AN ABSTRACT OF THE THESIS OF

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The Safety Analysis Report (SAR) of the Oregon State University TRIGA Research Reactor (OSTR) was prepared and used as a safety baseline for the reactor's operation since 1968. Although, in general, revision of the Safety Analysis Report of a research reactor is not a regulation requirement, it should be revised from time to time to include changes to the facility or procedures or update to current regulatory standards. The ANS 15.21 workgroup developed a draft standard format and content for safety analysis reports for research reactors. An area of this guidance, which was selected for this work's revision of OSTR-SAR, is the radiation protection program and waste management chapter. The Health Physics program of the facility was observed. The radiological data were obtained from the annual reports for more than 10 years of operation. The related data, such as meteorological data, were obtained and prepared for the analysis processes. The current federal regulation limits and recommendations were used as the references for dose assessments. The results show the OSTR has a sufficient

radiation protection program not only for the facility's workers, but also for the general public, and the program is in full compliance with the federal regulations. The dose estimation shows that the workers and general public can not receive and have not received doses in excess of regulatory limits from the normal operation of the OSTR.

Analysis and Proposed Revision of the Radiation Protection and Waste Management Programs as Described in the Oregon State University TRIGA Research Reactor Safety Analysis Report

By

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TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

LIST OF FIGURES

LIST OF TABLES

Analysis and Proposed Revision of the Radiation Protection and Waste Management Programs as Described in the Oregon State University TRIGA Research Reactor Safety Analysis Report

1. INTRODUCTION

A research reactor requires a Safety Analysis Report (SAR) not only to provide baseline of operation, but also to show the overall safety of the facility. In addition, the regulatory body needs the SAR for safety assessment, as a part of licensing procedures. In general, a SAR must be submitted to the regulatory body before a reactor is constructed, or prior to the initiation of reactor operation. However, any major modification of the facility during operation requires the revision of the SAR to ensure that the overall safety is still intact. Although it is not mandatory that the SAR for a research reactor have to be revised, periodic revision is recommended. [1]

The Oregon State TRIGA Reactor (OSTR) is a water-cooled, swimming pool type research reactor. It is capable of steady state operation up to a power level of 1.1 MW and can be operated on a pulse mode with a peak power level of 3000 MW. The major role of the OSTR is to serve a variety researchers in the fields of nuclear engineering applications and radiation protection for the OSU campus and other universities and colleges throughout the United States. The reactor was constructed in 1966 as a part of the Radiation Center (RC), located at the western site of the OSU campus. The first criticality was reached in March 1967 with a maximum power level of 250 kW with 20% enriched fuel. In 1969, the OSTR amended its license to operate at a power level of

1MW. The fuel was changed to be the FLIP fuel (70% enrichment) in 7 years later. In 1989, the reactor license further amended to permit operation up to 1.1 MW.

The OSTR and fissionable materials are the licensable materials which are subject to control by the US Nuclear Regulatory Commission (USNRC). The OSTR has a license, No.R-106, Docket No.50-243 for reactor operation. The other by-product materials, which include the irradiated and activated products from the reactor are licensed by the State of Oregon with the license No.ORE-90005.

A SAR is required for obtaining permission to operate from the USNRC. The SAR for the Oregon State University TRIGA research reactor (OSTR-SAR) was first prepared in 1968 by John C. Ringle, T V. Anderson, and Arthur G. Johnson. This SAR has been amended from time to time to reflect significant changes to facility. These revisions were submitted separately to the federal regulatory body, the USNRC.

The OSTR-SAR (1968) [2] consists of 5 main parts which are 1) Location and general feature of site, 2) Description of reactor building, 3) Reactor description, 4) Safety summary, and 5) Reactor administration and organization. There is not a specific part that concerned directly with the radiation protection program and waste management. In part 4, the SAR shows the entire reactor system safety. These are the reactor safety performances and parameters, radiation safety, and the reactor accident and impact assessments.

For radiation safety during normal operation, the dose estimation around the reactor was assessed. The conclusion reached was that, at the power of 1 MW, there was not any excessive dose rate existing around the reactor that might cause the reactor

workers doses to exceed any regulatory dose limits. However, the calculations of these dose estimations for the workers were not included in this report.

The airborne radioactive material released to environment was also assessed. ^{41}Ar was identified to be a main source of exposure to general public. The maximum concentration of ^{41}Ar at the point of release was calculated to be 4.0×10^{-6} $\mu Ci\ cm^{-3}$. However, the maximum permissible concentration for the release of ^{41}Ar at that time was limited to the level of 4.0×10^{-8} $\mu Ci\ cm^{-3}$. Because the estimated-release concentration was higher than the regulatory limit, the maximum dose at non-restricted boundary of facility was calculated by using specific meteorological data and the Gaussain plume model. The SAR reported that the maximum dose at the unrestricted boundary should be less than 45 mrem over the year. In 1972, the reactor stack height was changed from 55 feet to 65 feet 10 inches from ground level. This facility change resulted in the maximum dose being reduced to 15 mrem per year. In this calculation, only a prevailing direction of wind (from north) was used to calculate the dose at the three different atmospheric conditions.

In part 5, the organization and the responsibilities of various positions that concern reactor operation were described in detail. A description of the waste management program was not included in this report.

In 1970, the American Nuclear Society Standard Committee established Subcommittee ANS-15 for creating a standard for the operation of research reactors. By the assignment of this Subcommittee, the Work Group ANS 15.21 was established in 1991 for the responsibility of forming a standard guidance for research reactor SAR. The

ANS 15.21-Draft Standard Format and Content for Safety Analysis Reports for Research Reactors was used as a guideline in this thesis for revising the OSTR-SAR.

One chapter (chapter 11) of this draft standard that covers radiation protection and waste management programs was selected to be a topic for revising for the OSTR-SAR. The sequence of each topic was prepared by following the draft format style. The objective of this work is to draft a revision to the SAR to include requirements of this selected chapter. The details of standard format and content of the chapter are shown in Appendix A. It should be noticed that this chapter requires a safety analysis for normal operation, therefore radiological accident analysis including remedial action planing and dose assessment after accident are not included in this thesis.

The revision of the SAR shown in the "results section" of this thesis is divided into two main sections. These are the radiation protection and radioactive waste management programs. The first section includes the description and analysis of radiation sources, radiation protection program, ALARA program, radiation monitoring and surveying, radiation exposure control and dosimetry, contamination control, and environmental monitoring. The radiation dose assessment for the OSTR and general public from the reactor operation was included in radiation source topic. The second section includes radioactive waste management program, radioactive waste controls, and release of radioactive waste.

2. MATERIALS AND METHODS

The OSTR-SAR 1968 was studied with emphasis given to topics related to radiation protection. Chapter 11 of ANS 15.21 Draft Standard Format was thoroughly studied and the requirements in each section then were compared with the existing SAR. The effectiveness of the implementation of the SAR requirements was assessed by direct observation of the OSTR Health Physics program through the RHP 580 "Field Practice in Radiation Protection" class. This class gave the author an opportunity to learn, observe, and gain experience in the practice of Health Physics under the supervision of a Reactor Health Physicist. The procedures, records, audits, etc. of the Health Physics program were studied and a summary of the program is given in the Results section as a part of the revision SAR.

The OSTR radiological data, such as radiation levels on and off site, radiation doses at various sites of monitoring stations and the annual radioactive gaseous and liquid releases were obtained from the annual reports of the Oregon State University Radiation Center and TRIGA reactor (the OSTR annual reports). The annual reports for the year 1986 to 1997 were selected because, during these years, the OSTR Health Physics activities were more stable than the earlier years. This data then was averaged for use in next dose estimation step.

The doses to the OSTR radiation workers were estimated by using a time limit assumption. A radiation worker was assumed to work inside the reactor room for only eight hours per week and 50 weeks per year. A time limitation was applied for working in some "Radiation Areas" such as above the reactor pool. It was also assumed that the

workers have to work on the first floor of the reactor for the rest of time. The annual direct radiation and radiation from an airborne ^{41}Ar were calculated to obtain an estimation of the maximum total effective dose equivalent a worker could receive.

For public dose estimation, the receptors around the reactor were assigned. The nearest campus buildings from the reactor were used for this task. The name and location of these buildings are shown in Table 2 in the Result section. Three radiation source terms were considered to be the main sources that might result in exposure to general public. These are the direct radiation and radiation from gaseous and liquid effluents.

The average annual dose at the reactor fence and some areas at the RC were used as the reference points for assessment the public doses from direct radiation. The doses at the receptor points can be estimated directly from the inverse square law from the reference points.

The liquid effluent data were obtained from the annual reports. An average concentration value of each radionuclide was used to estimate the dose to public. It is unlikely that this liquid effluent pathway would result in exposure to the public, but for the most conservative estimation, the effluents were assumed to be consumed directly by individuals at the receptor points and the doses from such an intake were calculated.

For estimation of atmospheric dispersion of radioactive materials, site specific meteorological data are needed. Since such data are not available for the Corvallis area where the reactor located, the meteorological data of the Eugene airport (about 35 miles, south of Corvallis) were used. The data were obtained from the National Climate Data Center (NCDC), Asheville, NC. The data from NCDC is needed to be converted to a wind rose format in order to be used in this work. The NCDC wind data sets consist of

the three hours average of wind direction and speed and include the annual summary. An example of this data is shown in Appendix B. For creating the wind rose format, each of a three hours interval data have to be rearranged. By using a spreadsheet, a total of 5800 wind data points (obtained from July 1996 to August 1998) were regrouped according to their directions in 16 directions. Finally, a wind rose format that consists of the frequency and average speed of wind in each direction can be obtained. An example of these spreadsheets is shown in Appendix C. The averages of wind speed and frequency in each direction are shown in Appendix D and E respectively.

The COMPLY code was used for dose estimation. This is an U.S. Environmental Protection Agency (US.EPA) program. It is available from the EPA via their web site, http://www.epa.gov/radiation/assessment/software.html. The program is divided into four levels. The minimum input is required for lower levels, but a more complicated data set is necessary in the upper levels. Each level of the program can be initiated separately or can be run consecutively from the lowest level. However, if the users begin at the lowest level and find that the output (the annual effective dose) is complied with the federal regulation, the next level of analysis may not be necessary.

In level 1, the program input requirements include radionuclide name, concentrations or annual possession amounts, and release rates. The program will compare these inputs directly with the regulatory release limit. If the release concentration is found to be lower than the limit, the output will show the regulatory compliance of the facility. If the release is higher than the regulatory limit, then the users must go to the next levels. In level 2, the program calculates the radionuclide concentration by using an atmospheric dispersion model. Therefore, it requires more

input data such as release height, building height, stack diameter, volumetric flow rate, distance from source to the receptor, building width, and wind speed. The concentration of the diluted radionuclide is used to determine exposure to the receptor by direct radiation, inhalation and ingestion. This level assumes the receptor's entire food source is contaminated. The concentration of radionuclide in food is then calculated. At level 3 and 4, the locations of farms that produce food are identified and the site specific concentration are calculated. This generally has the effect of reducing the calculation dose from ingestion. However, there are two farms in level 3 (one for vegetable and one for milk and meat), but three farms for level 4 (one each for vegetable, milk and meat). In addition, only at level 4, the user can use wind rose data to determine dose in specific direction.

The program was written by using the Gaussian plume atmospheric dispersion model which is recommended in NCRP Report No.123I "Screening Models for Releases of Radionuclides to Atmosphere, Surface Water, and Ground" [3]. However, the COMPLY code has the highest level (level 4) which can be used with the wind rose data, while the NCRP report have only 3 levels. It should be noticed that, in this model, the height of the releasing stack is taken into account. If a stack's height is less than 2.5 times of building's height, the wake effect, the turbulence of wind due to the cross section of building which results the increasing of exposure near beyond the building, is being considered.

The ^{41}Ar is considered to be a main source of exposure to general public. The concentration of ^{41}Ar released over the past 12 years, 1986-1997, is available from the OSTR annual reports on annual stack releases [4]. The COMPLY program level 4 was

used for this work and the summary of input data is given in Table 1. The doses from direct radiation, liquid and gaseous effluents were summed to provide a conservative effective dose equivalent to the receptors around the reactor site.

Table 1. Input data for COMPLY analysis of the OSTR ^{41}Ar releases

Parameters	*Input values*
Radionuclide	^{41}Ar
Release Concentration*	3.4×10^{-8} μCi cm^{-3}
Stack Flow Rate	5.66 m^3 sec^{-1}
Release Rate	0.19 μCi s^{-1}
Stack Height	20.07 m
Building Height	14.63 m
Building Length	85.80 m
Building Width	57.07 m

*Average annual release concentration, 1986-1997

3. RESULTS: The Revision of the OSTR-SAR on Radiation Protection Program and Waste Management

3.1 The OSTR Radiation Sources

3.1.1 Solid Radiation Sources

3.1.1.1 OSTR fuels and control rods[5]

The OSTR core consists of the total of 82 FLIP fuels (FLIP= Fuel Lifetime Improvement Program), 3 control rods with fuel follower (safe, shim and regulating rod), and 1 control rod with void follower (transient rod). The total amount of ^{235}U in the core is 11.347 kg. Each fuel rod contains about 136 to 138 grams of uranium (70% enrichment, 8.5 wt-%). A fuel rod is 37.3 mm in diameter and 673.1 mm in length and each rod has a stainless steel cladding with a thickness of 0.51mm. Each fuel follower control rod has same amount of uranium, but a borated graphite is added at the top. A control rod is 1098.6 mm long, with same diameter and cladding thickness of fuel rod.

3.1.1.2 Neutron Source[5]

A 3 Ci of $^{241}Am/Be$ source has been used for reactor start up which gave a neutron emission rate about 6.0×10^6 neutrons per second. The source is contained in a cylindrical aluminum tube holder and placed in one of the outer most positions of core.

3.1.1.3 Solid Wastes

These wastes include dry solid and solidified liquid wastes generated from the operation and maintenance programs of the reactor. The average activity produced each year (from 1986 to 1997) is 1.25×10^{-3} Ci. The main radioisotopes found in these wastes are ^{46}Sc, ^{51}Cr, ^{54}Mn, ^{58}Co, ^{59}Fe, ^{60}Co, ^{65}Zn, and ^{152}Eu [4].

3.1.2 Liquid Radiation Sources

3.1.2.1 Primary Water Coolant.

The primary reactor water tank contains 4600 gallons of water that includes trace contaminants which are activated and become a liquid radiation source. After 3 hours of operation at the power of 1000 kW, the radioactivity of the water has been measured with an average of 1.75 $\mu Ci\ cm^{-3}$ [6]. The specific radionuclides found in the reactor tank water include ^{22}Na, ^{27}Mg, ^{41}Ar, ^{56}Mn, and other very trace isotopes. Because of the closed circuit of the primary water circulation, the radiation exposure from this source is limited to some specific areas, such as around coolant pipes, the demineralizer-tank. The worker exposure occurs only during the reactor maintenance operations.

3.1.2.2 Reactor Effluents.

All liquid effluents from the reactor facility are collected via a drainage system which includes a retention tank located underground at the north side of the Radiation Center. After the radioactive material concentration of the water is determined to be less than the regulatory limits for disposal, the water can be released to the sewer. The

primary radionuclide found in the effluents is ^{3}H (99.23%, averaged from the annual total activity) with the average concentration of 1.57×10^{-4} μCi cm^{-3}. Occasionally, traces of radionuclides such as ^{60}Co, ^{24}Na are identified, but the concentrations are less than 1.0×10^{-7} μCi cm^{-3} [4].

3.1.2.3 Liquid Wastes.

The liquid wastes generated from decontamination processes, such as the first rinse of water for cleaning the TRIGA tubes or laboratory glassware, are solidified via absorption and then are treated as solid waste.

3.1.3 Airborne Radiation Sources

3.1.3.1 Particulate Airborne Sources.

In normal operation of the OSTR, the only detectable particulate airborne radionuclides are the progenies of radon, such as ^{214}Pb and ^{214}Bi, with the concentrations of 1.0×10^{-9} to 3.0×10^{-11} μCi cm^{-3} [4]. These natural occurring sources are not the result of licensed activities.

3.1.3.2 Gaseous Airborne Sources.

There are two gaseous radioactive sources generated during reactor operation. ^{16}N is produced from the activation of oxygen in the reactor tank and ^{41}Ar is also activated from ^{40}Ar which dissolved in the primary water or from air trapped in experimental facilities. The 1968 SAR estimated the concentration of ^{16}N to be

7.34×10^{-6} μCi cm^{-3} at the surface of the tank, and 3.8×10^{-9} μCi cm^{-3} in saturation condition in the reactor room. Because of a very short half-life radionuclide, 7 seconds, and low concentration, ^{16}N was not considered to be a source of radiation exposure. The ^{41}Ar is produced from the experimental facilities, such as beam ports and thermal column, and reactor water. The maximum concentration of release at the stack was estimated in the 1968 SAR to be 4.0×10^{-6} μCi cm^{-3}, while the maximum concentration of release in current 10 CFR 20 is 1.0×10^{-8} μCi cm^{-3}. However there were some attempts for reducing this concentration that included purging the rotating rack with nitrogen and closing of vent valves to many of other facilities [7]. The actual concentration of release has been measured and found that the average concentration is 3.4×10^{-8} μCi cm^{-3} [4]. The concentration of ^{41}Ar in reactor room was also determined and the average of 2.2×10^{-7} μCi cm^{-3} is found [7]. This concentration is lower than the Derived Air Concentration (DAC) which equals to 3.0×10^{-6} μCi cm^{-3}.

3.2 Dose Estimation to OSTR Radiation Workers

The main radiation sources that might introduce dose to the radiation workers during reactor operation are the direct radiation from the reactor and the airborne radioactivity (^{41}Ar). At a reactor power of 1000 kW, there are some "hot spot" areas where radiation levels are quite high inside the reactor room. Some areas were classified to be the "High Radiation Areas". These areas would include a beam port when the shutter is opened. Access is generally not granted to people wishing to enter, therefore the occupational dose from these areas is not of concern. Some areas were classified to be "Radiation Areas". These areas are located around the reactor room. The details and

descriptions of these areas are specified in section 3.7. The maximum radiation field of OSTR is located at the reactor top, which has an average level of 100 mrem h^{-1} during operation[4]. The limitation of working time has been applied for personnel occupying this area. According to the Oregon State TRIGA Reactor Operation Procedures Number 6 (OSTROP 6), the Administrative and Personnel Procedures, the routine occupational occupancy on the reactor top is limited to 30 minutes per week when the reactor power level exceeds 100 kW. If we assume a working time for a worker is 40 hours per week and 50 weeks per year, then maximum dose that this worker might receive from working at the reactor top is 2500 mrem y^{-1}. If we apply this limitation for working in the other "radiation areas" where an average radiation field is 5 mrem h^{-1} such as the reactor's second deck[4], the dose from this situation is 125 mrem y^{-1}. Then if this worker spends the rest of working time (39 hours per week) in the reactor room where the average radiation level is less than 1 mrem h^{-1} [4], the annual dose from this case is less than 1950 mrem y^{-1}.

As mentioned in section 3.1.3.2, the ^{41}Ar dispersed in reactor room is considered to be a source of radiation exposure to the workers. The concentration of ^{41}Ar in the reactor room, under equilibrium condition (after 2 hours of reactor operation), is estimated to be 2.2×10^{-7} $\mu Ci\ cm^{-3}$ which equals to a dose of 366.7 mrem y^{-1}.

Therefore, the total of maximum dose estimated for a worker is less than 4941.7 mrem y^{-1}. It can be concluded that, in the worst case in which a radiation worker works in the reactor room for 40 hours per week, the annual estimation dose is still less than the regulatory dose limit (5000 mrem y^{-1}).

3.3 Dose Estimation to General Public

Radiation dose that might be introduced to general public from operation of a research reactor can be categorized from four radiation sources. These are direct radiation, solid, liquid and gaseous effluents. However, for OSTR operation, only direct radiation, and exposure to liquid, and gaseous effluents are considered to be radiation sources that might increase an amount of dose to general public. Because of an OSU waste management policy that does not allow users (including the OSTR) to dispose solid wastes onsite, the radiation from solid wastes is not considered. For public dose estimation, the receptors around the reactor were assigned as shown in Figure 1.

3.3.1 Dose from the Direct Radiation

The direct radiation from the reactor has been monitored by TLD dosimeters around the reactor site. The results of TLD monitoring program at the reactor fence and some locations of the Radiation Center were used to represent the direct radiation dose to non-restricted areas. The results from 12 years of monitoring show only 2 of 9 fence stations where located at North-North Eastern (NNE) direction (the MRCFE-4, 41.15 meter from reactor) and East-North-Eastern (ENE) direction (the MRCFE-3, 36.58 meter from reactor) have the significant doses above the background radiation level. The dose averages are 18.3 and 16.6 mrem y^{-1} respectively [4]. The TLD monitoring stations are described in the detail in section 3.6.1. Because the fence stations are not available in every direction, the dose at NNE direction was used as the reference dose for the North and North Eastern directions. There are not any significant doses above background level (NS) are observed in the other fence stations and the results of the selected location

dosimeters from the Radiation Center (MRCC series and MRCBRF) show a Non-Detectable dose (ND) [4]. The details of dosimeter locations are stipulated in section 3.6.1. Only the significant doses mentioned above were used to calculate the dose to the receptors. The summary of estimated-doses to receptors is shown in Table 2.

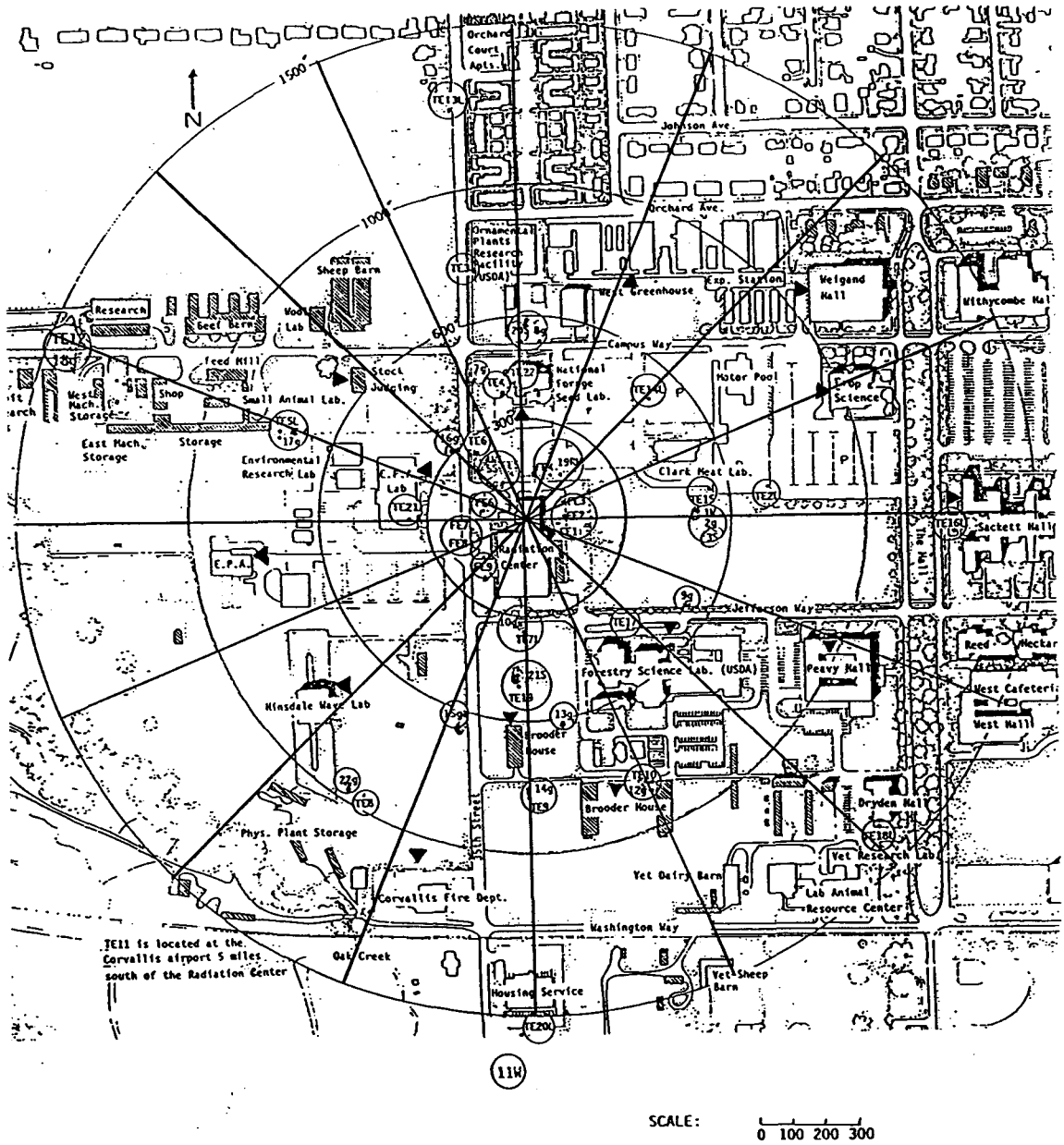


Figure 1. The designated-receptors around the OSTR (▲)

Table 2. The estimated-dose to the receptors by direct radiation from the OSTR

Direction from Reactor	Reference TLD Station	Average Dose at Station (mrem y^{-1})	Name of Receptor Building	Distance form Reactor to Receptor (m)	Estimated Dose to Receptor (mrem y^{-1})
N	MRCFE-4	18.3	National Forage Seed Lab.	114.3	2.37
NNE	MRCFE-4	18.3	West Green House	228.6	0.59
NE	MRCFE-4	18.3	Weigand Hall	320.0	0.30
ENE	MRCFE-3	16.6	Clark Meat Lab.	137.2	1.18
E	MRCFE-2	NS	Sackett Hall	342.9	0
ESE	MRCFE-1	NS	Peavy Hall	320.0	0
SE	MRCC-118	ND	Forestry Science Lab.	160.0	0
SSE	MRCC-106A	ND	Brooder House	251.5	0
S	MRCC-100	ND	Brooder House	182.9	0
SSW	MRCBRF	ND	Corvallis Fire Dept.	342.9	0
SW	MRCFE-9	NS	Hinsdale Wave Lab.	228.6	0
WSW	----	---	------	----	--
W	MRCFE-7	NS	EPA	251.5	0
WNW	MRCFE-6	NS	EPA Lab.	114.3	0
NW	MRCFE-5	NS	Stock Judging	182.9	0
NNW	----	---	-------	---	--

ND = Non Detectable dose
NS = Not Significant dose above background level

3.3.2 Dose from the Liquid Effluents

As mentioned in section 1.2, the OSTR liquid effluents are drained to a retention tank before released to the public sewage. More than 99 % of the total activity found in the effluent is ^{3}H which has an average concentration of 1.57×10^{-4} μCi cm^{-3}. The trace radionuclides are ^{60}Co, ^{24}Na, ^{99}Tc and etc., with average concentrations being less than 1.0×10^{-7} μCi cm^{-3} and the total activity of release was estimated to be 8.45×10^{-4} Ci per year [4] . Up to six radionuclides are occasionally detected but only in trace concentration. For the purpose of dose calculation, it is assumed that all of six minor radionuclides have the same concentration and that are equal to that of the measured ^{60}Co concentration. The ^{60}Co has the lowest value of Annual Limit of Intake (ALI) among the minor radionuclides released.

From the federal regulation, 10 CFR 20, the concentration of ^{3}H and ^{60}Co that can be released to public sewage is 1.0×10^{-2} μCi cm^{-3} and 3.0×10^{-5} μCi cm^{-3} respectively. However, the total activity must be less than 5 Ci per year for ^{3}H and 1Ci per year for other radionuclides. The regulation requires a fraction limit calculation, if more than one radionuclide is released. For this work, the calculation is shown as following.

$$\text{Fraction Limit} = \frac{^{3}\text{H Release Concentration}}{^{3}\text{H Release Limit}} + \frac{(^{60}\text{Co Release Concentration}) \times 6}{^{60}\text{Co Release Limit}}$$

$$= \frac{1.57 \times 10^{-4}}{1.0 \times 10^{-2}} + \frac{(1.0 \times 10^{-7}) \times 6}{3.0 \times 10^{-5}} = 0.036$$

The sum of the fraction limit is substantially less than unity. That means the release concentration is acceptable. In addition, the total activity released each year is also within the limits. Therefore the effluents can be allowed to be released to the public sewage. For the most conservative method of the dose estimation, we assume the receptors consume these effluents directly. From 10CFR20, the water consumption rate of a person is assumed to be 7.3×10^5 cm^3 y^{-1}. Therefore, the total of activity of ^3H and six times of ^{60}Co activity in a receptor's body was calculated to be 114.61 $\mu Ci\ y^{-1}$ and 0.438 $\mu Ci\ y^{-1}$ respectively. From the ALI values of each radionuclide, the dose from intake can be calculated, which equals to 7.16×10^{-3} rem y^{-1} for ^3H and 1.1×10^{-2} rem y^{-1} for ^{60}Co. The estimated -maximum total dose to general public from these liquid effluents is 1.82×10^{-2} rem y^{-1}.

3.3.3 Dose from the Gaseous Effluents

From the OSTR annual reports (1986-1997), the average release concentration of ^{41}Ar is 3.4×10^{-8} $\mu Ci\ cm^{-3}$ which is greater than the regulation limit (1.0×10^{-8} $\mu Ci\ cm^{-3}$). Therefore, the dose estimations to general public have to be determined. The same receptors were assigned in each 16 directions around the reactor. A meteorological data and the COMPLY code is used to predict the receptor's dose around the reactor site. The summary of wind data is shown in Table 3, and the dose estimation from the COMPLY code is shown in Table 4.

Table 3. The summary of wind data used for COMPLY code calculation

Wind From	Frequency	Average Speed (mile h^{-1})
Calm	0.171	Less than 3
N	0.152	8.060
NNE	0.019	5.721
NE	0.011	4.838
ENE	0.011	4.877
E	0.011	4.833
ESE	0.034	6.564
SE	0.080	6.911
SSE	0.118	8.315
S	0.129	7.864
SSW	0.066	7.894
SW	0.049	8.066
WSW	0.028	6.808
W	0.025	6.871
WNW	0.019	6.205
NW	0.034	7.514
NNW	0.052	8.325

Table 4. The estimated-dose from ^{41}Ar releasing calculated by the COMPLY code

Direction from Release Point	Name of Receptor Building	Distance from Release Point to Receptors (m)	Estimated dose (mrem y^{-1})
N	National Forage Seed Lab.	114.3	5.6×10^{-2}
NNE	West Green House	228.6	1.2×10^{-2}
NE	Weigand Hall	320.0	5.6×10^{-3}
ENE	Clark Meat Lab.	137.2	1.1×10^{-2}
E	Sackett Hall	342.9	3.0×10^{-3}
ESE	Peavy Hall	320.0	3.0×10^{-3}
SE	Forestry Science Lab.	160.0	1.0×10^{-2}
SSE	Brooder House	251.5	8.1×10^{-3}
S	Brooder House	182.9	3.7×10^{-2}
SSW	Corvallis Fire Dept.	342.9	3.0×10^{-3}
SW	Hinsdale Wave Lab.	228.6	3.3×10^{-3}
WSW	------	----	----
W	EPA	251.5	3.0×10^{-3}
WNW	EPA Lab.	114.3	1.8×10^{-2}
NW	Stock Judging	182.9	2.3×10^{-2}
NNW	-------	----	----

From these results, if we assume each receptor has a same probability to receive dose from liquid effluents, the total dose to each receptor from direct radiation, liquid and gaseous effluents can be estimated. The results are shown in Table 5.

Table 5. The total dose to the public from the OSTR operation

Direction from Reactor	Name of Receptor Building	Direct Radiation Dose (mrem y^{-1})	Dose from Liquid Effluent (mrem y^{-1})	Dose from Gaseous Effluent (mrem y^{-1})	Total Dose (mrem y^{-1})
N	National Forage Seed Lab.	2.37	18.20	5.6×10^{-2}	20.63
NNE	West Green House	0.59	18.20	1.2×10^{-2}	18.80
NE	Weigand Hall	0.30	18.20	5.6×10^{-3}	18.51
ENE	Clark Meat Lab.	1.18	18.20	1.1×10^{-2}	19.39
E	Sackett Hall	0	18.20	3.0×10^{-3}	18.20
ESE	Peavy Hall	0	18.20	3.0×10^{-3}	18.20
SE	Forestry Science Lab.	0	18.20	1.0×10^{-2}	18.21
SSE	Brooder House	0	18.20	8.1×10^{-3}	18.21
S	Brooder House	0	18.20	3.7×10^{-2}	18.24
SSW	Corvallis Fire Dept.	0	18.20	3.0×10^{-3}	18.20
SW	Hinsdale Wave Lab.	0	18.20	3.3×10^{-3}	18.20
WSW	---	---	---	---	---
W	EPA	0	18.20	3.0×10^{-3}	18.20
WNW	EPA Lab.	0	18.20	1.8×10^{-2}	18.22
NW	Stock Judging	0	18.20	2.3×10^{-2}	18.22
NNW	---	---	---	---	---

From the results in table 5, we can conclude that the total dose to the public from the OSTR operation is well below the regulation limits (100 mrem per year) and is dominated by the liquid release number.

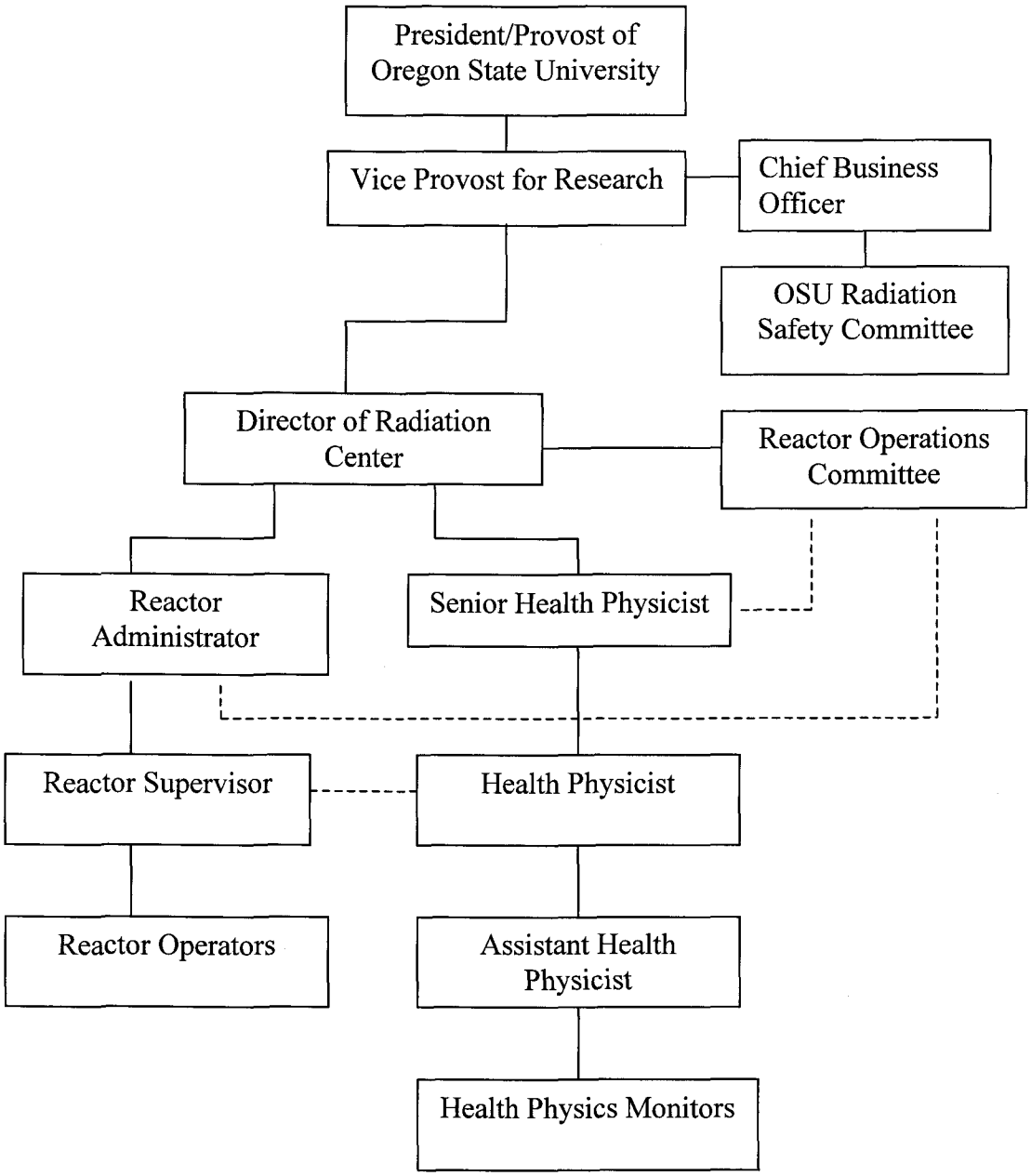
3.4 The OSTR Radiation Protection Program

Because the radiation generation machines and radioactive materials are being used for research purposes at the Oregon State University (OSU), by the requirements of the federal regulation, An OSU Radiation Safety Committee (OSU-RSC) was established. The major responsibilities of this committee are to create and implement the radiation safety policy that must be applied for all of radiation and radioactive materials utilization facilities around the campus. The OSU-RSC consists of at least 9 members from various kind of academic departments. These members are appointed by the Vice President for Finance and Administration. The OSU-RSC policy provides for the safety of staff, students, and the general public from radiation and radioactive material effluents. The ultimate goals of the policy are to control the doses to a level as low as reasonably achievable, to ensure that there is no risk of radiation shall be incurred unless the justification and optimization processes are introduced, and all of radiation applications must comply with federal and state laws[8]. A Radiation Safety Officer (RSO) is appointed to be the secretary of the committee and responsible for the radiation safety programs operation throughout the campus. In addition, each radiation facility requires a Program Director who has a primary responsibility for all of safety aspects in his program.

The Radiation Center (RC) was established to be the center of radiation and radioactive materials utilization of OSU. The Director of Radiation Center is assigned to be the Program Director for managing all activities in the RC and is responsible for the safe operation of the Oregon State TRIGA Reactor (OSTR). For the OSTR administration and operation, the Reactor Operating Committee (ROC) was also

established with at least 7 members with the expertise in Nuclear Engineering, Radiation Protection and another discipline of Engineering. In general, the members will be appointed for a year term by the Director in consultation with the Reactor Administrator and the Chairman of the ROC. One of the members must hold a Senior Reactor Operator license. The ROC meets on a quarterly basis. Subcommittees are established for the purpose of reviewing and auditing the reactor and radiation protection programs. The audit programs include review of reactor records and procedures, inspection of the reactor area, auditing of radioactive effluents, radiation dose, radiation survey, and radioactive materials transportation. In addition, the following auditing programs have been performed on annual basis: the OSTR active experiments, Emergency Response Plan, Physical Security Plan. The Radiation Center Health Physics Procedures are also audited but on a quarterly basis. The review programs are also tasked with the review of abnormal occurrences during OSTR operation and the review and approval of facility or procedures changing.

The summary of the organization structure is shown in Figure 2.



_____ Normal administrative channel
---------- Technical review, communications and assistance

Figure 2. The organization chart of the OSTR administration

The responsibilities of the main working groups are summarized as following.

1. OSU-RSC has responsibilities concerning about the policy making that respect with the overall radiation safety for OSU and establishing standards and regulations to implement the policy.

2. ROC is responsible for independent review, evaluation, and approval for the safe operation of the OSTR.

3. Radiation Center Director has duties as a chief administration person for RC and OSTR and to be the Program Director for OSTR operation.

4. Reactor Administrator is responsible for providing the guidelines, technical supports, and recommendations to the OSTR operation and coordination with Reactor Supervisor and Senior Health Physicist.

5. Reactor Supervisor and Reactor operators have duties on reactor operation and maintenance according to the requirement of license.

6. Senior Health Physicist is responsible for implementation the radiation safety policy to OSTR and managing the OSTR Health Physics program.

7. Health Physicist, Assistant Health Physicist, and Health Physics Monitor are responsible for routine Health Physics operations and other duties assigned by the Senior Health Physicist.

The line of command and administration between the reactor operation group and the Health Physics group is completely separated. The radiation safety programs are free to follow the policy of OSU-RSC. Any radiation safety concerns are reported directly to the Radiation Center Director. In addition, the Senior Health Physicist has authority to report directly to the OSU President 's office any undesired situations that involve with

the radiation safety, if the appropriate action is not taken by the Radiation Center Director.

The Radiation Protection plans and procedures have been established by the Health Physics group and are reviewed periodically by the ROC. These procedures, the Oregon State University Radiation Center Health Physics Procedures (RCHPP), describe step by step activities of the Health Physics group at OSTR and RC. A total of 36 procedures are being used at present. All Health Physics activities shall be supervised and approved by the Senior Health Physicist. These activities are kept on record and used for safety review and audit by the ROC.

The training program is one of the most useful tools for radiation protection. Therefore, the Health Physics group provides various levels of training program to anyone that might have activities related with radiation or radioactive materials within the RC and the OSTR. According to the RC regulations, a training program is required for everyone who works, occupies, studies, or performs research at the RC, no matter if his or her works are directly related with radiation. This basic training program, or namely the General Orientation, provides the general safety rules, access control system, basic knowledge about radiation, and the emergency procedures. There are other 5 parts of training program which the requirement depends on the characteristic of work at the RC and OSTR. The radiation workers, Nuclear Engineering and Radiation Health Physicist students, researchers and any individual who desire unescorted access to the room or area where radiation or radioactive materials are presented are required to pass to Part 2, the Radioactive Material User Orientation training program. This part provides the general concepts of radiation protection policy, the knowledge about the authorization of use and

possession of radioactive material, radiation surveys, protective equipment, inventory control, dose limitation, posting and labeling of radiation sign, personnel monitoring and record, radioactive management and emergency procedures. The part 3 training program is required for individuals who work on an unescorted basis in reactor restricted area. Part 4,5 and 6 are provided for other students, visitors, and temporary workers respectively.

3.5 The OSTR ALARA Program

The ALARA program can be implemented for two different radiological conditions, these are normal and abnormal or accident conditions. The normal condition can be described as the situation which the radiation source is under control and the exposure dose can be limited by some protective measures. The radiological accident might be declared whenever the radiation source is out of control and the exposure dose can be limited only by remedial action. [1]

For normal operation of the OSTR, the objectives of the ALARA program are to avoid unnecessary radiation exposure to the workers and general public, keep the dose as low as reasonably achievable, and to ensure that the individual dose will not exceed the limit.

To accomplish these objectives, the operation or the secondary dose limit has been set up, besides the primary or regulatory limits. The dose that individual might receive from a certain radiation activity is recorded and evaluated, then the limits are set up and used as the secondary limits. In some situation, the secondary limit might be set in higher than the primary one in a certain limit of time (a planed special exposure).

However, the OSU-RSC policy does not allow the concept of planned special exposure to be applied to the workers, and thus the OSTR secondary limits are not higher than the primary limits.

For ensuring the ALARA program is implemented, all of the following provisions are established and used as base line functions for OSTR Health Physicist group.

a) Protection all of RC and OSTR workers against unnecessary exposure
b) Preparing the instrument and equipment for personnel monitoring
c) On-site radiological monitoring and survey
d) Environmental radiological monitoring
e) Decontamination of personnel, equipment, and structures
f) Detecting and recording radioactivity release
g) Personnel dose record and evaluation
h) Radiation sources inventory
i) Radioactive waste management
j) Training program
k) Radiation Monitoring and Surveying

The review and audit programs are realized to be the keys to accomplish the ALARA concept. The routine operation records, any changing or modification of facility or procedures have been reviewed or audited to ensure that no unnecessary dose is introduced to the workers and public.

3.6 Radiation Monitoring and Surveying

3.6.1 Direct Radiation Monitoring

For the areas in reactor room where designated to be "High Radiation Areas", "Radiation Areas", and the areas where the potential exists that a high radiation area is likely to occur, the Area Radiation Monitors (ARMs) have been installed. A total of 12 ARMs have been positioned around these areas and the readouts can be observed at the reactor control room or at their own stations. The intermediate (alert warning) and high level alarms were set in each station. However, the alarm levels are different in each station depending on the background radiation during reactor operation and the maximum of radiation which can be allowed to present in these areas. The locations, range and the typical alarm level are shown in Table 6.

Table 6. The locations and alarm level of ARMs located in reactor room [7]

No.	Location	Intermediate Alarm (mR h^{-1})	High Alarm (mR h^{-1})
1	Beam Port #1	50	100
2	Beam Port #2 / Thermal column	50	100
3	Beam Port #3	50	100
4	Beam Port #4	50	100
5	Control Room	10	100
6	Reactor Top	750	1000
7	Reactor Top	750	1000
8	Fuel Storage Pits	10	20
9	Sample Handling Area	50	100
10	Pneumatic Transfer Terminal	50	100
11	Demineralized System Filter	50	100
12	Demineralizer Column	50	100

For the areas where significant gamma doses are likely to occur, or the areas where full or partial occupancy are assigned, with or without the ARMs, a manual radiation surveying program is implemented. The total of 9 positions within the reactor room and other 15 positions outside are surveyed by Health Physics Monitors on a daily, weekly and monthly basis. Portable "Ionization Chamber" survey meters are used for this task. Unlike the radiation monitoring by ARMs which usually inform whether the radiation is existing or not, the manual surveying gives information about dose which the occupants might receive. Therefore, an ionization chamber is considered to be an appropriate device. A neutron REM meter (NEMO) which consists of ^{6}LiI crystal and covered by a polyethylene moderator is used for neutron dose rate measurement.

The integrated dose both inside and outside reactor building have been observed by a numerous of TLD stations. The total of 13 stations is located within reactor building. A dosimeter at each station contains a standard personnel type beta-gamma film pack, plus a CR-39 plastic track-etch neutron detector in some specific key stations. With the same type of dosimeter, the other 47 monitoring stations is located around the RC area. These film packs will be exchanged quarterly. The locations of these TLD stations are shown in Figure 3.

There are total of 9 TLD stations are designated at the reactor fence and two types of TLD are used in each station, one owned OSU and the other supplied by an outside vendor. The OSU device consists of three Harshaw ^{7}LiF TLD700 chips in a plastic mount, while the other pack contains a $CaSO_4$ TLD. Both of these devices are exchanged on a quarterly basis. In addition, direct radiation surveys are carried out on a monthly basis at fence TLD positions. These measurements are made with a NaI detector (micro rem meter). The locations of fence dosimeter are shown in Figure 4.

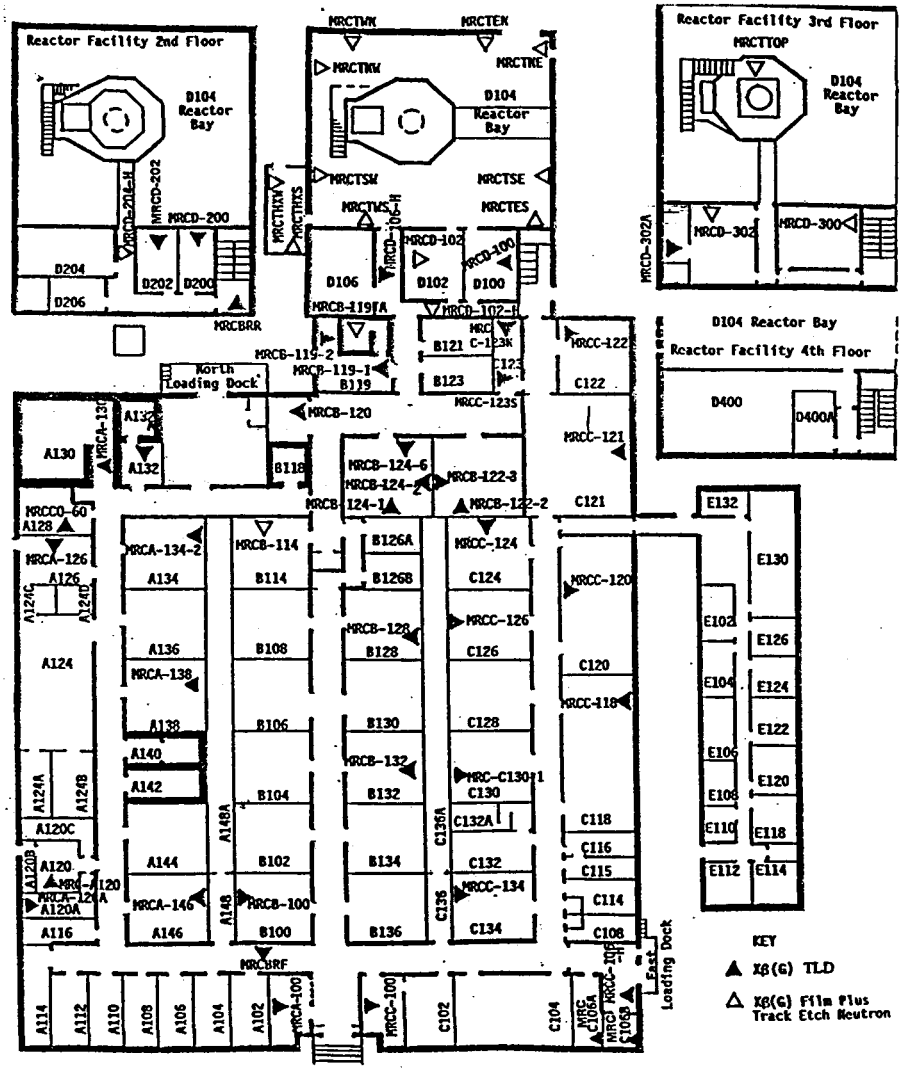


Figure 3. The location of the on-site TLD stations

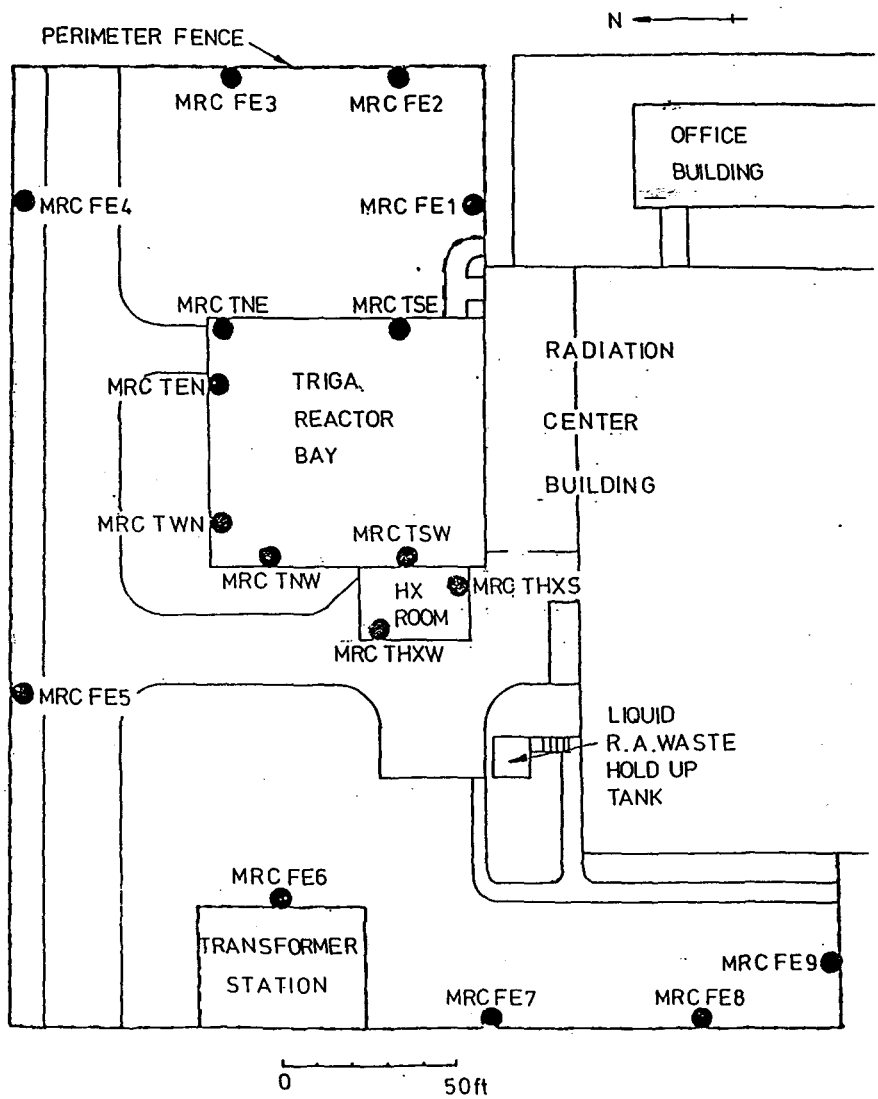


Figure 4. The locations of the fence TLD stations

3.6.2 Contamination Monitoring

The OSU strategies for dealing with contamination problem are to detect and decontaminate it as soon as possible. There are two different methods applied for contamination detection at the OSTR, these are direct and indirect method. The direct method is performed by the use of an opened-window GM detector (Pancake Type) to detect both fix and loose contamination on suspect surfaces. A thin window of detector can detect both beta and gamma contamination, but for the thick window, only gamma contamination can be determined. Once the contamination is found the contamination activity can be calculated by the following formula.

$$\text{Activity/area} = \frac{\text{Reading Contamination (cpm)} - \text{Background Count Rate (cpm)}}{\text{Yield (cpm dpm}^{-1}\text{)} \cdot \text{Area of Detector Window (cm}^2\text{)} \cdot 2.22 \times 10^6 \text{ dpm } \mu\text{Ci}^{-1}}$$

If the contaminated radionuclides are unknown, a standard source, ^{210}Bi, is used for obtaining the yield of detector. A reason of using this source is the beta energies of ^{210}Bi are quite similar with the beta energies of the reactor activation products.

In elevated radiation areas where the direct method is not effective, the indirect methods are used instead. There are two types of indirect contamination detection. The first type is called the quantity or gross contamination monitoring. This method is carried out by wiping a sanitary napkin across the suspect surface areas and then the napkin is monitored the radioactivity by a pancake style GM detector to determine if radioactive material is present. This method can show only the presence of contamination. If the actual activity per unit area is needed, the quality or analytical swipe method is used additionally. A filter paper is used to wipe across an area 100 cm^2 to pick up any loose

contamination. A gas flow proportional counter is used to detect the radioactivity on the surface of the filter paper. The contamination activity can be obtained by the following formula.

$$\text{Surface activity} = \frac{\text{Sample Count Rate} - \text{Background Count Rate}}{\text{Detector eff. (cpm dpm}^{-1}\text{)} \cdot \text{Area of smear (cm}^2\text{)} \cdot 2.22 \times 10^6 \text{ (dpm } \mu\text{Ci}^{-1}\text{)}}$$

If gamma ray emitting radioisotopes are present, a hyperpure germanium spectrometry is used for verifying the identity of radioisotopes and their activities.

3.6.3 Radioactive Airborne Monitoring

A Continuous Air Monitor (CAM) was installed at the reactor top to measure airborne radioactivity that might occur during the reactor operation. This device can also be used as an effective warning instrument to detect beginning of a radiological accident. This device is started and operates simultaneously with the reactor operation. The device consists of a plastic scintillation detector, filter papers, and an air pump system. Air above reactor top is pumped pass through a filter paper. The particulate radioactivity, if present, will be collected at the surface of the filter paper. The gross radioactivity is measured by such a detector which located in front of it. A filter paper will be replaced on daily basis, after the reactor is shut down. In normal operation, any particulate radioactivity above the background level has not been expected to exist The presence of such radioactivity can be referred to as an abnormal occurrence or radiological accident, such as the failure of fuel cladding, or the broken of activated-sample containers. The background radioactivity measured in this system is the radon concentration at the reactor

area. This level has been used as a reference level for the device operation. The CAM can be used for detection the gaseous radionuclides, but this feature is not being used.

A stack monitor is an addition instrument for measuring airborne radioactivity before it is released to environment. Air is taken from the reactor stack with the same rate of the stack's linear flow rate (an isokinetic sampling rate), passed a moving filter, then moved to a gas monitor chamber, and finally brought back to the stack. A detector located in front of the filter is used to detect any particulate radionuclides. The function of this system is similar to the CAM, but it has more advantages from a moving filter. The flow rate will not drop down due to particulate or dust accumulation at the surface of filter and a real time determination of puff release or a continuous release can be distinguished from moving filter system. The gas channel is used to detect any abnormal gaseous produced form the operation. The normal level is mainly resulted of ^{41}Ar production.

3.6.4 Primary Coolant Water and Liquid Effluent Monitoring

The primary water coolant is periodically collected and monitored the radioactivity. A hyperpure germanium spectrometer system is currently used for this task. Only the activation products are expected from the measurement. The presence of fission product energy peak might relate to an abnormal situation. The fission product peaks can be an indicator the failure of fuel cladding. For a real time measurement, a GM detector is placed in the cleanup loop for the primary water system. The output can be observed at control room and an alarm level has been set up.

The liquid discharges from the OSTR and the RC are drained to a hold up tank. The liquid samples can be collected for radioactivity measurement. The concentration of each radionuclide in the effluents is then compared with the federal regulatory limits for release to public sewage system.

3.6.5 Personnel Contamination Monitoring

As a major aspect in the OSTR radiation protection program, a personnel contamination monitoring procedures are implemented. Any individuals who exist from the radiation control areas or from the areas where radioactive materials are possessed or used are required to check themselves the contamination. The hands and shoe monitors and the portable contamination monitors are installed at the entrance areas of the reactor. The portable contamination monitors are also set up in the rooms or laboratories where the radioactive materials are used or stored. The radiation workers, students, or any individuals who work with radioactive materials are instructed by a Health Physicist to operate such a monitor. The general frisking procedures are posted at radiation workplaces and some common areas in the RC.

3.6.6 Personnel Dosimeter

According to the federal regulations, the personnel dosimeters are provided to radiation workers and any individuals who the dosimeters are required. For the reactor operators and Health Physicists, film badges or TLD badges are used as an integrated-dosimeter for beta, x-ray, and gamma dose measurement. The neutron doses are usually determined by the track-etch/albedo neutron dosimeters. The finger dosimeters (TLD)

and direct reading dosimeters (pocket ion chambers or digital pocket dosimeters) are also provided depending on the characteristic of works. The RC staff, researchers and students are provided with film badges or TLD badges and finger dosimeters. The neutron dosimeters are also provided, if their works are involved with neutron radiation. Film badges or TLD badges are also provided for the security persons. For visitors, the digital pocket dosimeters may be issued depending on the location of visiting.

3.6.7 The Radiation Monitor Calibration Program

To ensure that all of the OSTR radiation monitors are in an acceptable condition, a radiation monitor calibration program was established. The Radiation Center gamma calibration facility consists of two vertical wells which contain a 3 Ci and 100 mCi of ^{60}Co sources. Each source is connected with a chain for moving the source up and down at the specific distances. The activity of each source is re-calculated from time to time. The dose rate at each distance is measured annually by the use of a Victoreen Condenser R-meter. This meter is sent to an outside vendor for a standard re-calibration every year. The beta dose calibration is also required for ionization chambers. A depleted-uranium slab is used as a standard source. For GM detectors, the beta calibration is performed by the use of the various types of beta standard sources. The alpha survey meters are calibrated by the using of a calibrated-pulser and alpha standard sources. The neutron monitors are sent to an outside vendor for calibration. The neutron monitors are required to re-check after they were sent back. By the use of ^{60}Co and the Pu-Be sources, the monitors will be checked to ensure that the transportation does not make any impacts to

the monitors. In general, the survey meters are calibrated annually, however the re-calibration is required after the repairing of the instrument.

The ARM detectors have been calibrated annually by using of a ^{60}Co source. The checking of alarm setting is included in the program.

The CAM device is calibrated in both particulate and gaseous channels. For particulate channel, a ^{36}Cl standard source is used to determine the yield of detector. The air flowing rate is adjusted to meet a specific requirement and the alarm level is set. Only 0.06% of the Derived Air Concentration (DAC) of ^{138}Cs, which equals to 1.2×10^{-8} μCi cm^{-3} per one minute accumulation, is used as the reference level for setting. The purpose of this is to detect of an abnormal occurrence as quick as possible. [6]. For the gaseous channel, a certain amount of ^{41}Ar is injected to the channel to obtain the efficiency of detector (the count rates versus the ^{41}Ar concentration relationship). However, this channel has not been used, because of the presence of a very short half-life radionuclide, ^{16}N, which superimposes the present of ^{41}Ar and might cause the alarm during the reactor operation.

The calibration method for the CAM device is also being used for the stack monitor calibration, but the isokinetic-sampling rate calibration is required additionally. For the stack monitor, only 0.3% of the ^{138}Cs DAC (6.7×10^{-8} μCi cm^{-3} per one minute accumulation) is used for particulate alarm setting. This value is just above the background level of radon concentration in the reactor room. For gaseous channel, the alarm is set for ^{41}Ar at the level of 4.0×10^{-6} μCi cm^{-3} [6] (this value bases on the dose calculation at non-restrict areas, which is shown in 1968 SAR).

The primary water monitor is periodically calibrated. The calibration is performed by comparing the count rates of the monitor with the radioactivity of water obtained from the gamma spectrometry. The alarm is set at the twice of the normal radioactivity level.

The other monitors, such as pocket ion chambers and digital pocket dosimeters, are also calibrated. The accuracy of these instruments is determined. An instrument which has a detection error greater than 10% is not suggested to use, if the other good ones are available.

3.7 Radiation Exposure Control and Dosimetry

For more than 20 years of operation, it shows that the physical structure of the Oregon State reactor and reactor building can prevent unnecessary exposure to the vicinity areas. The reactor pool thickness and the volume of 6500 gallons of water are sufficient to protect reactor workers from radiation hazard. In addition, the concrete and steel structure of the reactor building can minimize the dose to the public as well.

The exposure control for the OSTR operation has been implemented by the personnel classification and accessing control. Any persons, such as employees, students, researchers or any individuals who have businesses with the RC, are classified in to two groups, the radiation and non-radiation groups. The radiation group is then divided into 6 subgroups base on their working characteristics. These groups are; 1) the facility operating personnel that includes reactor operators and Health Physicists, 2) the key facility research personnel, 3) the facility services maintenance personnel, 4) the laboratory class student, 5) the campus police and security personnel, and 6) the visitors.

An individual might be permitted to access some areas in the RC with or without escorts. The RC occupants are permitted unescorted-access to some specific areas depending on their works. For visitors, the escorted-access is required. The RC training program that mentioned in section 3.4 is a tool for individual classification. For entering to the reactor building during working hour, only the first subgroup is permitted to access by key issuing. The individuals in other groups might enter to the area, but the permission is required from the RC director or other authorized-persons.

The security devices have been installed at the main entrances of the reactor building. This measure is implemented not only for the reactor security purpose, but also for minimization the unnecessary exposure to the individuals.

In the reactor building, some areas were classified to be "High Radiation Areas", the areas where an individual might receive the dose equivalent in excess of 100 mrem in one hour at 30 cm from the radiation source or from any surface that the radiation penetrate. These areas, for example, are the beam port areas if the loading port and shutter are opened, and inside the block home of the experiment beam port, if the water shutter and sliding shutter are opened. The "Radiation Areas" also exist in the reactor building. These areas, by the definition in 10CFR20, are the areas where an individual might receive the dose equivalent in excess of 5 mrem in one hour at 30 cm from the radiation source or from any surface that radiation penetrate. These areas include the reactor top, bulk shield tank, thermal column, reactor water pipes, demineralize tank, and reactor bay. No radiation area is presented outside the reactor building and any non-restricted areas.

For external exposure control, no individual is allowed to enter to the high radiation area in any circumstances. The working time limitations are used to minimize exposure to the workers at some radiation areas. For example, at reactor top, a person might be allowed to work at this area for only 30 minute per week.

For internal exposure control, the capacity of 12,000 CFM input of fresh air is supplied in the reactor room and the same amount of air is exhausted from the reactor bay passing through four outlet ducts. The effluents are then released to a main reactor stack, approximately 20 meter high above ground. The air from the reactor beam ports, thermal column and rotating rack is conducted to an argon manifold to minimize the discharge of ^{41}Ar to the main exhaust fan and then released to the main stack. The large amount of air exchange rate can significantly minimize the hazard of internal exposure. However, the ^{41}Ar that is considered to be an influence radionuclide for airborne contamination is a noble gas, therefore no internal radiation hazard is involved.

The body contamination protection is a significant method to decrease the risk of both external and internal exposure. Some protective equipment, such as lab coats, rubber gloves, shoe covers, and in some case, coveralls are required for working at the reactor area. The examples of these tasks are the removal of samples from reactor pool or thermal column, the exchange of demineralize resin, the maintenance program of water system, and the exchange of air filter. The self contain air packs are not required in for normal OSTR operation. However, an amount of these devices were prepared for the emergency situation. The users training, routine check and calibration of these devices are the responsibilities of the Health Physics group.

The OSTR dose limitation system is certainly complied with the federal regulation. The dose limits are issued for different groups of people. The occupational dose limits are shown in Table 7.

Table 7. The OSTR occupational dose limits

Limit	Dose Description
5 rem	Total effective dose equivalent
50 rem	Sum of deep dose equivalent and committed dose equivalent to any individual organ , or tissue other than lens of eye
15 rem	Dose equivalent for lens of eye
50 rem	Shallow dose equivalent to the skin or any extremities

For minors, any individual under 18 years old, the dose limit is 10% of the previous occupational limits. The dose limit of 500 mrem to the embryo or fetus is provided for any declared-pregnant women. The total dose for general public is limited to 100 mrem in a year.

The various types of dosimeter that described in previous section are provided to OSTR staff and other radiation workers. However, by the regulatory criteria, these groups of people are also provided:

The person who have to entry into a high radiation area,

The person who might receive the deep dose equivalent exceeding 50 mrem in any one month,

The person who might receive the shallow dose equivalent exceeding 50 mrem in any one month, and

The minor or declared-pregnant woman who might receive the total effective dose equivalent exceeding 50 mrem in a year.

The personnel doses are assessed quarterly. The film badges or TLD are sent to an accreditation vendor to process. The results then posted and recorded regularity by a Health Physicist. The Senior Health Physicist is responsible to investigate any abnormal occurring dose, the dose that is much higher than an average dose of normal operation.

For internal dose assessment, if there is an evidence that the internal dose of individual exceeds 10% of the Annual Limit of Intake (ALI), an uptake monitoring program will be implemented. [7]

Any projects or research programs that might increase dose to radiation workers and general public must be approved by the Senior Health Physicist, before such program are initiated. These programs include new use or possess of any kind of radioactive materials, the modification or changing of facility, and changing of operation procedures. However, according to OSTR radiation safety policy and the ALARA program, a planed-special exposure that might increase the dose of the workers in excess of the regulatory limits are not permitted.

3.8 Contamination Control

The contamination can be found as area and object contamination. The TRIGA tubes that come from the rotating rack, and the irradiated sample containers (rabbits) that come from the pneumatic transfer system, are the primary sources of object contamination. The areas where these samples are retrieved are classified to be high potential contamination areas. These areas are, for example, around the reactor top, the

hoods at reactor bay, the TRIGA reactor washing machine, and the rabbit terminal fume hoods. The other sources of contamination are primary water and its treatment device such as ion exchange resin. The used-particulate filters are also considered to be the contamination items.

The OSTR policies with respect to the contamination control are the limitation of occurrence, the early detection, and the immediate decontamination. To accomplish these policies, a contamination control program was established. This program includes contamination-areas classification, contamination-areas monitoring, procedures for enter and exist these areas, personnel contamination monitoring, decontamination, and training program.

The potential contamination areas are classified. The contamination monitoring is then performed along with radiation monitoring program. The monitoring methods are already described in section 3.6.2. For radioactive materials laboratories, include the reactor, the basic protective clothing for the workers, such as lab coat and rubber gloves are required. Individuals are also required to check contamination on themselves before exist from these areas. A training program of Radiation Center provides the radiation workers to familiar with contamination monitors, frisking procedures, and basic decontamination procedures. As mentioned in previous section, at least a beta-gamma contamination monitor is available in each potential contamination room or area. If alpha radiation is involved, an alpha monitor is also provided.

Once the area contamination is found, decontamination procedures will be carried out immediately by a Health Physicist. If the contamination is a fix type, the contaminated-area might be covered with a protective material, such as plastic sheet. The

temporary closure of the area might be considered depending on the activity of contamination and the half-life of the radionuclide.

The items that wanted to bring outside from potential contamination area are suspected to be the contaminated- items. Contamination verification must be performed before such items are allowed to pass to the clean area. Both direct and indirect contamination measurements will be performed. If contamination is found, the complete decontamination and re-measurements are required. The failure of decontamination items, the fixed-contamination items, or the suspected-contamination items (due to the limit of detection) are not generally allowed to bring outside. They will be stored as the radioactive materials at the Radiation Center.

3.9 Environmental Monitoring

As mentioned in section 3.1, the major radiation sources of the OSTR operation that might impact to an environment around the reactor are the direct radiation, the gaseous and liquid effluents. The ^{41}Ar is considered to be a source of gaseous effluent from reactor stack. Although the release concentration is greater than regulatory limit, an environmental assessment shows that the annual dose to receptors around the reactor site is much less than the dose limit for general public. The liquid effluents produced from reactor operation are collected to a retention tank. The radioactivity of the effluents is measured before they are released to public sewage. The measurements show the total activity and concentration of the effluents are lower than applicable limits.

However, to ensure that the general public is safe from the OSTR operation, the environmental monitoring programs were established. The totals of 9 TLD stations are

located at the reactor fence for direct radiation measurement. The other 21 stations are located around the site at different distance from reactor to detect radiation from gaseous effluent. Each station is equipped with an OSU-TLD device and added at some specific stations with a TLD pack from an outside vender. An OSU-TLD consists of three ^{7}LiF TLD-700 chips in a plastic mount. The mount is contained in a polyethylene bottle which is placed in a PVC tube. Two $CaSO_4$ TLD chips of the outside vendor are packed and located at the designated- stations along with the OSU-TLD. Both OSU and outside vendor TLDs are exchanged quarterly.

The OSU-TLDs are processed by an OSTR Health Physicist, but for the outside vender TLDs, they are brought back to owner for processing. The direct radiation is also measured with a micro rem survey meter at each station in monthly basis.

The release of ^{41}Ar is monitored at all time, when the reactor is operated, by a stack monitor. The direct measurement of ^{41}Ar is performed periodically. The gas from stack is collected by using a vacuum flask and then is analyzed by a hyperpure germanium spectrometry.

The other environmental sampling and analyzing programs is also implemented. The water samples that include liquid discharge from the OSTR, rain water, tab water, soil samples and vegetable samples are collected quarterly. The total of 4 soil locations, 4 water locations, and 14 vegetable locations located around the reactor site are assigned.

The liquid discharges form OSTR, rain water, and tab water are analyzed by a Liquid Scintillation Counter (LSC) for ^{3}H measurement, and by a proportional counter for gross alpha and beta measurement. A gamma spectrometer is also used for identification the gamma emission isotopes. A liquid sample is divided for each

measurement. The raw water can be directly analyzed by a gamma spectrometer without pre-measuring treatment. For ^{3}H analysis, the samples must be filtered and added with the LSC solution. A part of filtered-water is then evaporated on a planchet. The planchet and filter paper are measured for gross alpha and beta activity.

An amount of soil within the plant root region is collected in a paper bag. The samples are dried in an oven for one day, then ground until find enough, then ashed in a muffle furnance. The ashes are removed to the planchets and added with an adhesive solution to fix the ash with the planchets. The prepared-samples are measured the radioactivity by a proportional counter. This method is used for plant analysis as well.

The background counting, efficiency, and lower limit of detection (LLD) of each counters are determined before samples are counted. The 95% of confidence level is applied for the analysis. If the net count of the samples is less than LLD, the report will show "Less than LLD" for each sample. However, if the net count is greater than LLD, the standard deviation of the count is required for reporting.

3.10 Radioactive Waste Management

One of primary responsibilities of OSU-RSC is to establish radioactive waste management policy. This policy is implemented for all OSU radiation laboratories including the OSTR. According to this policy, no radioactive waste is allowed to dispose by the users. The OSU-RSO is responsible for collect, treat and dispose all of the OSU radioactive wastes. The RSO is also responsible to establish the provisions and procedures about waste management and to provide the appropriate waste containers to the users.

For OSTR operation, radioactive wastes can be categorized into three types. These are solid, liquid and gaseous wastes. The solid wastes include contamination gloves, filter papers, wipe or smear devices, other protective clothing, air filters, absorbent materials, activated rabbits, and TRIGA sample tubes. All of these wastes are collected and segregated the active from the non-active items by a Health Physicist. The active wastes then are collected in 55 gallon-drum and transferred to the Radiation Center radioactive waste room. Each waste container is identified with a tag, which shows the details of wastes inside. The RSO is responsible for waste container examination, the final packaging and ship the wastes to an outside vender for disposal.

According to the OSU radiation safety regulations, drainage of liquid wastes to sink or public sewage is prohibited. This includes the first rinse of water from cleaning the contaminated items. The liquid waste form the RC laboratories are collected in the appropriate containers provided by the RSO. These wastes are segregated by radioactive half-life, to less than 30 days, 30 to 60 days, and greater than 60 days [8]. The liquid scintillation wastes are separately collected in specific containers. The liquid wastes will be absorbed with an appropriate material and then are shipped as solid wastes to the same company. The liquid scintillation wastes are shipped separately to the other company. The subsequence cleaning water and reactor effluent are drained into a retention tank. The samples of effluents are collected to determine the activity before drained to public sewage system. The limits of release follow the code of federal regulation (10CFR20). However, to accomplish the ALARA program that mentioned in previous section, the volume reduction for liquid wastes was introduced at the OSTR. The water effluent from

reactor, such as water from ion exchange resin, is recycled for using as reactor makeup water [4].

The release of gaseous and particulate radioactive materials is monitored by a stack monitoring system which described in section 3.6.3.

It can be concluded that, for the OSTR operation, an amount of liquid effluent, which the concentration is lower than an applicable limit is allowed to release to environment. The ^{41}Ar is considered to be only kind of gaseous waste. Although, its concentration is slightly higher than a limit of release, the dose estimation shows a small amount of dose to the public is increased and certainly lower than the regulatory dose limit. The solid and liquid wastes are not allowed to dispose by the users. The OSU-RSO is responsible to manage and dispose these kinds of wastes.

4. CONCLUSION

This study shows that the OSTR has an appropriate radiation safety program. The codes of federal regulations are used as a baseline for the program establishment. The Radiation Safety and Reactor Operation Committees are appointed to make the policies and look over the safety of operations. The OSTR radiation protection programs are implemented through the Health Physics group. The keys of success of these programs consist of three components. These are controlling, monitoring, and training programs. Each program is performed in a proper portion. In addition the policies, plans and procedures for the Health Physics activities are formally written and reviewed periodically.

In dose estimation part of this thesis, the annual doses to the OSTR workers seem to be high (but certainly do not exceed the limits). It should be realized that, because the activity of each worker can not be exactly estimated, the most conservative method, such as a full-occupancy time assumption, was used. However, this full-occupancy time of the workers in the reactor area is not likely to occur in real working activity. Therefore the actual doses to the workers are much lower than this estimation.

The most conservative method was also used for estimation the dose to the general public. This analysis assumes that an individual at each receptor area consumes the liquid effluents from reactor directly. This consumption, in reality, is unlikely to occur. Therefore, this assumption is used only to represent the receptor doses in the worst case scenario. From the analysis, the doses from the liquid effluents then overwhelm the other doses.

The meteorological data that used for gaseous dose estimation, from the EPA recommendation, should be averaged over five years, but the data used for this analysis was available for only two years. The error of dose estimation in each receptor might be found.

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2. John C. Ringle, T. V. Anderson, and Arthur G. Johnson, 1968. Safety Analysis Report for the Oregon State TRIGA Research Reactor.

3. National Council on Radiation Protection and Measurements, 1996. NCRP Report No. 123 I, Screening Models for Releases of Radionuclides to Atmosphere, Surface Water, and Ground.

4. Annual Report of the Oregon State University Radiation Center and TRIGA Reactor, 1986-1997.

5. T. V. Anderson, 1986. Oregon State TRIGA Reactor Training Manual, Volume 1 (Third Edition).

6. Oregon State University Radiation Center Health Physics Procedures, 1998. RCHPP 18, Maintenance and calibration Procedures for Radiation Protection Instrumentation (Including Operator Training Manual and Operating Procedures for the Radiation center Gamma Instrument calibration facility).

7. Oregon State TRIGA Reactor Training Manual, 1994. Volume 5, Radiological Protection.

8. Oregon State University Radiation Safety Manual, 1998.

APPENDICES

Appendix A: ANS 15.21 Chapter 11, SAR Standard Format and Content

In Chapter 11 of the safety analysis report (SAR), the applicant should discuss and analyze all radiological consequences related to normal operation of the reactor. In general, the design of function of structures, systems, and components, and all facility operations and materials authorized by the reactor operating license should be described in detail in other chapters of the SAR. This chapter should provide the principle discussions of the facility program to control radiation, expected exposures due to operation, maintenance, and use of the reactor. In this chapter the applicant should develop the methods for quantitative assessment of radiation dose in the restricted, controlled (if present), and unrestricted areas, should apply those methods to all applicable radiation sources related to the full range of operation, should describe the program and provisions for protecting the environment , and should provide the bases for analyzing radiological consequences from potential accidents addressed in detail in Chapter 13, "Accident Analysis."

In accordance with 10 CFR 20.1101, it is the responsibility of the licensee to develop, document, and implement a radiation protection program commensurate with the scope and extent of licensed activities and sufficient to ensure compliance with the regulations in 10 CFR Part 20. To the extent practicable, the licensee will also use procedures and engineering controls based on sound radiation protection principles to keep doses to occupational workers and members of the public as low as reasonably achievable (ALARA).

Waste materials resulting from maintenance, normal operations, or accident conditions at non-power reactors may contain radioactive isotopes. Such wastes are governed by the operating license, and it is necessary to control them like other licensed materials. At a non-power reactor, management and control responsibility for radioactive waste may be assigned to the organization responsible for reactor operations, with the radiation protection organization providing independent oversight for monitoring, assessing, and limiting risks related to radiation sources. In the alternative, licensee management could assign primary responsibility for handling and disposing of radioactive wastes to the radiation protection organization. In other case, the licensee should require procedures to ensure that radiation exposures and releases of radioactive material are adequately assessed and controlled. The SAR should discuss these issues, and provide the information necessary for NRC review. This format and content guidance for Chapter 11 integrates radioactive waste management and radiological protection in some sections, and provides separate sections for some information. The applicant should organize the functions and present the information as best suits the facility consistent with this guidance.

11.1 Radiation Protection

The following subsections provide guidance on the information the SAR should include in the description of the radiation protection program. The program is applied to the design of the reactor and its equipment, the reactor experimental facilities, reactor operations, design and use of associated laboratories, planning and procedures, and the instrumentation, techniques and practices employed to verify compliance with the

radiation dose limits and other applicable requirements specified in the regulations. Plans an the base used to develop procedures for assessing and controlling radioactive wastes and the ALARA program should be included. The responsibilities of the reactor facilities health physics organization, as well as other licensee radiation protection organizations (e.g., under a separate materials license), should be described. Facility organization charts should be included that show independence of the radiation protection function from the facility operations function.

This chapter should address all radiation sources and radioactive materials produced in the reactor and possessed or used within the reactor facility under the authorization of the reactor license. Other byproduct, special nuclear material (SNM), and source material possessed or used under the authorization of reactor license but not produced by reactor operation should be described. Program details should be given in the following subsections.

11.1.1 Radiation Sources

This section of the SAR should describe the source of radiation that are monitored and controlled by the radiation protection and radioactive waste programs. In general, the sources should be categorized as airborne, liquid, or solid as discussed in the sections that follow.

Include in this description a tabulation of all standard, check, and start-up sources categorized by isotopic composition, principle radiations (e.g., beta and gamma ray energies abundance > 10%), activity (curie content), neutron characteristics, geometry, physical and chemical form, and whether sealed or unsealed.

Also provide a tabulation of all fissile and fissionable materials, including fuel elements and assemblies, showing the status (fresh, in-core, interim storage, or spent), original enrichment, including uranium-235 (U-235) and total uranium (U) content, and current enrichment, including current U-235, total U, and total plutonium (Pu) (if appropriate).

Because of the varied nature of experimental programs, tabulation of the source strengths of irradiated experimental materials is not necessary in an SAR. However, the full range of source strengths expected to be encountered in the experimental program should be listed and discussed. Experimental protocols should provide detailed source data and be subject to the review of facility operations staff, the health physicist, and in the case of new experiments and specified deviations from previous experiments, the reactor review or audit committee. In evaluating all experiments, the applicant must also consider the requirements of 10 CFR 50.59.

Conservative estimates should be made of the quantities and types of radioactive wastes expected to result from reactor operations and use, based on previous or other similar reactor facility experience. Identification of such wastes should indicate which are associated with the operation and with the utilization of the reactor. Non-power reactor applicants have a tendency to provide overly conservative estimates; while of a conservative nature, estimates should also be realistic.

Where feasible, the SAR should include the physical and chemical form, amount, use storage conditions, and locations of all sources. In occupied or accessible areas, conservative estimates of external radiation fields should be given. An estimate of the maximum annual dose and collective doses to workers and the public should be given for

major and repetitive activities involving radiation. The applicant should discuss how the requirements of Subpart C of 10 CFR Part 20 (20.1201-20.1208) which contains regulations for occupational dose limits and Subpart D of 10 CFR Part 20 (20.1301-20.1302) which contain regulations for radiation dose limits for individual members of the public will be met. Regulations concerning compliance with dose limits for individual members of the public are given in 10 CFR 20.1302. Applicants that have licensed non-power reactors usually have historical information on radiation doses. They should discuss this information.

License conditions and, if applicable, technical specifications, concerning material possession limits, enrichment, material forms, and source strengths should be developed and analyzed in this and other Chapters, such as Chapter 4, "Reactor Description, of the SAR. These will control the use of the sources discussed above.

11.1.1.1 Airborne Radiation Sources

Airborne radioactive sources should be described in a manner suitable for designing worker protective measures and assessing and controlling workers doses. Airborne radionuclides are important because they typically are the principle source of radiation exposure to the public from a non-power reactor. A table should summarize the predicted concentrations and quantities of airborne radionuclides during the full range of normal operation (which includes maintenance activities) according to the areas that could be occupied by personnel. The applicant should estimate the release of airborne radionuclides to the environment. These releases should be used to determine consequences in the offsite environment. The applicant should discuss compliance with

the applicable regulation in 10 CFR Part 20. Note that while airborne radioactive sources from accidents are discussed in Chapter 13, the calculation methodologies developed in this chapter should be applicable to accident release analysis. Therefore, the models and assumptions used for the prediction and calculation of the dose rates and accumulative doses in both the restricted, controlled, if present, and unrestricted areas should be provided in detail. The guidance below gives an example of a description of appropriate methodology as illustrated for argon-41 (Ar-41), but is applicable to any airborne radionuclide, provided both internal and external dose delivery are accounted for.

The potential for Ar-41 production exists at most non-power facilities over the full range of normal operations, and Ar-41 could be the predominant radionuclides released to the unrestricted area. Ar-41 is produced when the argon-40 in air and air in solution in water is activated by neutrons. Ar-41 may be considered a radioactive waste produced by reactor operations. The specific source locations (e.g., primary coolant water, beam tubes, exposure rooms, and air-driven rabbit systems), predicted production rates, release mechanisms and rates, concentrations in occupied areas, possible personnel doses and dose rates, release points from the restricted area, dilution air (quantities and sources), quantities and concentrations predicted to be released, annual-average atmospheric conditions, diffusion and dispersion, predicted concentrations in unrestricted areas, and potential dose rates and annual doses, including gamma ray shine from elevated plumes should be addressed in detail.

For Ar-41, as well as other noble gases at non-power reactors it is acceptable to assume that all significant radiation risk is from external exposure to beta and gamma radiation. Other radionuclides, for example halogens or particulate, could cause internal

radiation risk by ingestion or inhalation. These doses should all be addressed, as applicable. The assumptions and methods should be conservative but physically realistic, and the validity of dose calculations should be assessed. Some non-power reactor applicants have used conservative assumptions and methods that have resulted in answers that while acceptable, are conservative by large factors.

Consideration should be given by the applicant to discussing the amount of conservatism in calculations. All assumptions should be justified, and sources of information should be adequately referenced. The calculations should address possible doses in the restricted areas and in the controlled, if applicable, and unrestricted areas. In the unrestricted areas, potential doses should be analyzed for the mainly exposed individual at the location of the nearest permanent residence, and at any locations of special interest, such as a classroom or campus dormitory. Due care should be applied if finite or non-uniform airborne distributions are intermingled with infinite cloud approximations within buildings or in realized gaussian plumes. Any such intermingling of models or assumptions should be justified. Similar discussions in this paragraph should address the production of airborne particulates, aerosols, vapors, and nitrogen-16 or other radionuclides. The discussion and calculations must show how facility design ensures that doses to the facility staff and the public will not exceed 10 CFR Part 20 limits and its ALARA requirements for effluents satisfied.

11.1.1.2 Liquid Radioactive Sources

The SAR should identify all expected liquid radioactive sources, such as reactor primary coolant, experimental solutions, reference sources, and fissile material. The SAR

should identify their origin and specify whether they result from reactor operations or the utilization program or exist for special purposes. Information should include radionuclides, concentrations, total curie strength, solubility, container characteristics, and planned release or disposition. Liquid radioactive wastes should be included. However, since the types of such wastes, their origins, and the source strengths will vary with time and the nature of the utilization program, only limiting descriptions of liquid wastes need be provided. The applicant should give an estimate of the quantity of liquid effluent released to the unrestricted environment. The applicant should discuss if credit is taken for dilution prior to release. The applicant should discuss compliance with the applicable sections of 10 CFR Part 20 such as 10 CFR 20.2003. Discuss any disposal of licensed material approved under 10 CFR 20.2002. Any storage or disposal facilities should be noted, with reference to their management, use, and the design basis of radiation protection capabilities.

11.1.1.3 Solid Radioactive Sources

The SAR should identify all expected solid radioactive sources, such as reactor fuel (spent, in-core, and fresh), calibration and test sources, experiment samples and facility components. The information should include, among other things, radionuclides, curie strengths, physical characteristics, and whether sealed or not-sealed. Solid radioactive waste should be included, but because the types and quantities will vary with time and the utilization program, only limiting descriptions of solid wastes need be provided. Provision for classifying, monitoring, storing, packaging, volume reduction prior to shipment, and disposing of solid radioactive wastes should be discussed. The

applicant should give an estimate of solid waste volume and radioactive content (in curies) expected to be removed from site on an annual basis. The applicant should discuss compliance with applicable sections of 10 CFR Parts 20, 61, and 71, and Department of Transportation regulations for transportation of radioactive material.

Discuss any capabilities or approvals received under NRC or State material licenses for onsite or offsite storage of solid radioactive wastes, including how the necessary characteristics of a restricted area are maintained. Discuss any disposal of licensed material approved under 10 CFR 20.2002.

This section should include the design bases for temporary, permanent, and installed shielding components at the facility, including utilization laboratory facilities, and radiation beams.

The following areas of the facility should be examined when developing the program for inventory and control of radiation sources:

- the exterior of the reactor biological shielding and reactor auxiliary locations (e.g., primary coolant system components and demineralizers) accessible to personnel
- the reactor experimental facilities, including beam ports, thermal columns, pneumatic or hydraulic transfer facilities, and all other irradiation facilities
- the radioactive material handling, preparation, packaging, and utilization facilities, including laboratories, hot cells, caves and storage and processing areas
- other extraneous sources, including, for example, neutron and gamma irradiation facilities, check and standard sources, neutron sources, fuel handling and storage facilities, experimental equipment storage facilities, and radioactive waste handling and storage facilities

11.1.2 Radiation Protection Program

In this section of the SAR, the applicant should describe the structure of the organization that administers the radiation protection program required by 10 CFR 20.1101, including information about staffing levels, positions of authority and responsibility, and position qualifications. The interfaces and interrelationships with other safety organizations, including the reactor facility operations staff, should be described. The information should include and discuss the charters, standards, procedures, and other documents that specify the authority and responsibilities of the organization, including authority to interdict perceived unsafe practices. The administrative plans and procedures that implement the facility policy, the overall program, and how the organization, policy, and program are designed for effective operation should be discussed. This discussion should describe the management policy governing the program and the allocation of policy-making responsibilities. Reference can be made to Chapter 12, "Conduct of Operations," if this information appears in that chapter.

The information should include the document control measures employed to ensure that the plans and procedures relative to the radiation protection program, including changes, are reviewed for adequacy, approved by authorized personnel, and distributed to and used by the applicable staff at the locations where radiation exposures could be encountered.

The radiation safety training program should be described in detail. This discussion should include the scope, and a summery of the content, of the training provided or required for all personnel, including facility-employed personnel, health

physics personnel, non-facility-employed research and service personnel, visitors, and security, fire, and other emergency personnel.

The applicant should describe in the SAR the purpose, organization, and functions of any review and audit committees with responsibilities relating to radiation safety, including the charter, responsibilities, frequency of meetings, audit responsibilities, scope of any reviews, and qualifications and requirements for committee members. A description of how each committee's work relates to the radiation safety organization and how the interface is achieved to ensure a comprehensive program should be included. If this information is discussed in Chapter 12, it can be referenced here.

A description of the program for conducting facility radiation safety audits of all functional elements of the radiation protection program to meet the requirements of 10 CFR 20.1101 (c) should be provided, identifying the scope of the audits, the bases for scheduling the audits, the qualifications of the auditors, the management level to which reports are send, and the process for following up on audit findings. The relationship of this program to any other self-assessment/internal appraisal program should be discussed. The bases for technical specifications related to facility radiation safety audits should be provided.

The system that examines that experiences of the radiation protection program and uses these experiences to improve the program and the facility design for radiation protection should be described. This system should also examine problem and incidents and develop lessons-learned, root causes, and effective corrective actions.

For activities not described in the SAR, or governed by procedures, a work control process such as the use of radiation work permits should be used. The applicant should discuss the control program used at the facility.

The applicant should describe the radiation safety program recordkeeping process, including record retention periods, accessibility, review, and archiving. Review of radiation safety records for accuracy and validity should be discussed. The use of records for developing trend analyses, informing management, planning radiation-related actions, and reporting to regulatory and other duty authorized entities should be discussed.

11.1.3 ALARA Program

In this section of the SAR the applicant should describe the ALARA program for the facility required by 10 CFR 20.1101. The description should include the basis for the program, and the management level and authority by which the facility ALARA policy is established. The applicant should discuss how this program is implemented to maintain doses of all personnel at the facility and releases of effluents to the unrestricted area ALARA. Provide and discuss the criteria used to determine how low the projected doses should be to permit task implementation (i.e., ALARA goals). The discussion should include methods to ensure that the radiation protection staff and their considerations of the facility ALARA program are specifically involved during review and approvals of design, in construction of facilities, in planning and implementing reactor utilization (experiment design and planning) and operation, in maintenance activities, and in the management and disposition of radioactive wastes.

11.1.4 Radiation Monitoring and Surveying

The program employed to routinely monitor workplaces and other locations accessible to people for identification and control of sources of radiation exposure should be described in this section, including the measures designed to ensure that monitoring of air, liquids, and solids is performed in all applicable areas. Also discuss the bases of the methods and procedures used for detecting and assessing contaminated areas, materials, and components, and describe the records kept to document the applicability, quality, and accuracy of monitoring methods, techniques, and procedures.

The SAR should provide summary descriptions of all radiation monitoring equipment employed throughout the facility, including locations and functions of each device and system. This description should also include sampling equipment for liquid and gaseous process and effluent streams. This discussion may be combined with (and appropriately cross-referenced to) the discussions in Chapter 7, "Instrumentation and Control Systems." The applicant should discuss the interface between the radiation monitoring system and engineered safety discussed in Chapter 6, "Engineered Safety Features," if any exist. Types of equipment should include systems of the following types (as appropriate to the facility):

- continuous air monitors (CAMs) including fixed and moving filter, and gaseous monitors
- portable survey instruments (radiation fields and contamination)
- remote area monitors (RAMs)
- samplers
- effluent monitors

- environmental monitors (provide details in Section 11.1.7)
- personal dosimeters
- portable monitors
- rad-waste storage monitors
- criticality monitors

The calibration of the radiation protection instrumentation, including the procedures and standards governing calibration, control of the calibration process, use of national standards, and verification should be described. This section should also describe the calibration equipment and discuss sensitivities to environmental and other conditions with respect to the calibration requirements. The program to ensure that routine periodic calibration is performed in a timely manner and the bases of calibration schedules should be described.

The applicant should describe in the SAR how routine monitoring provided at the facility is planned to ensure that radiation exposures to the public and workers or material releases can be detected, and discuss how the approach used for routine monitoring provides reasonable assurance that all radiation at and released from the site will be appropriately monitored.

Technical specifications and their bases related to the radiation monitoring equipment and procedures, as discussed in Chapter 14, "Technical Specifications," should be given and justified in this chapter.

11.1.5 Radiation Exposure Control and Dosimetry

Radiation exposure is controlled by controlling radioactive materials and effluent radioactive material releases. In this section of the SAR the applicant should describe the design bases for the equipment and procedures utilized for controlling exposure to personnel and releases of radioactive materials from the facility, and discuss how the facility structures, systems, and components are designed to provide assurance that there will be no uncontrolled effluent radioactive releases to the environment or to work areas. Some systems, such as containment/confinement and ventilation, may have been discussed in other chapters of the SAR; reference to those discussions in this chapter of the SAR is appropriate. The applicant should also discuss how the bases of radiation shielding, ventilation, and remote handling and decontamination equipment are designed to ensure that dose to the workers are maintained ALARA and within the applicable regulatory limits.

How the design of required entry control devices (i.e., alarms, signals, or locked entry ways) alerts workers to, or prevents entry into, high radiation and very high radiation areas should be described. The regulations in 10 CFR Part 20, Subpart G. " Control of Exposure From External Sources in Restricted Areas," contains requirements for control to high and very high radiation areas. It should be noted that 10 CFR 20.1601 (c) allows a licensee to apply to the Commission for approval of alternative methods for controlling access to high radiation areas if the licensee finds that the stated methods of control in the regulations would interfere with utilization programs. The application should contain a description of the proposed method along with a discussion of how the entrance or access point to high radiation areas will be controlled.

Personnel protective equipment and materials (e.g., anti-contamination clothing and respiratory equipment) employed in the facility should be discussed. Describe the facility conditions for which this personnel protective equipment should be employed. Also discuss whether respirators will be used at the facility. The use of respiratory protection equipment requires implementing and maintaining a respiratory protection program in accordance with the requirements of 10 CFR 20, Subpart H. If there will be a respiratory protection program, that program should be described as it relates to the minimum program requirements of 10CFR 20.1703.

The base and values for the expected annual radiation exposure for all locations of the facility should be discussed, including the exposure estimate for licensee-employed personnel, non-licensee-employed research and service personnel, and visitors. This discussion should include the exposure limits and controls for groups such as embryos, fetuses, declared pregnant women, minors, and students. The plans and procedures for exposure control and dosimetry during the full range of normal facility operation, potential accident conditions, rescue and recovery, and planned special personnel exposures (non-emergency) should also be discussed. Describe the dosimetry used for assessing external radiation exposures (e.g., whole body, extremities), including the frequency of dosimeter reading, administrative dose action levels, and the suitability of the dosimetry chosen with respect to the radiation sources anticipated and observed. Describe the same factors for how internal exposures and doses are assessed, evaluated, and controlled.

The applicant should describe the type of records retained to document the conditions under which individuals were exposed to radiation. The applicant should discuss the historical and current exposures to personnel and the associated trends.

11.1.6 Contamination Control

The applicant should discuss the plans and bases of procedures for radioactive contamination identification control, including methods established to assess the effectiveness of the contamination control program. The SAR information should include description and discussions of the plans and bases of procedures on the following topics, showing their relationship to regulatory requirements and ALARA concepts:

- a program for routine monitoring to detect and identify both fixed and loose contamination
- program to control access to contaminated areas, avoid further spreading of contamination, and remedy contaminated areas.
- personal monitoring and assessment of internal and external doses to personnel occupying or entering contaminated areas, and methods for appropriate surveying and "frisking" upon exit
- use of anti-contamination techniques to protect workers, and control and disposition of possibly contaminated clothing and materials
- procedures for monitoring and handling equipment and components intended for removal from contaminated areas that have not been decontaminated
- criteria for classifying contaminated material, equipment, and working areas, and managing, controlling, storing, and disposing of identified contamination

- training program for staff and visitors on the risks of contamination and on techniques for avoiding, limiting, and controlling contamination
- recordkeeping for contamination events, both for personnel and for locations, including records to be available for facility maintenance and for decommissioning
- the bases of technical specifications, if needed, applicable to contamination control: for example, limits on storage and handling of radioactive sources, especially unsealed ones; limitations on encapsulation of irradiated materials; and use of fume hoods and hot-waste drains

11.1.7 Environmental Monitoring

The applicant should describe the environmental monitoring program, including information relating to the following:

- Verification of compliance with commitments made in Environmental Impact Statements, Environmental Assessments, environmental reports, or other official documents, if applicable. Discussion of standards, if any, used in the environmental monitoring program
- For established programs, evaluation of the effectiveness of the program
- Identification of potential facility impacts on the environment and the evaluation of the need for remedial action or mitigation measures.
- Establishment of baselines for environmental quality, including data comparing pre-construction or pre-operational with operational environmental monitoring results.

The applicant should describe in the SAR the written plans and the bases of procedures for implementing the environmental monitoring program, and discuss the

document control measures employed to ensure that the plans and procedures, including changes, are reviewed for adequacy and approved by authorized personnel, and are distributed to and used at the appropriate locations throughout the facility.

`The environmental surveillance program and its bases should be described. Air, water, and land environments should be specifically discussed. These discussions should include information on at least the following topics:

- probable facility-related contaminants and pathways to people
- selection of sampling materials and locations
- sample collection methods and frequency
- sample analyses (analytical techniques) and sensitivities (detection limits)
- records of results and trends

11.2 Radioactive Waste management

Each facility that is licensed to operate or utilize a non-power reactor should establish a program and procedures that are designed to ensure that radioactive waste materials are identified, assessed, controlled, and disposed of in conformance with all applicable regulations and in a manner to protect the health and safety of the public and the environment. The magnitude and mature of the effort required should depend upon the size and complexity of both the reactor facility and its utilization programs. Therefore, the nature and details of the radioactive waste management program should also be commensurate with those factors. As noted previously, management of radioactive wastes could be an auxiliary function assigned to existing personnel, such as radiation protection or operations. Foregoing sections of this chapter have addressed the

program and procedures for controlling and assessing radiation exposures and doses at the facility due to all radiation and radioactive sources. This section should address the program and procedures for future managing sources classified as radioactive waste.

11.2.1 Radioactive Waste Management Program

In this section of the SAR, the applicant should discuss the radioactive waste management program philosophy and objectives. The applicant should describe the organizational structure within which the licensee will administer the reactor-related radioactive waste management program, including the organization and staffing levels, authorities and responsibilities, and position qualifications. The interfaces and interrelationships of facility organizations such as radiation protection and operations staff and the standards, charters, procedures, or other documents that specify the authority, duties, and responsibilities of the personnel in the radioactive waste management organization should be discussed. The policy governing the program, the allocation of policy-making responsibilities, and the administrative plans and procedures that implement the facility policy should be described. The overall program and how the organization, policy, and program lead to effective management of radioactive waste should be evaluated and described.

The SAR should describe the purpose, organization, and functions of any committees assigned responsibility for radioactive waste management oversight. The description should include each committee's charter, responsibilities, frequency of meetings, audit and review responsibilities, the scope of any audits or reviews, and qualifications and requirements for committee members. A description of how each

committee's work relates to the waste management organization and how the interface is achieved should be provided. If this information has already been described, reference that discussion.

The SAR should describe the waste management training program. This discussion should include the scope of facility waste management training, as well as specific training requirements for personnel associated with operation and use of the facility.

The SAR should describe the document control measures that ensure that the plans and procedures involving radioactive waste, including changes, are reviewed for applicability, approved by authorized personnel, and distributed to and used at the locations where waste management activities are controlled.

The SAR should describe the scope of waste management reviews and audits. This discussion should include the authority of waste management review and audit teams, the objectives and purposes for reviews and audits, and the bases for scheduling these reviews and audits.

The SAR should describe the radioactive waste management record-keeping process, including periods, accessibility, review, and archiving, and discuss any special review of waste management records for accuracy and validity. Records of radioactive wastes stored for the life of the facility or buried onsite should be discussed, as well as records for trend analysis.

The bases for any technical specifications related to the radioactive waste management program should be described.

11.2.2 Radioactive Waste Controls

The applicant should discuss the definition of radioactive waste, the point in any process that a radioactive component or material becomes waste, and the criteria for defining such waste. In the SAR, the applicant should describe the waste management program procedures that ensure that the radioactive wastes are identified and characterized appropriately, as noted above, and the bases of the procedures that ensure that radioactive wastes are adequately segregated from non-radioactive wastes. The plans and procedures for managing all forms of radioactive wastes generated during operations, research, and utilization of the reactor should be described. Radioactive wastes are radiation sources that should be described, along with other such sources, in Section 11.1 of the SAR.

The applicant should describe the plans and bases for procedures for managing gaseous and other airborne radioactive waste generated during operations, research, and utilization of the reactor, and radioactive waste off-gas collection systems designed to be utilized at the facility. A functional description and the location of each off-gas collection system should be provided. At many non-power reactors, the system for removal of gaseous radioactive waste is integral to the ventilation system for the facility and may have engineered safety functions. If these systems have been described in other chapters of the SAR, reference may be made to those discussions. For all off-gas and ventilation systems, describe the wastes produced by operation of the systems. Such items as filters and scrubbers, which collect and concentrate wastes, should be discussed to indicate the disposition of the radioactive material upon regeneration or replacement. If the radioactive materials enter other waste treatment systems, indicate how such transfers are

made and any possible chemical or radiological effects of the transfer. The operation of any gas-cleaning equipment and its designed performance should be discussed and included in this section. The bases of any applicable technical specifications that control these functions should be provided. Also describe all secondary radioactive residues that are generated during process treatment, their chemical and physical composition, and the modes for handling, controlling, and storing them.

The SAR should describe how liquid radioactive wastes are generated and where they enter the waste control and treatment systems. Such items as laboratory wastes, liquid spills, and cleanup solutions including detergent wastes should be discussed. Information about the projected inventory levels, interim and long-term storage, and processing of those streams to achieve volume reduction so solidification should be included. This discussion should include information about coolant clean-up systems and resin regeneration solutions and wastes, if applicable.

The objectives of the processes designed to treat radioactive or mixed liquid wastes should be described. Any backup and special safety features designed to ensure that the radioactive waste is contained during treatment should be described. A description of the designed equipment and systems, with appropriate engineering drawings to show the location of the equipment, flow paths, piping, valves, instrumentation, and other physical features should be provided, including all features, systems, or special handling techniques that prevent uncontrolled releases or personnel exposures.

The SAR should describe the plans and procedures for managing solid radioactive wastes generated during operations, research, and utilization of the reactor. This

description should include how solid radioactive materials are generated and where they enter the waste control and treatment systems. For solid radioactive wastes retained or stored onsite for the life of the facility, discuss the control methods used. Integrity and corrosion characteristics and the monitoring of the containment should be discussed, as well as the plan for disposing of these radioactive wastes when the facility is permanently decommissioned.

The SAR should describe the systems and equipment selected for identifying, segregating, and safety managing the solid, liquid, and gaseous radioactive waste that is generated, and should include appropriate engineering drawings showing the location of the equipment and associated features used for volume reduction, containment, and/or packaging, storage, and disposal. The SAR should also discuss the bases of procedures associated with operating treatment equipment, including performance tests, process limits, and the means for monitoring and controlling to meet these limits. The bases of applicable technical specifications that control these procedures and functions should be discussed. The methods and agents planned for all activities involving routine disposal or release to the environment of radioactive wastes generated in the facility should be described, as should methods used for packaging and shipping solid and liquid radioactive wastes to other facilities or agents for processing, storage, or other disposition.

The SAR should describe the radioactive waste minimizing program for the facility with respect to the following topics: (1) the specific numerical goals for reducing the volume or radioactivity of each waste steam; (2) the periodic assessment of reactor operations and experimental or utilization activities to identify opportunities to reduce or

eliminate the generation of wastes; (3) the continuing efforts to identify and, where cost-effective, implement waste reduction technologies; and (4) any periodic independent reviews performed to evaluate the effectiveness of programs to minimize radioactive waste.

11.2.3 Release of Radioactive Waste

The SAR should identify all radioactive waste materials for which controlled release to the environment or transfer to other parties for disposal is planned. This discussion should include the projected concentrations, forms, chemical compositions, and annual quantities of radioactive waste released under normal operating conditions.

All points where radioactive waste effluents are designed to be released from the facility to the environment should be identified, using a site map to locate the effluent release points and effluent monitoring equipment. Discussions and detailed analyses of potential radiological impact of radioactive waste effluents and the bases for continuous or intermittent monitoring should be provided in the earlier sections of Chapter 11. For liquid releases to the sanitary sewerage, the licensee shall ensure that the requirements of 10 CFR 20.2003 are met. The SAR should describe the systems and procedures designed to ensure that doses resulting from releases of radioactive effluents do not exceed applicable regulatory limits and ALARA goals.

Appendix B: An example of wind data obtained from NCDC

OBSERVATIONS AT 3–HOURLY INTERVALS

EUGENE, OR

MAY 1997 EUG WBAN # 24221

HOUR (LST)	≤12K FEET: SKY COVER	≤12K FEET: CEILING 100'S OF FT	SATELLITE: OBSERVATION TIME (LST)	SATELLITE: EFF CLD AMT Oktas	VISIBILITY (MILES)	WEATHER	TEMPERATURE °F: DRY BULB	TEMPERATURE °F: DEW POINT	TEMPERATURE °F: WET BULB	RELATIVE HUMIDITY (PCT)	WIND: SPEED (MPH)	WIND: DIRECTION TENS OF DEG	PRESSURE (INCHES,HG): STATION	PRESSURE (INCHES,HG): SEA LEVEL
					SUNRISE: 0504	MAY 01			SUNSET: 1915					
01	BKN	055			10.00		41	39	40	93	8	21	29.84	30.24
04	BKN	070			10.00		38	37	38	97	8	14	29.84	30.24
07	OVC	060			10.00		44	41	43	89	5	17	29.85	30.26
10	OVC	039			10.00		50	42	46	74	3	VR	29.84	30.24
13	BKN	060			10.00		56	40	48	55	7	17	29.79	30.19
16	SCT	NC			10.00		58	42	50	56	7	01	29.73	30.13
19	CLR	NC			10.00		49	45	47	86	5	34	29.69	30.09
22	CLR	NC			10.00		40	39	40	97	6	36	29.65	30.06
					SUNRISE: 0503	MAY 02			SUNSET: 1917					
01	CLR	NC			10.00		43	40	42	89	0	00	29.59	29.99
04	CLR	NC			10.00		40	38	39	93	7	10	29.55	29.95
07	FEW	NC			10.00		44	41	43	89	8	11	29.53	29.93
10	CLR	NC			10.00		52	43	48	72	5	VR	29.49	29.90
13	FEW	NC			10.00		64	48	55	56	5	VR	29.42	29.82
16	BKN	075			10.00		67	41	54	39	14	18	29.36	29.75
19	OVC	050			10.00		53	52	52	96	6	17	29.46	29.86
22	OVC	049			10.00		51	49	50	92	3	15	29.54	29.94
					SUNRISE: 0501	MAY 03			SUNSET: 1918					
01	SCT	NC			10.00		49	49	49	100	10	12	29.57	29.96
04	OVC	047			10.00		49	46	47	90	7	15	29.61	30.01
07	CLR	NC			10.00		50	47	48	89	8	13	29.65	30.05
10	OVC	120			10.00		56	47	51	72	5	VR	29.68	30.07
13	BKN	049			10.00		59	43	51	56	6	24	29.66	30.06
16	BKN	050			10.00	-RA	57	46	51	67	7	23	29.66	30.06
19	OVC	060			10.00		53	51	52	93	5	12	29.67	30.08
22	OVC	060			10.00		51	50	51	96	7	11	29.72	30.12
					SUNRISE: 0460	MAY 04			SUNSET: 1919					
01	BKN	065			10.00		50	47	48	89	10	13	29.75	30.15
04	BKN	120			10.00		47	44	46	90	6	13	29.78	30.18
07	SCT	NC			10.00		51	47	49	86	7	14	29.81	30.21
10	CLR	NC			10.00		61	47	54	60	13	18	29.81	30.20
13	FEW	NC			10.00		65	45	54	49	3	VR	29.78	30.18
16	CLR	NC			10.00		68	46	56	45	6	29	29.74	30.14
19	CLR	NC			10.00		61	53	57	75	7	25	29.72	30.12
22	CLR	NC			10.00		51	48	50	89	0	00	29.76	30.16
					SUNRISE: 0459	MAY 05			SUNSET: 1920					
01	CLR	NC			10.00		49	46	47	90	5	14	29.78	30.18
04	CLR	NC			10.00		50	47	48	89	6	15	29.76	30.16
07	CLR	NC			10.00		52	48	50	86	3	17	29.77	30.17
10	SCT	NC			10.00		63	48	55	58	8	22	29.76	30.15
13	SCT	NC			10.00		68	48	57	49	9	22	29.73	30.13
16	CLR	NC			10.00		68	47	57	47	14	26	29.71	30.11
19	BKN	080			10.00		60	48	54	65	13	25	29.73	30.12
22	FEW	NC			9.00		55	50	52	83	9	19	29.77	30.16
					SUNRISE: 0457	MAY 06			SUNSET: 1921					
01	OVC	029			10.00		55	51	53	87	12	21	29.77	30.16
04	OVC	060			10.00		51	50	51	96	7	14	29.79	30.19
07	BKN	047			10.00		52	50	51	93	6	17	29.84	30.24
10	BKN	025			10.00		58	47	52	67	9	18	29.86	30.27
13	CLR	NC			10.00		64	45	54	50	9	20	29.86	30.27
16	SCT	NC			10.00		65	39	52	39	9	28	29.85	30.25
19	CLR	NC			10.00		53	46	50	77	9	01	29.85	30.26
22	CLR	NC			10.00		46	43	45	89	8	35	29.88	30.29

HOUR (LST)	≤12K FEET: SKY COVER	≤12K FEET: CEILING 100'S OF FT	SATELLITE: OBSERVATION TIME (LST)	SATELLITE: EFF CLD AMT Oktas	VISIBILITY (MILES)	WEATHER	TEMPERATURE °F: DRY BULB	TEMPERATURE °F: DEW POINT	TEMPERATURE °F: WET BULB	RELATIVE HUMIDITY (PCT)	WIND: SPEED (MPH)	WIND: DIRECTION TENS OF DEG	PRESSURE (INCHES,HG): STATION	PRESSURE (INCHES,HG): SEA LEVEL
					SUNRISE: 0456	MAY 07			SUNSET: 1923					
01	CLR	NC			9.00		42	41	42	96	8	34	29.87	30.28
04	CLR	NC			10.00		40	39	40	97	5	02	29.87	30.28
07	BKN	010			10.00		46	44	45	93	9	36	29.85	30.26
10	CLR	NC			10.00		55	48	51	77	12	36	29.79	30.19
13	CLR	NC			10.00		64	50	56	61	13	35	29.71	30.11
16	CLR	NC			10.00		69	54	60	59	15	32	29.62	30.02
19	CLR	NC			10.00		60	54	57	80	6	31	29.58	29.98
22	CLR	NC			10.00		54	51	52	90	6	19	29.61	30.01
					SUNRISE: 0455	MAY 08			SUNSET: 1924					
01	CLR	NC			10.00		47	47	47	100	6	19	29.65	30.05
04	CLR	NC			10.00		45	44	45	97	6	20	29.68	30.08
07	CLR	NC			10.00		52	49	50	89	3	19	29.72	30.12
10	CLR	NC			10.00		65	52	58	63	0	00	29.72	30.12
13	CLR	NC			10.00		72	55	62	55	5	01	29.69	30.09
16	CLR	NC			10.00		73	53	61	50	9	01	29.65	30.05
19	CLR	NC			10.00		66	54	59	65	5	30	29.65	30.04
22	CLR	NC			10.00		52	50	51	93	6	15	29.68	30.08
					SUNRISE: 0453	MAY 09			SUNSET: 1925					
01	CLR	NC			10.00		48	47	47	96	6	16	29.72	30.11
04	CLR	NC			10.00		47	44	46	90	6	17	29.75	30.14
07	CLR	NC			10.00		52	48	50	86	6	14	29.79	30.19
10	CLR	NC			10.00		61	53	57	75	6	VR	29.80	30.20
13	CLR	NC			10.00		70	54	61	57	12	01	29.78	30.18
16	CLR	NC			10.00		73	55	62	53	12	36	29.74	30.14
19	CLR	NC			10.00		65	58	61	78	6	36	29.73	30.13
22	CLR	NC			10.00		55	51	53	87	8	01	29.77	30.17
					SUNRISE: 0452	MAY 10			SUNSET: 1926					
01	CLR	NC			10.00		49	47	48	93	7	36	29.77	30.17
04	CLR	NC			10.00		49	45	47	86	9	34	29.75	30.15
07	CLR	NC			10.00		55	49	52	80	8	36	29.77	30.16
10	CLR	NC			10.00		64	55	59	73	8	36	29.73	30.13
13	CLR	NC			10.00		71	59	64	66	9	36	29.66	30.06
16	CLR	NC			10.00		76	62	67	62	10	36	29.58	29.98
19	CLR	NC			10.00		72	64	67	76	3	01	29.54	29.93
22	CLR	NC			10.00		59	57	58	93	0	00	29.55	29.95
					SUNRISE: 0451	MAY 11			SUNSET: 1927					
01	CLR	NC			10.00		56	54	55	93	0	00	29.54	29.94
04	CLR	NC			10.00		55	50	52	83	3	30	29.54	29.93
07	CLR	NC			10.00		60	53	56	78	0	00	29.57	29.97
10	CLR	NC			10.00		72	57	63	60	8	01	29.57	29.97
13	CLR	NC			10.00		80	63	69	56	6	01	29.54	29.94
16	CLR	NC			10.00		82	67	72	60	7	35	29.48	29.88
19	CLR	NC			10.00		74	63	67	69	7	20	29.47	29.87
22	CLR	NC			10.00		64	55	59	73	12	20	29.52	29.91
					SUNRISE: 0450	MAY 12			SUNSET: 1928					
01	CLR	NC			10.00		59	54	56	83	8	20	29.54	29.93
04	CLR	NC			10.00		53	51	52	93	7	16	29.56	29.96
07	CLR	NC			10.00		60	54	57	80	9	20	29.57	29.96
10	CLR	NC			10.00		72	58	63	61	7	21	29.55	29.95
13	CLR	NC			10.00		78	56	65	47	9	23	29.54	29.93
16	CLR	NC			10.00		78	53	63	42	10	22	29.54	29.93
19	CLR	NC			10.00		67	48	57	51	9	24	29.57	29.96
22	CLR	NC			10.00		57	47	52	69	7	22	29.61	30.00

Appendix C: An example of wind data spreadsheet for obtaining wind rose format

	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	
	348.75-11.24	11.25-33.74	33.75-56.24	56.25-78.74	78.75-101.24	101.25-123.74	123.75-146.24	146.25-168.74	168.75-191.24	191.25-213.74	213.75-236.74	236.75-258.74	258.75-281.74	281.75-303.74	303.75-326.24	326.25-348.74	Calm
Nov-96	7	2	3	5	5	6	7	3	7	3	5	3	7	14	12	8	0
	3	10		5	5	9	7	5	6	5	3	5	14	3	5	7	0
	7				5	3	3	6	7	8	9	5	3	6	3	9	0
	5					7	7	6	5	6	10	3		7		7	0
	5					7	9	6	5	3	3	8				6	0
	3					5	12	5	5	3	13					12	0
	6					14	9	6	7	5	7					5	0
	3					12	7	15	7	7	5						0
	3						7	3	15	5							0
	3						6	5	9	8							0
	9						6	7	6	6							0
	6						8	6	5	5							0
	8						5	12	3	6							0
	7						7	9	7	6							0
	6						12	14	7	7							0
	5						5	17	5	14							0
	3						8	10	3								0
	5						5	10	6								0
	6						7	7	12								0
	8						7	14	8								0
	12						5	5	8								0
	14						5	17	9								0
	7						5	6	7								0
							8	5	6								0
							6	3	3								0
							6	5	9								0
							8	5	6								0
							9	10	10								0
							10	15	14								0
							10	6	5								0
							10	5	6								0
								9	10								0
								8	15								0
								15	14								0
									3								0
									6								0
									5								0
									16								0
									17								0
									6								0
									15								0
									12								0
									13								0
									5								0
N	23	2	1	2	3	8	32	36	45	17	8	5	3	4	3	7	36
AVG	6.130434783	6	3	5	5	7.875	7.375	8.388888889	8.022222222	6.117647059	6.875	4.8	8	7.5	6.666666667	7.714285714	0

Note: Calm wind (0) that shown in this table is the speed of wind which is lower than 3 mph

Appendix D: The averages of wind speed in each direction

	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	
	348.75-11.24	11.25-33.74	33.75-56.24	56.25-78.74	78.75-101.24	101.25-123.74	123.75-146.24	146.25-168.74	168.75-191.24	191.25-213.74	213.75-236.74	236.75-258.74	258.75-281.74	281.75-303.74	303.75-326.24	326.25-348.74	Calm
Apr-97	8.08	6.5		5	4.25	4.444	7.375	7.87	8.135	6.778	10.722	9.765	4.667	6.8	6.429	8.667	0
Apr-98	7.25	4.667	3	5	7	6.667	6.462	7.238	6.48	7.231	8.923	7.455	6.9	6.167	7.125	7.111	0
Aug-96	9.673	6.5	4	7	3	5.333	6.5	7.75	7.542	9.222	9.3	10.143	6.857	7.778	8.385	9.077	0
Aug-97	8.935	6.4		4	4	6.286	4.625	6.706	6.9	9.19	8.154	8.111	6.556	7.2	10.5	9.714	0
Aug-98	9.775	5.833	4.75	5	5	4.75	3	5.8	7.222	9.625	7.5	6.333	5.429	8.2	9.333	10.273	0
Dec-96	8.333	3	5		5.5	9.769	8.73	12.5	10.429	8.4	6.833	4.333	8.5		5.25	7.875	0
Dec-97	7.267	6.333	4	5	8	7.875	6.914	10.36	5.806	5.6	8.25	4	3	4.333	5.5	6.25	0
Feb-97	7.135	3		4		5.333	7	7.583	7.24	8.111	9.133	5.6	5.7	3	6.444	7.385	0
Feb-98	8.619	5.5	7.5	5	4.75	8.182	8.615	12.151	10.037	9.333	9	3	8.667	4	8.857	12	0
Jan-97	9	6.333		4	5	7.636	8.345	9.421	7.907	6.636	6.143	4	8	5.2	6.857	8.889	0
Jan-98	7	5	4	5.5	4.75	8.09	8.8	10.8	9.912	7.091	6.667		9	4.667	5	5.667	0
Jul-96	9.905	7.333	5	6	5	5.667	5.5	6.889	7.545	9.067	10.357	9.444	8	6.333	9.5	10.875	0
Jul-97	9.937	5.333	4	4	4	6	7.5	6.75	6.385	9.615	7.875	8.25	8.4	6.5	9.818	9.581	0
Jul-98	9.121	4.4		4.667	3		5	7.091	7.048	9.686	9.125	8.5	5.625	7.5	7.714	9.571	0
Jun-97	7.976	6.167	7		4	5.833	5.778	6.84	7.1	8.588	9	7.909	7.667	8.111	8.125	8.667	0
Jun-98	7.818	8.333	3	6	4.25		6.75	6	7.417	8.286	7.72	8.167	9.875	6	6.857	9.222	0
Mar-97	6.471	6.333	6		5.167	7.167	7.833	9.568	10.275	7.5	7.2	7.857	4.75	4	7.5	8.363	0
Mar-97	7.756	7	9	8	5	6	5.947	5.75	6.708	7.875	8.267	9.667	8.375	6.2	7.2	7.182	0
May-98	7.125	5	5.5	3	5	6.6	6.455	8.177	7.25	6.688	7.4	5.8	8.429	8.5	6.5	6.333	0
Nov-96	6.13	6	3	5	5	7.875	7.375	8.389	8.022	6.118	6.875	4.8	8	7.5	6.667	7.714	0
Nov-97	4.958	6	3	3	6.667	6.889	8.714	9.289	7.5	6.083		4.8	6.333	3.4	4.667	5.2	0
Oct-96	8.4	6	5		6.333	6.444	7.895	9.517	7.947	7.818	10	5.75	6.5	6.5	6.25	8	0
Oct-97	8.034	5	6		3	7.143	8.727	9.923	10.021	7.375	3.667	6	6	6.667	10.143	8.333	0
Sep-96	9.915	5.857	5	3.5	5.333	4.5	5.75	5.077	6.68	8.333	8.75	7.714	4.25	7.857	8.571	7.8	0
Sep-97	6.875	5.2	3		3	6.5	7.174	10.44	9.094	7.1	6.733	6	6.3	6.5	8.667	8.385	0
AVG	8.05952	5.72088	4.8375	4.877210526	4.833333333	6.564478261	6.91056	8.31516	7.86408	7.89396	8.066416667	6.80825	6.8712	6.204708333	7.51436	8.32536	0

Note: Calm wind (0) that shown in this table is the speed of wind which is lower than 3 mph

Appendix E: The frequencies of wind in each direction

	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW		
	348.75-11.24	11.25-33.74	33.75-56.24	56.25-78.74	78.75-101.24	101.25-123.74	123.75-146.24	146.25-168.74	168.75-191.24	191.25-213.74	213.75-236.74	236.75-258.74	258.75-281.74	281.75-303.74	303.75-326.24	326.25-348.74	Calm	sumN
Apr-97	25	2	0	2	4	9	16	23	37	18	18	17	3	5	7	6	29	221
Apr-98	24	6	1	1	1	3	13	21	25	13	13	11	10	6	8	9	56	221
Aug-96	49	4	2	3	1	6	4	8	24	9	20	7	7	9	13	26	43	235
Aug-97	31	5	0	3	2	7	8	17	30	21	13	18	9	5	6	14	40	229
Aug-98	71	6	4	1	2	4	1	5	9	8	6	3	7	5	18	22	55	227
Dec-96	27	1	1	0	2	13	37	62	42	10	6	3	2	0	4	8	20	238
Dec-97	30	6	2	2	3	21	35	25	31	10	4	2	2	3	6	12	51	245
Feb-97	37	2	0	2	0	6	18	24	25	18	15	5	10	1	9	13	37	222
Feb-98	21	6	2	1	4	11	39	53	27	9	9	1	3	3	7	3	22	221
Jan-97	29	3	0	2	1	11	29	38	43	11	7	2	2	5	7	9	47	246
Jan-98	11	2	2	2	4	22	45	55	34	11	3	0	2	3	5	6	35	242
Jul-96	63	3	1	1	2	3	4	9	22	15	14	9	1	3	16	32	38	236
Jul-97	63	6	2	3	2	2	4	12	13	13	8	4	10	4	11	31	46	234
Jul-98	58	5	0	3	1	0	4	11	21	32	16	8	8	2	7	7	35	218
Jun-97	42	6	1	0	2	6	9	25	20	17	15	11	12	9	8	9	35	227
Jun-98	44	12	1	1	4	0	4	16	24	28	25	6	8	8	7	9	27	224
Mar-97	17	3	4	0	6	12	24	44	51	16	10	7	4	2	6	11	19	236
Mar-97	41	5	1	1	3	12	19	24	24	16	15	9	8	5	5	11	35	234
May-98	24	2	2	1	5	5	11	34	32	16	20	5	7	2	6	9	48	229
Nov-96	23	2	1	2	3	8	32	36	45	17	8	5	3	4	3	7	36	235
Nov-97	24	6	1	1	3	9	28	38	28	12	0	5	3	5	6	5	62	236
Oct-96	30	2	2	0	3	9	19	29	38	22	12	8	6	2	8	13	36	239
Oct-97	29	2	1	0	2	7	22	39	48	16	3	5	6	6	7	6	46	245
Sep-96	47	7	2	4	3	4	16	13	25	12	12	7	4	7	14	10	42	229
Sep-97	24	5	2	0	1	6	23	25	32	10	15	4	10	4	6	13	51	231
SumSec	884	109	35	36	64	196	464	686	750	380	287	162	147	108	200	301	991	5800
Freq	0.152413793	0.018793103	0.006034483	0.006206897	0.011034483	0.033793103	0.08	0.118275862	0.129310345	0.065517241	0.049482759	0.027931034	0.025344828	0.018620690	0.034482759	0.051896552	0.170862069	1