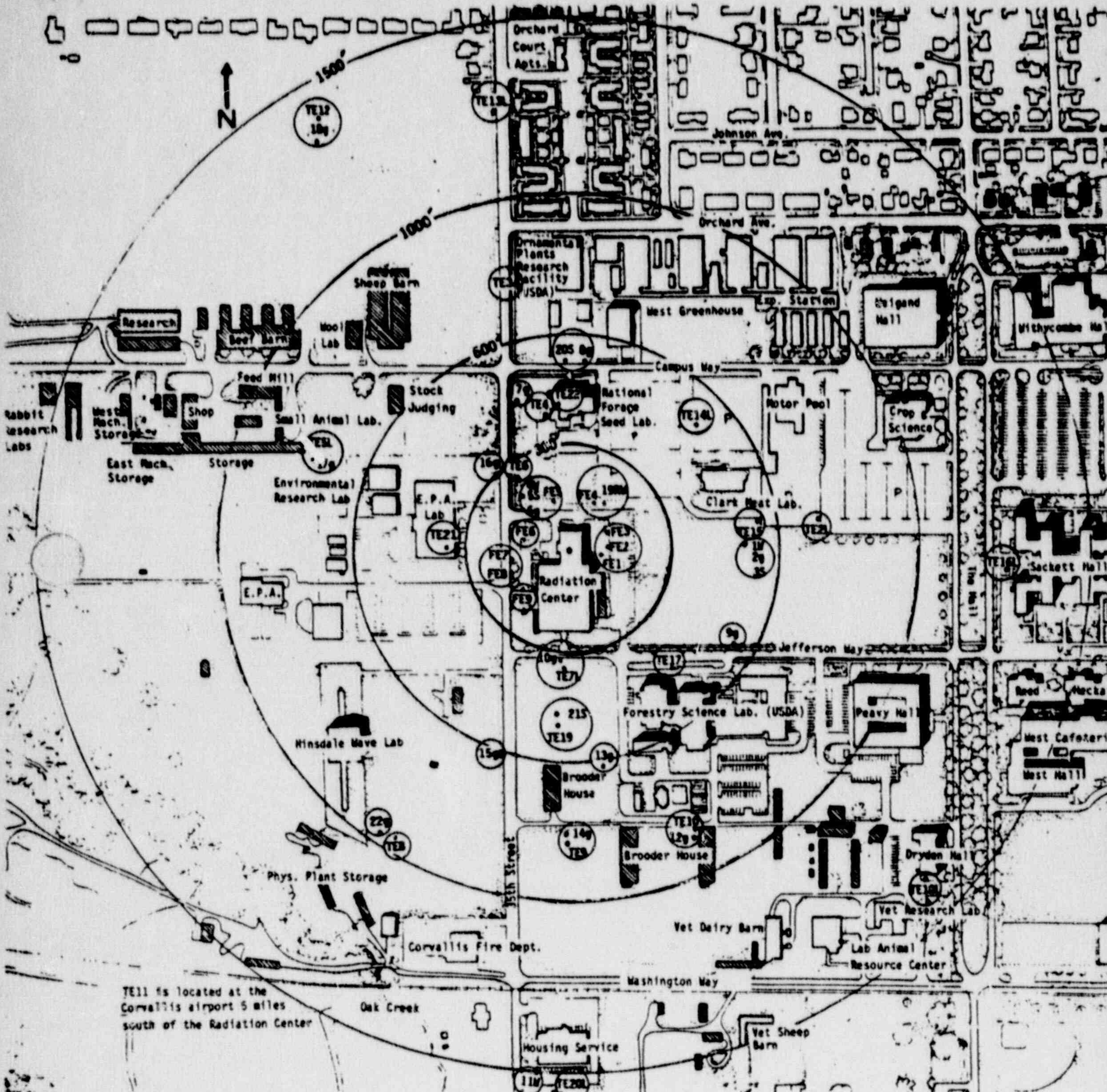
(1) Radiation Detection Company (R.D. Co.) TLD totals include their annual natural background contribution of 80 mrem for the reporting period. Average Corvallis area natural background using Radiation Detection Company TLDs totals 88 mrem for the same period.

(2) OSU fence totals include a measured natural background contribution of 72 ± 2 mrem.

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

Figure V.E.2

Monitoring Stations for the OSU TRIGA Reactor
For the Year July 1, 1988 through June 30, 1989



FE } Gamma
TE }
g Grass
S Soil
W Water
RW Rainwater

SCALE: 0 100 200 300

Table V.E.2

Total Dose Equivalent at the
Off-Site Gamma Radiation Monitoring Stations
for the Year July 1, 1988 through June 30, 1989

Off-Site Radiation Monitoring Station (1) (See Figure V.E.2)	Total Recorded Dose Equivalent Based on R.D. Co. TLDs (2) (mrem)	Total Recorded Dose Equivalent Based on OSU TLDs (3)(4) (mrem)	Total Calculated Dose Equivalent Based on the Annual Average μR/hr Exposure Rate (4) (mrem)
MRCTE-2L	---	69 ± 10 (5)	57 ± 21
MRCTE-3	106	69 ± 9 (5)	81 ± 17
MRCTE-4	96	85 ± 7	69 ± 19
MRCTE-5L	---	94 ± 13	81 ± 15
MRCTE-6	103	95 ± 11	81 ± 16
MRCTE-7L	---	82 ± 9	85 ± 10
MRCTE-8	111	72 ± 7 (5)	90 ± 14
MRCTE-9	108	91 ± 11	87 ± 14
MRCTE-10	91	84 ± 15	64 ± 15
MRCTE-11	90	58 ± 6 (5)	64 ± 27
MRCTE-12	103	100 ± 9	87 ± 20
MRCTE-13L	---	95 ± 5	76 ± 12
MRCTE-14L	---	60 ± 6 (5)	59 ± 16
MRCTE-15	97	65 ± 10 (6)	75 ± 14
MRCTE-16L	---	94 ± 9	80 ± 17
MRCTE-17	95	84 ± 7	68 ± 18
MRCTE-18L	---	102 ± 8	75 ± 16
MRCTE-19	108	101 ± 11	87 ± 13
MRCTE-20L	---	104 ± 14	75 ± 16
MRCTE-21	81	72 ± 4	45 ± 8
MRCTE-22	86	87 ± 9	54 ± 13

(1) Monitoring stations coded with an "L" contained one standard OSU TLD pack only. Stations not coded with an "L" contained, in addition to the OSU TLD pack, one R.D. Co. TLD monitoring pack.

(2) Radiation Detection Company TLD totals include their annual natural background contribution of 79 mrem for the reporting period. Average Corvallis area natural background using Radiation Detection Company TLDs totals 88 mrem for the same period.

(3) OSU off-site totals include a measured natural background contribution of 81 ± 7 mrem.

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

(5) The total dose equivalent for three quarterly monitoring periods only. The TLD packet was lost or stolen during one quarter.

(6) The total dose equivalent for three quarterly monitoring periods only. The TLD packet was mistaken for a pipe bomb and was removed and X-rayed by a bomb squad from the state of Oregon.

Table V.E.3

Annual Average Concentration of the Total Net Beta Radioactivity (Minus ^3H)
for Environmental Soil, Water, and Vegetation Samples
for the Year July 1, 1988 through June 30, 1989

Sample Location (See Figure V.E.2)	Sample Type	Annual Average Concentration of the Total Net Beta (Minus ^3H) Radioactivity (1)	Reporting Units
1-W	Water	(2) $2.91 \times 10^{-8} \pm 3.52 \times 10^{-9}$	µCi/cc
4-W	Water	(2) $2.91 \times 10^{-8} \pm 3.52 \times 10^{-9}$	µCi/cc
11-W	Water	(2) $2.91 \times 10^{-8} \pm 3.52 \times 10^{-9}$	µCi/cc
19-RW	Rainwater	(2) $2.91 \times 10^{-8} \pm 3.52 \times 10^{-9}$	µCi/cc
3-S	Soil	$9.07 \times 10^{-5} \pm 1.37 \times 10^{-5}$	µCi/gram of dry soil
5-S	Soil	$1.06 \times 10^{-4} \pm 1.32 \times 10^{-5}$	µCi/gram of dry soil
20-S	Soil	$6.23 \times 10^{-5} \pm 1.95 \times 10^{-5}$	µCi/gram of dry soil
21-S	Soil	$9.74 \times 10^{-5} \pm 1.25 \times 10^{-5}$	µCi/gram of dry soil
2-G	Grass	$3.23 \times 10^{-4} \pm 2.69 \times 10^{-5}$	µCi/gram of dry ash
6-G	Grass	$2.95 \times 10^{-4} \pm 2.74 \times 10^{-5}$	µCi/gram of dry ash
7-G	Grass	$4.04 \times 10^{-4} \pm 3.03 \times 10^{-5}$	µCi/gram of dry ash
8-G	Grass	$5.03 \times 10^{-4} \pm 3.17 \times 10^{-5}$	µCi/gram of dry ash
9-G	Grass	$3.29 \times 10^{-4} \pm 2.56 \times 10^{-5}$	µCi/gram of dry ash
10-G	Grass	$2.11 \times 10^{-4} \pm 2.76 \times 10^{-5}$	µCi/gram of dry ash
12-G	Grass	$2.59 \times 10^{-4} \pm 2.46 \times 10^{-5}$	µCi/gram of dry ash
13-G	Grass	$4.73 \times 10^{-4} \pm 3.28 \times 10^{-5}$	µCi/gram of dry ash
14-G	Grass	$2.45 \times 10^{-4} \pm 2.58 \times 10^{-5}$	µCi/gram of dry ash
15-G	Grass	$3.16 \times 10^{-4} \pm 2.82 \times 10^{-5}$	µCi/gram of dry ash
16-G	Grass	$3.46 \times 10^{-4} \pm 2.55 \times 10^{-5}$	µCi/gram of dry ash
17-G	Grass	$4.08 \times 10^{-4} \pm 3.07 \times 10^{-5}$	µCi/gram of dry ash
18-G	Grass	$4.58 \times 10^{-4} \pm 3.27 \times 10^{-5}$	µCi/gram of dry ash
22-G	Grass	$3.57 \times 10^{-4} \pm 2.88 \times 10^{-5}$	µCi/gram of dry ash

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

(2) Less than lower limit of detection of $2.91 \times 10^{-8} \pm 3.52 \times 10^{-9}$ µCi/cc.