Chapter 12

RADIATION PROTECTION

TABLE OF CONTENTS

Section Page

Chapter 12

RADIATION PROTECTION

TABLE OF CONTENTS (Continued)

Section Page

Chapter 12

RADIATION PROTECTION

TABLE OF CONTENTS (Continued)

Chapter 12

RADIATION PROTECTION

LIST OF TABLES

Chapter 12

RADIATION PROTECTION

LIST OF TABLES (Continued)

Chapter 12

RADIATION PROTECTION

LIST OF FIGURES

Chapter 12

RADIATION PROTECTION

LIST OF FIGURES (Continued)

Chapter 12

RADIATION PROTECTION

12.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES AND RADIATION EXPOSURES TO MEMBERS OF THE PUBLIC ARE AS LOW AS IS REASONABLY ACHIEVABLE (ALARA)

12.1.1 POLICY CONSIDERATIONS

Energy Northwest is committed to maintaining occupational and public radiation exposures as far below regulatory dose limits as is practical while performing all activities related to the operation of Columbia Generating Station (CGS) and the Independent Spent Fuel Storage Installation (ISFSI). This commitment is reflected in the Radiation Protection Program (RPP) that meets the requirements of 10 CFR 20 and provides for effective control of radiation exposure through

a. Management direction and support,

b. Establishment of radiation control procedures,

c. Consideration during design and modification of facilities and equipment, and

d. Development of good radiation control practices, including preplanning and the proper use of appropriate equipment by qualified, well trained personnel.

The radiation protection practices are based, when practicable and feasible, on Regulatory Guides 8.8, Revision 3, and 8.10, Revision 1. The Radiation Protection Program provides for the majority of the recommended actions in both regulatory guides, including the following:

a. Organization and position descriptions to ensure an adequate as low as is reasonably achievable (ALARA) program,
b. Exposure reduction program,
c. Cost-benefit analysis program, and
d. Exposure tracking program employing the “Radiation Work Permit.”

Procedures for personnel radiation protection are prepared consistent with the requirements of 10 CFR Part 20 and are approved, maintained, and adhered to for all operations involving personnel radiation exposure.

Energy Northwest organization is structured to provide assurance that the ALARA policy is effective in the areas described above. The following is a description of the applicable activities conducted by individuals or groups having responsibility for radiation protection.

a. The Plant General Manager, has the overall responsibility for approving the RPP and ALARA policy consistent with Energy Northwest and regulatory requirements, and for the radiological safety of all on-site personnel. This includes the responsibility for implementation of the ALARA program by plant staff. The Plant General Manager has ensured consideration of exposure reduction methods by adoption of the ALARA review practices described in Section 12.1.3. The Plant General Manager is responsible for the management of plant operational activities and for providing the Radiological Services Manager the resources and support necessary to implement the RPP. This includes the responsibility for ensuring that the ALARA program is not adversely affected by production oriented goals;

b. The Radiological Services Manager reports to the Plant General Manager and is responsible for implementing the RPP with the exception of those radiation safety duties for which the Assistant Radiological Services Manager/Radiation Protection Manager (RPM) is responsible. This individual provides organizational leadership and direction to the Assistant Radiological Services Manager/RPM in the management of the Radiation Protection department;

c. The Assistant Radiological Services Manager/RPM reports to the Radiological Services Manager. The Assistant Radiological Services Manager/RPM has direct access to the Plant General Manager in all matters relating to radiation safety, and has the responsibility and authority for ensuring that plant activities meet applicable radiation safety regulations and RPP requirements. Specific responsibilities are provided is Section 12.5.1;

d. The Radiological Operations Supervisor reports to the Radiological Services Manager. This individual provides supervision, leadership, and technical direction for implementation of the RPP;

e. The Health Physics (HP) Craft Supervisors report to the Radiological Operations Supervisor and implement the RPP through direct supervision of the plant HP Technicians. Areas of responsibility include characterization of plant radiological conditions, maintenance of radiological postings, detection and evaluation of radiological problems, and performance of facility and equipment decontamination, system flushes, and temporary shielding installation;

f. The Radiological Planning Supervisor reports to the Radiological Services Manager and is responsible for radiation exposure reduction. This is accomplished, in part, by providing input to the planning and coordination of plant activities. The Radiological Planning Supervisor makes recommendations for the control/elimination of radiological conditions that increase personnel exposure or the release of radioactivity;

g. The Radiological Support Supervisor reports to the Radiological Services Manager and provides technical support to the Radiation Protection organization. This individual provides additional assurance that regulatory requirements and industry guidance are incorporated into the program to ensure exposures are maintained ALARA.

In addition to the organization structure that has been provided for implementation of the ALARA policy, the following groups perform reviews to further ensure exposures are ALARA.

a. The Plant Operations Committee (POC) has been established and is functional. Its purpose is to serve as a review and advisory organization to the Plant General Manager in several areas, including radiological and nuclear safety. The RPM is a member of the POC and has direct input to this group on all radiological matters;

b. The Quality organization performs audits, surveillances, and assessments of plant operations. Audits, surveillances, and assessments will include verification of compliance with plant procedures, with regulations involving nuclear safety, and with operating license provisions. Results of these audits and surveillances will be submitted to both plant and corporate management. Since the system for ALARA review described in Section 12.1.3 provides for this consideration in all plant procedures, quality audits and surveillances will verify implementation of this principle;

c. The Corporate Nuclear Safety Review Board (CNSRB) provides a continuing appraisal of the ALARA policy. The Operational Quality Assurance Program Description (OQAPD) provides a description of this group's responsibilities and authority. By its charter, the CNSRB reviews radiological safety policies and programs and determines that these policies and programs are in compliance with NRC requirements. The CNSRB has the capacity for review in the area of radiological safety and has established a direct line of communication with plant management; and

d. The Senior Site ALARA Committee serves as a review and advisory organization to the Plant General Manager on occupational exposure to personnel. Committee membership, responsibilities, authorities, and records are prescribed in plant procedures.

The commitment to ALARA is implemented by employee training; audits, assessments, and reviews of the program; procedure development and reviews; enforcement of rules; and modifications to plant equipment or facilities where they will substantially reduce exposures at a reasonable cost. Management's commitment to the ALARA policy is further discussed in Section 12.5. This program meets all 10 CFR Part 20 regulations and considers the guidance of Regulatory Guides 8.8 and 8.10 in regard to policy considerations.

12.1.2 DESIGN CONSIDERATIONS

To ensure that personnel occupational radiation exposures are ALARA, extensive consideration is given to equipment design and locations, accessibility requirements, and shielding requirements. Many of these design objectives and considerations were established prior to the issuance of Regulatory Guide 8.8. However, the design of the plant substantially incorporates the recommendations provided in the regulatory guide. Design considerations that ensure occupational radiation exposures to personnel during normal operation and anticipated operational occurrences are ALARA are the following:

a. The facility is separated into controlled and uncontrolled areas based on anticipated radiation levels. The controlled areas of the facility are further defined by radiation zones established by personnel access requirements and are intended to limit radiation exposure to ALARA. The radiation zones provide guidance for the shielding design, contamination control, and ventilation flow pattern for the areas of anticipated personnel radiation exposure. See Section 12.3.1 for a description of the facility radiation zones.

b. Equipment location

    1. Several radiation sources on the 437 ft 0 in. level of the radwaste building are located with two sources in each cubicle. This arrangement maintains occupational exposures ALARA by use of the following alternate ALARA methods.

       The waste collector tank and floor drain collector tank are in the same cubicle. These tanks share redundant pumps and cross tie piping. If abnormal conditions occur and one or both of these tanks becomes a major source of radiation, either one of the pumps can be used to empty the tanks prior to any maintenance.

       The chemical waste tank and distillate tank share the same cubicle. These tanks are not expected to be major sources of radiation. Based on the source terms described in Table 11.2-1, the dose rate at 3 ft from the surface of these tanks normally does not exceed 0.1 mrem/hr. In addition, redundant pumps and cross tie piping permit the transfer of tank contents should abnormally high radioactivity levels occur.

Gas coolers and charcoal adsorbers share the same cubicle. These items have no moving parts and are highly reliable with no routine maintenance requirements. In addition, system redundancy and remote isolation capabilities eliminate the need for prompt entry into the cubicle. This permits the noble gases and radioiodines to significantly decay prior to entry.

Placing the preceding sources in shared cubicles does not result in increased occupational exposures.

2. Radioactive pipes are routed so that radiation exposure to plant personnel is minimized. The extent to which radioactive pipes are routed through normally accessible areas is minimized. Shielded pipe chases are utilized in normally accessible areas. Whenever possible, radioactive and nonradioactive pipes are kept separate for maintenance purposes.

3. Shielded valve stations are used where practical. To further minimize personnel exposure, remotely operated valves are used where practical. Normally operated manual valves in high radiation areas are provided with extension stems through a shield wall to a low radiation area.

4. Where practical, pumps are located in shielded areas outside cubicles containing radioactive components.

5. Where practical, local instrumentation readouts are routed to points outside shielding walls.

6. To minimize maintenance time and hence exposure, sufficient space is available in shield cubicles housing radioactive equipment (e.g., heat exchangers, demineralizers, etc.) to perform required tasks. Access platforms at intermediate levels are provided to enhance access to portions of equipment inaccessible from the floor.

7. Where possible, equipment and components which require frequent servicing are designed so that they can be removed, with minimum exposure, to appropriate low radiation areas.

8. Access to corridor C-125 on the 437 ft 0 in. level of the radwaste building will only be required for routine surveillance of equipment and nonroutine maintenance. Area radiation monitors 28 and 29 are located

in the corridor to detect abnormal radiological conditions and warn personnel if radiation levels are excessive. During reactor and radwaste operations, entry to this area will be under the direction of a Radiation Work Permit (RWP).

c. Shielding design is based on satisfying the radiation zone requirements utilizing design basis radiation sources for the shielding calculations. Shielding design is conservative since the design basis radiation sources are not expected to occur frequently.

   Penetrations through radiation shields are generally designed to prevent direct radiation streaming from a high radiation area to a low radiation area. Shielding discontinuities, such as shield plugs, are provided with offsets to reduce radiation streaming. Shield doors and labyrinths are used to eliminate radiation streaming through access openings in the cubicles.

d. Auxiliary systems that may become contaminated are designed with provisions for flushing or remote chemical cleaning prior to maintenance. This is accomplished by the following:

   1. Providing connections for the purpose of backflushing,

   2. Providing water connections to tanks containing spargers to allow for water injection to uncake contaminants, and

   3. Providing cross ties between redundant equipment and/or related equipment capable of redundant operation to allow removal of contaminated equipment from service.

e. The ventilation systems are designed to ensure control of airborne contaminants by providing air flow from areas of low radioactivity potential to areas of high radioactivity potential. Access to the systems is facilitated by the following:

   1. Filter access doors, which are sized to enhance the ease of performing maintenance, and

   2. Providing for periodic inservice testing of the equipment and filters.

f. Spread of contamination is minimized in the event spillage occurs by the following:

   1. Drains are provided in areas where equipment with large volumes of radioactive fluid is located. Drains are sized to conduct spillage to the appropriate liquid waste processing system;

   2. Floors and walls are protected with the appropriate coating to facilitate decontamination; and

   3. An equipment decontamination facility is provided to decontaminate tools and radioactive components.

g. While pipe runs are not sloped, those that carry radioactive fluids can be chemically decontaminated. Tank bottoms in the radwaste system are either sloped or dished. The exception is the waste surge tank located on el. 437 ft 0 in. level in the radwaste building.

h. Drain tap-offs are provided at low points in the piping systems.

i. Connections are placed above the centerline (top) of pipes when consistent with overall design requirements. Such connections are not always practical or desirable. For example, air pockets may form in an infrequently used line which is connected above the centerline (top) of another pipe.

j. Where practicable, the use of dead legs and low points in pipe systems are avoided, especially in pipes carrying demineralizer resins or concentrated radwaste.

k. T-connections in piping are minimized with the exception of

   1. Multiple flow paths, such as in the condensate filter demineralizer system, and

   2. Interconnected systems acting as mutual back ups, such as the floor drain and waste filter demineralizer systems.

l. Large pipe bend radii and piping elbows are used.

m. Butt welding by the open root method is used as described in Section 12.3.1.3.2.

n. Seal glands of pumps carrying concentrated radwaste or spent filter and demineralizer resins are flushed with condensate water during operation. After operation, the pumps are also flushed with condensate. Canned pumps are not used.

o. Where practicable, equipment used in the same process are located close together, resulting in short runs of interconnecting piping.

p. Pipes carrying slurries and resins are sized for turbulent flow to prevent any settling out of solids.

q. All equipment cubicles housing filters are equipped with removable ceiling plugs through which filter elements may be serviced or changed with the aid of tools to allow remote handling.

r. Operating experience from other BWR plants is periodically reviewed. Problems are reviewed and the plant design is checked to ensure that similar problems will not occur.

s. Design changes are reviewed by Radiation Protection.

12.1.3 OPERATIONAL CONSIDERATIONS

12.1.3.1 Procedures and Methods of Operation

A positive means of ensuring that occupational and public radiation exposures are ALARA has been incorporated into the Plant Procedures Manual (PPM) and Site Wide Procedure (SWP) preparation program. Procedures are formally reviewed for ALARA considerations as part of the approval process. The guidance provided by Regulatory Guide 8.8 is considered during this review. Additional information regarding ALARA considerations in procedures is provided in Section 12.5.3.2.

In addition to the above process, the Radiation Work Permit (RWP) process is used for individual tasks to identify radiological conditions and controls. This provides a means of prescribing precautionary measures, protective equipment, and other exposure reduction methods in each situation. Individual exposures, as determined by dosimeters, are recorded and provide a means of determining exposures for various tasks and work groups. This information is used for preplanning work, identifying sources, determining radiation levels and otherwise evaluating exposure problems.

Administrative controls ensure that occupational and public radiation exposures are as low as reasonably achievable. These are supplemented by the Radiation Protection staff through a program of surveillance, guidance, and investigation. A description of the program is outlined in Section 12.5 and includes the following aspects:

a. The Energy Northwest RPP includes procedures that provide for routine and special survey to determine sources and trends of exposure and for investigation to determine causes of normal and unusual exposure;

b. Plant procedures are formally reviewed by Radiation Protection for ALARA considerations when required;

c. Plant modifications that have radiological implications are reviewed by Radiation Protection to ensure that the changes incorporate ALARA considerations;

d. All maintenance tasks that involve radiological systems or controls are initially reviewed for ALARA and radiological concerns. For these tasks, a RWP is completed which describes the proposed actions. The RWP describes the conditions necessary such as survey requirements, surveillance, and protective apparel;

e. Prior to each scheduled maintenance and refueling outage, HP reviews the outage tasks and schedules and participates in outage planning for ALARA considerations; and

f. Adequate surveillance and supervision is provided to ensure that procedures are followed, prescribed precautions are taken, and radiation sources are identified.

12.1.3.2 Design Changes for ALARA Exposures

Operational requirements were considered in the original design of CGS for maintaining occupational exposures ALARA, and several design changes have been made since issuance of the PSAR and Regulatory Guide 8.8. These changes or additions were implemented as a result of review by both the architect-engineer and Energy Northwest personnel and include the following:

a. Revised offgas system valve design to prevent release of radioactive gases to building atmosphere,

b. Relocation of the counting room for lower background levels and adequate shielding,

c. Revised effluent monitoring capabilities to provide for more efficient monitoring,

d. Increased capability for in-plant continuous airborne radioactivity monitoring with remote readout and recording features,

e. Increased capability for the area radiation monitoring system to include remote readout/alarm and local alarm functions for the individual monitors,

f. Inclusion of supplied air stations throughout the plant for efficient respiratory protection,

g. Space and services provisions made for a decontamination facility and hot shop to reduce contact maintenance exposures and airborne radioactivity,

h. Revised penetration access design at sacrificial shield wall to reduce time required in this area,

i. Installation of overload protection on valve motor operators to minimize motor replacement in high dose rate areas,

j. Generated additional specification for replacement valve packing for selected valves to reduce time consumed in repacking,

k. Replaced hydraulic snubbers with mechanical snubbers to reduce maintenance requirements,

l. Provided method of venting the reactor vessel head through main steam line A to be ultimately filtered through the offgas system via the condenser, and

m. Made provisions for future connections to increase reactor water cleanup capacity during shutdown conditions to reduce radioiodine and particulate activity prior to vessel head removal and during outage periods.

New designs or design revisions are considered for exposure reduction as plant operation identifies problem areas.

12.1.3.3 Operational Information

Information from operating BWRs has been incorporated into the Radiation Protection procedures as discussed below:

a. Procedures have been written using guidance obtained from BWR operational experience reviews, from technical information obtained at power reactor conferences, and from participational experience in outages at operating BWRs;

b. Respiratory protection procedures incorporate proven practices from other nuclear facilities;

c. Typical procedures on survey methods, personnel monitoring, personnel dosimetry, and process/effluent radiological monitoring have been observed in the implementation stage at several operating reactor facilities. In addition, the procedures used at operating BWR plants have been reviewed. All of these were evaluated and applied in the procedure generating process;

d. Specific HP procedures or instructions have been written to furnish guidance on the following:

   1. The issuance, requirements, conditions, and controls of RWPs,

   2. The review process of plant procedures for ALARA considerations, and

   3. Methods for minimizing personnel exposures during RPV head removal, drywell entry, and conduct during emergencies.

12.2 RADIATION SOURCES

12.2.1 CONTAINED SOURCES

12.2.1.1 General

The design basis radiation sources considered are the following:

a. The reactor core,

b. Activation of structures and components in the vicinity of the reactor core,

c. Radioactive materials (fission and corrosion products) contained in system components,

d. Spent fuel, and

e. Radioactive wastes for offsite shipment.

The design basis process radiation sources used for shielding are based on the source terms given in Tables 11.1-1 through 11.1-5.

12.2.1.2 Reactor and Turbine Building

The reactor building sources include the following:

a. The reactor core,

b. Activated structures and components,

c. Components and equipment containing activation, fission, and corrosion products, and

d. Spent fuel.

12.2.1.2.1 Reactor Core Radiation Sources

During normal power operation, the reactor core radiation sources considered are the prompt fission neutrons and gamma rays, capture gamma rays, and fission product gamma rays. During shutdown, the reactor core radiation sources are the fission product gamma rays. The core is treated as a cylindrical volume source for use in the NRN+(1) and QAD+(2) codes. See Section 12.3.2 for details.

Table 12.2-1 lists the basic data required for reactor core source computations including volume fractions, power peaking factors, and core power density.

Table 12.2-2 presents the neutron multigroup flux at the reactor core-reflector boundary, 238 cm from the core centerline. The NRN computer code (Reference 12.2-1) is used in computing the flux. This code is based on a multigroup slowing down and diffusion system with the deep penetration source corrected by a multigroup removal source.

Table 12.2-3 lists the gamma ray energy spectrum for the reactor core during normal operation. The operating core energy spectrum represents the sum of the spectra for prompt fission, fission product, and to capture gamma rays. The postoperation fission product gamma ray energy spectrum in the core immediately after shutdown is listed in Table 12.2-4.

12.2.1.2.2 Process System Radiation Sources

12.2.1.2.2.1 Introduction. The following process systems govern the shielding requirements within the reactor and turbine buildings:

a. Recirculation (RRC),
b. Reactor water cleanup (RWCU),
c. Reactor core isolation cooling (RCIC),
d. Residual heat removal (RHR),
e. Fuel pool cooling and cleanup (FPC),
f. Main steam (MS) and the reactor feedwater system (RFW),
g. Traveling in-core probe (TIP), and
h. Offgas system (OG).

The locations of the equipment, which are part of these systems, are shown on the radiation zone drawings, Figures 12.3-5 through 12.3-18.

12.2.1.2.2.2 Recirculation System Sources. The coolant activation products, principally ^{16}N, are the dominant sources of radiation in the RRC system during normal operation. The ^{16}N introduced into this system decays significantly by the time it returns to the core. Therefore, decay is considered in the portions of this system where elapsed transit time is significant.

For shielding purposes, the pipes in the RRC system are treated as equivalent line sources. The ^{16}N source strength in these lines varies from 0.10 Ci/cm at the line from the reactor vessel to 0.02 Ci/cm at the header which distributes coolant flow back to the reactor. These sources are located within the drywell of the primary containment of the reactor building, from approximately el. 501 ft to el. 540 ft.

During shutdown, fission products become the dominant source. This shutdown source is considerably less than that of ^{16}N during normal operation. Shielding design is based on the ^{16}N source, which is more than adequate to shield against the fission product shutdown source.

12.2.1.2.2.3 Reactor Water Cleanup System Sources. The major sources of radiation in the portions of the RWCU system in the reactor building during normal operation are the coolant activation products, principally ^{16}N. The ^{16}N source strength (given in activity per unit length of line) in the RWCU system ranges from 1.00 x 10^{-3} Ci/cm in the lines flowing into the cleanup recirculation pumps to 4.68 x 10^{-8} Ci/cm at the exhaust of the tube side of the nonregenerative heat exchanger. Returning from the radwaste building, the ^{16}N source strength ranges from 3.08 x 10^{-10} Ci/cm to negligible (less than 10^{-14} Ci/cm).

The ^{16}N source strengths in the regenerative and nonregenerative heat exchangers are

a. Tube side of the regenerative heat exchanger: 2.69 x 10^{-6} Ci/cm^3,
b. Tube side of nonregenerative heat exchanger: 6.24 x 10^{-8} Ci/cm^3, and
c. Shell side of the regenerative heat exchanger: 1.70 x 10^{-14} Ci/cm^3.

These heat exchangers are treated as cylindrical sources for input into the QAD computer program (see Section 12.3.2). The portions of this system located in the radwaste building are discussed in Chapter 11. The RWCU pumps are located at el. 522 ft 0 in. and the RWCU regenerative and nonregenerative heat exchangers are located at el. 548 ft 0 in.

During shutdown, the fission products are the dominant radiation source. Since the shielding is designed to shield against the ^{16}N source during normal operation, it is more than adequate to shield against the shutdown fission product source.

12.2.1.2.2.4 Reactor Core Isolation Cooling System Source. The dominant source of radioactivity in the RCIC system is the ^{16}N present in the main steam used to drive the RCIC turbine. This ^{16}N source occurs when the RCIC turbine is tested on a monthly basis. The shielding design is based on this source. The transit time is assumed to be negligible in this system.

The resulting ^{16}N activity (given in activity/unit length of line) in the inlet line is 2.96 x 10^{-4} Ci/cm and in the outlet line, it is 6.57 x 10^{-5} Ci/cm. These sources are treated as equivalent line sources for shielding purposes.

The RCIC turbine source strength is 8.44 x 10^{-2} Ci of ^{16}N. It is treated as a point source since the distance to the dose points is much greater than the turbine dimensions. The RCIC pump and turbine is located at el. 422 ft 3 in. of the reactor building.

12.2.1.2.2.5 Residual Heat Removal System Sources. The RHR system radiation sources consist of the fission and corrosion products. Table 12.2-5 lists the gamma ray energy spectrum of the radionuclides in the RHR pumps, pipes, and heat exchangers 4 hr after shutdown. These sources are based on the RHR system operating in the normal reactor shutdown cooling mode. The fission and corrosion product isotope concentrations used are listed in Tables 11.1-2 through 11.1-4.

The RHR heat exchangers are located approximately from el. 559 ft 0 in. to el. 589 ft 0 in. on the west side of the reactor building. The RHR pumps are located at el. 422 ft 3 in. on the west side of the reactor building.

The pipes in this system are treated as equivalent line sources. The heat exchangers are treated as cylindrical sources.

12.2.1.2.2.6 Fuel Pool Cooling and Cleanup and System Sources. The primary sources of radioactivity in the spent fuel assemblies, which are stored in the fuel pool, are the fission products. Table 12.2-6 lists the gamma ray energy spectrum for the spent fuel sources for shutdown time of 2 days. The refueling area is located at el. 606 ft 10.5 in. in the reactor building.

These source terms are calculated using the Perkins and King data (Reference 12.2-2). The shielding calculations are done using the QAD point kernel code (Reference 12.2-3). The following assumptions are used in determining the shielding requirements:

a. After radioactivity has reached equilibrium in the fuel assemblies, it is assumed that the reactor is shut down and the whole core is moved, within 2 days, into the spent fuel pool;

b. The whole core and another one-fourth of a core from the last refueling are located by the north wall of the spent fuel pool to give the most conservative dose rate on the outside of the wall. Less water exists between the assembly racks and the north wall than between the assembly racks and any other side of the pool. The assemblies from past refuelings do not add to the shielding requirements because they have decayed for more than 1 year, they are shielded by pool water, and they provide self shielding; and

c. The water, racks, spent fuel, and other constituents that are located within the array of spent fuel assemblies are homogenized for the purpose of determining the required values of the linear attenuation coefficients.

The minimum depth of water needed to adequately shield the refueling area from the spent fuel assemblies is calculated. It is found that the elevated fuel assembly during the fuel transfer operation controls the dose rate to the refueling area and, hence, the water depth in the spent fuel pool. The fuel assembly is treated as a cylindrical source geometry for the purpose of computing the water depth.

The source strength used to determine the shielding requirements for the dryer-separator pool is based on a contact dose rate for the separator of 10 R/hr. The average gamma ray energy is approximately equal to 1 MeV.

12.2.1.2.2.7 Main Steam and Reactor Feedwater Systems Sources. In these systems, the dominant sources of radioactivity are the coolant activation products, principally ^{16}N. The following equipment is considered:

a. Moisture separators and reheaters (MSR),
b. Main condenser and hotwell,
c. Feedwater heaters, and
d. The piping associated with these systems.

The moisture separator and reheaters govern the shielding requirements on the operating floor (el. 501 ft 0 in.) of the turbine building. The ^{16}N source strengths for the first stage reheater tubes, the second stage reheater tubes, and the MSR plena are listed on Table 12.2-7. The first and second stage reheater tubes are approximated by rectangular parallelepipeds. The plena are divided into an array of rectangular parallelepipeds and cylinders, depending on their physical arrangement.

The ^{16}N source strength in the main condenser is 6.0×10^{-8} Ci/cm^3. This is based on the incoming flows from the exhaust of the reactor feed pump turbine, and the exhaust steam from the low pressure turbine. The main condenser is treated as either a truncated cone or infinite slab depending on the view angle and distance from the condenser to the dose point.

Since most of the ^{16}N exists as a noncondensable gas, it comes out of solution in the main condenser. Thus, the radiation sources in the hotwell are the fission or corrosion product radionuclides. Table 12.2-8 lists the gamma ray volumetric source strengths calculated from the hotwell. The hotwell is treated as an infinite slab.

The ^{16}N source strength of feedwater heater 6 listed in Table 12.2-9, governs the shielding requirements on the mezzanine floor of the turbine building (el. 471 ft 0 in.). The feedwater heater is treated as an array of rectangular parallel pipes and cylinders for input into QAD.

Table 12.2-10 lists the ^{16}N source strengths in selected steam piping in the MS and RFW systems.

12.2.1.2.2.8 Offgas Sources in the Turbine Generator Building. Table 12.2-11 lists the source strength from the offgas system equipment in the turbine building. Nitrogen-16 is the dominant radionuclide present in this system. The offgas equipment is located at el. 441 ft 0 in. of the turbine building.

12.2.1.2.2.9 Traveling In-Core Probe System Sources. The primary source of radiation in the TIP system is the ^{56}Mn in the traversing in-core probe (TIP) cable. The average source strength per unit length of cable is 3.27 x 10^4 µCi/cm. This is calculated using an exposure time of 864 sec. The average radioactivity emitted per unit length is calculated using a cosine distribution for the neutron flux in the axial direction of the core. The cable is treated as a line source for shielding purposes. The TIP components are located at el. 501 ft 0 in. of the reactor building.

12.2.1.2.2.10 Sources Resulting From Crud Buildup. Fission and corrosion product deposition is complex to describe analytically. In estimating the effects of this source, operating experience from other BWR plants is used (Reference 12.2-4); Section 12.4 discusses this in greater detail.

12.2.1.3 Radwaste Building

The radiation sources present in the radwaste building are discussed in Chapter 11.

12.2.1.4 Special Nuclear Materials

A list of all special nuclear materials with an activity greater than 100 mCi is given in Table 12.2-12.

12.2.2 AIRBORNE RADIOACTIVE MATERIAL SOURCES

12.2.2.1 General

Design features that limit the airborne radioactivity in normally occupied areas are incorporated into the plant. Areas that are designated as radiation Zones I and II are considered to be normally occupied areas.

The plant was designed so that the airborne radionuclide concentrations in normally occupied areas are well below the limits specified in 10 CFR Part 20. The criteria for Zones I and II are specified in 10 CFR 20, Appendix B to 20.1001-20.2401, Table 1, Column 3.

No radiation Zone I areas exist in the reactor or turbine generator building. The only Zone I areas found in the radwaste and control building are the main control room and the counting room shown on Figures 12.3-14 and 12.3-13, respectively. The main control room is located at el. 501 ft 0 in. It has a separate heating, ventilating, and air conditioning (HVAC) system which brings in fresh air from the outside. See Section 9.4.1 for further discussion. The counting room is located at el. 487 ft 0 in. As seen in Figure 9.4-3, the incoming ventilation flow into this room consists of fresh air. The outgoing air flow is to the area surrounding the counting room. It is concluded that the airborne concentration in the counting room is small.

See Section 12.2.2.3.5 for discussion on the contribution of sampling and radiochemical analysis on airborne radioactivity levels within this area.

12.2.2.2 Model for Computing the Airborne Radionuclide Concentration in a Plant Area

The model used for computing the airborne radionuclide concentration is based on the continuous leakage of a radioactive fluid into a plant area. The removal of radionuclides from this area is through ventilation exhaust and decay. The equation which yields the airborne radionuclide concentration in a plant area is:

$$C_i = \frac{A_i q_s (PF)_i}{\lambda i^v + q_a} \left(1 - \exp - (\lambda_i + q_a / V)\ \mathrm{t}\right) \qquad (12.2\text{-}1)$$

where: C_i = concentration of radionuclide i in a given plant area (μci/cm^3)

A_i = concentration of radionuclide i in the fluid (mCi/g)

q_s = rate of radionuclide leakage into an area (g/minute)

$(PF)_i$ = partition factor for radionuclide i (dimensionless)

λ_i = decay constant for isotope i (1/minute)

V = volume of area (cm^3)

q_a = HVAC air flow rate out of area (cm^3/minute)

t = time interval between start of leak and calculation of concentration (minute)

The equilibrium value of C_i is given by

$$C_i = \frac{A_i q_s (PF)_i}{\lambda_i\ V + q_a} \qquad (12.2\text{-}2)$$

Note that in all analyses where these two equations are used, the partition factor is set equal to 1.0. This yields conservative results.

12.2.2.3 Sources of Airborne Radioactivity

The potential sources of airborne radioactivity found in the plant are as follows:

a. Leakage from process equipment in radioactive systems, such as valves, flanges, and pumps,

b. Sumps, drains, tanks, and filter/demineralizer vessels which contain radioactive fluid,

c. Exhaust from relief valves,

d. Removal of reactor pressure vessel (RPV) head and associated internals,

e. Radioactivity released from sampling, and

f. Airborne radioactivity released from the spent fuel pool water and spent fuel movement.

Sections 12.2.2.3.1 through 12.2.2.3.6 discuss each of these sources and their effect on the airborne radionuclide concentration in normally occupied areas of the plant. All design features that serve to reduce the airborne radionuclide concentration are also discussed.

12.2.2.3.1 Effect of Leakage from Process Equipment in Radioactive Systems

Leakage into normally occupied plant areas from radioactive process systems is described by three parameters.

The first parameter is the location of the source of radioactive leakage. If this source is located in a Zone III or Zone IV radiation area, it does not contribute to the airborne radionuclide concentration in a normally occupied area because the air flow is from areas of potentially low radioactivity contamination to areas of potentially high radioactivity contamination. Any leakage of radioactive airborne contamination into a Zone III or Zone IV radiation area in the reactor, radwaste and control or turbine generator building is exhausted from that area through the HVAC system and is not transported into a normally occupied area. See the HVAC flow diagrams for the reactor, radwaste and control or turbine generator buildings, Figures 9.4-2, 9.4-3, and 9.4-6, and the radiation zone drawings, Figures 12.3-5 through 12.3-18.

Areas with multiple zone designation are regarded as having a high radioactivity contamination potential. Air from these areas is either exhausted to another Zone IV area or to the HVAC exhaust equipment. Thus, any radioactive leakage that occurs within an area with a multiple zone designation should not affect the airborne radionuclide concentration of a normally occupied area.

The second parameter is the rate at which the radioactive leakage is introduced into an area. Any system that operates continuously is potentially a greater source of airborne radioactivity than a system that operates periodically when radionuclide concentration and leakage rate are the same for each system. The operating pressure of the system is another consideration which affects the leakage rate. A system, such as the main steam system, is expected to have a higher leakage rate than a system which operates at atmospheric pressure. However, radioactive systems which operate at high pressure are generally located in Zone IV areas. Thus, these systems do not significantly contribute to the airborne radioactivity level in normally occupied areas. This is due to the HVAC air path which was discussed earlier.

The third parameter is the radionuclide concentration in the leaking fluid. A system that is leaking highly radioactive fluid such as reactor coolant or main steam is of greater concern, initially, than a system which is leaking condensate storage tank water. Those systems which can leak highly radioactive fluid are located in either a Zone III or Zone IV radiation area and the HVAC air flow paths do not transport this radioactivity to a low radiation area, as discussed earlier. The effect of systems which leak fluid with a low radionuclide concentration is discussed below.

A list of all radioactive systems found in the plant is provided in Table 12.2-13. Each radioactive system listed has been checked to determine its contribution to the airborne radioactivity levels in normally occupied areas. It is found that most of these systems are located in Zone III or Zone IV areas and do not contribute to the airborne radioactivity levels in normally occupied areas, due to HVAC design features as explained earlier. Some of the radioactive systems are located within areas which have a multiple radiation zone designation; e.g., Zone IV during system operation and Zone III during system shutdown. The systems in these areas do not contribute to the airborne radioactivity levels due to HVAC system design features, as explained earlier. Any system which is not entirely located in a Zone IV, Zone III, or multiple zone radiation area and which may contribute to airborne radionuclide levels in normally occupied areas is discussed in the following paragraphs. Those systems which are used only during loss-of-coolant accident (LOCA) conditions are not discussed. These include the high-pressure core spray (HPCS), low-pressure core spray (LPCS), and main steam leakage control (MSLC) systems. These systems are not regarded as being significant sources of airborne radioactivity since they are used only in a test mode or in an accident situation.

The major source of control rod drive (CRD) leakage which can contribute to the airborne radionuclide levels in a normally occupied area are the CRD pumps located in the CRD pump room. The CRD pump room is located between column lines 3.4/6 and N/N.8 at el. 422 ft 3 in. of the reactor building. Figure 4.6-5 shows suction flow for the CRD pumps is from the outlet of the condensate filter demineralizers or the condensate storage tank. Only one pump is in operation at any time and the leakage rate from this pump is assumed to be 50 gal/day. The incoming HVAC air flow to this area is approximately 3500 ft^3/minute. The mathematical model described in Section 12.2.2.2 is used to evaluate the airborne activity. The results are listed in Table 12.2-14. It is seen that the expected airborne radionuclide level is well below the derived air concentrations (DAC) values.

The major source of main condensate leakage which can contribute to the airborne radionuclide levels in a normally occupied area are the condensate pumps. These pumps are located between column lines D/F and 13/15 at el. 441 ft 0 in. of the turbine generator building. The suction flow from these pumps is taken from the main condenser hotwell. Since the suction side of the pumps is at a negative pressure to the surrounding area, no leakage from the pump seals should occur. However, leakage may occur from the valves and other equipment which are located on the discharge side of the pump. It is conservatively assumed that the leakage is 1 gpm. The airborne radionuclide concentration in the area where the condensate booster pumps and condensate pumps are located is listed in Table 12.2-15.

The offgas equipment which is radioactive is located in Zone IV areas. The exception is the afterfilter, which is located between column lines K.1/L.9 and 14.7/15.4 at el. 452 ft 0 in. of the radwaste building. This filter is not a source of airborne radioactivity. The reasons are that the filter operates at approximately atmospheric pressure and the radioactivity contained within the filter is small. See Section 11.3 for details on offgas system operation.

12.2.2.3.2 Effect of Sumps, Drains, Tank, and Filter Demineralizer Vents

The equipment drain (EDR), floor drain (FDR), and miscellaneous radwaste (MWR) systems are designed to collect and process various types of liquid radioactive waste generated in the reactor, radwaste, or turbine generator buildings, as explained in Section 11.2. In each of the aforementioned buildings, there is a network of EDR, FDR, and MWR drains which conduct radioactive liquid waste to an EDR, FDR, and MWR sump, respectively. These drains and sumps are not significant sources of airborne radionuclides for the following reasons:

a. Each of the EDR, FDR, and MWR sumps present in the reactor, radwaste, and turbine generator buildings is covered with a steel plate. The free volume in the sump is maintained at a negative pressure with respect to the surrounding area by use of a riser vent, which is connected to the HVAC system in the building of concern. The steel plate covering the sump does not provide an airtight seal. However, air is drawn into the sump, then through the riser vent and is exhausted to the HVAC system. Any radionuclides that escape into the free volume of the sump are discharged to the HVAC system and do not escape into the area surrounding the sump; and

b. The EDR, FDR, and MWR drains are connected to their respective sumps through the same riser vents discussed in the preceding paragraph. Some of the drains employ loop seals, which prevent radioactive gases from escaping into the areas around the location of the drains. Other drains do not employ loop seals, but since the riser vent is connected to the HVAC system, air will be drawn into the drain through the riser vent and out to the HVAC system.

The tanks and filter demineralizer vessels that contain significant inventories of radionuclides are vented to the HVAC system. These tanks and filter demineralizer vessels are located in Zone III or Zone IV radiation areas. Even if any airborne radionuclides were released from these tanks or filter demineralizers, there would be no effect on normally occupied areas due to the HVAC system design features, which are explained in Section 12.2.2.3.1.

12.2.2.3.3 Effect of Relief Valve Exhaust

The relief valves found in the various plant systems which can exhaust radioactive fluids are not considered to be a significant source of airborne radioactivity in normally occupied areas. The reasons are as follows:

a. All relief valves (except the main steam safety relief valves), which relieve pressure in the turbine main steam or bleed systems, exhaust directly to the condenser, and

b. All relief valves, which relieve pressure in a system which contains radioactive fluid, either exhaust directly into the suppression pool, an equipment or floor drain, or into a line which is part of the system in question.

With reference to the equipment or floor drain systems, the receiving point for relief valve discharges is in a radiation zone equivalent or higher than the equipment being relieved. For discharge back to the system, the same is true.

The main steam safety relief valves, which are located within primary containment, provide pressure relief to the main steam lines coming from the reactor. These valves exhaust directly to the suppression pool. The effect of main steam relief valve blowdown in normally occupied areas in secondary containment is analyzed by assuming that all radionuclides that are present in the main steam blowdown are released to the primary containment air. The radionuclide distribution within the free volume of the primary and secondary containment is assumed to be uniform. The radionuclide concentration in secondary containment becomes, in mCi/cm^3:

$$C_{sc,i} = \frac{R\, q_b t_b A_i}{q_v - R\, V_{sc}} (\exp - (R + \lambda_i)\ t \text{ - } \exp \text{ - } (\lambda_i + q_v / V_{sc}) t) \qquad (12.2\text{-}3)$$

where: R = primary containment leakage constant (1/minute)

q_b = main steam blowdown flow (g/minute)

t_b = duration of blowdown flow (minute)

q_v = ventilation flow rate out of secondary containment (cm^3/minute)

V_{sc} = volume of secondary containment (cm^3)

λ_i = decay constant for isotope i (1/minute)

t = time after blowdown event

$C_{sc,i}$ = airborne radionuclide concentration of radionuclide i in the secondary containment ($\mu Ci/cm^3$)

A_i = radionuclide concentration in blowdown fluid (μCi/g)

The value of t which yields the maximum value of $C_{sc,i}$ is

$$t = \frac{1}{R - q_v / V_{sc}} \ln \frac{R + \lambda_i}{\lambda_i + q_v / V_{sc}} \qquad (12.2\text{-}4)$$

The calculated results are based on the occurrence of a main steam isolation valve closure. This results in all 18 relief valves being actuated for a maximum duration of 40 sec. This event results in the maximum release of radionuclides to primary containment (see Table 5.2-3 and Sections 15.1 and 15.2). The values of the various parameters used in equations 12.2-3 and 12.2-4 are given as follows:

R = 0.5 vol. %/day (Section 3.8.2.3-1)

q_b = 1.6×10^7 lb/hr = 1.2×10^8 g/minute (Table 5.2-3)

t_b = 40 sec = 0.67 minute (Table 5.2-3)

q_v = 9.5×10^4 cfm (Table 11.3-6)

V_{sc} = $3.5 \text{ X } 10^6 \text{ ft}^3$ (Table 11.3-6)

The values of A_i are based on the information found in Section 11.1.

The maximum airborne concentration of the most important radionuclides in secondary containment from a main steam relief valve blowdown is listed in Table 12.2-16. The concentrations are far below the DAC criteria given in 10 CFR Part 20. It is concluded that the main steam relief valve blowdown has a negligible effect on the airborne radionuclide level in secondary containment.

12.2.2.3.4 Effect of Removing Reactor Pressure Vessel Head and Associated Internals

Experience at BWR plants has shown that an inventory of radioactive gases will accumulate inside the reactor vessel head between the time of shutdown and head removal. These gases consist primarily of the longer lived radioiodines and noble gases. To prevent these gases from being released to the refueling area, provisions are made so that the gases are vented to the HVAC system prior to reactor pressure vessel (RPV) head removal. These provisions include the ability to vent the RPV head through a 4-in. flexible hose connection to the reactor building sump vent filter units which contain particulate and charcoal filter components. The connection to the sump vent filter unit is shown in Figure 9.4-2.

Provisions are also made for clean water services to the RPV cavity area. This permits wetting of the RPV cavity and components to minimize airborne contamination. This is done prior to flooding the RPV cavity.

It is anticipated that RPV head and reactor internals removal will have a minimal effect on the airborne radionuclide level in the spend fuel area.

12.2.2.3.5 Effect of Sampling

The possibility of releasing radionuclides which could become airborne during sampling operations is recognized. Design features are incorporated into the sample system to limit the radionuclide release. Radioactive liquids that require frequent grab sampling are piped via sample tube lines to fume hoods located in sample rooms. Grab sampling will normally be accomplished in the fume hoods. During sampling, an inflow face air velocity of approximately 100 ft/minute will be maintained to sweep any airborne radioactive particles to the exhaust duct. Administrative control is further exercised in the form of procedures that are followed when process fluids are sampled. This minimizes the release of radioactive fluids and, hence, exposure to personnel during the sampling process.

12.2.2.3.6 Effect of Spent Fuel Movement

Experience at four operating BWR plants has shown that fuel movement does not present any unusual radiological problems. The expected level of radioactivity in the spent fuel pool water is listed in Table 3.5-7 of the Environmental Report. This activity is not expected to have any significant radiological effects on the airborne radionuclide levels above the spent fuel pool floor.

12.2.2.3.7 Effects of Solid Radwaste Handling Areas

The solid radwaste handling equipment contained Zone IV area between columns N.1 and S (Figure 12.3-11) are designed for remote operation. Entry for maintenance activities will normally entail shut down and flushing of systems and equipment. The solid radwaste handling equipment is operated primary from low background areas of Zone III between columns Q.1 and R.2 near the waste compactor.

The ventilation supply to this Zone III area is clean outside air with air flow into surrounding normally unoccupied areas. The only source of airborne radioactivity, the waste compactor, is vented directly to the filtered exhaust ventilation system.

Airborne radioactivity levels in the normally occupied solid radwaste handling areas will be ambient outside air concentrations.

12.2.2.3.8 Effects of Liquid Radwaste Handling Areas

Normally occupied liquid radwaste handling areas include the valve corridor (a Zone III area), the precoat room and the radwaste control room (Zone II areas) shown in Figure 12.3-12.

This valve corridor is supplied directly with outside air. Components that are operated from this corridor contain radioactive materials which are located in normally unoccupied valve and pump rooms which are served by separate ventilated supply and exhaust. The radwaste control room and the precoat rooms do not house components containing radioactive material.

Although not normally occupied, the possibility exists that entry into pump corridor (a Zone IV area between columns 11.2 and 12.2) (Figure 12.3-11) and valve and pump (East and West side) rooms (Zone IV areas of Figure 12.3-12) could be necessary while systems are operating.

The pump corridor of Figure 12.3-11 contains the highest concentrations of radioactive material and has the lowest air flow to volume ratio, so is taken as a worst case. Equilibrium airborne radioactivity concentration is calculated as described in Section 12.2.2.2 assuming a leak rate of 50 gal/day from the reactor water cleanup phase separator decant pump. The results are shown in Table 12.2-17.

12.2.3 REFERENCES

12.2-1 Aalto, E. et al., A Users Manual for the NRN Shield Design, AE-145, June 1964.

12.2-2 Perkins, J. F. and King, R. W., Energy Release from Decay of Fission Products, Nuclear Science and Engineering, Vol. 3, 1958 and Perkins, J. F., U.S. Army Missile Command Redstone Arsenal, Report No. RR-TR-63-11, July 1963.

12.2-3 Malenfant, Richard E., QAD: A Series of Point-Kernel General Purpose Shielding Program, National Technical Information Service, LA-3573, October 1966.

12.2-4 Butrovich, R. et al., Millstone Nuclear Power Station, Refueling/Maintenance Outage, GE NEDO-20761, Class 1, Fall 1974.

Table 12.2-1

Basic Reactor Data for Source Computations
(During Plant Operation)

Reactor thermal power		3486 MW
Overall average core power density		51.6 w/cm^3
Core power peaking factors		
At core center:	$\frac{P_{max}}{P_{ave\,Z}}$ (axial)	1.5
	$\frac{P_{max}}{P_{ave\,R}}$ (radial)	1.4
At core boundary:	$\frac{P_{max}}{P_{ave\,Z}}$ (axial)	0.5
	$\frac{P_{max}}{P_{ave\,R}}$ (radial)	0.7

Core volume fractions:

Material	Density (g/cm^3)	Volume Fraction
UO_2	10.4	0.254
Zr	6.4	0.140
H_2O	1.0	0.274
Void	0	0.332

Average water density between core and vessel below the core	0.74 g/cm^3
Average water-steam density above core	
In the plenum region	0.23 g/cm^3
Above the plenum (homogenized)	0.6 g/cm^3
Average steam density	0.036 g/cm^3

Table 12.2-2

Neutron Flux at Reactor Core-Reflector Boundary

Energy Range (MeV)	Neutron Flux (Neutrons/cm^2 sec)
17.9-14.0	7.13E9
14.0-12.0	2.37E9
12.0-10.0	1.79E10
10.0-9.0	2.37E10
9.0-8.0	4.69E10
8.0-7.0	1.17E11
7.0-6.0	3.45E11
6.0-5.0	6.57E11
5.0-4.0	1.23E12
4.0-3.0	2.34E12
3.0-2.5	2.04E12
2.5-2.0	1.27E12
2.0-1.5	2.97E12
1.5-1.0	5.63E12
1.0-0.7	3.18E12
0.7-0.5	3.92E12
0.5-0.3	4.15E12
0.3-0.1	5.62E12
0.1-0.03	3.50E12
0.03-0.01	2.31E12
1.0(-2)-1.0(-3)	3.76E12
1.0(-3)-1.0(-4)	3.07E12
1.0(-4)-1.0(-5)	2.40E12
1.0(-5)-1.0(-6)	1.94E12
1.0(-6)-1.0(-7)	1.50E12
1.05(-7)-thermal	2.58E12

Table 12.2-3

Reactor Core Gamma Ray Energy Spectrum
During Operation

Energy Range (MeV)	Mid-Range Energy (MeV)	Volumetric Energy Release Rate (MeV/cm^3 sec)
10.0-7.0	8.5	5.97E11
7.0-5.0	6.0	4.00E11
5.0-3.0	4.5	4.77E12
3.0-2.0	2.5	6.83E12
2.0-1.0	1.5	1.02E13
1.0-0.0	0.5	1.25E13

Table 12.2-4

Reactor Core Gamma Ray Spectrum
Immediately After Shutdown

Energy Range (MeV)	Average Energy (MeV)	Volumetric Energy Release Rate (MeV/cm^3 sec)
>2.60	3.00	8.70E11
2.60-2.20	2.40	4.24E11
2.20-1.80	2.00	2.63E11
1.80-1.35	1.58	1.15E12
1.35-0.90	1.13	1.40E12
0.90-0.40	0.65	1.46E12
0.40-0.00	0.20	2.83E11

Table 12.2-5

Fission Product Source in RHR Piping and Heat Exchangers 4 Hours After Shutdown

Energy Range (MeV)	Average Energy (MeV)	Energy Release (MeV/cm^3 sec)
4.00-3.00	3.50	2.30E2
3.00-2.60	2.75	3.70E2
2.60-2.20	2.40	8.20E2
2.20-1.80	2.00	1.30E3
1.80-1.35	1.58	4.80E3
1.35-0.90	1.13	5.00E3
0.90-0.40	0.65	9.20E3
0.40-0.10	0.25	1.80E3

Table 12.2-6

Gamma Ray Energy Spectrum For
Spent Fuel Sources (One Core)

Energy Range (MeV)	Average Energy (MeV)	Volumetric Energy Release Rate (MeV/cm^3 sec) 2 Days After Shutdown
>2.60	3.00	2.54E8
2.60-2.20	2.40	9.89E9
2.20-1.80	2.00	4.93E9
1.80-1.35	1.58	1.34E11
1.35-0.90	1.13	2.59E10
0.90-0.40	0.65	2.97E11
0.40-0.00	0.20	5.17E10

Table 12.2-7

Nitrogen-16 Source Strength in Main Steam
and Reactor Feedwater

Component	Radioactivity Concentration (Ci/cm^3)
Moisture separators and reheaters (MSR)	
Plena	3.48E-7
First stage reheater tube bundle (east end of MSR)	7.23E-7
First stage reheater tube bundle (west end of MSR)	5.91E-7
Second stage reheater tube bundle (east end of MSR)	1.43E-6
Second stage reheater tube bundle (west end of MSR)	1.14E-6

Table 12.2-8

Gamma Ray Energy Spectrum and Volumetric
Source Strength in the Hotwell

Group	Average Group Energy (MeV)	Volumetric Energy Release Rate (MeV/cm^3 sec)
1	3.50	3.82E1
2	2.80	7.92E1
3	2.40	1.43E2
4	2.00	1.24E2
5	1.57	3.94E2
6	1.12	3.00E2
7	0.65	6.71E2
8	0.20	8.26E1

Table 12.2-9

Nitrogen-16 Source Strength in
Feedwater Heater 6

	Radionuclide Concentration (Ci/cm^3)	
Feedwater Heater	Steam	Water
6	4.93E-7	8.40E-6

Table 12.2-10

Nitrogen-16 Source Strengths for Piping Associated With the Main Steam and Reactor Feedwater Systems

Point of Interest	Line Source (Ci/cm)
Input lines to high pressure turbine	7.60E-3
Extraction steam line from high pressure turbine to FWH 6A	7.50E-4
Crossunder piping from high pressure turbine to main steam relief	5.50E-4
Crossover piping from main steam relief to low pressure turbine	3.80E-4
Extraction steam line from low pressure turbine to FWH 1A	7.50E-5
Extraction steam line from low pressure turbine to FWH 2A	1.20E-4
Extraction steam line from low pressure turbine to FWH 3A	1.40E-4
Extraction steam line from low pressure turbine to FWH 4A	1.50E-4
Heater drain line from FWH 6A to FWH 5A	2.30E-5
Heater drain line from FWH 5A to FWH 4A	1.01E-6

Table 12.2-11

Offgas System Sources in the Turbine Generator Building

Component	^{16}N Source Strength (μCi/cm^3)
Steam jet air ejector (first stage)	2.3E0
Intercondenser	4.2E1
Steam jet air ejector (second stage)	3.6E0
Preheater	2.8E0
Recombiner	2.3E0
Offgas condenser	3.7E1
Water separator[a]	2.7E1

[a] The preheater, recombiner, offgas condenser, and water separator are located in the same room.

Table 12.2-12

Special Sources With Strength Greater Than 100 Millicuries

Isotope	Identification	Form	Quality (mCi)	Use/Location
^{241}AmBe	2-81-020	Solid	15,200	Calibration (EOF)
^{241}AmBe	2-82-050	Solid	2930	Neutron source (plant)
^{241}AmBe	2-86-073	Solid	1,020	WNP-1 source (plant)
^{137}Cs	2-79-012	Solid	186	Eberline calibration (EOF)
^{137}Cs	2-79-013	Solid	4,870	Eberline calibration (EOF)
^{137}Cs	2-79-016	Solid	1,970	Eberline calibration (EOF)
^{137}Cs	2-79-017	Solid	101,000	Eberline calibration (EOF)
^{137}Cs	2-79-033	Solid	790	Panoramic shepard cal (EOF)
^{137}Cs	2-83-097	Solid	7,050	ARM calibration (plant)
^{137}Cs	2-84-058	Solid	2,220	Shepard series 28 cal (EOF)
^{137}Cs	2-88-002	Solid	200	Victoreen model 878-W cal (plant)
^{85}Kr	2-84-019	Gas	200	Gas source-chemistry (plant)
^{238}PuBe	2-84-047	Solid	12,800	WNP-3 startup source (plant)
^{238}PuBe	2-84-048	Solid	12,900	WNP-3 startup source (plant)
^{55}Fe	2-94-020	Liquid	186	Chemistry calibration source (plant)

Table as of 11/6/97.

Table 12.2-13

List of Radioactive Piping and System Designations

Air removal (AR)

Bleed steam (BS)

Condensate filter/demineralizer (CPR)

Condenser vents and drains (CND)

Control rod drive (CRD)

Equipment drains radioactive (EDR)

Exhaust steam (ES)

Floor drains radioactive (FDR)

Fuel pool cooling (FPC)

Heater drains (HD)

Heater vents (HV)

High pressure core spray (HPCS)

Low pressure core spray (LPCS)

Main condensate before condensate demineralizers (COND)

Main steam (MS)

Main steam isolation valve leakage control system (MSLC)

Miscellaneous waste radioactive (MWR)

Offgas (OG)

Process sample radioactive (PSR)

Process vents (PVR)

Process waste radioactive (PWR)

Reactor core isolation cooling (RCIC)

Reactor recirculation (RRC)

Reactor water cleanup (RWCU)

Relief valve vents radioactive (VR)

Residual heat removal (RHR)

Table 12.2-14

Airborne Radionuclide Concentration in Control
Rod Drive Pump Area (el. 422 ft. 3 in. reactor building)

Radionuclide	Airborne Concentration C_i ($\mu Ci/cm^3$)	Derived Air Concentration (DAC)[a] (mCi/cm^3)	Ratio of C_i to DAC
^{131}I	1.9E-11	2E-8	1E-3
^{132}I	2.5E-12	3E-6	1E-6
^{133}I	2.2E-11	1E-7	2E-4
^{134}I	1.9E-12	2E-5	2E-7
^{135}I	8.7E-12	7E-7	1E-5
^{83}Br	3.3E-13	3E-5	1E-8
^{84}Br	6.3E-14	2E-5	3E-9
^{85}Br	1.3E-16	---	---

[a] 10 CFR 20, Appendix B to 20.1001-20.2401, Table I, Column 3.

Table 12.2-15

Airborne Radionuclide Concentration in
Condensate Pump Area (el. 441 ft. 0 in. turbine generator building)

Radionuclide	Airborne Concentration C_i (μCi/cm^3)	Derived Air Concentration (DAC)[a] (mCi/cm^3)	Ratio of C_i to DAC
^{131}I	4.2E-10	2E-8	2E-2
^{132}I	3.8E-9	2E-6	3E-3
^{133}I	2.9E-9	1E-7	2E-2
^{134}I	7.4E-9	2E-5	4E-4
^{135}I	4.2E-9	7E-7	6E-3
^{83}Br	4.8E-10	3E-5	2E-5
^{84}Br	8.2E-10	2E-5	4E-5
^{85}Br	3.2E-10	1E-7	3E-3

[a] 10 CFR 20, Appendix B to 20.1001-20.2401, Table I, Column 3.

Table 12.2-16

Airborne Radionuclide Concentration in Secondary Containment from a Main Steam Relief Valve Blowdown

Radionuclide	Airborne Concentration C_i (μCi/cm^3)	Derived Air Concentration (DAC)[a] (mCi/cm^3)	Ratio of C_i to DAC
^{131}I	3.0E-11	2E-8	2E-3
^{133}Xe	5.0E-10	1E-4	5E-6

[a] 10 CFR 20, Appendix B to 20.1001-20.2401, Table I, Column 3.

Table 12.2-17

Airborne Radionuclide Concentration in
Liquid Radwaste Handling Area

Radionuclide	Airborne Concentration C_i ($\mu Ci/cm^3$)	Derived Air Concentration (DAC)[a] (mCi/cm^3)	Ratio of C_i to DAC
^{140}Ba	5.8E-10	6E-7	1E-3
^{140}La	6.5E-10	6E-7	1E-3
^{239}Np	2.2E-10	9E-7	2E-3
^{58}Co	9.8E-10	3E-7	3E-3
^{89}Sr	4.8E-10	6E-8	1E-2
^{99}Mo	2.6E-10	6E-7	4E-4
^{99M}Tc	1.7E-10	6E-5	3E-6
^{132}Te	1.5E-10	9E-8	2E-3
^{131}I	9.2E-10	2E-8	4E-2
^{132}I	2.4E-10	3E-6	1E-4
^{133}I	4.1E-10	1E-7	4E-3
^{135}I	1.8E-10	7E-7	2E-4

[a] 10 CFR 20.

12.3 RADIATION PROTECTION DESIGN FEATURES

12.3.1 FACILITY DESIGN FEATURES

Columbia Generating Station plant incorporates the design objectives and the design features guidance given in Regulatory Guide 8.8 to the extent discussed in Section 12.1.1. Examples of these features are discussed in Section 12.3.1.3.

Figures 12.3-1 through 12.3-18 show the general arrangement for each of the plant buildings. In addition, these figures show the shielding arrangement, radiation zone designations for both normal operation and shutdown conditions, controlled access areas, personnel and equipment decontamination areas, location of the health physics facilities, location of area radiation monitors, location of radwaste control panels, location of the counting room, and location of the onsite laboratory for analysis of chemical and radiochemical samples. The counting room is located at el. 487 ft 0 in. of the radwaste and control building (see Figure 12.3-13). The design basis radiation level within the counting room is 0.1 mrem/hr during normal operation.

Plant areas, as identified in Section 12.3 figures, are categorized as design radiation zones according to expected maximum radiation levels and anticipated personnel occupancy with consideration given toward maintaining personnel exposures ALARA and within the standards of 10 CFR 20.

12.3.1.1 Radiation Zone Designations

The design basis criteria used for each zone are given below, and the plant layout including major equipment, locations, and radiation zone designations are shown in Figures 12.3-5 through 12.3-18.

For purposes of radiation exposure control, a restricted area is defined in 10 CFR 20, paragraph 20.1003, and plant procedures.

Zone	Maximum Dose Rate (mrem/hr)	Design Bases Criteria
I	≤1.0	Unlimited occupancy.
II	≤2.5	Unlimited occupancy for plant personnel during the normal work week.
III	100.0	Design base occupancy less than 1 hr per week. Posted zones and controlled entries.
IV	Unlimited	Positive access control. Controlled entry and occupancy.

Each access point to every Zone IV area may be secured by locked door or other positive control method while it is a "high radiation area." Occupancy of such areas is limited in frequency and duration and entrance must be authorized in advance by the Radiological Operations Supervisor or his representative.

An area survey of radiation levels will be conducted prior to first entry of Zone IV areas to determine the maximum habitation time.

12.3.1.2 Traffic Patterns

Access to plant areas is shown in Figures 12.3-1 through 12.3-4. Access control and traffic patterns in the plant have been evaluated to maintain personnel radiation exposures ALARA and to minimize the spread of contamination.

Normal entry into the plant is as follows:

a. Personnel normally access the plant Protected Area through the security Protected Area Access Point (PAAP).

b. The main Radiologically Controlled Area (RCA) normally includes the reactor building, turbine generator building, radwaste building, and diesel generator building. Normal access to these areas is through one of two Health Physics control points located at each end of the main plant corridor.

12.3.1.3 Radiation Protection Design Features

Section 12.1.2 discusses the design objectives for the plant. Examples showing the incorporation of these objectives are provided in the following sections.

12.3.1.3.1 Facility Design Features

Filters and Demineralizers

Liquid radioactive waste and other process streams containing radioactive contaminants are processed through filters and demineralizers. The pressure-precoat type of filter is used in the major fluid processing systems. Cartridge type filters are used in a few select instances such as in the control rod drive (CRD) system. In the case of demineralizers, either the pressure-precoat or deep-bed type of demineralizer is employed.

Each filter and demineralizer is located in a shielded cubicle at el. 487 ft 0 in. of the radwaste and control building. These filters and demineralizers are accessible through the ceiling of the cubicle by removing the shielding plug at el. 507 ft 0 in. This minimizes the radiation exposure to plant personnel from adjacent sources. After removal of the shielding plug, the filter or demineralizer can be serviced remotely by use of special tools designed for this purpose. If it is necessary, the filter or demineralizer can be removed from the cubicle to a work area for servicing. There are overhead cranes provided for the purpose of shielding plug and filter or demineralizer vessel removal.

Each pressure precoat type filter or demineralizer has its own support equipment such as a holding pump, process control valves, and precoating equipment. The holding pump and process control valves associated with each filter or demineralizer is located in the valve gallery. This valve gallery is located on the east side of the radwaste-control building, el. 467 ft 0 in., and is a Zone III radiation area (Figure 12.3-12).

The holding pump and motor-operated valves can be operated from control panels located in Zone III radiation areas. Manually operated process control valves are operated by use of reach rods that pass through the shielding walls into the corridor. This corridor is a Zone III radiation area. With the exception of instrument root valves, all pumps and valves can be remotely operated. Instrument root valves are normally open and are not used during normal plant operation. The filter or demineralizer precoat equipment and associated controls, which includes metering equipment, are located in a Zone III radiation area to provide for maintenance access. Each filter or demineralizer may be backwashed with condensate water, chemically cleaned, or purged with air from a remote control panel. Each deep-bed demineralizer has its own support equipment. A gravity feed subsystem transfers the fresh resins from the resin addition tank to the demineralizers. The spent resins are transferred hydropneumatically to the spent resin tank.

All piping routed to and from filter or demineralizer cubicles is located in the valve gallery area or in shielded pipe tunnels. The pipes are routed to avoid unnecessary bends, thus minimizing possible crud buildup.

Specific examples of filters or demineralizers that incorporate the aforementioned design features are the waste collector filter and waste collector demineralizer. A typical layout is shown in Figure 12.3-19.

<u>Tanks</u>

All tanks that contain radioactive liquids and solids are categorized into different groups depending on their level of radioactivity. Tanks that contain significant levels of radioactivity are located at el. 437 ft-0 in. of the radwaste and control building.

The tanks that contain radioactive backwash and resins are located in shielded cubicles. These include the condensate phase separator tanks-condensate backwash receiving tank, spent resin tanks waste sludge phase separator tanks, and the reactor water clean up (RWCU) phase separator tanks. These tanks are constructed of either stainless steel or epoxy-lined carbon steel.

The tanks that contain unprocessed liquid radioactive waste, such as the waste collector, floor drain collector, and chemical waste tanks are not located in individually shielded cubicles. However, as described in Section 12.1.1, administrative controls will be used to keep doses to plant personnel ALARA. The waste collector and the floor drain collector tanks are constructed of carbon steel. The chemical waste tanks are stainless steel.

To measure the liquid level in the aforementioned tanks, a bubbler-type level indicator, coupled with a pressure transducer, provides for remote monitoring of the liquid level.

All tanks described above are vented to the radwaste building heating, ventilating, and air conditioning (HVAC) exhaust system as described in Section 12.2.2. This limits the release of airborne radionuclides to the area surrounding the tank.

Pumps

Pumps handling spent demineralizer resins are shielded from the phase separator tank. An example of this is the waste sludge discharge pump which is shielded from the waste sludge phase separator by concrete and block wall. A shutoff valve in the suction line keeps the pump isolated when it is not in use. Following the discharge of a sludge batch from the phase separator tanks, the pump and its associated piping is automatically flushed with condensate water. Thus, when it is not in use, the pump is free of sludge. A condensate supply, controlled by a valve electrically linked to the pump, maintains an external sealing barrier, preventing sludge leakage past the shaft seal during pump operation.

Heat Exchangers

Heat exchangers handling radioactive fluids are designed to limit occupational exposures. An example is the cooler condensers whose function is to condense moisture from the offgas process stream. The cooler condensers are located in a separate cell in the radwaste building, as shown in Figure 12.3-11. No personnel access is required during plant operation. The associated valves, other than those used during maintenance operations, are remotely operated. The glycol coolant which flows through the tube side is kept at a higher pressure than the offgas flowing through the shell side of the condenser. Thus, a tube failure will not result in radioactivity contaminating the glycol cooling loop. Condensate water that is separated from the offgas is returned, via a water loop seal, to the radioactive sump for processing. The 16-ft deep loop seal prevents escape of radioactive gas through the drain connection. An enlarged discharge section in the loop seal protects it against siphoning. The enlarged discharge section also provides for automatic loop seal restoration should its contents be displaced by a temporary pressure surge. Figure 12.3-20 shows schematically the cooler condenser loop seal arrangement.

Recirculation Pumps

The decontamination concentrator bottoms recirculation pump serves the decontamination solution concentrator. The pump is made of 316 stainless steel to minimize the corrosive effects of miscellaneous chemical waste fluid. The pump is located in a separate cell from the concentrating equipment it serves. A steam/air sparger at the bottom of the concentrator can be used to flush the pump prior to any maintenance operations. The internals of the pump may be removed without disturbing the connecting piping. A double mechanical seal with clean water circulating through it prevents leakage of process liquid past the shaft seal.

The decontamination concentrator bottoms recirculation pump is not used. There are no plans to use the pump.

Evaporators

The decontamination solution concentrators use steam to boil off water from the incoming chemical waste, as described in Section 11.2. The heating steam is supplied from an evaporator. As shown in Figure 12.3-21, steam generated from demineralized water flows in a closed loop through the shell side of the evaporator and the shell side of the concentrator heating element. The steam is then circulated to the condensate return tank to be pumped back to the evaporator. The steam in the concentrator heating element is at a higher pressure than the decontamination solution flowing through the tube side. Furthermore, the auxiliary steam system, which provides heating steam to the tube side of the evaporator, is at a higher pressure than the shell side. This arrangement provides an effective barrier against contamination of the house auxiliary steam.

The decontamination solution evaporator system is deactivated. There are no plans to use the system.

Valve Gallery and Valve Operating Stations

Valves handling radioactive fluids and requiring manual operation are located in a valve gallery at el. 467 ft 0 in. of the radwaste and control building. These valves are operated from behind a shielding wall with the aid of reach rods. Reach rod penetrations are not in the line-of-sight with major radiation sources, such as resin traps. In addition, the reach rod wall penetrations are grouted about the reach rod assembly, and steel plates are added on both sides of the penetration to minimize radiation exposure. A typical application of reach rods and details of a shielding wall penetration is shown in Figure 12.3-19.

The operating stations for motor-operated valves are located in Zone III radiation areas.

Sampling Areas

The location of the sampling areas within the plant is discussed in Section 9.3. Design features of sample areas that reduce occupational exposure are discussed in Section 12.2.2.3.5.

Ventilation Filters and Filter Trains

Filters that are installed as part of the HVAC units in the Columbia Generating Station plant are located in an accessible area. Selected filter units are designed so that it is possible to introduce a plastic bag over the filter through a service access opening in the HVAC unit. For other filter units, the filter can be pulled into a plastic bag as the filter is removed from the HVAC unit.

Hydrogen Recombiners

The hydrogen recombiners for the offgas system are located in the turbine-generator building. These recombiners are single-pass devices which do not require process control valves. They are located in a shielded cell and do not require personnel access during operation. Temperature and pressure in the recombiners are remotely monitored. The recombiners and associated piping are designed to withstand an internal explosion.

12.3.1.3.2 Design Features That Reduce Crud Buildup

Design features and considerations are included to reduce radioactive nickel and cobalt production and buildup. For example, the primary coolant system consists mainly of austenitic stainless steel, carbon steel, and low alloy steel components. Nickel content of these materials is low. Nickel and cobalt contents are controlled in accordance with applicable ASME material specifications. A small amount of nickel base material (Inconel 600) is employed in the reactor vessel internal components. Inconel 600 is required where components are attached to the reactor vessel shell, and coefficient of expansion must match the thermal expansion characteristics of the low-alloy vessel steel. Inconel 600 was selected because it provides the proper thermal expansion characteristics, adequate corrosion resistance and can be readily fabricated and welded. Alternate low nickel materials which meet the above requirements and are suitable for long term reactor service are not available. Hardfacing and wear materials having a high percentage of cobalt are restricted to applications where no alternate materials of equally good characteristics are available.

To minimize crud buildup, lines carrying slurries in the reactor water cleanup system and in the radwaste system use extensively self-flushing valves. Furthermore, all of the pipe joints of 2.5 in. and above in size in the reactor water cleanup (RWCU) and radwaste systems are butt welded. Butt welding is also used in the equipment drain processing (EDR) and floor drain processing (FDR) systems for lines of 1 in. and above, except for two 1-in. flow control valves, one 1.5-in. flow control valve, two 2-in. socket welded ball valves, one 1-in. socket welded ball valve, and four 3-in. flanged butterfly valves. Lines of 2.5-in. and larger in the offgas system are also butt welded.

The recirculation system is equipped with decontamination flanges for decontamination of the recirculation pump and associated hardware. Also the cleanup, emergency core cooling system (ECCS), and radwaste systems are provided with decontamination connections to enable decontamination of their hardware. In addition, the RWCU and condensate process filter demineralizers reduce the amount of activation products in these systems. Boiling water reactors (BWRs) do not use high temperature filtration.

Depleted zinc is injected into the feedwater system to displace cobalt from the corrosion product layer in the recirculation system. Iron is injected into the feedwater system to cause the displaced cobalt to adhere to the surface of the fuel rods. This has caused a reduction of exposure rates from the recirculation system.

12.3.1.3.3 Field Routing of Piping

All code Group A piping is dimensioned in detail. Other piping, 2 in. in diameter and smaller, carrying radioactive fluids, are not dimensioned in detail but are required to be in specified space envelops for shielding and in-plant exposure considerations. Such piping runs have their inception points and terminal points dimensionally located. These inception or terminal points may be an outlet or inlet of a pump, a junction with a larger pipe, or a tank nozzle. Also located dimensionally are all pipe penetrations through a wall, ceiling, or floor. Radioactive piping routed through lower radiation zones is enclosed within a shielded tunnel when warranted by high expected radiation levels. Radioactive piping 2.5-in. in diameter and larger is detailed dimensionally.

12.3.1.3.4 Design Features That Reduce Occupational Doses During Decommissioning

Many of the design facilities which presently exist in the plant can be used to minimize occupational exposure during decommissioning, whether decommissioning is accomplished through mothballing, entombment, removal/dismantling, or any combination of the above alternatives. Such facilities include those used for handling and for offsite shipment of fresh fuel, spent fuel, more sources, contaminated filter elements, resins, and other radioactive wastes. The radioactively contaminated spent fuel pool water can be removed by a portable pump placed into the pool after all spent fuel has been removed from the site and any decommissioning use of the pool is finished. The portable pump will discharge into the skimmer surge tank from where the existing decontamination procedure can be implemented.

The number of man rems due to the airborne radioactivity, that may be introduced by the handling of radioactively contaminated systems, as well as the number of man rems due to contact with the same systems, can be reduced by first decontaminating them. Means exist in the present design where radioactively contaminated systems can be decontaminated chemically by remote control and flushed. The plant has a hot machine shop and a hot instrument shop located in the radwaste building where contaminated equipment can be decontaminated under controlled conditions. Provisions have been made in the same building for a future decontamination facility with expanded features.

If decommissioning is accomplished by mothballing, the above provisions will reduce to low levels the occupational radiation exposure to personnel. According to Regulatory Guide 1.86, this method involves "putting the facility in a state of protective storage." In general, the facility may be left intact except that all fuel assemblies and the radioactive fluids and waste should be removed from the site.

If entombment is chosen as the method of decommissioning, the previously described plant design facilities are adequate to accomplish the tasks with low occupational radiation exposure to personnel. The additional requirements described in Regulatory Guide 1.86 for "sealing all the remaining highly radioactive or contaminated components (e.g., the pressure vessel and reactor internals) within a structure integral with the biological shield after having all fuel assemblies, radioactive fluids and wastes, and certain selected components shipped offsite" can be met. The necessary modifications to the Columbia Generating Station primary containment structure are shown in Figure 12.3-22.

Low occupational radiation exposure to personnel can be achieved if the decommissioning method adopted is that of immediate removal/dismantling of the plant. For example, the laydown area, el. 606 ft 10.5 in., in the reactor building is provided with a drainage system. There the drywell head and the reactor pressure vessel head can be decontaminated if it is required. Furthermore, there is ample space to erect a contamination control envelope around them during their segmentation for transportation and burial. The plant design accommodates the cutting of the reactor internal under water and within contamination control envelopes. The turbidity of the water can be reduced by using the RWCU system. The only modification that might be required to the existing system could be the installation of a strainer for the removal of large filings or other large size contaminants. The highly radioactive pieces can be transferred under water to the cask loading area in the spent fuel pool by methods similar to loading spent fuel. The airborne radioactivity generated by the dismantling process can be removed by the filtration systems that the contaminated control envelopes can have and/or by the standby gas treatment system (SGTS).

12.3.1.4 Radioactive Material Safety

12.3.1.4.1 Materials Safety Program

Columbia Generating Station has a program to ensure the safe storage, handling, and use of sealed and unsealed special nuclear source and byproduct materials. Included in the program are procedures for the following:

a. Receiving and opening shipments as required by 10 CFR 20.1906,

b. Storage of licensed materials as required by 10 CFR 20.1801 and 20.1802,

c. Inventory and control of radioactive materials,

d. Posting of radioactive material storage areas and tagging of source,

e. Leak tests - sources are checked for leakage or loss of material at least semiannually, and

f. Disposal - all licensed material disposals are in accordance with 10 CFR Part 20 requirements or by transfer to an authorized recipient as provided in 10 CFR Parts 30, 40, or 70.

12.3.1.4.2 Facilities and Equipment

Facilities are provided for handling unsealed sources, such as the liquid standard solutions used for calibration of plant instrumentation. The radiochemical laboratory is equipped with a negative pressure fume hood with filtered exhaust. The hood work surface is designed to withstand heavy weights so that shielding can be provided in the form of lead brick. Drains from the fume hood are routed to the liquid radwaste system.

Remote handling tools are used as needed for movement of the high level sealed sources from their normal storage containers. Shielding to reduce personnel exposure is provided for these sources when they are not in use and to the extent practicable while they are in use.

Portable radiation and contamination monitoring instrumentation is provided, as described in Section 12.5.2, for surveillance to maintain control of the sources.

12.3.1.4.3 Personnel and Procedures

The Columbia Generating Station Assistant Radiological Services Manager/Radiation Protection Manager (RPM) is responsible for the control and monitoring of sealed and unsealed source and byproduct materials. The Nuclear Material Manager appointed by the Engineering Manager is accountable for special nuclear materials (SNM). The Chemistry Technical Supervisor is responsible for the minimization of radioactive waste and the preparation, offsite shipment, and disposal of radioactive materials and radwaste. Monitoring during handling of these materials is provided by Radiation Protection. Experience and qualifications of Radiation Protection personnel are described in Section 13.1.

Health Physics requirements and instructions to personnel involved in handling byproduct materials are included in implementing procedures.

12.3.1.4.4 Required Materials

Any byproduct, source, or special nuclear materials in the form of reactor fuel, sealed neutron sources for reactor startup, sealed sources for reactor instrument and radiation monitoring equipment calibration, or as fission detectors, will be limited to the amounts required for reactor operation or specific calibration purposes except as noted in the facility operating license.

12.3.2 SHIELDING

12.3.2.1 General

The radiation shielding design is in compliance with all NRC regulations concerning permissible radiation doses to individuals in restricted and nonrestricted areas. The guidance provided in Regulatory Guide 1.69 on concrete radiation shields is followed to the extent discussed in Section 1.8. The guidance provided in Regulatory Guide 8.8 on radiation protection is followed to the extent discussed in Section 12.1.2. The plant design and layout optimizes personnel and/or equipment protection. Shielding is supplemented with whatever administrative control procedures are necessary to ensure regulatory compliance. These control procedures include area access restrictions, occupancy limitations, personnel monitoring requirements, and radiation survey practices. Other criteria and considerations are listed in Section 12.1.2.

The shielding design is evaluated under the following conditions of plant operation:

a. Operation at design power, including anticipated operational occurrences,

b. Shutdown conditions, with radiation from the subcritical reactor core, spent fuel assemblies, and other sources discussed in Section 12.2, and

c. Postaccident conditions, including those accident occurrences analyzed in Chapter 15. Emphasis is placed on control room habitability.

The majority of the shielding calculations performed are of the "bulk shielding" type. Ordinary concrete, having a density of about 150 lb/ft^3, is used for shielding except for special applications. In special applications, water, steel, high density concrete, lead, and permali JN P/3% boron are used.

The effects of mechanical or electrical penetrations in shield walls on radiation exposure to personnel is minimized by locating penetrations to preclude direct view of radiation sources through the penetration. The effect of penetrations in shield walls is also minimized by keeping penetration openings to the smallest practicable size. Penetrations are located away from immediate areas with personnel access. When these criteria cannot be implemented, penetrations are offset.

Access into shielded areas is, in general, by labyrinths. Labyrinths are located to preclude direct personnel radiation exposure. Where labyrinths are not practicable, shield doors are used. Knock-out walls for equipment removal are constructed of brick arranged in staggered rows to preclude direct streaming.

Portable and removable shielding devices are used when practical and feasible. Portable shielding devices are easily moved from one location to another. Removable shielding devices are normally used at specific locations and can be removed when necessary. The reactor vessel to fuel pool transfer passage is a location where removable shielding is employed primarily for the protection of personnel working in the drywell. Personnel evaluation of the affected drywell area may be employed instead of, or in conjunction with, the above mentioned shielding.

12.3.2.2 Methods of Shielding Calculations

Standard methods are used in computing the required shielding thickness for a given source. These methods are described in References 12.3-1 through 12.3-4. Specific methods of calculation and the computer codes used in the shielding design are discussed below.

The NRN computer code (Reference 12.3-5) is used to determine the shielding requirements for the core generated neutrons and to calculate the thermal neutron flux used to determine captured gamma sources outside the core. This code is based on a multigroup slowing down and diffusion system corrected by a multigroup first flight or removal neutron source. The neutron cross sections used with this code are from Oak Ridge National Laboratory with modifications which have resulted from comparisons with data in BNL-325 (Reference 12.3-6) and the ENDF/B data libraries.

The QAD-BR computer code, which is based on the QAD-P5 code (Reference 12.3-7), is the basic code used to determine shielding requirements for gamma ray sources. This code provides gamma flux, dose rate, energy deposition, and other quantities which result from a point by point representation of a volume distributed source of radiation. Attenuation coefficients for water, iron, and lead, used in this program are taken from the Engineering Compendium for Radiation Shielding (Reference 12.3-1). Concrete attenuation coefficients are taken from ANL-6443 (Reference 12.3-8).

Any shielding requirements which are not determined with the QAD code were determined by using the methods discussed in the Reactor Shielding Design Manual (Reference 12.3-2). The various sources are reduced to their basic geometric configuration (line, disc, cylinder, sphere, etc.) and the corresponding equations are solved to find the dose. The Taylor exponential form of the buildup factor is used in these equations. All required data is taken from Reference 12.3-1.

The criteria for penetration acceptability is based on the radiation zone levels in the areas separated by the wall where the penetration is located. The penetration is analyzed if the penetration passes from Radiation Zone IV to Zones III, II, or I; or, if the penetration passes from Radiation Zone III to Zones II or I. This analysis considers both the scattered as well as the direct radiation. The direct component is calculated using the point source attenuation equation, as described in the Reactor Shielding Design Manual (Reference 12.3-2). The scattered component is calculated using the Chilton-Huddleston equations (Reference 12.3-9). Compensatory shielding (e.g., labyrinths, steel plate, lead wool) is used as needed, to reduce radiation streaming through penetrations and to protect against localized "hot spots."

The general dose rate in each plant area, including contributions from radiation streaming, satisfies the design dose rate specified for that area.

Entrances to shielding cubicles are designed to prevent source radiation from passing directly through the opening. Whenever possible, a labyrinthine entrance is designed to reduce the emerging radiation to a level compatible with the access requirements outside the cubicle. The scattered and direct components of the emerging radiation are calculated by the above methods.

12.3.2.3 Shielding Description

12.3.2.3.1 General

The description of the shielding throughout the entire plant is summarized within the following sections. These descriptions are to be used in conjunction with the radiation zone maps, Figures 12.3-5 through 12.3-18, to locate the process equipment which is shielded and to determine the design dose rate.

12.3.2.3.2 Reactor Building

The sacrificial shield is an ordinary concrete structure 2 ft thick, lined on the inside and outside by steel plates of a minimum thickness of 0.5 in. each. The sacrificial wall extends between el. 519 ft 2.25 in. and 567 ft 4.5 in.

The biological shield wall protects station personnel in the reactor building from radiation emanating from the reactor vessel. The dose rate at the outer face of the biological shield as well as above the shield plug (above the reactor vessel) is, except at penetrations, less than 2.5 mrem/hr during normal reactor operation. The reactor core is the primary source of radiation, and it is used in computing the above dose rate. The wall is in the shape of a shell of the frustum of a cone, and its composition is ordinary concrete at least 5 ft thick. Inside the biological wall exists the primary containment vessel which has the same shape as the wall.

The primary containment vessel is made of 0.75 in. minimum steel plate. The ^{16}N contained in the recirculation system, the main steam lines, and the water in the vessel below the core along with the fission process in the core constitute the major sources of radiation used to determine the radial dose rate. The shielding arrangement for the other major sources in the reactor building is shown on Figures 12.3-15 through 12.3-18.

Personnel evacuation of the affected drywell area(s) and/or employing removable shielding at the fuel pool passage are two methods used for personnel protection in the drywell during fuel handling operations. The shielding is designed such that radiation levels are no greater than 100 R/hr at contact. Portable locally alarming radiation monitors and/or direct Health Physics monitoring are employed for additional personnel protection.

12.3.2.3.3 Turbine Building

In the turbine building, ^{16}N constitutes the major source of radiation and basis for shielding design. It is contained in the turbines, moisture separator reheaters, and the feedwater heater system that are located in the turbine access areas at el. 501 ft 0 in. and 471 ft 0 in. These areas are surrounded by ordinary concrete walls at least 3.5 ft thick. The dose rate to the areas outside these walls is less than 2.5 mrem/hr.

The walls which surround the turbine-generator access area at el. 501 ft 0 in. extend up to 524 ft 0 in. This minimizes the effects of direct radiation streaming at the site boundary.

The shielding arrangement for the major sources of radiation in the turbine building, including those discussed above, are shown in Figures 12.3-5 through 12.3-10.

12.3.2.3.4 Radwaste Building

The shielding arrangement for the major sources in the radwaste building are shown in Figures 12.3-11 through 12.3-14.

12.3.3 VENTILATION

The plant ventilation systems for the different areas of the plant are designed to meet the requirements of 10 CFR Parts 20 and 50. Gaseous wastes will be released in a controlled manner to environs such that during plant operation, onsite and offsite radioactivity levels are ALARA. The design features which limit and reduce the airborne radioactivity are as follows:

a. In the reactor, radwaste, and turbine generator buildings the air flow is from areas of low airborne radioactivity potential to areas of higher airborne radioactivity potential. This serves to isolate and segregate airborne radioactivity which may be released due to equipment failure or malfunction and leakage from fluid systems;

b. To prevent radioactivity buildup, all ventilation air is supplied to the reactor, turbine, and radwaste buildings on a once through basis;

c. All cubicles housing equipment which handles radioactivity contaminated material are ventilated at a minimum rate of three volume air changes per hour;

d. All sinks and chemical laboratory work areas where radioactive samples or materials are handled are provided with exhaust hoods to protect operating personnel from airborne contaminants;

e. All liquid equipment leaks which are potential sources of airborne radioactivity in the reactor building are collected in the reactor building equipment drain system. The drain system is maintained at a negative pressure with respect to the balance of the building to protect operating personnel from airborne leakage generated in the system sumps. All exhaust air drawn from the reactor building equipment drain system is filtered by absolute particulate and charcoal filters. The particulate and charcoal filters minimize the release of contaminated particulates and iodine; and

f. The primary containment purge system reduces airborne radioactivity within the drywell to acceptable levels prior to entry of working personnel. The level of radioactivity released to the environment during normal primary containment purge will not exceed the requirements of 10 CFR Part 20 Appendix B. When airborne radiation levels in the primary containment are too high to allow direct purging to the atmosphere through the reactor building exhaust, purge air at a reduced flow rate is passed through the SGTS prior to exhaust. In this latter mode, airborne iodine and particulates are removed from the purge exhaust air prior to release;

The air cleaning systems which utilize special filtration equipment to limit airborne radioactive contaminants are

a. Standby gas treatment system (see Section 6.5),
b. Control room emergency filtration system (see Sections 9.4 and 6.4),
c. Reactor building sump vent exhaust filter system (see Section 9.4), and
d. Radwaste building exhaust filtration system (see Section 9.4).

In addition, small local absolute particulate filters are used to locally filter the effluent from sample sink hoods and chemical hoods. These small filter units are all described in Section 9.4.

The SGTS filter units discussed in Section 6.5.1 and the control room emergency filter units discussed in Sections 6.4 and 9.4.1 are the only engineered safety feature (ESF) filtration systems. A detailed evaluation of these units with respect to Regulatory Guide 1.52 is given in Section 1.8. The balance of the filtration systems meet the intent of Regulatory Guide 1.52 with the following deviations:

a. Reactor building sump vent filter units

   These units are composed of demisters, an electric heater, a medium efficiency prefilter, an absolute particulate (HEPA) filter and tray type adsorber filters composed of 2-in. deep charcoal beds in a sheet metal housing. These units do not have absolute particulate filters downstream of the adsorber section as described in Regulatory Guide 1.52.

   The 5-ft spacing between filter frames, as required in Regulatory Guide 1.52, is not provided in these units. These units are of low capacity (1000 cfm), and personnel access into the units for servicing and testing is not possible. Access panels are provided for servicing and removal of all unit components. Sufficient space is provided between elements to permit removal of any element without disturbing any other element.

b. Radwaste building exhaust filter units

   These three units are composed of medium efficiency prefilters, absolute particulate (HEPA) filters and centrifugal fans in a sheet metal housing. Unit capacity is 42,000 cfm, which is in excess of the 30,000 cfm maximum capacity recommended in Regulatory Guide 1.52. These units are composed of a 5 filter high by 8 filter wide array. Permanent service platforms are provided on both sides of the HEPA filters at an intermediate height for operating personnel during filter testing and service.

   Absolute particulate (HEPA) filter and charcoal adsorber filters will be tested periodically to ensure continued filter efficiency as discussed in Sections 6.5.1.4, 9.4.2.4, and 9.4.3.4.

Figure 12.3-23 (el. 572 ft 0 in.) shows the layout of the SGTS filter units. The concrete wall between the units serves as both a fire wall and missile barrier between the two filter trains. Access doors, 20 in. x 50 in., are provided into each plenum section between unit elements. Ample aisle space is provided outside the units for ease of access for personnel to perform tests and maintenance. Charcoal test canisters are provided as shown in Figure 12.3-23. There are

12 test canisters per 4-in. deep filter bed. Dioctylphthalate (DOP) and freon injection and detection ports are provided as shown.

12.3.4 IN-PLANT AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING INSTRUMENTATION

12.3.4.1 Criteria for Necessity and Location

The objectives of the in-plant area radiation and airborne radioactivity monitoring systems are to

a. Warn of excessive gamma radiation levels in areas where nuclear fuel is stored or handled,

b. Provide operating personnel with a record and indication in the main control room of gamma radiation levels at selected locations within the various plant buildings,

c. Provide information to the main control room so that decision may be made with respect to deployment of personnel in the event of a radiation accident or equipment failure,

d. Assist in the detection of unauthorized or inadvertent movement of radioactive material within the various plant buildings,

e. Provide local alarms at selected locations where a substantial change in radiation levels might be of immediate importance to personnel frequenting the area,

f. Monitor areas having a high potential for increase gamma radiation levels where personnel may be required to work,

g. Supplement other systems including process radiation leak detection or building release detection in detecting abnormal migrations of radioactive materials from process streams,

h. Monitor the general conditions in the reactor building following an accident, and

i. Furnish information for making radiation surveys.

No credit is taken for the operability of the in-plant area radiation and airborne radioactivity monitors in the event of an accident. However, the probability is high that many or all of the 13 area monitors in the reactor building will be operable. These monitors have local sensors in separate physical locations within the reactor building. The wiring from the local sensors is run to the main control room in cable runs that have Seismic Category I qualified supports. The electrical supply for the area monitor indicator and control room alarms in the main control room are powered from a critical 120-V ac supply. The local audible alarms are powered from a local ac supply and its loss would not impact control room readings. The sensors are designed to maintain operability and accuracy in atmosphere up to 95% RH, and temperature to 50°C.

12.3.4.2 Description and Location

a. Area radiation monitors

Area radiation monitors consist of local detector alarm units and main control room mounted indicator trip units, alarms, and recorders. Redundant criticality monitors are located in the reactor building new fuel storage pit as recommended by Regulatory Guide 8.12. When practicable and feasible the guidance in Regulatory Guide 8.12 has been followed. Major items in Regulatory Guide 8.12 have been addressed and include

1. Employing two detectors in the new fuel vault,

2. Emergency planning that includes both the Emergency Plan and the Emergency Plan Implementing Procedures, evacuation routes, assembly areas, and yearly drills, and

3. Surveillance testing of criticality alarm systems, including procedures that address methods and frequency of testing.

The criticality monitor alarm setpoints are determined to meet 10 CFR 70.24(a)(1) using the calculational methodology of ANSI/ANS 8.3-1979, Appendix B. However, the sensitivity design guideline of 1 msec for response and alarm discussed in Section 5.3 of ANSI/ANS 8.3-1979 involves highly enriched material. Thus, the 1 msec guideline is not practicable nor applicable at Columbia Generating Station.

Other detector locations have been selected in accordance with good operating practice and from past operating experience with similar plants. Detector locations are shown in Figures 12.3-5 through 12.3-18. Annunciations are given in the main control room and locally at the sensors when radiation levels exceed a predetermined level. Point indication and recording are provided for in the main control room. Local detectors are wall-mounted approximately 7 ft off the floor. The detectors have sufficient cable length to be taken from their normal positions to floor level for insertion into calibrating chambers to verify instrument accuracy. Direct-reading dosimeters, worn by individuals in radiologically controlled areas, provide protection in areas where area radiation monitors have not been installed.

An additional area radiation monitor is installed on the refueling bridge during bridge operation to provide personnel protection. This monitoring system provides local indication, visual, and audible alarm.

There is no area radiation monitor in the area of the radwaste building where the waste containers are filled and stored. Waste containers will normally be processed either "in cask" or in the shielded waste storage bay.

The location and ranges of the 31 area radiation monitors are given in Table 12.3-1. Table 12.3-2 lists the maximum background radiation levels for the area radiation monitors in the reactor building based on design basis calculation.

b. Airborne radiation monitors

Airborne radioactivity monitoring for plant personnel protection and surveillance uses fixed location, continuous particulate monitors which include continuous iodine samplers; portable continuous particulate monitors with continuous iodine samplers positioned at specific work sites; and particulate and iodine grab samples taken before and during specific jobs.

Movable local alarming continuous air monitors are placed at predetermined plant locations for personnel protection and to substantiate the quality of the plant breathing atmosphere. The monitors have local readouts (charts) and have radioiodine sampling capabilities.

The installed continuous particulate monitoring system was designed for responsive personnel protection and plant surveillance. The four installed particulate monitors measure the airborne particulate activity levels in the radwaste, reactor, and turbine building ventilation exhaust, and furnish alarm and recording signals to the main control room. These units draw approximately 3 cfm air sample through the particulate filter which is monitored by a shielded beta detector with an efficiency of approximately 30%. The resultant response of the system is an increase of about 350 cpm for 1 hr of sampling at a 1×10^{-10} $\mu Ci/cm^3$ concentration. External gamma radiation will increase the background by 70 cpm/mrem/hr.

The actual ability of a ventilation exhaust monitoring system to detect the airborne particulate concentration in a specific space is dependent on the following factors:

1. Flow rate ratio (flow of air from a specific confined space/flow rate of bulk ventilation system exhaust),

2. Particulate activity and its half-life of the bulk ventilation system exhaust air,

3. Radionuclide composition in the specific confined space, and

4. The energy of the beta radiation from the radionuclide composition.

Normal plant conditions are expected to yield a bulk ventilation exhaust air concentration (primarily short-lived fission product daughters and natural activity half-life about 20 minutes) of 1-3 x 10^{-10} μCi/cm^3. This will reach an equilibrium on the sample filter of about 500 cpm. The MPC_a for normal plant airborne contamination is expected to be greater than 6 x 10^{-8} μCi/cm^3. At this MPC_a concentration a 1-hr accumulation (one MPC_a-hr) will equal 2.0 x 10^5 cpm. Applying a dilution factor of 270:1, the 1-hr accumulation will equal 750 cpm. This is a worst case dilution that considers the reactor building TIP Drive room at 400 cfm in the 105,000 cfm building bulk exhaust flow. Therefore, the ventilation monitoring system will easily detect 10 MPC_a-hr on all locations.

Local particulate constant air monitoring instruments and a comprehensive (particulate, noble gases, and iodine) grab air sampling program complement the plant air sampling program. Under these conditions, corrective actions will be taken and an assessment by portable sampling system results and portable monitoring activities will establish activity levels in all occupied areas which have potential for abnormal airborne activity.

In the radwaste building, the potentially contaminated areas normally entered by people would be those corridors adjacent to radioactive liquid and gaseous waste processing systems equipment such as demineralizers, concentrators, waste storage tanks, recombiners, dryers, moisture separators, and charcoal holdup vessels. Assuming that exfiltration from any one of the process systems to a normally entered corridor was sufficient to attain MPC_a levels for ^{137}Cs in that corridor, the dilution ratio would approach a factor of 10 to 100. For the worst case (100 to 1 ratio of bulk ventilation flow rate to corridor flow rate), ^{137}Cs at MPC_a (6 x 10^{-8} μCi/cm^3) would be detected within 1 hr on the continuous particulate monitor. If exfiltration from a process vessel cubicle was sufficient to produce MPC_a levels in an adjoining corridor, it is more probable that the normal cubicle flow rate input to the bulk ventilation flow would produce a prior distinguishable countrate ramp.

In the turbine building, airborne contamination is most likely to arise from nuclear steam leaks or offgas processing systems piping. Heater bay areas, steam jet air ejector rooms, turbine areas, and steam-driven feedwater pump areas have the potential for airborne contamination. The areas have individual ventilation exhaust rates in excess of 5000 cfm and ^{137}Cs MPC_a concentrations originating in these areas would give a continuous air monitor response ramp which is distinguishable within 1 hr.

Each of the continuous particulate monitors has an associated iodine sampling cartridge which is counted regularly for baseline and surveillance information. This cartridge and iodine sampled/collected with portable sampling devices will be analyzed in the plant laboratory counting facility when abnormal airborne activity levels are signaled by a continuous particulate monitoring system. At a 5 cfm air flow rate through an iodine sampling cartridge, iodine present at an occupational MPC_a concentration of 9 x 10^{-9} µCi/cm^3 would be quantitatively observable within a 1-minute sample interval. Bases for this assessment are a 10-minute count time on a 12-15% Ge(Li) detector system having an overall efficiency of about 1% when source and geometry considerations are included. The information presented for detecting one MPC_a concentration for ^{137}Cs in areas having a low ventilation flow rate can also be applied to the iodine case. One MPC_a of iodine can be ascertained within a 1-hr sampling period with a dilution factor of greater than 100. The routine weekly analysis of integrated iodine samples of radwaste, reactor, and turbine building ventilation air will permit observation of small iodine inputs. When these inputs are significant, a particulate and iodine sampling program is initiated to establish the source point.

Continuous particulate monitoring is reasoned to be the most responsive personnel protection and internal plant surveillance mechanism available. In addition, all tasks with potential for generating airborne contamination will be performed only when authorized by a radiation work permit (RWP).

The RWP assesses the radiological hazards, establishes additional monitoring and sampling requirements and, if necessary, specifies required engineering control and/or respiratory protection.

During outages, the above airborne monitoring system will be augmented by additional iodine sampling (continuous and grab) on the refueling floor since airborne iodine concentrations are known to become significant at this time.

12.3.4.3 Specification for Area Radiation Monitors

The area radiation monitoring system is shown as a function block diagram in Figure 12.3-24. Each channel consists of a sensor and a converter unit, a combined indicator and trip unit, a shared power supply, and a shared multipoint recorder. All channels also have a local meter and visual alarm auxiliary unit mounted near the sensor.

Each monitor has an upscale trip that indicates high radiation and a downscale trip that indicates instrument failure. These trips actuate alarms but cause no control function. The trip circuits are set so that a loss of power annunciates in the control room.

The type of detector used is a Geiger-Mueller tube responsive to gamma radiation over an energy range of 80 KeV to 7 MeV. Detector ranges are given in Table 12.3-1.

The calibrating frequency is once every 18 months using standard sources with National Institute of Standards and Technology (NIST) traceability. This ensures accuracies of (+) or (-) 20% over the detection interval.

An internal trip test circuit, which is adjustable over the full range of the trip circuit, is provided. The test signal is fed into the indicator and trip-unit input so that a meter reading is provided in addition to a real trip. High-range radiation alarm trip circuits for high level and criticality monitors are of the latching type and must be manually reset at the front of the control room panel. The trip circuits for all other area radiation monitors are of the nonlatching type.

12.3.4.4 Specification for Airborne Radiation Monitors

The airborne particulate monitors contain scintillation detectors with count ratemeters. Means for remote recording and alarm annunciation are provided for in the main control room. Sample collectors consist of shielded, fixed particulate, filter-type air collectors. The calibration frequency will occur at least annually and after major maintenance. Instrument response checks will be made at least monthly. Monitors will be calibrated using standard radioactive sources in the same geometry as the location of the particulate filter or by collection and analysis of mixtures present at known air flow rates. Particulate monitors are provided in the reactor, turbine, and radwaste buildings. The monitors are located so as to monitor the exhaust air from that building prior to any filtration. In addition, charcoal sampling cartridges are installed in each monitor for laboratory analysis of iodine.

Each of the four channels of the airborne radioactivity monitors has an independent local visual and audible alarm and all share a common annunciator alarm window in the main control room. High radioactivity or equipment failure will generate an alarm signal. No automatic system functions are performed by the alarm signals.

12.3.4.5 Annunciators and Alarms

The annunciators are of the window type with pushbutton switches for sound acknowledgment and light reset functions. There are no automatic system actions performed by the area radiation monitors. The annunciator alarm windows are located in the main control room.

Area monitors have local/remote alarms that sound on exceeding their high radiation alarm settings and remote alarms to indicate instrument failure (see Figure 12.3-24). Monitors located in the reactor building near the fuel pool and in the new fuel areas have individual high radiation alarm windows. The remainder of the area monitor units in the reactor building have a common annunciator window for high radioactivity. Area monitors in the turbine building and the radwaste building each have a common building high radioactivity alarm window. All the area monitors have one common alarm window for instrument failure.

The two area monitors that are used as criticality detectors are located in the new fuel vault. These monitors have a range of 10^{+2}-10^{+6} mrem/hr and alarm on either of a fully adjustable upscale or downscale trip point. The alarm setpoint and bases are given in the Licensee Controlled Specifications.

12.3.4.6 Power Sources, Indicating and Recording Devices

The area radiation monitor power supply units, indicating devices (except local alarms), and the recorders are all located on panels in the control room and receive power from the 120-V ac instrument power supply bus. The local audio alarms are supplied from a local 120-V ac instrument distribution panel. The recorder is a multipoint, strip chart type, which compiles a permanent record of inputs from the area radiation monitors.

12.3.5 REFERENCES

12.3-1 Jaeger, R. G. et al., Engineering Compendium on Radiation Shielding, Volume 1, Shielding Fundamentals and Methods.

12.3-2 Rockwell, T., Reactor Shielding Design Manuals, 1st Edition, D. Van Nostrand Co., Inc., New York, 1956.

12.3-3 Goldstein, H., Fundamental Aspects of Reactor Shielding, Addison-Wesley Publishing Co., Inc., Reading, 1959.

12.3-4 Blizard, E. P., Reactor Handbook, Vol. III, Part B, Shielding.

12.3-5 Aalto, E. et al., A Users Manual for the NRN Shield Design, AE-145, June 1964.

12.3-6 Hughes, D. J. and Schwartz, R. B., Neutron Cross Sections, BNL 325, 2nd Edition, 1958.

Hughes, D. J., Magurno, B. A. and Brussel, M. K., Neutron Cross Sections, BNL 325, wnd. ed., Supplement 1, 1960.

Stehn, John R. et al., Neutron Cross Sections, BNL 325, 2nd. Edition, Supplement 2, 1964.

12.3-7 Malenfant, Richard, QAD-A Series of Point Kernel General Purpose Shielding Program, U.S. Department of Commerce, LA-3573, Springfield, VA, 1966.

12.3-8 Walker, R. L., and Grotenhuis, M., A Summary of Shielding Constants for Concrete, ANL-6443, November 1961.

12.3-9 Chilton, A. B. and Huddleston, C. M., A Semiemperical Formula for Differential Dose Albedo for Gamma Rays on Concrete, Nuclear Science and Engineering, 17, 419-424, 1963.

Table 12.3-1

Area Monitors

Station	Location	Building Level (ft)	Range (mrem/hr)
1	Reactor building fuel pool area	606	10^2-10^6
2	Reactor building fuel pool area	606	1-10^4
3	Reactor building new fuel area	606	10^2-10^6
3A	Reactor building new fuel area 2	606	10^2-10^6
4	Reactor building control rod hyd equipment area E	522	1-10^4
5	Reactor building control rod hyd equipment area W	522	1-10^4
6	Reactor building equipment access area S	572	1-10^4
7	Reactor building neutron monitor system drive mechanical area	501	1-10^4
8	Reactor building SGTS filters area	572	1-10^4
9	Reactor building northwest RHR pump room	422	1-10^4
10	Reactor building southwest RHR pump room	422	1-10^4
11	Reactor building northeast RHR pump room	422	1-10^4
12	Reactor building RCIC pump room	422	1-10^4
13	Reactor building HPCS pump room	422	1-10^4
14	Turbine building turbine front standard	501	1-10^4
15	Turbine building entrance	441	1-10^4
16	Turbine building reactor feed pump area 1A	441	1-10^4
17	Turbine building reactor feed pump area 1B	441	1-10^4
18	Turbine building condensate pump area	441	1-10^4

Table 12.3-1

Area Monitors (Continued)

Station	Location	Building Level (ft)	Range (mrem/hr)
19	Main control room	501	$1\text{-}10^4$
20	Radwaste building valve room E	467	$1\text{-}10^4$
21	Radwaste building valve room W	467	$1\text{-}10^4$
22	Radwaste building sample room	487	$1\text{-}10^4$
23	Reactor building CRD pump room 10	422	$1\text{-}10^4$
24	Reactor building equipment access area (W)	471	$1\text{-}10^4$
25	Radwaste building hot machine shop	487	$1\text{-}10^4$
26	Radwaste building contaminated tool room	467	$1\text{-}10^4$
27	Radwaste building waste surge tank area	437	$1\text{-}10^4$
28	Radwaste building tank corridor area north	437	$1\text{-}10^4$
29	Radwaste building tank corridor area south	437	$1\text{-}10^4$
30	Radwaste building radwaste control room	467	$1\text{-}10^4$
32	Reactor building NE entrance	471	$10^{-1}\text{-}10^4$
33	Reactor building NW entrance	501	$10^{-1}\text{-}10^4$
34	Reactor building eastside	606	$10^{-1}\text{-}10^4$
35[a]	Reactor building refueling bridge	606	0.1-2000

[a] Item 35 is installed at its dedicated location on the refueling bridge prior to bridge operation. Alarm settings for all of the above monitors will be selected to provide indication of any abnormal increase in radiation levels while minimizing false alarms.

Table 12.3-2

Maximum Design Basis Background Radiation
Level for Area Monitors

ARM	Building Level (ft)	Maximum Design Basis Background Level (mrem/hr)
ARM-RE-1	606	100
ARM-RE-2	606	100
ARM-RE-3	606	100
ARM-RE-3A	606	100
ARM-RE-4	522	3000
ARM-RE-5	522	500
ARM-RE-6	572	100
ARM-RE-7	501	100
ARM-RE-8	572	100
ARM-RE-9	422	4000
ARM-RE-10	422	4000
ARM-RE-11	422	100
ARM-RE-12	422	3000
ARM-RE-13	422	100
ARM-RE-23	422	3000
ARM-RE-24	471	100
ARM-RE-32	471	100
ARM-RE-33	501	100
ARM-RE-34	606	100

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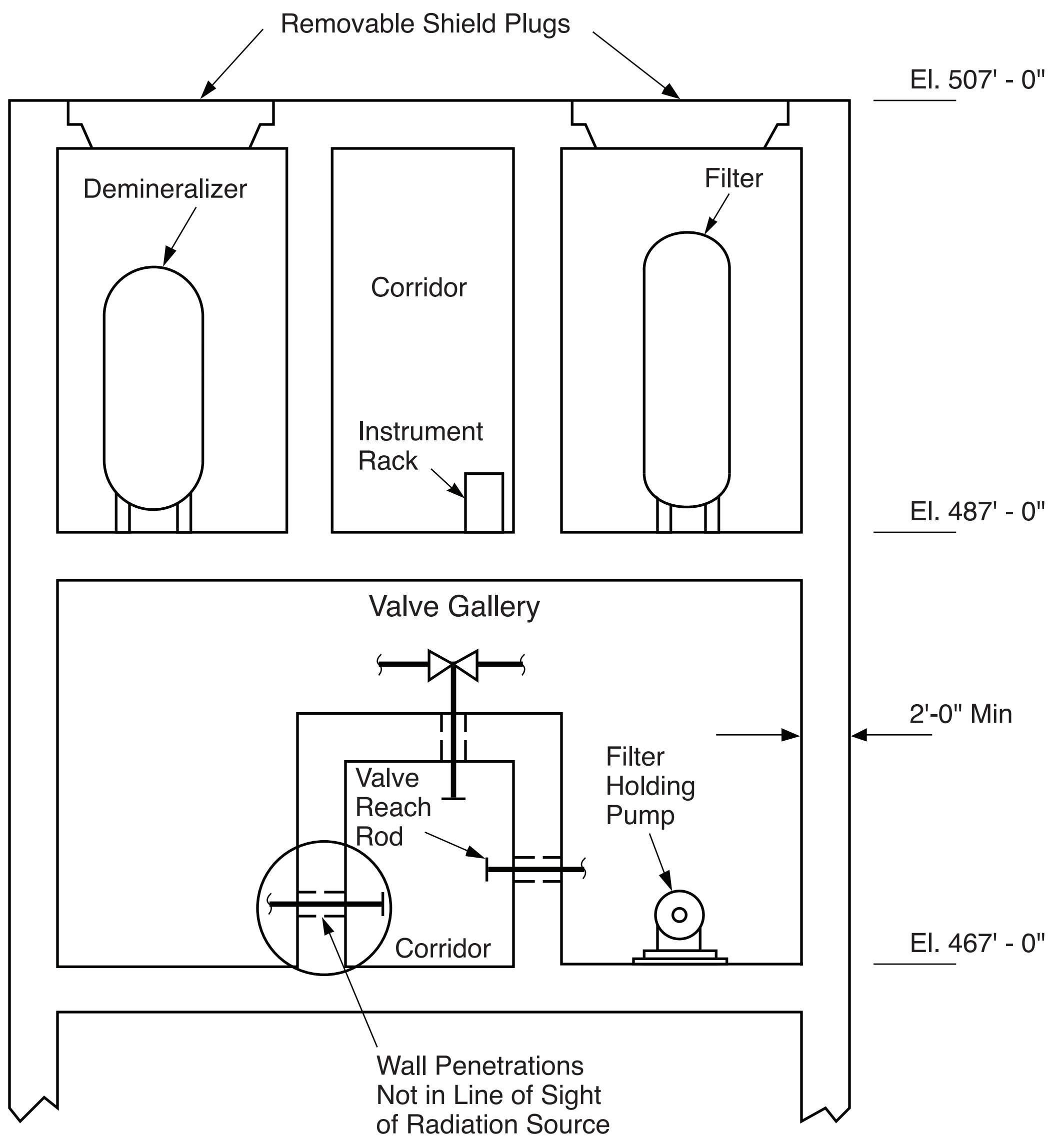
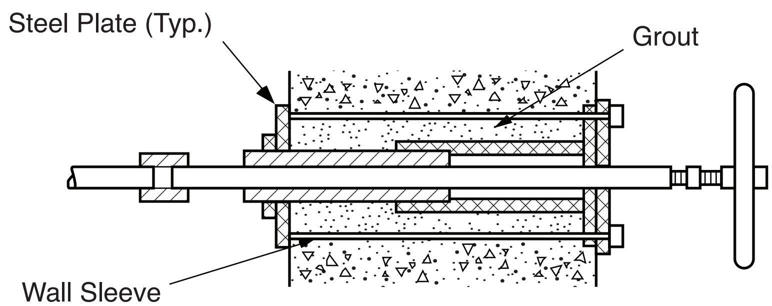
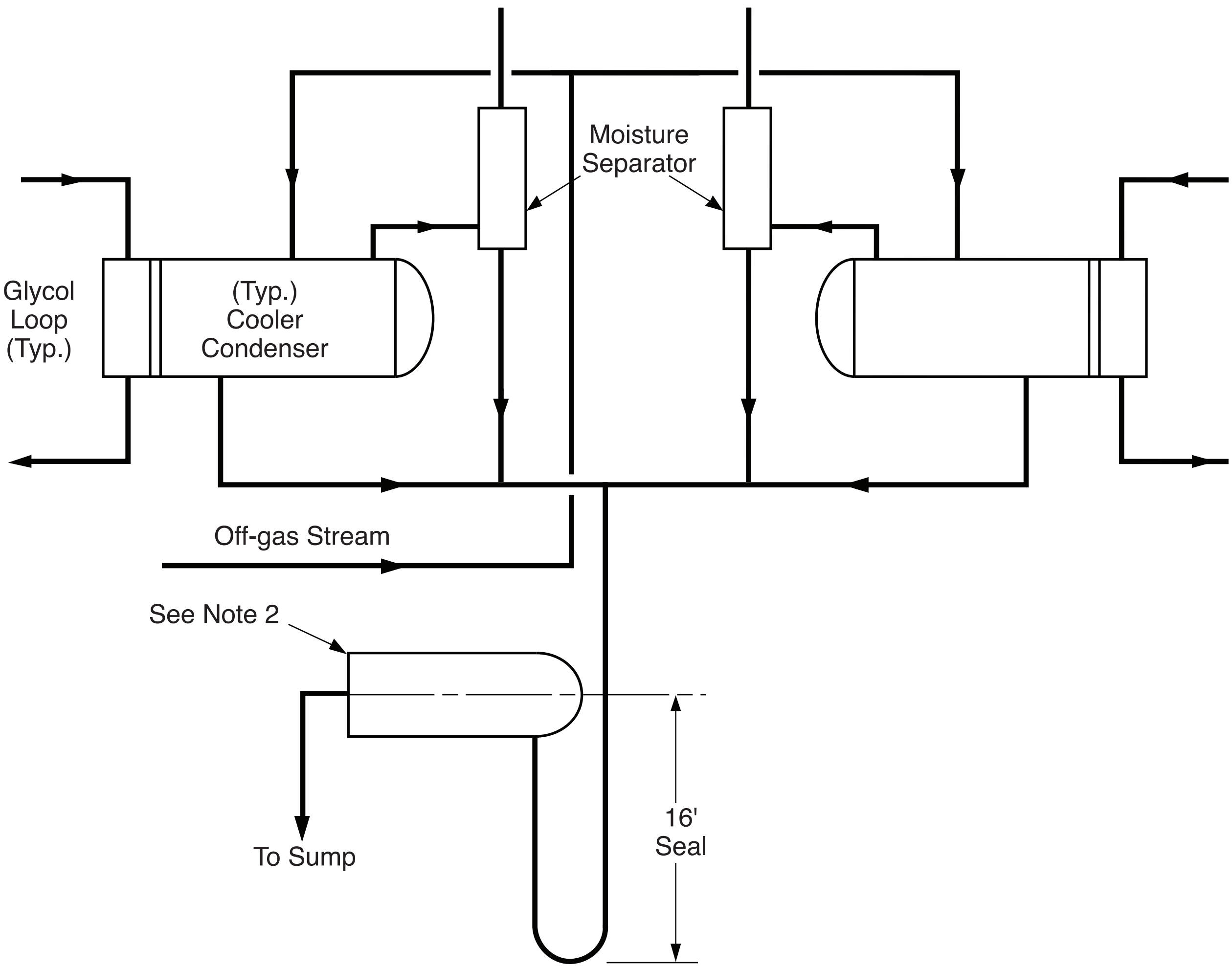




Penetration Detail

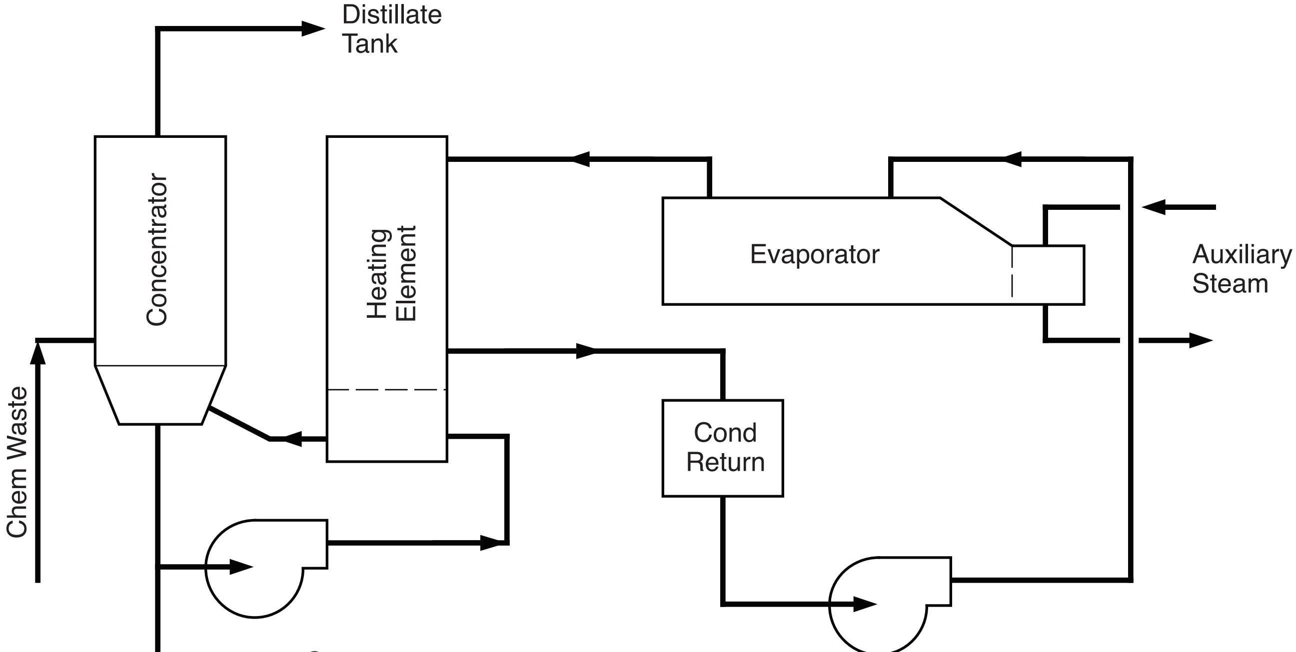
Columbia Generating Station Final Safety Analysis Report	**Arrangement of Filtration and Demineralization Equipment (Typical)**
	Draw. No. **900547.52** | Rev. | Figure **12.3-19**



Notes:

1. Valves and instrumentation are not shown to prevent clutter.

2. Overflow volume in the enlarged discharge section is sufficient to restore loop seal following a pressure surge.

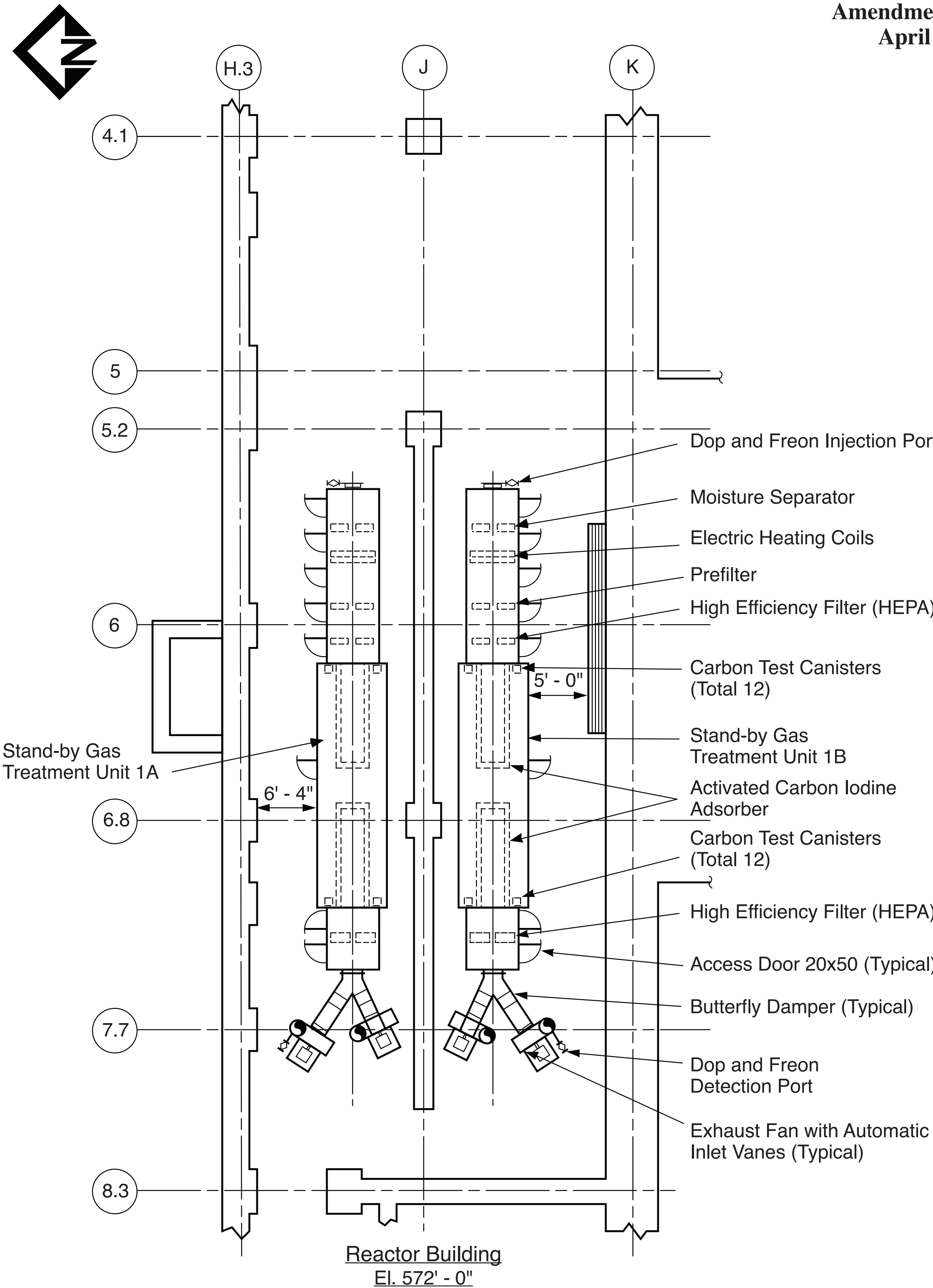


Columbia Generating Station
Final Safety Analysis Report

Decontamination Concentrator Steam Supply Arrangement

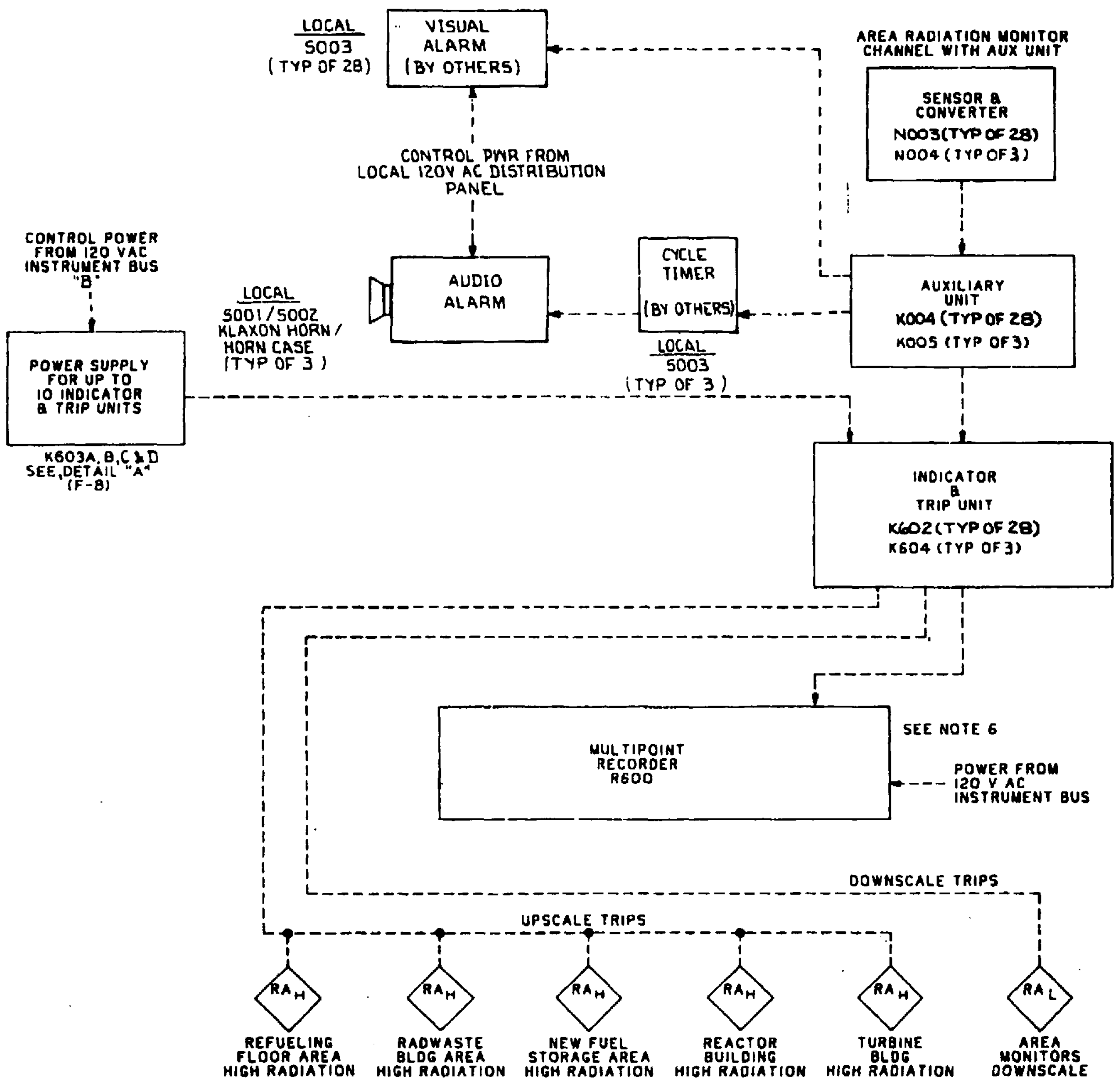
Draw. No. 900547.54 | Rev. | Figure 12.3-21

Figure Not Available For Public Viewing



Columbia Generating Station Final Safety Analysis Report	Layout of the Standby Gas Treatment System Filter Units		
	Draw. No. 900547.56	Rev.	Figure 12.3-23

GENERAL ELECTRIC
IED
919D693AD AREA RADIATION MONITORING SYS
HANFORD 2
23BX173AD
FCF
239X247AD (D21-1010)

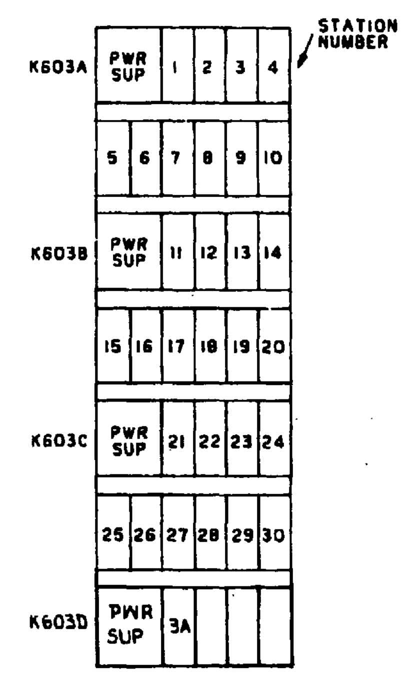


RANGE	QUANTITY (NOTE 1)		
MR / HR	SENSOR & CONVERTER	AUX UNIT	INDICATOR & TRIP UNIT
.01-100			
0.1-1000			
$1\text{-}10^4$	N003 (28)	K004 (28)	K602 (28)
$10^2\text{-}10^6$	N004 (3)	K005 (3)	K604 (3)

TABLE 1

NOTES:

1. RANGES CHOSEN BY PURCHASER ARE SHOWN ON D21-3050
2. REMOVED
3. REMOVED
4. CHANNEL CALIBRATED BY USE OF CALIBRATION UNIT K003. (NOT SHOWN)
5. FOR LOCATION & IDENTIFICATION OF INSTRUMENTS SEE INSTRUMENT DATA SHEET D21-3050.
6. RECORDERS ARE LOCATED IN PROCESS RADIATION RECORDER VERTICAL BOARD. (H13-P600)



INDICATOR & TRIP UNITS FOR EACH STATION
PANEL D21-P600 (H13-P673)
DETAIL "A"

REFERENCE DOCUMENTS

		MPL ITEM NO.
1.	DES SPEC SPECIAL WIRE & CABLE	A62-4010
2.	DES SPEC AREA RADIATION MONITORING.	D21-4010

F

Columbia Generating Station
Final Safety Analysis Report

Block Diagram - Area Radiation Monitoring System

Draw. No. 02D21-04,1,1 | Rev. 6 | Figure 12.3-24

Form No. 960690

12.4 DOSE ASSESSMENT

12.4.1 DESIGN CRITERIA

The criteria for the dose to plant personnel during normal operation and anticipated operational occurrences including refueling, are based on the requirements discussed in 10 CFR Part 20. The design radiation levels during normal operation and refueling are shown in Figures 12.3-5 through 12.3-18. In areas such as the control room and offices, the maximum dose rate does not exceed 1.0 mrem/hr (Zone I radiation level). For personnel who work in controlled radiation areas, radiation Zone II through IV in Figures 12.3-5 through 12.3-18, administrative controls ensure that doses do not exceed the requirements of 10 CFR Part 20.

12.4.2 PERSONNEL DOSE ASSESSMENT BASED ON BWR OPERATING DATA

The italicized information is historical and was provided to support the application for an operating license.

12.4.2.1 General

In general, data (Reference 12.4-1) from operating boiling water reactors (BWRs) have shown that the man-rem exposures to plant personnel are primarily due to the corrosion product isotopes. Of the corrosion product isotopes, ^{60}Co is believed to be the single most important radionuclide.

A review of the data from operating reactors was performed in References 12.4-6 and 12.4-7. Based on this it was concluded that the shielding design, which assumes the GE BWR design base source terms, was adequate to account for the additional radioactivity that will deposit in the lines due to crud.

Chemical cleaning connections were also installed on a number of systems. A chemical cleanup can be performed to reduce the deposits of crud and minimize the increase in radiation levels if needed. Section 12.3.1.3.2 addresses the design features that were incorporated to reduce the buildup of crud.

The variables that have been found to affect plant personnel exposure include the following:

a. *The BWR plants show an increase in total personnel exposure during the first few years of operation,*

b. *The need to minimize plant downtime requires that inspection and repair tasks must be started immediately after plant shutdown when the dose rates from short-lived radionuclides can be significant,* *c. Plant design and equipment layout has a significant effect on personnel dose. Section 12.3.1 discusses the design features used to minimize plant personnel exposure,*

d. Training and experience of plant workers,

e. The extent of maintenance operations required for a specific year, and

f. The extent that a utility uses non-regular or contractor personnel.

12.4.2.2 Personnel Dose from Operating BWR Data

References 12.4-1 through 12.4-5 provide a tabulation of personnel exposures for operating BWRs. Table 12.4-9 tabulates the average personnel exposure for operating BWRs for the period 1969 through 1980. References 12.4-4 and 12.4-5 provide more recent information. The assessments of personnel exposures summarized in Section 12.4.2.3 include this more recent information.

12.4.2.3 Occupancy Factors, Dose Rates, and Estimated Personnel Exposures

A summary of the total estimated man-rem doses broken down by major function is given in Table 12.4-1. More detailed breakdowns are presented in Tables 12.4-2 through 12.4-8 for each of the seven major functions given in Table 12.4-1. These tables are based on the more recent information obtained from industry operating experience. The data from Table 12.4-9 is given for comparison purposes only.

The results of the total estimated man-rem doses will be discussed with reference to six occupational groups as follows:

a. Group 1 - This group includes maintenance personnel such as mechanical, electrical, instrument craftsmen, and Foreman. There are approximately 128 people is this group. Tables 12.4-4 and 12.4-8 provide the functional breakdown of exposures for this occupational group. As can be seen from the tables, 433 total man-rem may be expected.

Routine and special maintenance operations which include control rod drive repairs, residual heat removal (RHR) repairs, snubber maintenance, etc., account for approximately 60% of the average annual personnel dose. One to two rem per year per person is projected for the station maintenance personnel for a maximum total of 256 man-rem per year. Accordingly, the remaining 175 man-rem per year would be expected to be received by non-station maintenance personnel. As discussed in Section 12.3.1, the equipment layout *and design and shielding design are such that the exposures are as low as is reasonably achievable (ALARA).*

b. *Group 2 - This group includes plant operations personnel composed of supervisors, control room staff and plant equipment operators. There are approximately 60 people in this group. Tables 12.4-2, 12.4-3, 12.4-5, and 12.4-6 show the total estimated man-rem for this group. As can be seen, the total is approximately 130 man-rem per year or approximately 2.2 rem per year per man. Personnel in this group will be performing routine and non-routine operation and surveillance, waste processing and refueling operations. In plant operations, personnel are expected to receive approximately one to two rem per year per man for a maximum total of 120 man-rem per year. The remaining 10 man-rem per year may be expected to be received by non-station personnel. As part of this total, the supervisors and control room staff are expected to receive an exposure of less than 500 mrem/yr.*

c. *Group 3 - This group includes health physics and chemistry personnel. There are approximately 53 people in this group. If the plant chemistry personnel spend 1% of their time collecting samples in Zone III sampling stations. They will receive a maximum dose of 723 mrem/yr. Assuming the remainder of their time is spent in Zone I and Zone II areas, the total dose is between 1 and 2 rem per person. The health physics personnel conduct radiation surveys and support maintenance activities which require continuous and pre-job radiation surveys. The exposure to these health physics personnel ranges from 2 to 3 rem/yr. This is based on experience from operating plants. Assuming a dose of 3 rem per person per year and considering 35 health physics people in the group, the total is 105 man-rem per year. Since this group covers virtually all functions delineated in Tables 12.4-2 through 12.4-8, this 105 man-rem is considered to be spread out across all the functions.*

d. *Group 4 - This group includes engineers and technical supervisors. There are approximately 27 people in this group. Personnel in this group will spend most of their time in Zone I areas where exposures are less than 500 mrem/yr. Table 12.4-7 indicates approximately 153 man-rem per year will be experienced for inservice inspection. Plant technical personnel will have a supervising roll in this operation with non-station personnel performing the inspection operations. This supervisory roll will take the personnel into all zone levels during ISI activities and this roll is expected to result in exposure from 1 to 2 rem/yr. Thus, the projected dose estimate for the 27 people in this group is 54 man-rem per year, the balance being accounted for in the non-station personnel.*

e. *Group 5 - This group includes station supervisors such as health physics and chemistry supervisors, shift supervisors, etc. There are approximately 24 people in this group. Station personnel will supervise Group 1 and Group 2 personnel. Their dose is approximately the same as personnel in these groups. With a projected dose estimate of 1 rem per year per person with 24 people in the group, the total dose is 24 man-rem per year.*

f. *Group 6 - This group includes administrative and management personnel. There are approximately 31 people in this group. Personnel in this group spend their time in Zone I radiation areas. The projected dose estimates will be less than 500 mrem/yr. With 31 people in this group and a 500 mrem per man per year the total dose is 15.5 man-rem per year.*

As seen from Table 12.4-1, the total estimated man-rem exposure is 715 man-rem. Groups 3, 5, and 6 are considered to be spread over all the functions. These groups constitute only 15% of the total exposure in any case.

12.4.3 INHALATION EXPOSURES

Airborne radionuclide concentrations in normally occupied areas are, as discussed in Section 12.2.2, well below the limits set by 10 CFR Part 20 and thus inhalation exposures are negligible. In areas where engineering controls or operational procedures do not reduce the airborne radionuclide concentrations sufficiently, additional measures such as access control, limiting exposure time (DAC hours), and respiratory protection devices are used to maintain the total effective dose equivalent (TEDE) ALARA.

12.4.4 SITE BOUNDARY DOSE

Steam handling equipment on the turbine operating floor can contribute to the site boundary dose in two ways: through a direct component and through an air-scattered “skyshine” component. Since the ^{16}N bearing equipment is known, it can be shielded to reduce the direct component. The “skyshine” component reaches the site boundary as a result of those gamma rays which are directed such that they bypass any intercepting shield walls and are scattered by the air to the site boundary.

The calculated results show that the skyshine dose will have its greatest effect on a dose point approximately 1950 m north of the turbine building. The skyshine dose at this point will be approximately 4 mrem/yr. This result is based on a plant capacity factor of 80%.

The main contributors to this dose and their contribution (in percent) are the south moisture-separator reheater (MSR) which contributes 60%, the north MSR which contributes 20%, the cross over lines which contribute 10%, and the turbines and feedwater heaters which contribute 10%.

The dose estimate was computed from a model that represents the ^{16}N gamma leakage by point isotopic sources. This model uses the output from the COHORT Code (Reference 12.4-3) which gives the airscattered dose as a function of distance and source ray angle.

The site boundary dose from liquid and gaseous effluents are discussed in Sections 11.2.3 and 11.3.3.

12.4.5 REFERENCES

12.4-1 *Atomic Industrial Forum, Compilation and Analysis of Data on Occupational Radiation Exposure Experienced at Operating Nuclear Power Plants, September 1974.*

12.4-2 *Ninth Annual Occupational Radiation Exposure Report, NRC, NUREG-0322, Washington, D.C., October 1977.*

12.4-3 *Tenth Annual Occupational Radiation Exposure Report, NRC, NUREG-0463, Washington, D.C., October 1978.*

12.4-4 *Occupational Radiation Exposure at Light Water Cooled Power Reactors, Annual Report 1977, NRC, NUREG-0482, Washington, D.C., April 1977.*

12.4-5 *Occupational Radiation Exposure at Commercial Nuclear Power Reactors, Annual Report 1979 and 1980, Volumes 1-2, NRC, NUREG-0713, December 1981.*

12.4-6 *NRC Seventh Annual Occupational Radiation Exposure Report 1974, NUREG-75/108, November 1975.*

12.4-7 *Atomic Industrial Forum, Compilation and Analysis of Data on Occupational Radiation Exposure Experienced at Operating Nuclear Power Plants, September 1974.*

Table 12.4-1

Summary of Occupational Dose Estimates

		Man-rem/yr
1.	*Routine operation and surveillance*	*53*
2.	*Nonroutine operation and surveillance*	*15*
3.	*Routine maintenance*	*288*
4.	*Waste processing*	*15*
5.	*Refueling*	*48*
6.	*Inservice inspection*	*153*
7.	*Special maintenance*	*145*
	Total	*717*

Table 12.4-2

Occupational Dose Estimates During Routine Operations and Surveillance

	Activity	Average Dose Rate (mrem/hr)	Exposure Time (hr)	Number of Workers	Frequency	Dose (man-rem/yr)
1.	Walking	0.5	0.5	2	1/shift	= 0.54
2a.	Checking					
	Railroad access	1	1	2	1/shift	= 2.2
	Change rooms	1	1	2	1/shift	
	Relay room	1	1	2	1/shift	
	Motor generator sets	1	1	2	1/shift	
	Battery room	1	1	2	1/shift	
	Computer room	1	1	2	1/shift	
	Switch gear room	1	1	2	1/shift	
	Air conditioning equip.	1	1	2	1/shift	
	Recirc. motor gen.	1	1	2	1/shift	
	RBCCW heat	1	1	2	1/shift	
	Exchangers	1	1	2	1/shift	
	Emergency air comp	1	1	2	1/shift	
	RBCCU pumps	1	1	2	1/shift	
	RBCCW expansion	1	1	2	1/shift	
	Tank	1	1	2	1/shift	
2b.	Mech. vac. pumps	10	0.5	2	1/shift	= 11
	CRD pumps	10	0.5	2	1/shift	
	CRD hydraulic	10	0.5	2	1/shift	
	Cont. units	10	0.5	2	1/shift	
	Refueling floor	10	0.5	2	1/shift	
	CRD filters	10	0.5	2	1/shift	
	RUCV demmo resin	10	0.5	2	1/shift	
	Tanks	10	0.5	2	1/shift	
	RNP pumps	10	0.5	2	1/shift	
	SRMP pumps	10	0.5	2	1/shift	
	Air coolers	10	0.5	2	1/shift	
	IVST racks	10	0.5	2	1/shift	
2c.	CRD storage and repair	15	0.2	2	1/shift	= 3.3
	SGTS	15	0.2	2	1/shift	
	HPCI turbine and pump	15	0.2	2	1/shift	
2d.	RWCU heat exchangers	50	0.1	1	1/shift	= 5.5
	RHR heat exchangers	50	0.1	1	1/shift	
	Acid purple and turbine	50	0.1	1	1/shift	

Table 12.4-2

Occupational Dose Estimates During Routine Operations and Surveillance (Continued)

	Activity	Average Dose Rate (mrem/hr)	Exposure Time (hr)	Number of Workers	Frequency	Dose (man-rem/yr)
	Checking (continued)					
2e	Demin precoat tank	0.2	0.5	1	1/shift	= 0.1
	Precoat pump	0.2	0.5	1	1/shift	
	Waste sample pump	0.2	0.5	1	1/shift	
	Floor drain sample room	0.2	0.5	1	1/shift	
	Waste surge pump	0.2	0.5	1	1/shift	
	Equip. drain sump pump	0.2	0.5	1	1/shift	
	Waste surge pump	0.2	0.5	1	1/shift	
	Waste precoat pump	0.2	0.5	1	1/shift	
	Waste sludge disch. pump	0.2	0.5	1	1/shift	
	Waste filter aid pump	0.2	0.5	1	1/shift	
	Chemical waste pump	0.2	0.5	1	1/shift	
	Floor drain coll. pump	0.2	0.5	1	1/shift	
2f.	Chemical waste tank	50	0.5	1	1/shift	= 1.3
	Spent resin pump	50	0.5	1	1/shift	
	Cond. phase decant pump	50	0.5	1	1/shift	
	Cond. phase sludge	50	0.5	1	1/shift	
	Discharge mixing pump	50	0.5	1	1/shift	
2g.	Floor drain demin.	8	2	1	1/shift	= 0.8
	Waste hopper	8	2	1	1/shift	
	Floor drain filter	8	2	1	1/shift	
2h.	Turbine inst. and controls	0.5	1	2	1/shift	= 1.1
	Gen. CO_2 units	0.5	1	2	1/shift	
	Station air comp.	0.5	1	2	1/shift	
	Heater feed pumps	0.5	1	2	1/shift	
	Demin. pumps and valves	0.5	1	2	1/shift	
	MTG lubrication system	0.5	1	2	1/shift	
	Hatch area above demin. tanks	0.5	1	2	1/shift	
	H_2 seal 2.1 equip.	0.5	1	2	1/shift	
	Health shell pull space	0.5	1	2	1/shift	
	BCCW heat expansion and pumps	0.5	1	2	1/shift	

Table 12.4-2

Occupational Dose Estimates During Routine Operations and Surveillance (Continued)

	Activity	Average Dose Rate (mrem/hr)	Exposure Time (hr)	Number of Workers	Frequency	Dose (man-rem/yr)
	Checking (continued)					
2i	TBCCW expansion tank	5	0.3	1	1/shift	= 1.6
	Ventilation equipment	5	0.3	1	1/shift	
	Demin. precoat and resin tanks	5	0.3	1	1/shift	
	Demin. precoat pumps	5	0.3	1	1/shift	
	Sump pumps	5	0.3	1	1/shift	
	Reactor feed pump turbine	5	0.3	1	1/shift	
	Lub. system	5	0.3	1	1/shift	
	MTG lub oil cooler	5	0.3	1	1/shift	
	Main gen. and exciter	5	0.3	1	1/shift	
	MTG utilizer activators	5	0.3	1	1/shift	
	Stop and throttle valves	5	0.3	1	1/shift	
	Circ. water isol. valves	5	0.3	1	1/shift	
2j.	Heater drain pumps	50	0.2	1	1/shift	= 11.0
	Heater drain flash tanks	50	0.2	1	1/shift	
	Condense water box	50	0.2	1	1/shift	
	Reactor feed pumps and turbines	50	0.2	1	1/shift	
2k.	Drain coolers	15	0.5	1	1/shift	= 14.6
	Feed water heaters	15	0.5	1	1/shift	
	Reheater seal tank	15	0.5	1	1/shift	
	Gland steam condenser	15	0.5	1	1/shift	
	Main turbine	15	0.5	1	1/shift	
	Reheater separators	15	0.5	1	1/shift	
					Total	53.04

Table 12.4-3

Occupational Dose Estimates During Nonroutine Operations and Surveillance

	Activity	Average Dose Rate (mrem/hr)	Exposure Time (hr)	Number of Workers	Frequency	Dose (man-rem/yr)
1.	Operation of equipment:					
1a.	Traversing in-core probe system	2	2	2	3/yr	0.02
1b.	Safety injection system	5	1	1	1/month	0.06
1c.	Feedwater pumps and turbine	1	1	1	1/week	0.05
1d.	Instrument calibration	2	1	1	1/day	0.73
2.	Collection of radioactive samples:					
2a.	Liquid system	10	0.5	1	1/day	1.83
2b.	Gas system	5	0.5	1	1/month	0.03
2c.	Solid system	10	0.5	1	4/yr	0.01
2d.	Radiochemistry	1	1	2	1/day	0.73
2e.	Radwaste operation	3	8	3	1/week	3.75
2f.	Health physics	5	2	2	1/day	7.30
					Total	14.50

Table 12.4-4

Occupational Dose Estimates During Routine Operations and Surveillance

	Activity	Average Dose Rate (mrem/hr)	Exposure Time (hr)	Number of Workers	Frequency	Dose (man-rem/yr)
1.	Minor repairs reactor building	1	20	2	1/week	2.1
2.	Ventilation and air conditioning	0.5	20	1	1/week	0.5
3.	Control rod drive repair	15	200	6	1/yr	18
4.	Reactor water cleanup pump	180	35	3	1/yr	19
5.	Reactor water cleanup valve and heat exchanger	110	45	6	1/yr	30
6.	Residual heat removal system	200	27	8	1/yr	43
7.	Safety relief valves	80	30	5	1/yr	12
8.	Main steam isol. valves	75	100	6	1/yr	45
9.	Recirc. pumps	200	50	3	1/yr	30
10.	Snubber inspector and repair	75	100	5	1/yr	37.5
11.	Misc. turbine bldg. repairs	2	8	1	1/day	5.8
12.	Reactor feed pumps and turbine	10	40	2	2/yr	0.8
13.	Drain coolers	2	40	2	1/yr	0.16
14.	Steam jet air ejectors	10	40	2	2/yr	1.6
15.	Offgas system	2	40	2	6/yr	0.96
16.	MTG actuator	5	40	1	1/yr	0.24
17.	Heater drain flash tanks	2	40	1	1/yr	0.08
18.	Condenser water box	5	20	1	1/yr	0.1
19.	Annual turbine inspection	3	120	10	1/yr	3.6
20.	Misc. radwaste pump repairs	25	40	2	4/yr	8.0
21.	Misc. radwaste valve repairs	10	40	2	6/yr	4.0
22.	Filter and demin.	65	30	3	1/yr	5.9
23.	Centrifuge	50	8	2	4/yr	3.2
24.	Evaporation	85	50	3	1/yr	12.8
25.	Turbine instr. and control	2	10	1	1/week	1.0
26.	Waste solidification	2	40	2	2/yr	0.32
27.	Area monitors	20	40	2	2/yr	0.32
28.	Operate laundry facility	0.5	40	3	1/day	2.2
					Total	288.2

Table 12.4-5

Occupational Dose Estimates During Waste Processing

Activity	*Average Dose Rate (mrem/hr)*	*Exposure Time (hr)*	*Number of Workers*	*Frequency*	*Dose (man-rem/yr)*
Radwaste control room	*.5*	*8*	*1*	*1/shift*	*4.4*
Sampling and filter changing	*15*	*8*	*1*	*1/week*	*6.2*
Panel operator insp. and testing	*1*	*2*	*1*	*1/day*	*0.73*
Operation of waste and packaging equipment	*2*	*16*	*2*	*1/week*	*3.3*
				Total	*14.6*

Table 12.4-6

Occupational Dose Estimates During Refueling

	Activity	Aveg. Dose Rate (mrem/hr)	Exposure Time (hr)	Number of Workers	Frequency	Dose (man-rem/yr)
1.	Opening/closing reactor pressure vessel	60	40	10	1/yr	24
2.	Fuel preparation	10	24	2	1/yr	0.48
3.	Refueling	10	100	15	1/yr	15
4.	Fuel handling	2.5	100	4	1/yr	1.0
5.	Fuel sipping	10	120	6	1/yr	7.2
					Total	47.7

Table 12.4-7

Occupational Dose Estimates During Inservice Inspection

	Activity	Average Dose Rate (mrem/hr)	Exposure Time (hr)	Number of Workers	Frequency	Dose (man-rem/yr)
1.	Removal/replacement of insulation	150	80	4	1/yr	48
2.	Installation/removal and ladders	50	40	4	1/yr	8
3.	Inspecting inside drywell	150	80	6	1/yr	72
4.	Recorder data	50	80	6	1/yr	24
5.	Inspecting outside drywell	5	50	2	1/yr	0.5
					Total	153

Table 12.4-8

Occupational Dose Estimates During Special Maintenance

Activity	Average Dose Rate (mrem/hr)	Exposure Time (hr)	Number of Workers	Frequency	Dose (man-rem/yr)
Sparger replacement	800	60	5	Should not be necessary	---
CRD replacement	260	35	5	1/yr	45.5
Turbine overhaul	5	250	20	1/5 yr	5
Servicing in-core detectors	15	50	3	1/yr	2.3
Offgas charcoal sys. overhaul	20	100	2	1/20 yr	0.2
Special maintenance reactor water clearup sys.	150	100	8	1/10 yr	12
Misc. piping repairs	80	100	10	1/yr	80
				Total	145.0

Table 12.4-9

Summary of Annual Information Reported by Commercial Boiling Water Reactors

1969 - 1980

Year	Number of Reactors Included	Annual Collective Doses (man-rems)	No. of Workers With Measurable Dose	Gross MW-Yrs Electric Generated	Average Dose Per Worker (rems)	Average Collective Dose Per Reactor (man-rems)	Average No. Personnel With Measurable Doses Per Reactor	Average Man-rems Per MW-yr	Average MW-yrs Generated Per Reactor	Average Rated Capacity (MWe) Net
1969	3 (2)	586 (300)	290[a]	192	1.03[a]	195	145[a]	3.1	64	112
1970	6 (4)	764 (510)	1,321[a]	912	0.39[a]	127	330[a]	0.8	152	267
1971	7 (5)	1,784 (1,069)	1,873[a]	1,038	0.57[a]	255	375[a]	1.4	187	339
1972	10 (7)	2,858 (2,130)	2,258[a]	3,058	0.94[a]	286	323[a]	0.9	306	434
1973	12	4,564	5,340	3,394	0.85	380	445	1.3	283	459
1974	14	7,095	8,769	4,059	0.81	507	626	1.7	290	513
1975	18	12,611	14,607	5,789	0.86	701	812	2.2	321	611
1976	23	12,626	17,859	8,586	0.71	549	776	1.5	373	647
1977	23	19,042	21,388	9,098	0.89	828	930	2.1	396	645
1978	25	15,096	20,278	11,774	0.74	604	811	1.3	471	669
1979	25	18,322	25,245	11,671	0.73	733	1,010	1.6	467	669
1980	26	29,530	34,094	10,868	0.87	1,136	1,311	2.7	418	664

[a]During the years 1969 through 1972, all plants reported collective doses but a few did not submit the number of personnel that received measurable doses. The number of reactors that did report doses and number of workers is given in parentheses in the second column. The collective doses shown in parentheses in the third column, as well as the noted numbers in the remaining columns, are all based on the data submitted by the number of reactors shown in parentheses. This correction, and others, changed some of the values from those appearing in earlier NUREG documents.

12.5 RADIATION PROTECTION PROGRAM

12.5.1 ORGANIZATION

Radiation Protection, under the direction of the Radiological Services Manager, implements the Radiation Protection Program (RPP).

Health Physics (HP) is audited for compliance to regulations and to ensure that occupational and public radiation exposures are as low as is reasonably achievable (ALARA). Regulatory Guide 1.8 and ANSI 18.1-1971 have been followed in the selection of HP personnel. Energy Northwest pre-employment practices include screening to determine that plant employees are trustworthy, fit, and qualified to perform their duties safely. The experience and qualifications of the personnel responsible for the RPP and for handling and monitoring radioactive materials including special nuclear source and byproduct materials, are described in Sections 12.5 and 13.1. Also, Section 13.1 describes the minimum qualification requirements for specific plant personnel, using the criteria outlined in Regulatory Guide 1.8 and ANSI 18.1-1971.

The Plant General Manager reports to the Vice President, Operations and has the overall responsibility for the RPP. The Plant General Manager is responsible for ensuring that personnel, facilities, and other resources required to implement the RPP are available. This includes ensuring that the authority to implement an effective RPP is delegated through the management structure, ensuring the program receives the active support of all Energy Northwest personnel, and ensuring production goals, maintenance activities, and work schedules do not adversely affect the ability to provide proper radiological controls. In turn, all plant personnel share the responsibility for ensuring personal radiological safety and are required to follow the rules and procedures established for radiological safety. Specific responsibilities regarding ALARA are described in Section 12.1.

The Operations Manager reports to the Plant General Manager and has the responsibility for ensuring the independence of the RPP from plant operational pressures. The Operations Manager provides the Radiological Services Manager the support necessary for the effective implementation of the RPP.

The Radiological Services Manager reports to the Plant General Manager and is responsible for the implementation of the RPP with the exception of those radiation safety duties for which the Radiation Protection Manager (RPM) is responsible. The RPM has direct access to the Plant General Manager in all matters relating to radiation safety. The RPM meets the qualifications defined in Regulatory Guide 1.8, and provides the experience and expertise necessary to implement the RPP. The RPM is an assigned duty and not a defined position in the organization. RPM duties and responsibilities may be assigned to any of the Radiation Protection management or supervisory positions described in this Chapter.

The Radiological Services Manager supports the safe, reliable, and economic operations of the plant within applicable laws, standards, codes, regulations, and Energy Northwest policies.

The Radiological Planning Supervisor reports to the Radiological Services Manager and directs the activities and monitors the performance of the Radiological Planning Group. This group is responsible for performing ALARA reviews and evaluations to support HP Operations.

The Radiological Support Supervisor reports to the Radiological Services Manager and directs the activities and monitors the performance of the Radiological Support Group. This group provides technical support for all aspects of the RPP.

The Radiological Operations Supervisor reports to the Radiological Services Manager and directs the activities and monitors the performance of the Health Physics Craft Supervisors and Rad Material Control/Rad Waste Supervisor.

The Health Physics Craft Supervisors report to the Radiological Operations Supervisor and direct the activities and monitor the performance of HP Technicians. Health Physics Craft Supervisors are responsible for ensuring conditions that have the potential for causing exposure to radiation are identified, posted, and controlled.

The Rad-Material Control and Rad-Waste Supervisor reports to the Radiological Operations Supervisor and is responsible for providing immediate supervision, leadership and technical support to the laborers in the areas of equipment and area decontamination, radioactive material control and inventory and anti-contamination laundry; process, package and transport of radioactive waste material, including mixed waste.

Each individual who has unescorted access to Columbia Generating Station restricted areas is responsible for ensuring personnel radiation safety. This includes strict compliance with radiation protection requirements, procedures, and good radiological work practices. In addition, individuals with escort duties are responsible for the radiological safety of visitors.

12.5.2 EQUIPMENT, INSTRUMENTATION, AND FACILITIES

This section describes the equipment, instrumentation, and facilities available for implementation of the RPP and the criteria used for selection of the instrumentation and equipment. The guidance provided by Regulatory Guides 8.3, 8.4, 8.6, and 8.28 has generally been followed with exceptions noted as follows:

a. Regulatory Guide 8.3, “Film Badge Performance Criteria” will be followed if film badges are used in the plant program; however, other dosimeter types, such as optically stimulated luminescent (OSL) dosimeters, are used as the Dosimeter of Legal Record (DLR) for compliance with 10 CFR 20.1501, 20.1502, and 20.2206;

b. Regulatory Guide 8.4 is implemented for the selection of direct-reading pocket dosimeters as defined in Section 2 of ANSI N13.5-1972 except for C.2.b, which states, "The calibration/response test result should not exceed ±10% of an exposure from a source traceable to the National Bureau of Standards."* This is accepted on the minus side, but is considered excessively stringent on the positive side. Since the error on the positive side results in exposure conservatism to the worker, +20% is a more reasonable limit for rejection of a pocket dosimeter. Vendor literature will be accepted as documentation that performance standards specified in Regulatory Guide 8.4 are met. Continued use of direct-reading pocket dosimeters will be based on their ability to perform acceptably under test conditions for temperature and humidity described in approved Health Physics Instructions as follows: ±2% drift per 24 hr at -10°C and any percent humidity; +2% drift per 24 hr at 50°C and 95% humidity; and +20% and -10% of 80% of calibrated full scale;

c. Regulatory Guide 8.6, "Standard Test Procedure for Geiger-Mueller Counters," will be used as applicable. This guide references ANSI N42.3-1969 (ANSI/IEEE Standard 309-1970) for twelve different tests to Geiger-Mueller counters. Energy Northwest will develop tests and procedures to ensure that Geiger-Mueller tube characteristics are appropriate for planned (or intended) applications. These tests may incorporate plateau characteristics, dead time, efficiency, and operating environment;

d. The majority of direct-reading dosimeters at Columbia Generating Station are electronic dosimeters with audible-alarm capabilities. A program for their appropriate use requires that conditions under which they may not perform adequately be discussed, as well as describing performance specifications which are met.

Electronic dosimeters will not be used to circumvent the initial meter survey required prior to work in an area. Radiation Protection will assign electronic dosimeters only when the working environment is suitable for their use. Individuals required to wear an electronic dosimeter will be provided appropriate instructions either in training or at the time of issue to minimize the risk of improper use. Regulatory Guide 8.28 endorses, with one exception, the performance specifications indicated in ANSI N 13.27 "Performance Specifications for Pocket - Sized Alarming Dosimeters/Ratemeters."

* National Institute of Standards and Technology.

Since credit is taken for the audible-alarm capabilities of the electronic dosimeter, the Energy Northwest program for use of audible-alarm dosimeters complies with Regulatory Guide 8.28 with the following exceptions:

1. Requirement: Section C.2.c of the Regulatory Guide requires a source check of audible-alarm dosimeters each day before use.

   Energy Northwest position: Electronic dosimeters equipped with an automatic electronic test to ensure detector function are not subject to the requirement for a source check each day before use.

2. Requirement: Section C.2.b(1) of the Regulatory Guide specifies that alarm dosimeters should not be used when the alarm may not be heard, such as (a) in a high noise environment, (b) when the user has a pronounced hearing loss, (c) when the user is wearing mufflers over the ears, or (d) when the sound from the dosimeter would be muffled by heavy clothing worn over the dosimeter.

   Energy Northwest position: Electronic dosimeters will be allowed for use when their alarm may not be heard except when they are used to fulfill the alarming dosimeter function described in Technical Specification 5.7. When used in accordance with the Technical Specification, alternative methods of warning are required when the audible alarm may not be heard. Alternative methods include, but are not limited to the following: (1) vibrator, (2) ear phone, (3) flashing light clearly visible to the worker.

The program outlines the performance requirements for electronic dosimeters and details the exception to the ANSI N13.27 criteria, while ensuring that reliable electronic dosimetry is used to facilitate exposure control and the ALARA concept.

12.5.2.1 Criteria for Selection

a. Radiation and contamination survey instrumentation: This equipment was selected to cover the wide range requirements extending from picocurie quantity measurements in the laboratory to the thousand R/hour ranges necessary for emergency dose rate determinations. The laboratory instrumentation was chosen to provide capability for the quantitative and qualitative analyses required to identify and measure the radionuclides encountered in a power reactor. The portable instrumentation includes low level detection capabilities for alpha, beta, and gamma contamination and wide ranges of dose rate measuring instruments for beta, gamma, and neutron radiation. The criteria for quantity selection were to provide adequate available counting time for anticipated demand in the laboratory and sufficient portable instruments to cover normal operational and emergency requirements in all areas of the Columbia Generating Station facility;

b. Airborne radioactivity monitoring: The basic criteria for selection of this equipment were to provide a means for determining radioactive airborne effluents released from the plant, and to effectively monitor airborne radioactivity levels within the plant environs. Provisions have been made for continuing response monitoring of noble gases discharged from gaseous release points from the reactor, radwaste, and turbine building, and for continuous sampling of radioiodines and particulates at these same locations. Internal plant air monitoring instrumentation is used within these buildings with readout locally and in the control room;

c. Area radiation monitoring: This system was designed to provide continuous surveillance of radiation levels throughout the plant with local alarm at predetermined levels, local indication, and control room annunciation and recording. Functions of the system include warning of excessive gamma radiation levels in fuel storage and handling areas, detection of unauthorized or inadvertent movement of radioactive materials in the plant, local alarms to warn personnel in an area of a substantial increase in radiation levels, provision for supervisory information in the control room so that correct decisions may be made in the event of a radiation incident, backup to other systems for detection of abnormal migrations of radioactive materials in or from the process streams, and providing a permanent record of gamma radiation levels at selected locations within the various plant buildings; and

d. Personnel monitoring: Personnel dosimetry devices were chosen to provide a record of exposure received by occupationally exposed individuals at the site who are likely to receive, under normal or accidental conditions, exposures greater than 10% of applicable 10 CFR 20 limits.

   Personnel dosimeter badges (DLRs) containing an OSL dosimeter or other acceptable dosimetry provide the primary legal record of exposure received by personnel. Each person requiring monitoring for record is assigned a badge, which is recorded with the wearer’s identification. Results from the badge and the period of exposure are recorded on a document kept as a legal Energy Northwest record. Badges used will be capable of recording exposure over a range of at least 40 mrem to greater than 1000 rem.

   Persons being monitored may be required to wear other dosimetry assigned by the Radiation Protection staff, such as direct reading dosimeters, integrating dose meters, extremity badges, and finger rings.

12.5.2.2 Facilities

Radiation Protection facilities at Columbia Generating Station include the following:

a. Personnel decontamination showers and sinks are located in the radwaste building (487 ft level). Temporary change areas are set up as necessary in areas of the plant to localize and prevent the spread of contamination while performing maintenance activities. Small inventories of protective clothing are stored in the emergency relocation centers, operation and radwaste control rooms, and strategic locations throughout the plant;

b. Monitoring equipment for personnel [e.g., friskers and installed personnel monitors (IPMs)] and tools/personal items [friskers and small article monitors (SAMs)] are provided at the radiological access control areas and various areas within the plant to survey for radioactive contamination;

c. Facilities for personnel exposure monitoring and protection, which include:

    1. Internal dosimetry,
    2. Respiratory protection testing;

d. Medical first aid facilities are equipped to provide care for injuries, including those with radioactive contamination involved; and

e. Facilities for equipment and tool decontamination exist in the radwaste, turbine, and reactor buildings. The locations and facilities are

    1. Radwaste building

    The general decontamination area is shown in Figure 12.3-12, approximate column location Q.4-13.6 at the 467 ft 0 in. level. Facilities include curbing, sink, monorail hoist, and drains. At the 487 ft 0 in. level, Figure 12.3-13, column coordinates R.2-14.0, tools and small equipment can be decontaminated in the hot machine shop. Facilities in the hot machine shop include a bench space and drains. Also, there is a personnel (male/female) decontamination station at the 487 ft 0 in. level, column coordinates K.1-15.9. This facility contains sinks, showers, and a decontamination kit.

2. Turbine building

   Figure 12.3-5, columns H-9.5, el. 441 ft 0 in. identifies the turbine building decontamination area. Facilities include a monorail, curb, sink, shower, and drains.

3. Reactor building

   The head washdown area is shown in Figure 12.3-18 at column coordinates N-5.8 at the 606 ft 10 in. level and contains a curb and drain.

   The CRD room area, Figure 12.3-16, columns M-3.4, 501 ft 0 in. elevation contains a sink, monorail, bench, and storage vault.

4. The office of the RPM is located in the service building. Health Physics Craft Supervisors and HP Technicians are located in locations to provide for ready access by other plant workers and an area to generate and process records.

5. A hot machine shop and a hot instrument shop are provided in the radwaste building for work on contaminated equipment under controlled conditions. A HEPA-filtered vacuum system is installed in the hot machine shop to control airborne radioactivity while working on radioactive equipment. Portable HEPA-filtered vacuum systems are also available.

6. A laboratory complex is provided in the radwaste building consisting of a sample room, hot radiochemistry laboratory, and a counting room where radioactive samples will be qualitatively and/or quantitatively analyzed.

f. A protective clothing storage and distribution facility inside the protected area fence, but outside the power block.

Radiation Protection facilities at the Plant Support Facility/Emergency Operations Facility:

The Energy Northwest Plant Support Facility (PSF) is located 0.75 mile southwest of Columbia Generating Station. It is designed and equipped to provide emergency capabilities in support of Columbia Generating Station and for support of Columbia Generating Station during normal plant operations and maintenance. Support facilities important to HP include:

a. Portable radiation monitoring equipment calibration,
b. Radiological training.

The instrument calibration laboratory is located in the extreme northwest corner of the lower level. It contains an irradiation cubicle that is shielded on all sides and above by 2 ft of concrete. The cubicle entrance is protected by a labyrinth and a lockable gate. Larger open sources are stored and used in the cubicle. The shielded cubicle together with administrative controls such as procedures, radiation work permits, and surveys ensure that calibration laboratory operation will not result in radiation areas in surrounding spaces.

Calibrations are performed in accordance with approved procedures and are traceable, either directly or indirectly, to the National Institute of Standards and Technology (NIST). Available sources are listed in Table 12.2-12.

12.5.2.3 Equipment

Radiation Protection equipment, other than instrumentation, is described in the following:

a. Protective clothing and accessories are provided for personnel required to work in contaminated areas. Clothing requirements for a particular task or area are prescribed by Radiation Protection based on the actual or potential conditions. Available clothing includes, but is not limited to:

   1. Coveralls and laboratory coats,
   2. Gloves - rubber and/or cotton,
   3. Head covers,
   4. Foot protection, and
   5. Plastic suits - with or without supplied air.

b. Respiratory protection equipment is provided for personnel when it is not practicable to apply process controls or other engineering controls to control airborne radioactive contamination. The decision to use respiratory protection equipment is based on maintaining the TEDE ALARA. The respiratory protection program is conducted per the requirements of 10 CFR 20.1701, 1702, 1703, and 1704. Exposure is limited to derived air concentrations (DAC) and annual limit on intake (ALI) values specified in Appendix B, Table 1 of 10 CFR 20. Allowance is made for use of respiratory protective equipment, as prescribed in 20.1703, in limiting an individual's intake of airborne radioactive materials. Among the types of equipment used are:

   1. Full face air purifying respirators,
   2. Airline supplied full face masks (pressure demand regulated), and
   3. Self contained breathing apparatus (pressure demand regulated).

c. Air sampling equipment, in addition to the continuous air monitors, includes high and low volume portable air samplers, low volume constant air samplers, and air samplers with a self-contained power source. Collection media (filters) employed are capable of collecting particulate and radioiodine samples; and

d. Emergency equipment and supplies are maintained in lockers at strategic locations within the plant. These lockers are to be used for a rapid initial response and are not intended to provide the resources for a long term recovery operation. Equipment is stored for field team use at the EOF. Four-wheel drive vehicles, automobiles, and survey kits are available for use by the field team. Locations and types of emergency equipment are listed in the Emergency Plan.

Other than emergency supplies, the primary storage areas for radiation protection equipment are the two Radiation Protection control areas located in the service and radwaste buildings. Temporary storage facilities are set up in localized areas as required.

12.5.2.4 Instrumentation

Typical plant portable radiological instrumentation is described in Table 12.5-1. All of this instrumentation is calibrated at least semiannually when in use except the Condenser R-meter which is calibrated annually. Electronic calibrations of instrument components are performed using test equipment traceable to the NIST. Overall calibration of radiation measuring instruments is performed using radioactive standards traceable to a recognized source in a known, reproducible geometry. Calibration of low level radiation detection instruments is done with a pulse generator.

12.5.3 PROCEDURES

Section 12.1.3 described a process that was incorporated into the preparation and revision of plant procedures which provide a positive method of ensuring Radiation Protection input and ALARA consideration into radiation exposure related activities. The intent of this process is to incorporate the general guidance of appropriate regulatory guides plus the previous experience of power reactor radiation protection work into all applicable plant procedures.

12.5.3.1 Personnel Control Procedures

The Plant Procedures Manual contains the administrative procedures for control of access to radiation areas, high radiation areas, and very high radiation areas. This includes control of time spent within these areas by all plant workers. Basically, the procedures limit entry to these areas to time required for necessary operational maintenance and surveillance activities only. The primary tool used to ensure control and to maintain TEDE ALARA at Columbia

Generating Station is the Radiation Work Permit (RWP). All work performed at Columbia Generating Station in radiologically controlled areas is performed in accordance with an RWP, with the exception of specific activities identified in the implementing plant RWP procedure. The RWP provides current data on radiation levels within the area of interest, any restrictions on allowable work time, protective clothing and respiratory protective requirements, information on special tools or equipment needed, special radiation safety and personnel monitoring requirements, and any other special instructions or radiological hold points necessary. A section of the RWP is used to incorporate the criteria given in Regulatory Guides 8.2, 8.8, and 8.10 into each individual task, even though it has already been included in job procedures through the system for ALARA consideration described in Section 12.1.3. All RWPs require approval from Radiation Protection supervision prior to starting work. In addition, all personnel who perform activities covered by the RWP are required to read, understand, and document their understanding as specified by implementing plant procedures. There are two types of RWPs:

a. Specific RWP is issued for the performance of a particular task or function which falls outside the limitations imposed for General RWPs, and

b. General RWP may be issued to cover repetitive (routine) functions in areas where radiological conditions are known and stable.

In addition to the administrative controls used at Columbia Generating Station, certain physical controls are established which restrict entry to radiation areas, high radiation areas, and very high radiation areas. Radiation areas are posted as required by 10 CFR 20, and high radiation areas and very high radiation areas are locked or otherwise controlled as specified by this same regulation and Technical Specifications.

The plant security control system complements both the administrative and physical entry restraints by allowing access only to personnel with authorization to be in specific plant areas.

12.5.3.2 As Low As Is Reasonably Achievable Procedures

The procedures and processes described below are in addition to those described in Sections 12.1.3 and 12.5.3.1.

They have been developed to ensure that occupational radiation exposures are maintained ALARA. A primary goal of the Columbia Generating Station RPP is the control and reduction of individual and collective radiation exposures. This goal is achieved through training and comprehensive job planning and reviews as follows:

a. Training - ALARA training is required by plant procedures to be included in all applicable radiation training courses. The training applies to individuals whose duties require working with radioactive materials, entering radiation areas, directing the work of others in radiologically controlled areas, or planning work and preparing procedures for work in radiologically controlled areas. ALARA training is commensurate with the individual's duties, responsibilities, and radiation exposure potential. Workers who may enter restricted areas are given specific instructions about prenatal exposure risks to the developing embryo and fetus. This instruction includes the information provided by Regulatory Guide 8.13;

b. Job Planning and Reviews - Plant procedures specify that each job, involving exposure to radiation and/or radioactive materials, receives job planning and/or an ALARA review. The extent of the review is determined by an evaluation of the radiological risks involved. The preliminary job planning and review identifies the need for pre-job briefings and/or job planning meetings to coordinate work efforts and to familiarize personnel with the work and exposure reduction techniques. Pre-job planning may include the following:

1. Job history reviews,
2. Determination of radiological conditions,
3. Determination of exposure estimate, and
4. Interface with planners, schedulers, job supervisor, ALARA.

Additional ALARA reviews are performed by the Senior Site ALARA Committee. The Senior Site ALARA Committee has been developed to ensure participation by a range of plant personnel and provide for an appropriate level of management involvement and direction in ALARA issues. The Senior Site ALARA Committee serves as a review and advisory organization to the Plant General Manager on occupational radiation exposure to personnel.

Plant procedures provide requirements for committee membership, responsibilities, authority, and records of meetings and actions. The Senior Site ALARA Committee is responsible for the review of plant and departmental exposure goals and reviewing and assessing the effectiveness of the radiation exposure control program and the ALARA Program. The Senior Site ALARA Committee may create Working Groups to provide dose reduction methods for tasks which have a significant potential for dose reduction.

As part of pre-job review, ALARA job planning meetings are conducted when significant exposure savings or increased contamination control may result. The planning meetings may include the job supervisor, job planner, Radiation Protection supervision, HP technicians, and key workers. These meetings are used to ensure worker familiarity with procedures, work locations, RWP requirements, unusual hazards, and job-specific ALARA techniques to be employed. The use of mock-ups or dry runs may result from these meetings.

In addition to pre-job ALARA reviews, plant procedures have provisions for work-in-progress ALARA reviews and post-job ALARA reviews. These reviews are coordinated by Radiation Protection and may include discussions with individuals who performed the work, HP technicians, engineers, job supervisors, designers, or others as appropriate;

c. Remote handling tools and/or equipment - use of special tools/equipment for remote handling of radioactive equipment is factored into each applicable work activity; and

d. Exposure records are maintained in a manner that will allow Radiation Protection to tabulate and correlate exposure results to identify problem areas with individuals or activities.

Plant procedures are evaluated to determine the need for an ALARA review. An ALARA review is required for new procedures in which the actions take place in a radiologically controlled area or involve handling radioactive material.

An ALARA review is required for procedure revisions which:

a. Cause entry into a radiation area, high radiation area, or high-high radiation area,
b. Cause opening a contaminated or potentially contaminated system,
c. Significantly increase dose rates, or
d. Significantly increase exposure received.

Radiation Protection requirements, prerequisites, precautions, and ALARA considerations are incorporated into these procedures during the procedure review and approval process. In addition, an RWP is issued for those activities having radiological implications. Special activities such as inservice inspection (ISI), outage, and refueling are reviewed by the Senior Site ALARA Committee and Radiation Protection as appropriate. Special precautions, prerequisites, or requirements may be incorporated into the plant procedures based on these reviews.

12.5.3.3 Radiological Survey Procedures

A radiological survey is defined as an evaluation of radiological conditions and potential hazards. When appropriate, such an evaluation includes a physical survey (e.g., direct radiation and/or contamination surveys).

Routine surveys are conducted in various areas throughout the site to identify, monitor, and control sources of radiation and contamination. Routine surveys are performed on a frequency based on a consideration of potential radiological hazards, personnel occupancy and radiological stability. Included in routine surveys are the following daily checks:

- Check the plant area radiation monitoring system, and
- Check inservice portal, air and other continuously operating radiation/contamination monitors.

Abnormal changes in background and abrupt unexplained increases are investigated.

Nonroutine surveys are performed as the need and conditions dictate. The frequency and extent of these surveys should be determined based on historical data, on the conditions and activities taking place in the area, and on ALARA considerations.

Surveys of normally inaccessible, unoccupied areas are performed after each shutdown or prior to entry into these areas. Postings and survey record sheets are updated as conditions dictate.

Instructions relating to radiation surveys are provided in the Energy Northwest Radiation Protection Procedures.

12.5.3.4 Procedures for Radioactive Contamination Control

This section describes the bases and methods used for the monitoring and control of radioactive contamination on personnel, material, and surfaces.

a. Bases: The methods used for the monitoring and control of Columbia Generating Station licensed radioactive material are based on 10 CFR 20.1101(b), 20.1501, NRC Circular 81-07, and industry-accepted practices. Tools, equipment, and other items with detected quantities of licensed radioactive materials will not be unconditionally released. Detection levels will be based on the ALARA principle.

b. Methods: Personnel and materials will be surveyed in accordance with 10 CFR 20.1501. When physical surveys are performed, they will be conducted using industry-accepted, calibrated, detection instruments and with techniques that are appropriate to the level of risk. Other sections of Chapter 12 cover the selection criteria for contamination survey instruments, contamination monitoring facilities, protective clothing, contamination and radiation controls established through the RWP program, contamination monitoring surveys, ALARA with respect to contamination, and control of airborne radioactive material.

Instructions for monitoring and control of contamination are maintained in the Radiation Protection Procedures.

12.5.3.5 Procedures for Control of Airborne Radioactivity

Evaluation of airborne radioactivity concentrations is done procedurally by several methods. Routine airborne surveys consist of observing the continuous air monitors located in various areas of the plant and also the effluent monitors. These observations are supplemented by grab samples taken on a routine basis and by laboratory analysis of selected particulate and charcoal filters used on the continuous monitors. Special airborne surveys are made with portable samplers when a continuous air monitor indicates increases in airborne radioactivity or to evaluate conditions in a specific area or on a specific job.

The portable air sampling equipment consists of both high and low volume collectors with appropriate media for collecting particulates and radioiodines. These samplers are used for both spot evaluations by collection of grab samples and longer term evaluations by use of low volume samplers to collect over the period of a specific job or activity. Laboratory analysis is made of air samples for gross radioactivity and, where warranted, for specific isotope identification and quantification to determine and record airborne concentrations.

Selected numbers of the routine air samples collected are analyzed for specific isotope content to ensure that the DAC levels are not being approached. Special samples are taken for this purpose whenever unexplained increases occur on continuous air monitors or when gross activity levels indicate there is a potential for exceeding the value specified in 10 CFR 20, Appendix B, Table 1, Column 3 of any isotope present in the mixture.

Airborne radioactive iodine monitoring includes integrated sample collection and laboratory analysis plus portable sampling and analysis. Portable and stationary sampling encompasses iodine collection on charcoal and/or silver zeolite cartridges of nominal dimension of 2-in. disc diameter by 1-in. thickness at calibrated flow rates. Duration of sampling is determined by the anticipated ambient concentration levels whereas a nominal sampling period in excess of 5 minutes is selected to minimize sampling errors. Where gross noble gas concentrations exist, the sample cartridge may be purged in the laboratory with clean filtered air to minimize noble gas interferences. The cartridge will be sealed in a clean plastic bag and taken to the analytical laboratory counting room for analysis.

Areas are barricaded and posted as airborne radioactivity areas whenever average concentrations in that area exceed 0.3 DAC of the values specified in 10 CFR 20, Appendix B to Parts 20.1001-20.2401, Table 1, Column 3. The use of respiratory protection equipment is evaluated when a significant potential for an airborne hazard exists, or when entering an area of unmonitored, unknown airborne contamination.

Various methods for control and reduction of airborne activity are incorporated into Energy Northwest Radiation Protection Procedures, which include proper use of the ventilation system, use of specially designed equipment to collect radioactive airborne contaminants, methods for reducing and containing contamination to prevent it becoming airborne, and methods for cleanup of primary water prior to opening this system.

The respiratory protection program is designed to meet the requirements of 10 CFR 20.1701, 1702, 1703, and 1704. Procedures for fitting, training, maintenance, and testing of the respiratory protection equipment are included. All equipment is required to have appropriate National Institute for Occupational Health and Safety (NIOSH)/Mine Safety and Health Administration (MSHA) approval if available. Unless the requirements are met, the protection factors are not used. Unapproved equipment may be used in some instances where reduction of intake of radioactive material will result, but no protection factor is taken for its use. An example of this is use of charcoal cartridges in atmospheres where radioiodines are present to reduce the inhalation of these materials.

12.5.3.6 Radioactive Material Control Including Special Nuclear Materials (SNM)

Columbia Generating Station has implemented a program to ensure safe radioactive material control which include:

a. Procedures and training for receiving and shipping radioactive materials in accordance with 10 CFR 20.1906,

b. Procedures and training for storing licensed materials in accordance with 10 CFR 20.1801 and 1802,

c. Procedures and training for shielding, handling, and inventory control of sealed and unsealed radioactive sources and SNM,

d. Procedures and training for posting and/or labeling radioactive materials in accordance with 10 CFR 20 requirements,

e. Procedures and training for leak testing sealed radioactive sources in accordance with Technical Specifications, and

f. Procedures and training for disposal of all licensed radioactive materials in accordance with 10 CFR 20, 10 CFR 30, 10 CFR 40, 10 CFR 61, or 10 CFR 70.

g. Procedures and training for activities associated with dry storage cask loading and unloading of spent fuel and the operation of the Independent Spent Fuel Storage Installation for storage of spent fuel in accordance with 10 CFR 72.

Inherent in the above mentioned procedures is direction for handling liquid standard solutions used for calibration of plant instrumentation which include ventilation control, shielding, waste collection, contamination control, and monitoring.

Plant procedures assign the responsibility for control and monitoring of sealed and unsealed sources and byproduct materials to the RPM. The Engineering Manager is responsible for overall implementation of control of SNM. This is accomplished through a Nuclear Material Manager who is appointed in writing by the Engineering Manager. The Chemistry Technical Supervisor is responsible for minimization of radioactive waste and the preparation, offsite shipment, and disposal of radioactive materials and radwaste. Monitoring during handling of nuclear materials is provided by Radiation Protection, as appropriate.

12.5.3.7 Personnel Dosimetry Procedures

Section 12.5.2 describes the monitoring devices used to provide the primary legal records of exposure incurred by personnel and additional equipment used to backup and supplement this data. Records of radiation exposure are maintained for each individual for whom personnel monitoring is required by 10 CFR 20.1502. Reports of required monitoring are documented on NRC Form 5 or electronic media containing all the information required by NRC Form 5. Energy Northwest provides these individual radiation exposure records pursuant to the provisions of 10 CFR 19.13.

For monitored individuals, a determination of prior occupational dose is made per the requirements of 10 CFR 20.2104. This includes the dose received during the current year at Columbia Generating Station and other nuclear facilities. This exposure history is documented on NRC Form 4 or equivalent.

All individuals who are monitored for external radiation exposure are monitored for internally deposited radioactivity as follows:

a. Initial, performed prior to the individual entering any radiologically controlled area. Monitoring may be either quantitative (whole body count) or qualitative (passive monitoring).

b. When a worker formally declares pregnancy, a whole body count is performed.

c. At termination of employment at Columbia Generating Station, if the individual has been monitored for external radiation exposure. Monitoring may be either quantitative (whole body count) or qualitative (passive monitoring).

d. Whenever an individual causes an alarm of the passive whole body contamination monitors (portal monitors) and internal deposition is suspected, a quantitative bioassay (whole body count) is performed.

e. Special bioassays as determined by the RPM.

Energy Northwest monitors all adult workers who are likely to exceed 10% of the 10 CFR 20 annual occupational radiation exposure limits for adults. Radiation exposure monitoring for minors (individuals less than 18 years of age) is required if they are likely to exceed 0.1 rem deep dose equivalent in a year. Monitoring for declared pregnant women is required if they are likely to exceed 0.1 rem deep dose equivalent during the entire pregnancy. Monitoring is not required for visitors who enter a restricted area, since they are not likely to exceed 10% of the annual limit. However, confirmatory monitoring of visitors may be performed if directed by Radiation Protection supervision. The determination of whether an individual is likely to exceed 10% of the 10 CFR 20 limit, and thus require monitoring, is based on a prospective evaluation. An evaluation is not required for each individual, but is based on employees with similar job functions.

For internal exposure, monitoring is required if an adult worker is likely to receive, in 1 year, an intake in excess of 10% of the applicable 10 CFR 20 annual limit. The need for internal exposure monitoring of individuals is based on a prospective evaluation which will be updated whenever there is an indication that there has been significant fuel failure.

Since ^{90}Sr and ^{3}H are not measurable by whole body counting, in-vitro bioassay (urinalysis) will be performed when the plant radiation surveillance program indicates a potential need. All results obtained from in-vivo and in-vitro bioassay will be evaluated and become part of the individual's record, as appropriate.

Energy Northwest complies with the adult occupational dose limits identified in 10 CFR 20.1201. An individual is allowed to exceed these 10 CFR 20 exposure limits only in exceptional situations where the dose received is in accordance with the conditions of a planned special exposure, as specified in 10 CFR 20.1206. Records of planned special exposures are maintained and retained per the requirements of 10 CFR 20.2105. Written reports of planned special exposures are submitted to the NRC per the requirements of 10 CFR 20.2204.

In addition to the 10 CFR 20 dose limits, Energy Northwest uses administrative exposure hold points to maintain exposures ALARA. Plant procedures allow an individual to exceed an administrative hold point but, only if a prior approved dose extension is obtained.

Procedurally, DLR badges are processed for radiation workers semiannually at a minimum but may receive interim processing if an abnormal exposure is suspected. Pocket dosimeters and other auxiliary monitoring devices are used to maintain an estimate of an individual's dose during the interim period between processing of DLRs. The use of auxiliary monitoring devices as a permanent record of an individual's dose is restricted to times when DLRs are lost or damaged or give a false result. When a large discrepancy exists between the two devices, it must be established that the DLR is in error before the auxiliary monitoring device result is assigned as the permanent record.

Plant supervisors are notified of their assigned worker's exposure status and are responsible for maintaining these and their own exposure to ALARA and within specified limits.

Personnel dosimeters, that require processing to determine the radiation doses, are processed and evaluated in accordance with the requirements of the National Voluntary Laboratory Accreditation Program (NVLAP).

12.5.3.8 Radiation Protection Surveillance Program

The practices incorporated into the overall structure to ensure that the RPP is maintained at a high level and upgraded to meet new requirements and problems are the following:

a. Section 12.1.1 describes the organization structured to provide assurance that the ALARA policy is effective. It is also pointed out in this section that the plant RPP has several levels of review from a performance standpoint;

b. Section 12.1.3 describes the process for review of plant procedures for ALARA consideration;

c. The RWP program and other records previously described provide a valuable source of information and are used to determine where the occupational radiation exposures are occurring and as a means of review for possible methods of exposure reduction;

d. The Radiological Services Manager and his staff work on an individual and group basis with other plant organizations to determine what their principal sources of exposure are and to look for methods of reducing these exposures;

e. Procedures provide for routine maintenance, calibration, and testing of all radiation instrumentation and equipment. New equipment will be added as necessary for replacement and to supplement that existing. Written procedures are provided for use of equipment where required;

f. Plant facilities are routinely reviewed for possible improvements from a radiation protection standpoint. Section 12.1.3 describes several changes that have been incorporated into plant design for this purpose. Other considerations are additional shielding where practicable, improved ventilation control, additional equipment, and increased physical restriction;

g. The routine and special surveys previously described point out levels of radioactive contamination in plant areas. The Columbia Generating Station staff is committed to maintaining a clean plant and considers it routine procedure to reduce levels of contamination whenever such action will not result in an increase of occupational radiation exposure to personnel;

h. One aspect that is considered important and used in implementing the RPP is the incorporation of previous reactor and power reactor experience in this area. Previously successful methods, procedures, and equipment are used whenever possible; and

i. Training of all personnel who work in the plant in radiation safety practices is mandatory and given a high priority by Energy Northwest and Columbia Generating Station Management. The Training Manager, in conjunction with the Radiological Services Manager, is responsible for development of all training programs, including radiation safety indoctrination.

   Radiation Protection assists in this training by providing instructors for some phases. The degree of training provided each plant worker is dependent on his function and degree of responsibility; however, all radiation workers in the plant are provided training considered necessary or required for their position. The training programs provided are designed to meet the requirements of 10 CFR 19.12 and the guidance of Regulatory Guides 1.8, 8.8, 8.10, 8.27, and 8.29. Clarifications, elaborations, and exceptions in using the above mentioned regulatory guides are located in the Energy Northwest Procedures.

Table 12.5-1

Health Physics Instrumentation

Type	Number Available	Radiation Detected	Sensitivity Range
Ion chamber dose rate survey meter	9	Beta, gamma	0-5E4 mR/hr (gamma) 0-2.E4 mrad/hr (beta) cfx 5R/hr
High range ion chamber dose rate survey meter	2	Gamma	0-1.999E7 mR/hr
Telescoping dose rate survey meter	3	Gamma	0-1.0E3 R/hr
Neutron dose rate survey meter	2	Neutron	0.1-5.0E3 mrem/hr
Contamination survey meter with end window or pancake GM probe	20	Beta, gamma	0-5.0E4 cpm or 0-5.0E5 cpm
Contamination survey meter with scintillation detector	2	Alpha	0-5.0E5 cpm
			0
Condenser R-meter	1	Gamma	0-100 R
Direct reading pocket dosimeters	300	Gamma	0-999 rem
Direct reading pocket dosimeters	200	Gamma	Various ranges 0-500 mR