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SECTION 12

12.0 RADIATION PROTECTION

12.1 SHIELDING

12.1.1 Design Objectives

Radiation shielding is designed to perform three primary functions:

a. To ensure that during normal operation the radiation dose to operating personnel and to the general public is within the dose limits set forth in the Code of Federal Regulations Title 10, Part 20 (10CFR20) and 10CFR50.

b. To ensure that operating personnel are adequately protected in the event of a reactor accident and to preclude undue hazard to the general public.

c. To minimize activation of components so as to reduce personnel dose during refueling, maintenance, and inspection operations.

The shielding design is based on operating at the original design core power level of 2772 MWt with reactor coolant activity levels corresponding to 1 percent failed fuel. The shielding design is governed by the calculated source strengths and the desired accessibility of the particular space.

In addition to the radiation protection afforded by the shielding facilities provided for in the plant design, every effort was undertaken to achieve radiation dose levels which are as low as reasonably achievable.

License Amendment No. 278 increased core rated thermal power by 1.63% from 2772 MWt to 2817 MWt, based on the use of more accurate instrumentation for heat balance measurement. Based on an instrument uncertainty of 0.37% using the Caldon LEFM CheckPlus™ instrumentation, core power could be as high as 2827 MWt. The plant shielding evaluations utilized source terms based on 102% of 2772 MWt (2827 MWt). Therefore, no additional changes to shielding are required to meet current regulatory requirements as a result of the power uprate.

12.1.1.1 Radiation Exposure of Materials and Components

No regulations similar to those established for the protection of individuals exist for materials and components. Materials are selected on the basis that radiation exposure does not cause significant changes in their physical properties which could adversely affect their operation during the design life of the station. Materials for equipment required to operate under accident conditions are selected on the basis of the additional exposure received.

12.1.1.2 Specific Design Values

The material used for most of the station shielding is ordinary concrete and concrete block with a bulk density of about 143 lb/ft^3. Only in a very few instances is lead, steel, or water utilized as primary shielding materials.

12.1.1.3 Radiation Zoning and Access Control

Table 12.1-1 identifies the design radiation zones used for the Davis-Besse Station and the design accessibility limits associated with each zone. The following definitions only apply to personnel access requirements for differing radiation areas as used for the Davis-Besse station.

a) Unrestricted Area means an area where the access is neither limited nor controlled by the licensee.

b) Radiologically Controlled Area (RCA) means an area where the access is limited by licensee for the purpose of protecting individuals against undue risk from exposure to radiation and radioactive material.

c) Radiation Area means an area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 0.005 rem in one hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates. Radiation Areas are included in RCAs.

d) High Radiation Area means an area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 0.1 rem in one hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates. High Radiation Areas are included in RCAs.

e) Locked High Radiation Area means an area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 1.0 rem in one hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates. Locked High Radiation Areas are included in RCAs.

f) Very High Radiation Area means an area, accessible to individuals, in which radiation levels could result in an individual receiving an absorbed dose in excess of 500 rads in one hour at one meter from a radiation source or from any surface that the radiation penetrates. Very High Radiation Areas are included in RCAs.

In case of emergency, personnel will be allowed to use those escape routes which provide maximum safety and minimum exit time.

12.1.2 Design Description

Figures 12.1-1 through 12.1-9 give the radiation zones for normal station operating conditions and various other conditions including a loss-of-coolant accident. These dose rates are not expected to occur during normal operation since the design is for 1% failed fuel and the normally expected amount of failed fuel is 0.1%. These drawings provide scaled layouts of the entire operating station and indicate the expected radiation levels for the various station conditions.

Figures 1.2-10 and 1.2-11 provide cross sections of the stations at various locations, indicating the thicknesses of various floors and ceilings.

The anticipated average dose rates vary within each zone and with the various station operating conditions. In general, the dose rates in the vicinity of process equipment are normally expected to be at least an order of magnitude lower than indicated by the applicable radiation

zones. This is due to the extremely conservative assumptions made concerning the quantity of failed fuel. The direct dose at the site boundary is negligible both in the case of normal operation and under accident conditions. Therefore the shielding design is based on conservative assumptions ensuring adequate radiation protection to the general public and to the operating personnel as well.

12.1.2.1 General Descriptions and Evaluations

12.1.2.1.1 Shield Building

The walls of the concrete shield building are nominally 2-1/2 feet thick, and the dome is 2 feet thick.

Shield wall thicknesses are usually determined in 6-inch increments of ordinary concrete, such that the dose rate at the point in an area with the highest expected radiation levels (e.g. just outside the shield wall opposite the midpoint of a tank) is less than the dose limit for the zone in that area. The radiation zones are indicated in Figures 12.1-1 through 12.1-9.

Occupancy times are determined by the zone designations. A worker occupying an area for the designated occupancy time would receive a dose less than the limits of 10CFR20 since he would not be exposed to the maximum dose rate at all times.

Sources in the tanks and pipes, etc. are based on normal operation with 1 percent failed fuel. These sources are listed in Table 12.1-2 as the maximum curie activity of each component. For the corrosion products, the average levels were used, since the maximum values here represent the maximum possible crud levels and not the expected values. The crud levels given as maximums in Table 12.1-2 are not design conditions for shielding. For abnormal operation or maintenance, portable shielding will be used.

The shield building serves two main shielding purposes:

a. During operations, it shields the surrounding station structures and yard areas from radiation originating at the reactor vessel and the primary loop components. Together with additional shielding in the interior and in the containment vessel, the concrete shell is designed to reduce radiation levels outside the shell to below 0.25 mrem/hr in uncontrolled areas.

b. In the event of an accident, the shielding will reduce station and off-site radiation intensities, emitted directly from released fission products, to acceptable emergency levels. The concrete roof of the shield building effectively reduces contributions to skyshine.

12.1.2.1.2 Containment Vessel

The containment vessel consists of a 1-1/2 inch thick free standing steel shell with a 4-1/2 foot annulus between it and the shield building. During operation, some areas inside the containment vessel are inaccessible because of high dose rates. The reactor vessel, which is the major radiation source (during power operations), is surrounded by a heavy concrete biological shield. A concrete shield also surrounds equipment that carries reactor coolant water. The heavy concrete primary shield surrounding the reactor is designed to:

a. Attenuate the neutron flux to minimize the activation of components and structural steel.

b. Limit the radiation level after shutdown to permit access to the reactor coolant system equipment.

c. Reduce, in conjunction with the secondary shield, the radiation level from sources within the reactor vessel and the reactor coolant system to allow limited access to the containment during normal operation.

The primary shield consists of a minimum of 5 ft of reinforced concrete surrounding the reactor vessel. The cavity between the primary shield and the reactor vessel is air cooled to prevent overheating and dehydration of the concrete primary shield wall. The reactor vessel is insulated with stainless steel insulation to reduce heating even further.

Secondary shielding is provided to reduce the radiation from various components of the reactor coolant system to levels which allow limited access to the containment during normal operation and to supplement the primary shielding. The secondary shield consists of a minimum of 4 ft of reinforced concrete surrounding the reactor coolant piping, the reactor coolant pumps, the steam generators and the pressurizer. Nitrogen-16 and neutrons are the major sources of radioactivity in the reactor coolant during operation and determine the thickness of the secondary shield.

Inside the containment vessel, the refueling canal is filled with water during refueling operations. This shielding is designed for personnel protection during storage of activated reactor internals and for protection during movement of spent fuel elements.

The combined effect of the primary shield, the secondary shield, the containment vessel, and the shield building is designed to reduce the radiation level outside the shield building to less than 0.25 mrem/hr during full power operation.

12.1.2.1.3 Auxiliary Building

The function of the auxiliary building shielding is to protect personnel working near various system components, such as those in the makeup and purification system, the radioactive waste processing systems, the sampling systems, and the spent fuel pool cooling system. Controlled access to the auxiliary building is allowed during reactor operation. Each equipment compartment is individually shielded to reduce the radiation level inside it and inside adjacent compartments below predetermined levels. The amount of shielding required for a given compartment is thus determined by the access requirements of that compartment and the adjacent compartments. These accessibility requirements are shown on the radiation zone diagrams, Figures 12.1-1 through 12.1-9.

All radiation areas can be reached through service corridors which are entered from a radiologically controlled area access control point. Radiation doses are further limited by use of manually operated valves equipped with reach rods or valves that are remotely operated. Gauges and other instruments which need visual checking from time to time can be inspected from the corridors or on the local or central control boards. The different systems are isolated from each other by individually shielded chambers. Systems can be isolated for maintenance or repair with no significant radiation contribution from other systems in their shielded chambers.

Heavy concrete shielding is provided around the waste gas decay tanks wherever they are adjacent to accessible areas.

The counting rooms have been shielded or selectively located to reduce background radiation as much as possible. For the room containing the ventilation system, concrete block is used to shield against Radiation from the ventilation filters. The design radiation levels with 1% failed fuel, as depicted in Figures 12.1-1 through 12.1-6, at valve operating stations are indicated on the radiation zone drawings.

12.1.2.1.4 Control Room

The shielding for the control room is designed to ensure that the direct dose after a maximum hypothetical accident contributes an insignificant amount to the overall dose, which does not exceed 5 rem whole body over the duration of the accident. Under normal operating conditions, the control room is a radiation zone A area. Figure 12.1-10 shows a layout of the control room, and Figure 12.1-11 shows a scaled isometric view of the control room.

The control room ceiling is 1 foot thick, and the control room mechanical equipment room, which is above the control room, has a 1 foot 6 inch thick roof. Therefore, for shielding purposes the control room roof is 2 feet 6 inches thick.

12.1.2.1.5 Fuel Handling Area Shielding

Fuel handling area shielding is designed to facilitate the removal and transfer of spent fuel assemblies from the reactor vessel to the spent fuel facility. It is designed to protect personnel against the radiation emitted from the spent fuel and control rod assemblies.

The refueling canal above the reactor vessel is flooded to elevation 601.5 to provide a temporary water shield above the components being withdrawn from the reactor vessel. The water height is approximately 23.5 feet above the reactor Vessel flange. This height assures a minimum of 102 inches of water above the active portion of a withdrawn fuel assembly at its highest point of travel.

Upon removal from the reactor vessel, the fuel is moved to the spent fuel pool by the fuel transfer Mechanism, via the fuel transfer tube. Concrete and lead shielding is provided around the reactor and the steam generator for personnel protection during refueling. The spent fuel in the auxiliary building is permanently flooded to provide a minimum of 108 inches of water above a fuel assembly being withdrawn from the fuel transfer tube and rack. All spent fuel pool, cask pit, and transfer pit penetrations are more than 108 inches above the stored fuel assemblies so that a failure in any penetration cannot drain the pool to the point where there is insufficient shielding for the spent fuel.

The radiation dose rates in the accessible rooms adjacent to the spent fuel pool were re-analyzed when the pool was reracked in Cycle 13 to provide 1624 spent fuel storage locations. The analysis (Holtec Report No. HI-992294) concluded no changes to the radiation zone designations described in the USAR were required.

12.1.2.1.6 Turbine Building

The turbine building is fully accessible with radiologically controlled areas established as needed.

In the event of a Maximum Hypothetical Accident (MHA) or a steam generator tube leak, access to secondary systems would be radiologically controlled.

12.1.2.1.7 General Station Yard Areas

The radiation shielding design of the shield building and auxiliary buildings protects all station yard areas from excessive radiation exposure.

Penetrations through shield walls are made at angles to prevent streaming. Tight sleeves are installed around all penetration leads, or steel collars are installed where necessary. If possible, the piping or conduit makes a 90° bend after penetrating the shield, and bulk shielding is installed behind the bend to stop streaming.

12.1.2.1.8 Old Steam Generator Storage Facility

The Old Steam Generator Storage Facility (OSGSF) provides long term storage for two (2) Once Through Steam Generators (OTSGs), two (2) Reactor Coolant System (RCS) Hot Leg Piping sections and one (1) Reactor Vessel Closure Head (RVCH) (with Control Rod Drive Mechanisms and Service Support Structure), that have been removed from service. The facility is located outside of the Protected Area, but is within the boundary of the Owner Controlled Area (OCA), as shown in Figure 1.2-12.

The components with higher levels of fixed and/or loose surface contamination, the OTSGs, sections of Hot Leg Piping, and the RVCH, will be housed in a reinforced concrete shielded section of the structure with a labyrinth vestibule and lockable personnel access door. Building drains for the shielded portion of the structure will be collected and monitored.

The radiation source terms used to calculate the thicknesses of the concrete walls and concrete roof were conservatively determined using bounding survey dose rate values with a radionuclide activity distribution obtained from reactor coolant system radioactive crud samples. The total source term is 2.42 Ci. The isotopes in the source term which are greater than 100 millicuries (mCi) are Co-58-2.02 Ci, Cr-51 – 153 mCi and Nb-95 – 101 mCi.

The OSGSF has been designed such that the dose rates at the exterior of the facility are within station designated dose rate limits which are more restrictive than the dose rate limits of 10CFR20. The OTSG and RVCH storage areas have been designated as Zone D Radiologically Controlled Areas and the vestibule as a Zone C area, as shown in Figure 12.1-13.

The collapse of the Old Steam Generator Storage Facility (OSGSF), a non-Seismic Class I structure, due to a design basis earthquake would release loose surface contamination from the OTSG, RCS Hot Leg Piping sections and RVCH/CRDMs and Service Structure which are stored in the facility. This would result in an unmitigated dose at the nearest site boundary of 88 mrem (TEDE) whole body and 29 mrem thyroid. These doses are a small fraction of the acceptance criteria recommended for onsite low level radiological waste storage facilities. Generic Letter 81-38 recommends exposure limits for the public of less than a tenth of the 10CFR100 limits (i.e., a whole body dose of less than 2.5 rem and a thyroid dose of less than 30 rem from iodine exposure).

12.1.3 Source Terms

12.1.3.1 Process Equipment

The following information relating to radwaste source terms and estimated activities was written as part of the initial safety analysis and is now considered to be historical design information. Analysis for this section included an equilibrium refueling cycle of 277 effective full power days. Beginning with Cycle 5, the refueling cycle has been extended, resulting in an increase in the quantities of some isotopes and reduction of others. This change does not significantly affect plant releases or pose any difficulty in operating within current regulatory requirements.

Estimates of the total quantity of principal nuclides, with the exception of tritium, in the major pieces of process equipment that contain significant radioactivity are given in Table 12.1-2. Included are both the maximum and average values of the radioisotopic inventories expected to be present. When used in an estimate, values for the isotopic specific activities in reactor coolant are taken as the highest levels attained in the primary system during an equilibrium fuel cycle.

Activity levels in smaller pieces of process equipment (pumps, valves, etc.) and in interconnecting piping are not included in Table 12.1-2. They are determined, as needed, by assuming that the item under consideration is filled with fluid identical to the effluent from the nearest upstream major piece of equipment and in some instances, to the letdown from the primary system.

12.1.3.2 Process Piping

Normally, no process piping is field-run. When it is necessary to add piping, layouts for such piping are planned and drawn up, and then reviewed by the engineering staff for adequacy prior to approval. Once approved, the layouts are used in conjunction with existing process piping layouts and radiation zone diagrams to guide field personnel in the safe and proper installation of the new piping.

12.1.3.3 Spent Fuel Assembly

The source term for a spent fuel assembly is given in Table 15A-3.

12.1.3.4 Radioactivity Stored Outside

Sources of radioactivity normally stored outside the station buildings, other than the Dry Fuel Storage Facility are small amounts contained in the low level radwaste storage areas and in the borated water storage tank. There is no dose at the unrestricted area boundary (site boundary) associated with these areas.

The Dry Fuel Storage Facility (DFSF) is being used under the licensing provisions of 10CFR72, Subparts K and L. The radiological impact of this facility has been analyzed and the site boundary dose remains below the 10CFR100 guidelines.

See Section 12.1.2.1.8 for information on the Old Steam Generator Storage Facility (OSGSF).

12.1.3.5 Turbine Building Nitrogen-16 Gamma Dose

Since Davis-Besse utilizes a PWR, the contribution of N-16 to the gamma dose within the turbine building is negligible.

12.1.4 Area Monitoring

The area radiation monitors installed for station personnel protection are listed in Table 12.1-3. The areas selected and number of points monitored have been coordinated with the station radiation access control requirements to provide the operating personnel with a knowledge of the areas containing high radiation levels. In general, area radiation monitors are placed in areas where radiation levels could feasibly increase due to postulated occurrences. Refer to Section 11.4 for design and maintenance details common to all radiation monitors.

Each monitor channel consists of three remotely located, interconnected subsystems:

1. the detector, located in the monitored area,
2. the local readout unit, located in or adjacent to the monitored area, and
3. the control and power supply instrumentation, located in the control cabinet room.

In conjunction with the local indicators, each measurement provides a readout in the control cabinet room (an indicator on the respective control unit) and, in some cases, in the main control room. Automatic recording functions are performed on all of the area radiation measurements.

Alarm indication is provided at the local readout unit (audible and visual) and in the control room (visual and by the Fire Detection System). Alarm indication is provided for both high radiation levels and channel circuit failure. All area radiation monitors have "trip test function" switches. These switches allow for testing of the alarm and trip functions and meter response.

The detectors, Geiger-Mueller detectors and ionization chamber detectors, have a design range large enough to measure the dose races resulting from postulated fuel handling accidents and sufficient resolution to respond to normal operating conditions. The detector and its measurement circuitry are designed to read upscale when exposed to energy levels greater than their full scale values.

The power supply for each monitor is located in its respective control unit. The power is taken from an uninterruptable AC bus servicing the control cabinet room. The essential systems are powered from an essential AC instrumentation bus. All detectors are electrically isolated from the general station ground.

The two area monitors on the fuel handling area exhaust duct redirect the flow to the emergency ventilation system upon detecting a high activity level. These are the only area monitors performing an engineering function other than alarm initiation.

The new fuel area has received an exemption from the provisions of 10CFR70.24 requiring a criticality alarm system and, therefore, it does not have a criticality alarm. The spent fuel pool area complies with 10CFR50.68(b), therefore, it also does not have a criticality alarm system. (The spent fuel racks are designed for a maximum nominal initial enrichment of 5.05 percent U^{235} to meet the requirement for compliance with 10CFR50.68(b), the maximum nominal initial

enrichment is limited to 5.00% U^{235}.) The evaluations for both the 10CFR70.24 exemptions and 10CFR50.68(b) compliance took credit for area radiation monitors (RE 8417, RE 8418, RE8427 and RE8426).

The two containment vessel high range monitors (RE4596A & RE4596B) are designed to withstand the containment environment and are capable of operation during and after a LOCA.

In addition to the area radiation monitors, area radiation surveys are conducted on a weekly basis at designated locations in the office and support buildings, turbine building , auxiliary building, and containment vessel (when appropriate). If these routine surveys indicate that radiation levels are higher than expected the frequency of surveys is increased and investigations conducted to determine cause for the abnormal level.

Records of these area radiation surveys and investigations are maintained in accordance with accepted industry records retention practices and the conditions of the plant operating license.

Area monitors are source-checked on a regular basis. Calibration is performed with a portable unit with a point source which has a constant correlation to the standards of the National Institute of Standards and Technology (formerly National Bureau of Standards). Instruments found to be out of calibration are recalibrated. Area monitors are calibrated as required by Technical Specifications, if applicable. Other area monitors are calibrated each refueling interval.

12.1.5 Operating Procedure

12.1.5.1 Introduction

Keeping Total Effective Dose Equivalent (TEDE) as low as reasonably achievable is the responsibility of every individual. It is the primary responsibility of the Training Section to educate the individual. The Radiation Protection Section develops operating procedures to ensure that TEDE is kept as low as reasonably achievable during station operation and maintenance.

The Radiation Protection procedures used at Davis-Besse were based on USNRC documents including standards, regulatory guides, reports, NCRP and ICRP recommendations, and selected documentation relating to health physics. The present procedures reflect the experience gained by the operation of Davis-Besse.

12.1.5.2 Training

Radiological control training classes are given to radiation workers assigned to the station. Other workers are given a radiological control orientation session or remain in the company of qualified personnel. Radiation Protection procedures are available to all personnel.

12.1.5.3 Procedures

In order to minimize exposure of personnel to radiation, within the station as well as on the site property, proper signs, roping and tagging are used.

A radiation work permit procedure is used, which controls exposure to radiation when working in designated radiologically controlled areas.

A radiation surveillance program is carried out within the restricted and controlled areas so that the radiation levels are known in areas where personnel are working.

A personnel monitoring program, administered by the Radiation Protection Section, determines exposures to radiation workers at the station in compliance with 10CFR20.

Dosimetry is available for visitors who have to enter radiologically controlled areas.

12.1.6 Estimates of Dose

12.1.6.1 Design Doses

Peak external dose rates are considered as the maximum dose rates present in designated access zone area as given in Subsection 12.1.1.3 and shown in Figures 12.1-1 through 12.1-9, and Figure 12.1-13. These dose rates are not expected to occur during normal operation since the design is for 1% failed fuel and the normally expected amount of failed fuel is 0.1%. The highest dose rates in the station occurs in Zone E areas and in rooms containing equipment and piping handling highly radioactive fluids.

The annual doses received by the station operating personnel are well below the limits of 10CFR20 since the station shielding and radiation access control are based on maximum coolant activities. Since the average isotopic concentrations are considerably less than the maximum, the average dose is less.

The control room is a Zone A area and hence has a maximum design dose rate of 0.25 mrem/hr, corresponding to maximum coolant activity. Therefore, annual doses in these areas, considering occupancy factors, are within the limits of 10CFR20.

Annual unrestricted area boundary (site boundary) and population doses from containment vessel purges and turbine building sources are given in Tables 12.1-4 and 12.1-5 respectively.

The estimated peak external dose rate at the unrestricted area boundary (site boundary) from normal operating conditions is 0.00057 mrem/hr. The estimated peak external annual dose at the site boundary under normal operating conditions is 5 mrem.

The estimated number of yearly man-rem from the plant as designed is 200. This dose is entirely due to plant operations and routine maintenance.

Table 12.1-5A gives typical doses received by personnel in operating plants.

12.1.6.2 Method of Shielding Design

Shield wall thicknesses were determined by using basic shielding data and equations. The methods were taken from the "Reactor Shielding Design Manual," by T. Rockwell and "The Engineering Compendium of Radiation Shielding." Data is taken from the Table of the Isotopes "Reactor Physics Constants", ANL-5800, XDC-59-8-1-9 and other pertinent texts. Radiation sources were determined as indicated in Subsection 12.1.3.

Geometric models included standard cylindrical shapes of tanks, demineralizers, filters, pipes, etc. Special cases requiring different geometric assumptions were determined in individual cases. Sources were divided into energy bins for the different gamma energies.

Shielding in an area is done by considering the access requirement in an area and then by providing shielding for all sources of radiation in that area so as to achieve the desired accessibility.

12.1.6.3 Estimated Annual Dose

Table 12.1-5A gives typical doses received by personnel in operating plants. Those areas described in Table 12.1-5A that have radiation levels greater than 100 mrem/hr are classed as radiation Zone E areas, as described in Subsection 12.1.1.3, and listed in Table 12.1-1. These areas have administrative controls for access.

The estimated annual exposure from the plant as designed is 200 man-rem. This number is a typical value for relevant operating plants as shown in Table 12.1-5A.

TABLE 12.1-1

Radiation Zones

Zone Designation	Design Dose Rate (mrem/hr on a 40 hr/week basis)	Description
A	≤0.25	Uncontrolled, unlimited access.
B	≤2.5	Controlled, limited access (40 h/week).
C	≤ 15	Controlled, limited access for routine tasks (6-2/3 h/week).
D	≤ 100	Controlled, limited access for short periods (1 h/week).
E	> 100	Controlled occupancy for very short periods. Normally inaccessible.
E1	≤ 1000	Controlled occupancy for very short periods. Areas locked at 1000 mrem/hr.
E2	≤ 10,000	Occupancy only under controlled conditions, areas locked.
E3	≤ 100,000	Very likely will never be occupied, areas locked.
E4	>100,000	Will never be occupied, areas locked.

TABLE 12.1-2

Estimates of Principal Nuclides in Process Equipment

Isotope	Total Activity in Reactor Coolant Drain Tank (Ci)		Total Activity in Clean Waste Receiver Tank (Ci)		Total Activity in Clean Waste Monitor Tank (Ci)		Total Activity in Concentrates Storage Tank (Ci)	
	Max.	Ave.	Max.	Ave.	Max.	Ave.	Max.	Ave.
Corrosion Products								
Cr-51	9.15×10^{-3}	9.15×10^{-3}	0.15	0.15	1.63×10^{-5}	8.11×10^{-6}	0.11	0.11
Mn-54	1.05×10^{-3}	1.05×10^{-3}	1.65×10^{-2}	1.65×10^{-2}	1.87×10^{-6}	9.31×10^{-7}	1.25×10^{-2}	1.25×10^{-2}
Fe-55	3.62×10^{-2}	3.62×10^{-2}	0.57	0.57	6.42×10^{-5}	3.21×10^{-5}	0.44	0.44
Fe-59	1.05×10^{-3}	1.05×10^{-3}	1.65×10^{-2}	1.65×10^{-2}	1.87×10^{-6}	9.31×10^{-7}	1.25×10^{-2}	1.25×10^{-2}
Co-58	5.50×10^{-2}	5.50×10^{-2}	0.87	0.87	9.75×10^{-5}	4.88×10^{-6}	0.66	0.66
Co-60	2.95×10^{-4}	2.95×10^{-4}	4.63×10^{-3}	4.63×10^{-3}	8.79×10^{-7}	8.79×10^{-7}	3.52×10^{-3}	3.52×10^{-3}
Zr-95	7.20×10^{-2}	7.20×10^{-2}	1.14	1.14	1.29×10^{-4}	6.40×10^{-5}	0.86	0.86
Gaseous Radionuclides								
Kr-83m	0.72	7.15×10^{-2}	1.13×10^{-3}	1.13×10^{-4}	2.53×10^{-4}	2.53×10^{-5}	-	-
Kr-85m	3.80	0.38	5.97×10^{-3}	5.97×10^{-4}	1.35×10^{-3}	1.35×10^{-4}	-	-
Kr-85	7.82	0.79	1.23×10^{-2}	1.23×10^{-3}	2.77×10^{-3}	2.77×10^{-4}	-	-
Kr-87	2.08	0.21	3.27×10^{-3}	3.27×10^{-4}	7.36×10^{-4}	7.36×10^{-5}	-	-
Kr-88	6.65	0.67	1.05×10^{-2}	1.05×10^{-3}	2.36×10^{-3}	2.36×10^{-4}	-	-
Xe-131m	5.58	0.56	8.78×10^{-3}	8.78×10^{-4}	1.98×10^{-3}	1.98×10^{-4}	-	-
Xe-133m	7.05	0.71	1.11×10^{-2}	1.11×10^{-3}	2.50×10^{-3}	2.50×10^{-4}	-	-
Xe-133	617.39	61.74	0.98	9.71×10^{-2}	0.22	2.19×10^{-2}	-	-
Xe-135m	2.30	0.23	3.62×10^{-3}	3.62×10^{-4}	8.14×10^{-4}	8.14×10^{-5}	-	-
Xe-135	12.60	1.26	1.99×10^{-2}	1.99×10^{-3}	4.47×10^{-3}	4.47×10^{-4}	-	-
Xe-138	1.27	0.13	2.00×10^{-3}	2.00×10^{-4}	4.49×10^{-4}	4.49×10^{-5}	-	-

TABLE 12.1-2 (Continued)

Estimates of Principal Nuclides in Process Equipment

Isotope	Total Activity in Reactor Coolant Drain Tank (Ci)		Total Activity in Clean Waste Receiver Tank (Ci)		Total Activity in Clean Waste Monitor Tank (Ci)		Total Activity in Concentrates Storage Tank (Ci)	
	Max.	Ave.	Max.	Ave.	Max.	Ave.	Max.	Ave.
Ionic Radionuclides								
Rb-88	6.70	0.67	1.06	0.11	1.19×10^{-5}	8.79×10^{-7}	7.98×10^{-3}	7.98×10^{-4}
Sr-89	1.12×10^{-2}	1.12×10^{-3}	1.76×10^{-3}	1.76×10^{-4}	8.79×10^{-7}	8.79×10^{-7}	1.34×10^{-3}	1.34×10^{-4}
Sr-90	3.62×10^{-4}	3.62×10^{-5}	5.70×10^{-3}	5.70×10^{-4}	8.79×10^{-7}	8.79×10^{-7}	4.32×10^{-7}	4.32×10^{-8}
Sr-91	7.07×10^{-2}	7.07×10^{-3}	1.12×10^{-2}	1.12×10^{-3}	8.79×10^{-7}	8.79×10^{-7}	8.42×10^{-3}	8.42×10^{-4}
Sr-92	2.17×10^{-2}	2.17×10^{-3}	3.40×10^{-3}	3.40×10^{-4}	8.79×10^{-7}	8.79×10^{-7}	2.58×10^{-5}	2.58×10^{-6}
Y-90	2.51×10^{-2}	2.51×10^{-3}	3.94×10^{-2}	3.94×10^{-3}	8.79×10^{-7}	8.79×10^{-7}	2.99×10^{-4}	2.99×10^{-5}
Y-91	0.15	1.41×10^{-2}	0.23	2.21×10^{-2}	2.49×10^{-6}	8.79×10^{-7}	1.68×10^{-3}	1.68×10^{-4}
Mo-99	10.22	1.03	16.07	1.61	1.81×10^{-4}	9.05×10^{-6}	0.13	1.22×10^{-2}
I-131	8.01	0.81	1.26	0.13	9.49×10^{-4}	4.74×10^{-5}	9.54×10^{-3}	9.54×10^{-3}
I-132	5.58	0.56	0.89	8.88×10^{-2}	6.59×10^{-4}	3.30×10^{-6}	6.65×10^{-3}	6.65×10^{-3}
I-133	9.38	0.94	1.48	0.15	1.11×10^{-4}	5.54×10^{-5}	1.12×10^{-2}	1.12×10^{-2}
I-134	1.14	0.12	0.18	1.79×10^{-2}	1.35×10^{-4}	6.71×10^{-6}	1.36×10^{-3}	1.36×10^{-4}
I-135	4.72	0.48	0.75	7.41×10^{-2}	5.57×10^{-4}	2.79×10^{-5}	5.61×10^{-3}	5.61×10^{-4}
Cs-134	10.15	1.02	15.98	1.60	1.81×10^{-4}	9.05×10^{-4}	0.13	1.21×10^{-2}
Cs-136	1.84	0.19	2.89	0.29	3.25×10^{-5}	1.63×10^{-4}	2.19×10^{-2}	2.19×10^{-3}
Cs-137	31.49	3.15	49.52	4.96	5.58×10^{-4}	2.80×10^{-5}	0.38	3.75×10^{-2}
Cs-138	1.82	0.19	2.87	0.29	3.23×10^{-5}	1.62×10^{-4}	2.17×10^{-2}	2.17×10^{-3}
Ba-137m	29.01	2.91	45.62	4.57	5.14×10^{-4}	2.58×10^{-5}	0.35	3.46×10^{-2}
Ba-139	0.19	1.89×10^{-2}	2.97×10^{-2}	2.97×10^{-3}	8.79×10^{-7}	8.79×10^{-7}	2.25×10^{-4}	2.25×10^{-5}
Ba-140	1.40×10^{-2}	1.40×10^{-3}	2.20×10^{-3}	2.20×10^{-4}	8.79×10^{-7}	8.79×10^{-7}	1.67×10^{-5}	1.67×10^{-6}
La-140	5.58×10^{-3}	5.58×10^{-4}	8.78×10^{-4}	8.78×10^{-5}	8.79×10^{-7}	8.79×10^{-7}	6.65×10^{-6}	6.65×10^{-7}
Ce-144	1.29×10^{-3}	1.29×10^{-4}	2.02×10^{-4}	2.02×10^{-5}	8.79×10^{-7}	8.79×10^{-7}	1.53×10^{-6}	1.53×10^{-7}

TABLE 12.1-2 (Continued)

Estimates of Principal Nuclides in Process Equipment

Isotope	Total Activity in[1] Miscellaneous Waste Evaporator Storage Tank (Ci)		Total Activity in Waste Gas Surge Tank (Ci)		Total Activity in Waste Gas Decay Tank (Ci)		Total Activity in Filters (Ci)	
	Max.	Ave.	Max.	Ave.	Max.	Ave.	Max.	Ave.
Corrosion Products								
Cr-51	236.03	40.60	-	-	-	-	59.01	59.01
Mn-54	27.05	4.66	-	-	-	-	6.77	6.77
Fe-55	933.33	160.92	-	-	-	-	233.34	233.34
Fe-59	27.05	4.66	-	-	-	-	6.77	6.77
Co-58	1417.83	243.60	-	-	-	-	354.46	354.46
Co-60	7.65	1.32	-	-	-	-	1.92	1.92
Zr-95	1857.59	318.89	-	-	-	-	464.40	464.40
Gaseous Radionuclides								
Kr-83m	-	-	66.41	6.65	66.41	6.65	-	-
Kr-85m	-	-	353.10	35.31	353.10	35.31	-	-
Kr-85	-	-	2100.0	210.0	2100.0	210.0	-	-
Kr-87	-	-	193.08	19.31	193.08	19.31	-	-
Kr-88	-	-	618.74	61.88	618.74	61.88	-	-
Xe-131m	-	-	518.32	51.84	518.32	51.84	-	-
Xe-133m	-	-	654.37	65.44	654.37	65.44	-	-
Xe-133	-	-	57338.22	5733.83	57338.22	5733.82	-	-
Xe-135m	-	-	213.81	21.39	213.81	21.39	-	-
Xe-135	-	-	1172.68	117.27	1172.68	117.27	-	-
Xe-138	-	-	117.92	11.80	117.92	11.80	-	-

[1]Miscellaneous Waste Evaporators are no longer used to process radioactive fluids.

TABLE 12.1-2 (Continued)

Estimates of Principal Nuclides in Process Equipment

Isotope	Total Activity in[1] Miscellaneous Waste Evaporator Storage Tank (Ci)		Total Activity in Waste Gas Surge Tank (Ci)		Total Activity in Waste Gas Decay Tank (Ci)		Total Activity in Filters (Ci)	
	Max.	Ave.	Max.	Ave.	Max.	Ave.	Max.	Ave.
Ionic Radionuclides								
Rb-88	398.61	11.82	-	-	-	-	-	-
Sr-89	0.67	1.98×10^{-2}	-	-	-	-	-	-
Sr-90	2.16×10^{-4}	6.40×10^{-4}	-	-	-	-	-	-
Sr-91	4.21	0.13	-	-	-	-	-	-
Sr-92	1.29	3.83×10^{-2}	-	-	-	-	-	-
Y-90	1.50	4.43×10^{-2}	-	-	-	-	-	-
Y-91	8.36	0.25	-	-	-	-	-	-
Mo-99	608.24	18.02	-	-	-	-	-	-
I-131	476.85	14.13	-	-	-	-	-	-
I-132	332.17	9.85	-	-	-	-	-	-
I-133	558.05	16.54	-	-	-	-	-	-
I-134	67.62	2.01	-	-	-	-	-	-
I-135	280.50	8.32	-	-	-	-	-	-
Cs-134	603.82	18.02	-	-	-	-	-	-
Cs-136	109.25	3.25	-	-	-	-	-	-
Cs-137	1874.92	55.51	-	-	-	-	-	-
Cs-138	108.37	3.22	-	-	-	-	-	-
Ba-137m	1727.29	51.23	-	-	-	-	-	-
Ba-139	11.22	0.34	-	-	-	-	-	-
Ba-140	0.84	2.47×10^{-2}	-	-	-	-	-	-
La-140	0.34	9.85×10^{-3}	-	-	-	-	-	-
Ce-144	7.65×10^{-2}	2.28×10^{-3}	-	-	-	-	-	-

[1]Miscellaneous Waste Evaporators are no longer used to process radioactive fluids.

TABLE 12.1-2 (Continued)

Estimates of Principal Nuclides in Process Equipment

Isotope	Total Activity in Boric Acid Evaporator (Ci)		Total Activity in [1] Waste Evaporator (Ci)		Total Activity in Makeup Tank (Ci)		Total Activity in Letdown Cooler (Ci)	
	Max.	Ave.	Max.	Ave.	Max.	Ave.	Max.	Ave.
Corrosion Products								
Cr-51	0.14	0.14	236.03	49.45	3.14×10^{-2}	3.14×10^{-2}	6.27×10^{-4}	6.27×10^{-4}
Mn-54	1.53×10^{-2}	1.53×10^{-2}	27.05	5.67	3.59×10^{-3}	3.59×10^{-3}	7.18×10^{-5}	7.18×10^{-5}
Fe-55	0.53	0.53	933.33	195.99	0.13	0.13	2.48×10^{-3}	2.48×10^{-3}
Fe-59	1.53×10^{-2}	1.53×10^{-2}	27.05	5.67	3.59×10^{-3}	3.59×10^{-3}	7.18×10^{-5}	7.18×10^{-5}
Co-58	0.80	0.80	1417.83	296.69	0.19	0.19	3.77×10^{-3}	3.77×10^{-3}
Co-60	4.28×10^{-3}	4.28×10^{-3}	7.63	1.60	1.01×10^{-3}	1.01×10^{-3}	2.02×10^{-5}	2.02×10^{-5}
Zr-95	1.05	1.05	1857.59	388.39	0.25	0.25	4.94×10^{-3}	4.94×10^{-3}
Gaseous Radionuclides								
Kr-83m	1.04×10^{-4}	1.04×10^{-5}	-	-	52.46	5.25	4.90×10^{-2}	4.90×10^{-3}
Kr-85m	5.51×10^{-4}	5.51×10^{-5}	-	-	278.67	27.87	0.26	2.60×10^{-2}
Kr-85	1.14×10^{-3}	1.14×10^{-4}	-	-	573.73	57.38	0.54	5.36×10^{-2}
Kr-87	3.01×10^{-4}	3.01×10^{-5}	-	-	152.45	15.25	0.15	1.43×10^{-2}
Kr-88	9.64×10^{-4}	9.64×10^{-5}	-	-	488.12	48.82	0.46	4.56×10^{-2}
Xe-131m	8.10×10^{-4}	8.10×10^{-5}	-	-	258.62	25.87	0.39	3.83×10^{-2}
Xe-133m	1.03×10^{-3}	1.03×10^{-4}	-	-	326.43	32.65	0.49	4.83×10^{-2}
Xe-133	8.96×10^{-2}	8.96×10^{-3}	-	-	28619.59	2861.96	42.31	4.24
Xe-135m	3.34×10^{-4}	3.34×10^{-5}	-	-	106.44	10.65	0.16	1.58×10^{-2}
Xe-135	1.83×10^{-3}	1.83×10^{-4}	-	-	583.89	58.39	0.87	8.64×10^{-2}
Xe-138	1.84×10^{-4}	1.84×10^{-5}	-	-	58.74	5.88	8.69×10^{-2}	8.69×10^{-3}

[1] Waste Evaporator is no longer used to process radioactive fluids.

TABLE 12.1-2 (Continued)

Estimates of Principal Nuclides in Process Equipment

Isotope	Total Activity in Boric Acid Evaporator (Ci)		Total Activity in [1] Waste Evaporator (Ci)		Total Activity in Makeup Tank (Ci)		Total Activity in Letdown Cooler (Ci)	
	Max.	Ave.	Max.	Ave.	Max.	Ave.	Max.	Ave.
Ionic Radionuclides								
Rb-88	0.98	9.71×10^{-2}	970.96	14.39	22.94	2.30	0.46	4.59×10^{-2}
Sr-89	1.63×10^{-3}	1.63×10^{-4}	1.63	2.41×10^{-2}	3.84×10^{-2}	2.84×10^{-3}	7.67×10^{-4}	7.67×10^{-5}
Sr-90	5.26×10^{-5}	5.26×10^{-6}	5.26×10^{-2}	7.79×10^{-4}	1.25×10^{-3}	1.25×10^{-4}	2.49×10^{-5}	2.49×10^{-6}
Sr-91	1.03×10^{-2}	1.03×10^{-3}	10.25	0.16	0.25	2.43×10^{-2}	4.85×10^{-3}	4.85×10^{-4}
Sr-92	3.14×10^{-3}	3.14×10^{-4}	3.14	4.66×10^{-2}	7.41×10^{-2}	7.41×10^{-3}	1.49×10^{-3}	1.49×10^{-4}
Y-90	3.64×10^{-2}	3.64×10^{-3}	3.64	5.40×10^{-2}	8.59×10^{-2}	8.59×10^{-3}	1.72×10^{-3}	1.72×10^{-4}
Y-91	0.21	2.04×10^{-2}	20.36	0.31	0.49	4.81×10^{-2}	9.62×10^{-3}	9.62×10^{-4}
Mo-99	14.82	1.49	1481.62	21.94	35.00	3.50	0.70	7.00×10^{-2}
I-131	1.17	0.12	1161.56	17.21	27.44	2.75	0.55	5.49×10^{-2}
I-132	0.81	8.10×10^{-2}	809.14	12.00	19.12	1.92	0.39	3.83×10^{-2}
I-133	1.36	0.14	1359.35	20.14	32.12	3.22	0.65	6.43×10^{-2}
I-134	0.17	1.65×10^{-2}	164.71	2.45	3.90	0.39	7.79×10^{-2}	7.79×10^{-3}
I-135	0.69	6.84×10^{-2}	683.27	10.13	16.15	1.62	0.	3.23×10^{-2}
Cs-134	14.71	1.48	1470.83	21.94	34.75	3.48	0.70	6.95×10^{-2}
Cs-136	2.67	0.27	266.12	3.96	6.29	.63	0.13	1.26×10^{-2}
Cs-137	45.68	4.57	4567.10	67.61	107.89	10.89	2.16	0.22
Cs-138	2.64	0.27	263.96	3.92	6.24	0.63	0.13	1.25×10^{-2}
Ba-137m	42.08	4.21	4207.49	62.40	99.40	9.9	1.99	0.20
Ba-139	2.74×10^{-2}	2.74×10^{-3}	27.34	4.07	0.65	6.46×10^{-2}	1.30×10^{-2}	1.30×10^{-3}
Ba-140	2.03×10^{-3}	2.03×10^{-4}	2.03	3.01×10^{-2}	4.79×10^{-2}	4.79×10^{-3}	9.57×10^{-4}	9.57×10^{-5}
La-140	8.10×10^{-4}	8.10×10^{-5}	0.81	1.20×10^{-2}	1.92×10^{-2}	1.92×10^{-3}	3.83×10^{-4}	3.83×10^{-5}
Ce-144	1.87×10^{-4}	1.87×10^{-5}	0.19	2.77×10^{-3}	4.41×10^{-3}	4.41×10^{-4}	8.81×10^{-5}	8.81×10^{-6}

[1] Waste Evaporator is no longer used to process radioactive fluids.

TABLE 12.1-2 (Continued)

Estimates of Principal Nuclides in Process Equipment

Isotope	Total Activity in Pressurizer Quench Tank Cooler (Ci) Max.	Ave.
Corrosion Products		
Cr-51	2.09×10^{-4}	2.09×10^{-4}
Mn-54	2.40×10^{-5}	2.40×10^{-5}
Fe-55	8.27×10^{-4}	8.27×10^{-4}
Fe-59	2.40×10^{-5}	2.40×10^{-5}
Co-58	1.26×10^{-3}	1.26×10^{-3}
Co-60	6.72×10^{-6}	6.72×10^{-6}
Zr-95	1.65×10^{-3}	1.65×10^{-3}
Gaseous Radionuclides		
Kr-83m	1.64×10^{-2}	1.64×10^{-3}
Kr-85m	8.67×10^{-2}	8.67×10^{-3}
Kr-85	0.18	1.79×10^{-2}
Kr-87	4.75×10^{-2}	4.75×10^{-3}
Kr-88	0.16	1.52×10^{-2}
Xe-131m	0.13	1.28×10^{-2}
Xe-133m	0.17	1.61×10^{-2}
Xe-133	14.11	1.42
Xe-135m	5.25×10^{-2}	5.25×10^{-3}
Xe-135	0.29	2.88×10^{-2}
Xe-138	2.90×10^{-2}	2.90×10^{-3}
Ionic Radionuclides		
Rb-88	0.16	1.53×10^{-2}
Sr-89	2.56×10^{-4}	2.56×10^{-5}
Sr-90	8.27×10^{-6}	8.27×10^{-7}
Sr-91	1.62×10^{-3}	1.62×10^{-4}
Sr-92	4.94×10^{-4}	4.94×10^{-5}
Y-90	5.73×10^{-4}	5.73×10^{-5}
Y-91	3.21×10^{-3}	3.21×10^{-4}
Mo-99	0.24	2.34×10^{-2}
I-131	0.19	1.83×10^{-2}
I-132	0.13	1.28×10^{-2}
I-133	0.22	2.15×10^{-2}

TABLE 12.1-2 (Continued)

Estimates of Principal Nuclides in Process Equipment

Isotope	Total Activity in Pressurizer Quench Tank Cooler (Ci) Max.	Ave.
Ionic Radionuclides (Continued)		
I-134	2.60×10^{-2}	2.60×10^{-3}
I-135	0.11	1.08×10^{-2}
Cs-134	0.24	2.32×10^{-2}
Cs-136	4.20×10^{-2}	4.20×10^{-3}
Cs-137	0.72	7.20×10^{-2}
Cs-138	4.16×10^{-2}	4.16×10^{-3}
Ba-137m	0.67	6.63×10^{-2}
Ba-139	4.31×10^{-3}	4.31×10^{-4}
Ba-140	3.19×10^{-4}	3.19×10^{-5}
La-140	1.28×10^{-4}	1.28×10^{-5}
Ce-144	2.94×10^{-5}	2.94×10^{-6}

TABLE 12.1-2 (Continued)

Estimates of Principal Nuclides in Process Equipment

Isotope	Total Activity in Misc. Waste Drain Tank (Ci)		Total Activity in Detergent Waste Drain Tank (Ci)		Total Activity in Misc. Waste Monitor Tank (Ci)	
	Max.	Ave.	Max.	Ave.	Max.	Ave.
Corrosion Products						
Cr-51	236.03	13.95	263.03		5.50×10^{-2}	4.55×10^{-4}
Mn-54	27.05	1.60	27.05	Gross Ave.	6.30×10^{-3}	5.21×10^{-5}
Fe-55	933.33	55.29	933.33	Activity	0.22	1.80×10^{-3}
Fe-59	27.05	1.60	27.05	2.77×10^{-4}	6.30×10^{-3}	5.21×10^{-5}
Co-58	1417.83	83.70	1417.83		0.34	2.72×10^{-3}
Co-60	7.65	0.46	7.65		1.77×10^{-3}	1.47×10^{-5}
Zr-95	1857.59	109.57	1857.59		0.44	3.56×10^{-3}
Gaseous Radionuclides						
Kr-83m	-	-	-	-	-	-
Kr-85m	-	-	-	-	-	-
Kr-85	-	-	-	-	-	-
Kr-87	-	-	-	-	-	-
Kr-88	-	-	-	-	-	-
Xe-131m	-	-	-	-	-	-
Xe-133m	-	-	-	-	-	-
Xe-133	-	-	-	-	-	-
Xe-135m	-	-	-	-	-	-
Xe-135	-	-	-	-	-	-
Xe-138	-	-	-	-	-	-

TABLE 12.1-2 (Continued)

Estimates of Principal Nuclides in Process Equipment

Isotope	Total Activity in Misc. Waste Drain Tank (Ci)		Total Activity in Detergent Waste Drain Tank (Ci)		Total Activity in Misc. Waste Monitor Tank (Ci)	
	Max.	Ave.	Max.	Ave.	Max.	Ave.
Ionic Radionuclides						
Rb-88	136.96	4.06	74.62		8.90×10^{-4}	1.32×10^{-5}
Sr-89	0.23	6.80×10^{-3}	0.13		1.49×10^{-6}	3.30×10^{-7}
Sr-90	7.41×10^{-3}	2.20×10^{-4}	4.04×10^{-3}		3.30×10^{-7}	3.30×10^{-7}
Sr-91	1.45	4.29×10^{-2}	0.79		9.39×10^{-6}	3.30×10^{-7}
Sr-92	0.45	1.32×10^{-2}	0.25		2.88×10^{-6}	3.30×10^{-7}
Y-90	0.52	1.53×10^{-2}	0.28		3.33×10^{-6}	3.30×10^{-7}
Y-91	2.88	8.53×10^{-2}	1.57		1.87×10^{-5}	3.30×10^{-7}
Mo-99	208.99	6.19	113.86		1.36×10^{-3}	2.01×10^{-5}
I-131	163.85	4.86	89.26		5.34×10^{-3}	7.91×10^{-5}
I-132	114.14	3.39	62.18		3.73×10^{-3}	5.50×10^{-5}
I-133	191.74	5.69	104.46		6.23×10^{-3}	9.23×10^{-5}
I-134	23.24	0.69	12.66		7.55×10^{-4}	1.12×10^{-5}
I-135	96.38	2.86	52.51		3.13×10^{-3}	4.65×10^{-5}
Cs-134	207.47	6.19	113.03		1.35×10^{-3}	2.01×10^{-5}
Cs-136	37.54	1.12	20.45		2.44×10^{-4}	3.63×10^{-4}
Cs-137	644.21	19.08	350.95		4.19×10^{-3}	6.20×10^{-5}
Cs-138	37.24	1.11	20.29		2.42×10^{-4}	3.59×10^{-4}
Ba-137m	593.48	17.61	323.32		3.86×10^{-3}	5.74×10^{-5}
Ba-139	3.86	0.12	2.11		2.51×10^{-5}	3.73×10^{-7}
Ba-140	0.29	8.48×10^{-3}	0.16		1.86×10^{-6}	3.30×10^{-7}
La-140	0.12	3.39×10^{-3}	6.22×10^{-2}		7.41×10^{-7}	3.30×10^{-7}
Ce-144	2.63×10^{-2}	7.82×10^{-4}	1.44×10^{-2}		3.30×10^{-7}	3.30×10^{-7}

TABLE 12.1-2 (Continued)

Estimates of Principal Nuclides in Process Equipment

Isotope	Total Activity in Demineralizers (Ci)		Total Activity in Spent Resin Storage Tank (Ci)		Total Activity in Degasifier (Ci)	
	Max.	Ave.	Max.	Ave.	Max.	Ave.
Corrosion Products						
Cr-51	236.03	236.03	236.03	236.03	4.19×10^{-3}	4.19×10^{-3}
Mn-54	27.05	27.05	27.05	27.05	4.80×10^{-4}	4.80×10^{-4}
Fe-55	933.33	933.33	933.33	933.33	1.66×10^{-2}	1.66×10^{-2}
Fe-59	27.05	27.05	27.05	27.05	4.80×10^{-4}	4.80×10^{-4}
Co-58	1417.83	1417.83	1417.83	1417.83	2.52×10^{-2}	2.52×10^{-2}
Co-60	7.65	7.65	7.65	7.65	1.35×10^{-4}	1.35×10^{-4}
Zr-95	1857.59	1857.59	1857.59	1857.59	3.30×10^{-2}	3.30×10^{-2}
Gaseous Radionuclides						
Kr-83m	-	-	-	-	1.42	0.15
Kr-85m	-	-	-	-	7.52	0.76
Kr-85	-	-	-	-	15.48	1.55
Kr-87	-	-	-	-	4.13	0.42
Kr-88	-	-	-	-	13.17	1.32
Xe-131m	-	-	-	-	11.06	1.11
Xe-133m	-	-	-	-	13.96	1.40
Xe-133	-	-	-	-	1222.90	122.29
Xe-135m	-	-	-	-	4.56	0.46
Xe-135	-	-	-	-	24.96	2.50
Xe-138	-	-	-	-	2.52	0.26

TABLE 12.1-2 (Continued)

Estimates of Principal Nuclides in Process Equipment

Isotope	Total Activity in Demineralizers (Ci)		Total Activity in Spent Resin Storage Tank (Ci)		Total Activity in Degasifier (Ci)	
	Max.	Ave.	Max.	Ave.	Max.	Ave.
Ionic Radionuclides						
Rb-88	621.98	62.20	621.98	62.20	3.07	0.31
Sr-89	197.45	19.75	197.45	19.75	5.13×10^{-3}	5.13×10^{-4}
Sr-90	18.65	1.87	18.65	1.87	1.66×10^{-4}	1.66×10^{-5}
Sr-91	6.58	0.66	6.58	0.66	3.24×10^{-2}	3.24×10^{-3}
Sr-92	2.02	0.21	2.02	0.21	9.91×10^{-3}	9.91×10^{-4}
Y-90	2.33	0.24	2.33	0.24	1.15×10^{-2}	1.15×10^{-3}
Y-91	40.98	4.10	40.98	4.10	6.43×10^{-2}	6.43×10^{-3}
Mo-99	949.16	94.92	949.16	94.92	4.68	0.47
I-131	745.07	74.51	745.07	74.51	3.67	0.37
I-132	518.32	51.84	518.32	51.84	2.56	0.26
I-133	871.42	87.15	871.42	87.15	4.30	0.43
I-134	105.61	10.57	105.61	10.57	0.521	5.21×10^{-2}
I-135	437.33	43.74	437.33	43.74	2.16	0.22
Cs-134	4200.21	420.03	4200.21	420.03	4.65	0.47
Cs-136	200.80	20.08	200.80	20.08	0.85	8.41×10^{-2}
Cs-137	8900.0	890.0	8900.0	890.0	14.43	1.45
Cs-138	169.43	16.95	169.43	16.95	0.84	8.43×10^{-2}
Ba-137m	8900.0	890.0	8900.0	890.0	13.29	1.33
Ba-139	17.53	1.76	17.53	1.76	8.64×10^{-2}	8.64×10^{-3}
Ba-140	22.97	2.30	22.97	2.30	6.40×10^{-3}	6.40×10^{-4}
La-140	0.52	5.19×10^{-2}	0.52	5.19×10^{-2}	2.56×10^{-3}	2.56×10^{-4}
Ce-144	34.01	3.41	34.01	3.41	5.89×10^{-4}	5.89×10^{-5}

TABLE 12.1-2 (Continued)

Estimates of Principal Nuclides in Process Equipment

Isotope	Total Activity in Seal Return Cooler (Ci)		Total Activity in Decay Heat Cooler (Ci)		Total Activity in Fuel Pool Cooler (Ci)	
	Max.	Ave.	Max.	Ave.	Max.	Ave.
Corrosion Products						
Cr-51	1.05×10^{-4}	1.05×10^{-4}	9.41×10^{-3}	9.41×10^{-3}	2.30×10^{-3}	2.30×10^{-3}
Mn-54	1.20×10^{-6}	1.20×10^{-6}	1.08×10^{-3}	1.08×10^{-3}	2.63×10^{-4}	2.63×10^{-4}
Fe-55	4.14×10^{-4}	4.14×10^{-4}	3.72×10^{-2}	3.72×10^{-2}	9.09×10^{-3}	9.09×10^{-3}
Fe-59	1.20×10^{-6}	1.20×10^{-6}	1.08×10^{-3}	1.08×10^{-3}	2.63×10^{-4}	2.63×10^{-4}
Co-58	6.28×10^{-3}	6.28×10^{-4}	5.65×10^{-2}	5.65×10^{-2}	1.39×10^{-2}	1.39×10^{-2}
Co-60	3.36×10^{-4}	3.36×10^{-6}	3.03×10^{-4}	3.03×10^{-4}	7.40×10^{-5}	7.40×10^{-5}
Zr-95	8.23×10^{-4}	8.23×10^{-4}	7.40×10^{-2}	7.40×10^{-2}	1.81×10^{-2}	1.81×10^{-2}
Gaseous Radionuclides						
Kr-83m	8.16×10^{-3}	8.16×10^{-4}	0.74	7.34×10^{-2}	-	-
Kr-85m	4.34×10^{-2}	4.34×10^{-3}	3.90	0.39	-	-
Kr-85	8.92×10^{-2}	8.92×10^{-3}	8.03	0.81	-	-
Kr-87	2.38×10^{-2}	2.38×10^{-3}	2.14	0.22	-	-
Kr-88	7.59×10^{-2}	7.59×10^{-3}	6.84	0.69	-	-
Xe-131m	6.38×10^{-2}	6.38×10^{-3}	5.74	0.58	-	-
Xe-133m	8.05×10^{-2}	8.05×10^{-3}	7.24	0.73	-	-
Xe-133	7.06	0.71	634.59	63.46	-	-
Xe-135m	2.63×10^{-2}	2.63×10^{-3}	2.36	0.24	-	-
Xe-135	0.15	1.44×10^{-2}	12.95	1.30	-	-
Xe-138	1.45×10^{-2}	1.45×10^{-3}	1.31	0.14	-	-

TABLE 12.1-2 (Continued)

Estimates of Principal Nuclides in Process Equipment

Isotope	Total Activity in Seal Return Cooler (Ci)		Total Activity in Decay Heat Cooler (Ci)		Total Activity in Fuel Pool Cooler (Ci)	
	Max.	Ave.	Max.	Ave.	Max.	Ave.
Ionic Radionuclides						
Rb-88	7.65×10^{-2}	7.65×10^{-3}	6.89	0.69	1.69	0.17
Sr-89	1.28×10^{-4}	1.28×10^{-5}	1.15×10^{-2}	1.15×10^{-3}	2.81×10^{-3}	2.81×10^{-4}
Sr-90	4.14×10^{-6}	4.14×10^{-7}	3.73×10^{-4}	3.73×10^{-5}	9.10×10^{-5}	9.10×10^{-6}
Sr-91	8.08×10^{-4}	8.08×10^{-5}	7.27×10^{-2}	7.27×10^{-3}	1.78×10^{-2}	1.78×10^{-3}
Sr-92	2.47×10^{-4}	2.47×10^{-5}	2.23×10^{-2}	2.23×10^{-3}	5.44×10^{-3}	5.44×10^{-4}
Y-90	2.87×10^{-4}	2.87×10^{-5}	2.58×10^{-2}	2.58×10^{-3}	6.30×10^{-3}	6.30×10^{-4}
Y-91	1.61×10^{-3}	1.61×10^{-4}	0.15	1.45×10^{-2}	3.53×10^{-2}	3.53×10^{-3}
Mo-99	0.12	1.17×10^{-2}	10.50	1.05	2.57	0.26
I-131	9.15×10^{-2}	9.15×10^{-3}	8.24	0.83	2.02	0.21
I-132	6.38×10^{-2}	6.38×10^{-3}	5.74	0.58	1.41	0.15
I-133	0.11	1.08×10^{-2}	9.64	0.97	2.36	0.24
I-134	1.30×10^{-2}	1.30×10^{-3}	1.17	0.12	0.29	2.86×10^{-2}
I-135	5.39×10^{-2}	5.39×10^{-3}	4.85	0.49	1.19	0.12
Cs-134	0.12	1.16×10^{-2}	10.43	1.05	2.55	0.26
Cs-136	2.10×10^{-2}	2.10×10^{-3}	1.89	0.19	0.47	4.61×10^{-2}
Cs-137	0.36	3.60×10^{-2}	32.37	3.24	7.92	0.80
Cs-138	2.08×10^{-2}	2.09×10^{-3}	1.88	0.19	0.46	4.58×10^{-2}
Ba-137m	0.34	3.32×10^{-2}	29.82	2.99	7.29	0.73
Ba-139	2.16×10^{-3}	2.16×10^{-4}	0.20	1.94×10^{-2}	4.74×10^{-2}	4.74×10^{-3}
Ba-140	1.60×10^{-4}	1.60×10^{-5}	1.44×10^{-2}	1.44×10^{-3}	3.51×10^{-3}	3.51×10^{-4}
La-140	6.38×10^{-5}	6.38×10^{-6}	5.74×10^{-3}	5.74×10^{-4}	1.41×10^{-3}	1.41×10^{-4}
Ce-144	1.47×10^{-5}	1.47×10^{-6}	1.33×10^{-3}	1.33×10^{-4}	3.23×10^{-4}	3.23×10^{-5}

TABLE 12.1-3

Area Radiation Monitors

Detector Location	Measurement Range
Containment Vessel	
Containment vessel (2 detectors) (RE2387/RE2389)	0.1 mRem/hr to 10^7 mRem/hr
Containment vessel high range detectors (RE4596A & RE4596B)	1.0 Rad/hr to 10^8 Rad/hr
Auxiliary Building Elevation 545	
Maintenance work room 109	0.1 mRem/hr to 10^4 mRem/hr
Decay heat cooler room 113	0.1 mRem/hr to 10^4 mRem/hr
Waste gas valve access room 122	0.1 mRem/hr to 10^4 mRem/hr
Reactor coolant and radwaste sample room 106	0.1 mRem/hr to 10^4 mRem/hr
Emergency core cooling system rooms 105 and 115	0.1 mRem/hr to 10^4 mRem/hr
Misc. waste evap. room 116	0.1 mRem/hr to 10^4 mRem/hr
Miscellaneous waste detergent drain tank room 125	0.1 mRem/hr to 10^4 mRem/hr
Auxiliary Building Elevation 565	
Corridor 227	0.1 mRem/hr to 10^4 mRem/hr
Corridor 209	0.1 mRem/hr to 10^4 mRem/hr
Mech. penetration room No. 1	0.1 mRem/hr to 10^4 mRem/hr
Mech. penetration room No. 2	0.1 mRem/hr to 10^4 mRem/hr
Purification demineralizer valve room 212	0.1 mRem/hr to 10^4 mRem/hr
Reactor coolant makeup pump room 226	0.1 mRem/hr to 10^4 mRem/hr
Auxiliary Building Elevation 585	
Mech. penetration room No. 3	0.1 mRem/hr to 10^4 mRem/hr
Mech. penetration room No. 4	0.1 mRem/hr to 10^4 mRem/hr
Fuel handling area space 300	0.1 mRem/hr to 10^4 mRem/hr
Hot Shop Room 302	0.1 mRem/hr to 10^7 mRem/hr

TABLE 12.1-3 (Continued)

Area Radiation Monitors

Detector Location	Measurement Range
Auxiliary Building Elevation 603	
Corridor 411	0.1 mRem/hr to 10^4 mRem/hr
Hot instrument shop room 406	0.1 mRem/hr to 10^4 mRem/hr
Hot laboratory room 424	0.1 mRem/hr to 10^4 mRem/hr
Containment personnel lock area 426	0.1 mRem/hr to 10^4 mRem/hr
Containment equipment hatch	0.1 mRem/hr to 10^4 mRem/hr
Spent fuel pool area 224	0.1 mRem/hr to 10^7 mRem/hr
Auxiliary Building Elevation 623	
Control room	0.1 mRem/hr to 10^4 mRem/hr
Control cabinet room	0.1 mRem/hr to 10^4 mRem/hr
Fuel handling area exhaust	0.1 mRem/hr to 10^7 mRem/hr
Turbine Building Elevation 565	
Condensate polishing demineralizer holdup tanks room 248	0.1 mRem/hr to 10^4 mRem/hr
Turbine Building Elevation 585	
Condensate polishing demineralizer backwash receiving tank room 334	0.1 mRem/hr to 10^4 mRem/hr
Turbine Building Elevation 603	
Condensate polishing demineralizer Area 432	0.1 mRem/hr to 10^4 mRem/hr

TABLE 12.1-4

Dose from Containment Vessel Purges

PATH	% FAILED FUEL	ANNUAL SITE BOUNDARY DOSE (mrem) THYROID	WHOLE BODY	POPULATION DOSE (Man-rem)
Containment vessel purges (50,000 cfm filtered)	0.1	7.4×10^{-3}	0.027	0.21
Containment vessel purges (50,000 cfm filtered)	1.0	7.4×10^{-2}	0.270	2.1

TABLE 12.1-5

Dose from Air Ejectors

PATH	% FAILED FUEL	PRIMARY TO SECONDARY LEAKAGE GPD	ANNUAL SITE BOUNDARY DOSE (mRem) THYROID	WHOLE BODY	TOTAL POPU-LATION DOSE (man-rem)
Turbine Building					
Steam jet air ejector and ventilation system	0.1	100	1.7×10^{-3}	0.18	1.37
Steam jet air ejector and ventilation system	0.1	720	1.2×10^{-2}	1.3	9.9
Steam jet air ejector and ventilation system	1.0	100	1.7×10^{-2}	1.8	13.7

TABLE 12.1-5A

Typical Doses to Personnel in Other Operating Plants at Date of Davis-Besse License

Annual man-rem:

Yankee Rowe	202
San Onofre	110
*	52
*	75
*	126
*	62
*	106
*	285
*	1000
Indian Point 2	1120
Connecticut Yankee	600

Inservice Inspection Doses:

Component	man-rem
Reactor head	1.1
Reactor vessel	7.1
Pressurizer	1.66
Steam generator	2.275
Reactor coolant piping	3.235
TOTAL	15.37

*Unidentified reactors given in “Water Reactor Plant Contamination and Decontamination Requirements” by D. H. Charlesworth, American Power Conference, 1971.

TABLE 12.1-5A (Continued)

Typical Doses to Personnel in Other Operating Plants at Date of Davis-Besse License

Component Radiation Dose Rates:	Reactor A	Reactor B	Reactor C
Primary loop piping (outside)	200-350 mrem/hr	50 mrem/hr	
Primary loop piping (inside)	1-13 rem/hr		
Steam generator plenum	(inside shutdown) 15 rem/hr	(outside operating) 10 rem/hr	
Reactor vessel head (outside)	400-500 mrem/hr		
Reactor vessel head (inside)	15 rem/hr		
Reactor vessel nozzles (outside)	15-20 mrem/hr		
Reactor vessel nozzles (inside)	5 rem/hr		
Seal water filter	165 rem/hr	140 mrem/hr	900 mrem/hr
Letdown ion exchanger	> 1000 rem/hr	800 mrem/hr	100 rem/hr
Letdown filter	600 rem/hr	1200 mrem/hr	50 rem/hr
Primary drain tank	100-500 mrem/hr	35 mrem/hr	
Liquid waste monitor tank		2 mrem/hr	20 mrem/hr
Compactor		3 mrem/hr	
Waste gas flash tank		19 mrem/hr	
Radwaste process filter		85 mrem/hr	
Liquid waste holdup tank		120 mrem/hr	
Liquid waste holdup pump (area)		8 mrem/hr	
Charging pump (area)		5 mrem/hr	
Seal water heat exchanger		50 mrem/hr	
Radiochemistry lab drain tank (area)		140 mrem/hr	
Top of steam generator		1 mrem/hr	
Side of steam generator		5 mrem/hr	
Side of pressurizer		90 mrem/hr	
Top of pressurizer		30 mrem/hr	
Reactor coolant pump		33 mrem/hr	
RHR heat exchanger	500 mrem/hr	120 mrem/hr	30-50 mrem/hr
RHR pump (area)		40 mrem/hr	
Containment (outside secondary shield)			50-200 mrem/hr
Boric acid drums	200 mrem/hr	200 mrem/hr	
Shutdown cooling (area)	100 mrem/hr		
Spent fuel pool heat exchanger (area)	30 mrem/hr		
Waste gas storage tank			100 mrem/hr

Removed in Accordance with RIS 2015-17

DAVIS-BESSE NUCLEAR POWER STATION

RADIATION ZONES FOR NORMAL
OPERATION- ELEVATION 545'-0"
A-11
FIGURE 12.1-1

REVISION 26
JUNE 2008

Removed in Accordance with RIS 2015-17

DAVIS-BESSE NUCLEAR POWER STATION

RADIATION ZONES FOR NORMAL OPERATION
ELEVATION 565

A12

FIGURE 12.1-2

REVISION 20
DECEMBER 1996

Removed in Accordance with RIS 2015-17

DAVIS-BESSE NUCLEAR POWER STATION

RADIATION ZONES FOR
NORMAL OPERATION - ELEVATION 585
A-13
FIGURE 12.1-3

REVISION 29

DECEMBER 2012

DB 08-30-12 DFN=/JJ/RASDGN/A13.DGN/CIT

Removed in Accordance with RIS 2015-17

DAVIS-BESSE NUCLEAR POWER STATION

RADIATION ZONES FOR NORMAL OPERATION
ELEVATION 603'-0"
A-14
FIGURE 12.1-4

REVISION 30
OCTOBER 2014

DB 02-24-14 DFN=/J/RASDGN/A14.DGN/CIT

Removed in Accordance with RIS 2015-17

DAVIS-BESSE NUCLEAR POWER STATION

RADIATION ZONES FOR NORMAL OPERATION
ELEVATION 623
A-15
FIGURE 12.1-5

REVISION 29

DECEMBER 2012

DB 08-31-12 DFN=/J/RASDGN/A15.DGN/C1T

Removed in Accordance with RIS 2015-17

DAVIS-BESSE NUCLEAR POWER STATION

RADIATION ZONES FOR NORMAL
OPERATION - ELEVATION 643'
A-16
FIGURE 12.1-6

REVISION 31

OCTOBER 2016

DB 03-21-16 DFN=/G/ARCH/A16.DGN/TIF

SPACE AND ACCESS TABULATION OF RADIATION ZONES

SPACE NO.	BUILDING	GENERAL ELEVATION	SPACE DESIGNATION	PLANT CONDITIONS N	LOCA	HS	R	M	ST	SPACE SHOWN ON DWG NO.	REMARKS
100	AUXILIARY BLDG	545'-0"	SEE NOTE 5	E1	E	E1	E1	E1	E1	A-11	
101	AUXILIARY BLDG	545'-6"	SEE NOTE 5	E1	E1	E1	E1	E1	E1	A-11	
102	AUXILIARY BLDG	545'-0"	SEE NOTE 5	E4	E4	E4	E4	E4	E4	A-11	
103	AUXILIARY BLDG	545'-0"	SEE NOTE 5	E2	E	E2	E2	E2	E2	A-11	
104	AUXILIARY BLDG	545'-0"	SEE NOTE 5	E1	E4	E1	E1	E1	E1	A-11	
105	AUXILIARY BLDG	545'-0"	SEE NOTE 5	E1	E4	E1	*E	E1	E1	A-11	*SEE NOTE 3
106	AUXILIARY BLDG	545'-0"	SEE NOTE 5	D	E4	D	D	D	D	A-11	
107	AUXILIARY BLDG	545'-0"	SEE NOTE 5	E1	E	E	E	C	E	A-11	
108	AUXILIARY BLDG	545'-0"	SEE NOTE 5	E1	E	E	E	C	E	A-11	
109	AUXILIARY BLDG	545'-0"	SEE NOTE 5	E1	E1	E1	E1	E1	E1	A-11	
110	AUXILIARY BLDG	545'-0"	SEE NOTE 5	C	E2	C	C	C	C	A-11	
111	AUXILIARY BLDG	545'-0"	SEE NOTE 5	E1	E	E	E	C	E	A-11	
112	AUXILIARY BLDG	545'-0"	SEE NOTE 5	C	C	C	C	C	C	A-11	
113	AUXILIARY BLDG	545'-0"	SEE NOTE 5	E1	E4	E2	E	E2	E2	A-11	
114	AUXILIARY BLDG	545'-0"	SEE NOTE 5	D	D	D	D	D	D	A-11	
115	AUXILIARY BLDG	545'-0"	SEE NOTE 5	E1	E4	E1	E	E1	E1	A-11	
116	AUXILIARY BLDG	545'-0"	SEE NOTE 5	E1	E	E	E	C	E	A-11	
117	AUXILIARY BLDG	545'-0"	SEE NOTE 5	E1	E	E	E	C	E	A-11	
118	AUXILIARY BLDG	545'-0"	SEE NOTE 5	D	D	D	D	D	D	A-11	
119	AUXILIARY BLDG	545'-0"	SEE NOTE 5	E2	E	E	E	C	E	A-11	
120	AUXILIARY BLDG	545'-0"	SEE NOTE 5	E1	E	E	E	E*	E	A-11	*SEE REMARKS FOR RM 121
121	AUXILIARY BLDG	545'-0"	SEE NOTE 5	E1	E	E	E	E*	E	A-11	*THIS ROOM WILL BE A ZONE "C" IF ALL TANKS ARE EMPTY
122	AUXILIARY BLDG	545'-0"	SEE NOTE 5	D	D	D	D	D	D	A-11	
123	AUXILIARY BLDG	545'-0"	SEE NOTE 5	E1	E	E	E	C	E	A-11	
124	AUXILIARY BLDG	545'-0"	SEE NOTE 5	E1	E	E	E	C	E	A-11	
125	AUXILIARY BLDG	545'-0"	SEE NOTE 5	D	E	D	D	D	D	A-11	
126	AUXILIARY BLDG	545'-0"	SEE NOTE 5	E2	E	E	E	C	E	A-11	
—	AUXILIARY BLDG	545'-0"	SEE NOTE 5	C	C	C	C	C	C	A-11	
104A	AUXILIARY BLDG	545'-0"	SEE NOTE 5	D	E2	D	D	D	D	A-11	
—	AUXILIARY BLDG	545'-0"	SEE NOTE 5	C	C	C	C	C	C	A-11	
—	AUXILIARY BLDG	555'-0"	SEE NOTE 5	D	D	D	D	D	D	A-11	
—	AUXILIARY BLDG	555'-0"	SEE NOTE 5	C	C	C	C	C	C	A-11	
—	AUXILIARY BLDG	555'-0"	SEE NOTE 5	C	C	C	C	C	C	A-11	
										A-11	
106A	AUXILIARY BLDG	545'-0"	SEE NOTE 5	E1	E4	E1	E1	C	E1	A-11	
117A	AUXILIARY BLDG	545'-0"	SEE NOTE 5	D	E	D	D	D	D	A-11	
104A	AUXILIARY BLDG	545'-0"	SEE NOTE 5	E1	E1#	E1#	E1#	E1#	E1#	A-11	# ASSUMED TO BE SAME AS SURROUNDING ROOMS
110A	AUXILIARY BLDG.	555'-0"	SEE NOTE 5	D	D#	D#	D#	D#	D#	A-11	# ASSUMED TO BE SAME AS SURROUNDING ROOMS
113A	AUXILIARY BLDG	555'-0"	SEE NOTE 5	E2	E2	E2	E2	E2	E2	A-11	
200	AUXILIARY BLDG	565'-0"	SEE NOTE 5	D	E	D	D	D	D	A-12	
201	AUXILIARY BLDG	565'-0"	SEE NOTE 5	D	E	D	D	D	D	A-12	
202	AUXILIARY BLDG	565'-0"	SEE NOTE 5	D	E	D	D	D	D	A-12	
203	AUXILIARY BLDG	565'-0"	SEE NOTE 5	D	E	D	D	D	D	A-12	
204	AUXILIARY BLDG	565'-0"	SEE NOTE 5	D	E	D	D	D	D	A-12	
205	AUXILIARY BLDG	565'-0"	SEE NOTE 5	E2	E	E2	E2	D	E2	A-12	
206	AUXILIARY BLDG	565'-0"	SEE NOTE 5	C	E	C	C	C	C	A-12	
207	AUXILIARY BLDG	565'-0"	SEE NOTE 5	D	E	D	D	D	D	A-12	
208	AUXILIARY BLDG	565'-0"	SEE NOTE 5	E1	E4	E1	E1	E1	E1	A-12	TO OBTAIN THIS ZONE, TEMPORARY SHIELDING WILL BE NEEDED DURING OCCUPANCY
209	AUXILIARY BLDG	565'-0"	SEE NOTE 5	C	E4	C	*C	C	C	A-12	*SEE NOTE 4
210	AUXILIARY BLDG	565'-0"	SEE NOTE 5	E4	E4	E4	E4	E4	E4	A-12	
211	AUXILIARY BLDG	565'-0"	SEE NOTE 5	E1	E4	E1	E1	D*	E1	A-12	*TEMPORARY SHIELDING AROUND OTHER PIPES WILL BE NEEDED
212	AUXILIARY BLDG	565'-0"	SEE NOTE 5	D	E4	D	D	D	D	A-12	
102	AUXILIARY BLDG	545'-0"	SEE NOTE 5	E1	E	E1	E1	D	E1	A-12	
214	CTMT BLDG	565'-0"	SEE NOTE 5	E2	E4	E2	B	B*	E2	A-12	* DURING REFUELING ONLY
215	CTMT BLDG	565'-0"	SEE NOTE 5	E2	E4	E2	C*	C*	E2	A-12	* SEE NOTE FOR SPACE NO. 214
216	CTMT BLDG	565'-0"	SEE NOTE 5	E3	E4	E3	E3	E3	E3	A-12	
217	CTMT BLDG	565'-0"	SEE NOTE 5	D	E4	D	D	D	D	A-12	
218	CTMT BLDG	565'-0"	SEE NOTE 5	E3	E4	E3	E3	E3	E3	A-12	
219	CTMT BLDG	563'-6"	SEE NOTE 5	E3	E4	E3	E3	C*	E3	A-12	* DRAINED

SPACE NO.	BUILDING	GENERAL ELEVATION	SPACE DESIGNATION	PLANT CONDITIONS N	LOCA	HS	R	M	ST	SPACE SHOWN ON DWG NO.	REMARKS
220	CTMT BLDG	565'-0"	SEE NOTE 5	D	E4	D	D	D	D	A-12	
221	AUXILIARY BLDG	576'-6"	SEE NOTE 5	C*	E	C	D	C	C	A-12	# "D" IF MOVING FUEL
222	AUXILIARY BLDG	583'-3"	SEE NOTE 5	E3	E	E3	E3	C	E3	A-12	
223	AUXILIARY BLDG	563'-6"	SEE NOTE 5	E3	E	E3	E3	C	E3	A-12	
224	AUXILIARY BLDG	563'-6"	SEE NOTE 5	E3	E	E3	E3	C	E3	A-12	A ZONE "C" DURING MAINTENANCE PROVIDED NO SPENT FUEL IN THE POOL
225	AUXILIARY BLDG	565'-0"	SEE NOTE 5	E1	E	E1	*D	D	E1	A-12	*SEE NOTE 3
—	CTMT BLDG	565'-0"	SEE NOTE 5	E1	E4	E1	B/D	*B/D	E1	A-12	* DURING REFUELING
227	AUXILIARY BLDG	565'-0"	SEE NOTE 5	C	E4	C	C	C	C	A-12	
228	AUXILIARY BLDG	565'-0"	SEE NOTE 5	E4	E4	E4	E4	D	E4	A-12	
119	AUXILIARY BLDG	545'-0"	SEE NOTE 5	E1	E	E1	E1	D	E1	A-12	
230	AUXILIARY BLDG	565'-0"	SEE NOTE 5	D	E	D	D	D	D	A-12	
231	AUXILIARY BLDG	565'-0"	SEE NOTE 5	D	E	D	D	D	D	A-12	
232	AUXILIARY BLDG	565'-0"	SEE NOTE 5	D	—	D	D	B*	D	A-12	* THE MAJOR PORTION OF THE DOSE WILL COME FROM FLUID IN THE PIPES
233	AUXILIARY BLDG	565'-0"	SEE NOTE 5	E4	E4	E4	E4	E4	E4	A-12	
234	AUXILIARY BLDG	565'-0"	SEE NOTE 5	E1	E	E	E1	D	E1	A-12	
235	AUXILIARY BLDG	565'-0"	SEE NOTE 5	E1	E	E1	E1	D	E1	A-12	
236	AUXILIARY BLDG	565'-0"	SEE NOTE 5	D/E1	E4	D/E1	D/E1	D/E1	D/E1	A-12	SEE REMARKS FOR RM 208
226A	AUXILIARY BLDG	565'-0"	SEE NOTE 5	E1	C	E1	E1	E1	E1	A-12	
237	AUXILIARY BLDG	565'-0"	SEE NOTE 5	A	E2	A	A	A	A	A-12	
123	AUXILIARY BLDG	545'-0"	SEE NOTE 5	E1	E	E1	E1	C	E1	A-12	
124	AUXILIARY BLDG	545'-0"	SEE NOTE 5	E1	E	E1	E1	C	E1	A-12	
240	AUXILIARY BLDG	565'-0"	SEE NOTE 5	D	E	D	D	C	D	A-12	
241	AUXILIARY BLDG	565'-0"	SEE NOTE 5	C	E	C	C	C	C	A-12	
242	AUXILIARY BLDG	565'-0"	SEE NOTE 5	C	E	C	C	C	C	A-12	
243	AUXILIARY BLDG	565'-0"	SEE NOTE 5	E1	E	E1	E1	C	E1	A-12	
244	AUXILIARY BLDG	565'-0"	SEE NOTE 5	E1	E	E1	E1	C	E1	A-12	
245	TURBINE BLDG	566'-0"	SEE NOTE 5	A	E	A	A	A	A	A-12	
—	AUXILIARY BLDG	563'-0"	SEE NOTE 5	C	E	C	C	C	C	A-12	
—	AUXILIARY BLDG	565'-0"	SEE NOTE 5	D	E	D	D	D	D	A-12	
—	CTMT BLDG	565'-0"	SEE NOTE 5	D	E4	B	B	B*	B	A-12	* DURING REFUELING
—	CTMT BLDG	555'-0"	SEE NOTE 5	E3	E4	D	B	B*	B	A-12	* DURING REFUELING
—	AUXILIARY BLDG	565'-0"	SEE NOTE 5	C	B	C	C	C	C	A-12	
—	AUXILIARY BLDG	565'-0"	SEE NOTE 5	C	B	C	C	C	C	A-12	
246	TURBINE BLDG	567'-0"	SEE NOTE 5	A	B	A	A	A	A	A-12	
247	TURBINE BLDG.	567'-0"	SEE NOTE 5	A	B	A	A	A	A	A-12	
248	TURBINE BLDG.	567'-0"	SEE NOTE 5	E3	B	E3	E3	E3	E3	A-12	CAN BE ZONE "E3" WITH FAILED FUEL LEAKAGE
249	TURBINE BLDG	576'-3"	SEE NOTE 5	A	B	A	A	A	A	A-12	
250	TURBINE BLDG	545'-0"	SEE NOTE 5	A	B	A	A	A	A	A-12	
251	TURBINE BLDG	565'-0"	SEE NOTE 5	A	B	A	A	A	A	A-12	
252	TURBINE BLDG	567'-0"	SEE NOTE 5	A	B	A	A	A	A	A-12	
253	TURBINE BLDG	562'-6"	SEE NOTE 5	A	B	A	A	A	A	A-12	
213	CTMT BLDG	565'-0"	SEE NOTE 5	E3	E4	E3	E3	E3	E3	A-12	
250	AUXILIARY BLDG.	565'-0"	SEE NOTE 5	A	E3	A	A	A	A	A-12	
254	TURBINE BLDG	567'-0"	SEE NOTE 5	A	B	A	A	A	A	A-12	
389	LLRW	574'-0"	SEE NOTE 5	E4/E4						A-12	
—	CTMT ACC FAC	585'-0"	CAF FL. EL. 585'-0"	A						A-13	
—	CTMT ACC FAC	603'-0	CAF FL. EL. 603'-0"	A						A-14	
—	CTMT ACC FAC	618'-0"	CAF FL. EL. 618'-0"	A/D*						A-15	* DURING HIGH INTEGRITY CONTAINER (HIC) LIFTS
—	CTMT ACC FAC	633'-0"	CAF FL. EL. 633'-0" AND ABOVE	A/D*						A-16	* DURING HIGH INTEGRITY CONTAINER (HIC) LIFTS
OSGSF-1	OSGSF	584'-1"	SEE DRAWING A-3000	C	C	C	C	C	C	A-3000	
OSGSF-2	OSGSF	583'-6"	SEE DRAWING A-3000	D	D	D	D	D	D	A-3000	
OSGSF-3	OSGSF	583'-6"	SEE DRAWING A-3000	D	D	D	D	D	D	A-3000	
300	AUXILIARY BLDG	585'-0"	SEE NOTE 5	D	E	D	D	D	D	A-13	
300A	AUXILIARY BLDG	585'-0"	SEE NOTE 5	D	E	D	D	D	D	A-13	
300B	AUXILIARY BLDG	585'-0"	SEE NOTE 5	D	E	D	D	D	D	A-13	
301	AUXILIARY BLDG	585'-0"	SEE NOTE 5	D	E	D	D	D	D	A-13	
302	AUXILIARY BLDG	585'-0"	SEE NOTE 5	E2	E2	E2	E2	E2	E2	A-13	SEE NOTE 4, DWG. A-15
303	AUXILIARY BLDG	585'-0"	SEE NOTE 5	E1	E4	E1	F1	E1	F1	A-13	SEE REMARKS FOR RM 208
304	AUXILIARY BLDG	585'-0"	SEE NOTE 5	D	E4	D	*D	D	D	A-13	*SEE NOTE 4
305	AUXILIARY BLDG	585'-0"	SEE NOTE 5	D	E	D	D	D	D	A-13	ZONE WILL BE HIGHER DURING DECONTAMINATION
306	AUXILIARY BLDG	585'-0"	SEE NOTE 5	C	E	C	C	C	C	A-13	
222	AUXILIARY BLDG	563'-6"	SEE NOTE 5	E3	E	D	D	C	D	A-13	
223	AUXILIARY BLDG	563'-6"	SEE NOTE 5	E3	E	D	D	C	D	A-13	
224	AUXILIARY BLDG	563'-6"	SEE NOTE 5	E3	E	E3	E3	C*	E3	A-13	*PROVIDED THERE IS NO SPENT FUEL PRESENT
318	AUXILIARY BLDG	585'-0"	SEE NOTE 5	C	E2	C	C	C	C	A-13	
—	CTMT BLDG	585'-0"	SEE NOTE 5	E2	E4	E2	B/D	*B/D	E2	A-13	* DURING REFUELING
312	AUXILIARY BLDG	585'-0"	SEE NOTE 5	D/E1	E2	D/E1	D/E1	D/E1	D/E1	A-13	
313	AUXILIARY BLDG	585'-0"	SEE NOTE 5	D	E2	D	D	D	D	A-13	
314	AUXILIARY BLDG	585'-0"	SEE NOTE 5	E1	E4	E1	E1	E1	E1	A-13	
315	CTMT BLDG	578'-0"	SEE NOTE 5	D	E4	D	B*	B*	D	A-13	*PROVIDED THERE ARE NO HOT INSTRUMENTS IN THE TANK
216	CTMT BLDG	565'-0"	SEE NOTE 5	E3	E4	E3	E3	E3	E3	A-13	

NOTES

AREAS DESIGNATED AS A ZONE "D" OR LOWER DURING AN MHA WILL MEET THE REQUIREMENTS OF *10CFR50 APPENDIX A CRITERION 19* (5 REM WHOLE BODY DOSE DURING A 30 DAY PERIOD).

HOWEVER, IT WILL BE APPROXIMATELY 14 HOURS BEFORE AREAS DESIGNATED AS A ZONE "D" WILL BE RECEIVING AN EXPOSURE RATE OF 2.5 M REM/HR OR LOWER.

FOR RADIATION ZONE DRAWING NOTES SEE DWG A-11

3. AREAS 105, 225, WILL BE A ZONE "E" DURING SPENT FUEL TRANSFER.
4. IMMEDIATELY ADJACENT TO THE SEISMIC GAP IN AREAS 209 AND 304 WILL BE A ZONE "E" DURING SPENT FUEL TRANSFER.
5. FOR SPACE DESIGNATIONS SEE DRAWINGS A-3 THRU A-9.

LEGEND:

N - NORMAL PLANT OPERATION @ POWER
LOCA - LOSS OF COOLANT ACCIDENT
HS - HOT STANDBY
R - REFUELING
M - MAINTENANCE OR COMPONENT TESTING
ST - SYSTEMS TESTING

ZONE	DOSE RATE	
A	≤ 0.25	MR/HR
B	≤ 2.5	MR/HR
C	≤ 15	MR/HR
D	≤ 100	MR/HR
E	> 100	MR/HR
E1	≤ 1000	MR/HR
E2	≤ 10	R/HR
E3	≤ 100	R/HR
E4	> 100	R/HR

DAVIS-BESSE NUCLEAR POWER STATION

RADIATION ZONES TABULATION SHEET 1

A-17

FIGURE 12.1-7

REVISION 30

OCTOBER 2014

DB 10-07-14 DFN=/J/RASDGN/A17.DGN/CIT

SPACE AND ACCESS TABULATION OF RADIATION ZONES

SPACE NO.	BUILDING	GENERAL ELEVATION	SPACE DESIGNATION	PLANT CONDITIONS N	LOCA	HS	R	M	ST	SPACE SHOWN ON DWG NO.	REMARKS
216	CTMT. BLDG.	586'-0"	SEE NOTE 2	E5	E	E4	E5	E5	E5	A-13	
216	CTMT. BLDG.	583'-8"	SEE NOTE 2	E5	E	E3	E5	D*	E5	A-13	*DRAINED
316	CTMT.BLDG.	564'-8"	SEE NOTE 2	D	E4	D	B	B*	D	A-13	*DURING REFUELING
317	CTMT. BLDG.	564'-8"	SEE NOTE 2	D	E4	B	A	B/D*	E	A-13	B-DURING REFUELING D-DURING NORMAL OPERATION
318	AUXILIARY BLDG	595'-8"	SEE NOTE 2	B	E3	B	B	B	B	A-13	
319	AUXILIARY BLDG	568'-8"	SEE NOTE 2	B	E3	B	B	B	B	A-13	
320	AUXILIARY BLDG	595'-8"	SEE NOTE 2	A	E	A	A	A	A	A-13	
320A	AUXILIARY BLDG	595'-8"	SEE NOTE 2	A	E	A	A	A	A	A-13	
317A	CTMT. BLDG.	565'-8"	SEE NOTE 2	B	E4	D	B	D	D	A-13	
318A	AUXILIARY BLDG.	595'-0"	SEE NOTE 2	B	E2	B	B	B	B	A-13	
319A	AUXILIARY BLDG.	595'-0"	SEE NOTE 2	B	E1	B	B	B	B	A-13	
321	AUXILIARY BLDG	586'-8"	SEE NOTE 2	B	E	B	B	B	B	A-13	
321A	AUXILIARY BLDG	583'-8"	SEE NOTE 2	A	E	A	A	A	A	A-13	
322	AUXILIARY BLDG	585'-8"	SEE NOTE 2	B	E	B	B	B	B	A-13	
323	AUXILIARY BLDG	585'-0"	SEE NOTE 2	A	E3	A	A	A	A	A-12	
324	AUXILIARY BLDG	586'-8"	SEE NOTE 2	A	E3	A	A	A	A	A-13	
325	AUXILIARY BLDG	585'-0"	SEE NOTE 2	A	E1	A	A	A	A	A-13	
326	HEATER BAY	586'-0"	SEE NOTE 2	A	E	A	A	A	A	A-13	
327	AUXILIARY BLDG	585'-0"	SEE NOTE 2	A	B	A	A	A	A	A-13	
328	AUXILIARY BLDG	586'-8"	SEE NOTE 2	A	B	A	A	A	A	A-13	
329	AUXILIARY BLDG	568'-8"	SEE NOTE 2	A	B	A	A	A	A	A-13	
330	TURBINE BLDG	586'-0"	SEE NOTE 2	A	B	A	A	A	A	A-13	
331	TURBINE BLDG	568'-8"	SEE NOTE 2	B	E	B	B	B	B	A-13	
331A	TURBINE BLDG	565'-0"	SEE NOTE 2	A	B	A	A	A	A	A-13	
333	TURBINE BLDG	585'-8"	SEE NOTE 2	A	E	A	A	A	A	A-13	
334	TURBINE BLDG	586'-8"	SEE NOTE 2	A*	B	A*	A*	A*	A*	A-13	*B COULD BE ZONE "C" OR HIGHER WHEN THERE IS STEAM BELLOWS LEAKAGE
335	TURBINE BLDG	585'-8"	SEE NOTE 2	A	B	A	A	A	A	A-13	
336	TURBINE BLDG	585'-8"	SEE NOTE 2	A	B	A	A	A	A	A-13	
337	TURBINE BLDG	585'-8"	SEE NOTE 2	A	B	A	A	A	A	A-13	
338	TURBINE BLDG	568'-8"	SEE NOTE 2	A	B	A	A	A	A	A-13	
339	TURBINE BLDG	585'-8"	SEE NOTE 2	A	B	A	A	A	A	A-13	
340	OFFICE BLDG	585'-8"	SEE NOTE 2	A	B	A	A	A	A	A-13	
341	OFFICE BLDG	585'-8"	SEE NOTE 2	A	B	A	A	A	A	A-13	
341A	FIRE SHED	582'-0"	SEE NOTE 2	B						A-15	
343	OFFICE BLDG	585'-8"	SEE NOTE 2	A	B	A	A	A	A	A-13	
344	OFFICE BLDG	585'-8"	SEE NOTE 2	A	B	A	A	A	A	A-13	
345	OFFICE BLDG	585'-8"	SEE NOTE 2	B	B	B	B	B	B	A-13	
—	AUXILIARY BLDG	568'-8"	SEE NOTE 2	D	E	D	D	D	D	A-13	
—	AUXILIARY BLDG	568'-8"	SEE NOTE 2	C	B	C	C	C	C	A-13	
—	AUXILIARY BLDG	585'-8"	SEE NOTE 2	A	B	A	A	A	A	A-13	
348	TURBINE BLDG.	568'-8"	SEE NOTE 2	A	B	A	A	A	A	A-13	
—	AUXILIARY	586'-0"	SEE NOTE 2	A	B	A	A	A	A	A-13	
—	AUXILIARY	568'-8"	SEE NOTE 2	C	C	C	C	C	C	A-13	
—	CTMT. BLDG	565'-8"	SEE NOTE 2	D	E4	D	D	D*	D	A-13	*SEE REMARKS FOR SPACE NO. 317
—	OFFICE BLDG	585'-8"	SEE NOTE 2	A	B	A	A	A	A	A-13	
—	OFFICE BLDG	568'-0"	SEE NOTE 2	A	B	A	A	A	A	A-13	
—	OFFICE BLDG	568'-0"	SEE NOTE 2	A	B	A	A	A	A	A-13	
—	OFFICE BLDG	568'-0"	SEE NOTE 2	A	B	A	A	A	A	A-13	
334A	TURBINE BLDG.	585'-8"	SEE NOTE 2	C/D	C/D	C/D	C/D	C/D	C/D	A-13	
338B	TURBINE BLDG.	585'-0"	SEE NOTE 2	A	B	A	A	A	A	A-13	
338C	TURBINE BLDG.	568'-0"	SEE NOTE 2	A	B	A	A	A	A	A-13	
340A	OFFICE BLDG	604'-8"	SEE NOTE 2	A	B	A	A	A	A	A-13	
347	TURBINE BLDG.	586'-8"	SEE NOTE 2	A	B	A	A	A	A	A-13	
348	PUMP HOUSE	585'-0"	SEE NOTE 2	A	B	A	A	A	A	A-13	
432	TURBINE BLDG.	588'-1"	SEE NOTE 2	A	B	A	A	A	A	A-13	
385	LLRW	585'-0"	SEE NOTE 2	D/E1						A-15	
386	LLRW	585'-0"	SEE NOTE 2	D/E1						A-15	
387	LLRW	585'-0"	SEE NOTE 2	D/E1						A-15	
388	LLRW	585'-0"	SEE NOTE 2	D/E3						A-15	
389	LLRW	585'-0"	SEE NOTE 2	D/E3						A-15	
390	LLRW	585'-0"	SEE NOTE 2	D/E3						A-15	
391	LLRW	585'-0"	SEE NOTE 2	D/E3						A-15	
392	LLRW	585'-0"	SEE NOTE 2	D/E3						A-15	
393	LLRW	585'-0"	SEE NOTE 2	D/E3						A-15	
394	LLRW	585'-0"	SEE NOTE 2	D/E3						A-15	
395	LLRW	585'-0"	SEE NOTE 2	D/E3						A-15	
396	LLRW	585'-0"	SEE NOTE 2	D/E3						A-15	
397	LLRW	585'-0"	SEE NOTE 2	D/E3						A-15	
398	LLRW	585'-0"	SEE NOTE 2	D/E3						A-15	
401	PSF	603'-0"	SEE NOTE 2	B						A-14	
400	PSF	603'-0"	SEE NOTE 2	B						A-14	
400A	PSF	603'-0"	SEE NOTE 2	B						A-14	
402	PSF	603'-0"	SEE NOTE 2	B						A-14	
403	PSF	603'-0"	SEE NOTE 2	B						A-14	
410	PSF	603'-0"	SEE NOTE 2	B						A-14	
350	AUXILIARY BLDG	603'-0"	SEE NOTE 2	B	E	D	D	D	D	A-14	
351	AUXILIARY BLDG	603'-0"	SEE NOTE 2	B	E	B	B	B	B	A-14	
400	AUXILIARY BLDG	603'-0"	SEE NOTE 2	C	E	C	C	C	C	A-14	
401	AUXILIARY BLDG	603'-0"	SEE NOTE 2	C/D	E	C/D	C/D	C/D	C/D	A-14	
402	AUXILIARY BLDG	603'-0"	SEE NOTE 2	C	E4	C	C	C	C	A-14	
404	AUXILIARY BLDG	603'-0"	SEE NOTE 2	C	E3	C	C	C	C	A-14	
405	AUXILIARY BLDG	603'-0"	SEE NOTE 2	C	E	C	C	C	C	A-14	

SPACE NO.	BUILDING	GENERAL ELEVATION	SPACE DESIGNATION	PLANT CONDITIONS N	LOCA	HS	R	M	ST	SPACE SHOWN ON DWG NO.	REMARKS
406	AUXILIARY BLDG	603'-0"	SEE NOTE 2	D	E	D	D	D	D	A-14	
407	CTMT. BLDG	603'-0"	SEE NOTE 2	E2/E3	E4	E2/E3	B	B*	E2/E3	A-14	*DURING REFUELING
219	CTMT BLDG	585'-8"	SEE NOTE 2	E2	E4	E2	E2	E2	E4	A-14	
218	CTMT. BLDG	568'-8"	SEE NOTE 2	E2	E4	E2	E2	E4	E2	A-14	
410	CTMT. BLDG	568'-8"	SEE NOTE 2	D	E4	D	B	B*/D	B	A-14	SEE REMARKS FOR SPACE NO. 317
411	AUXILIARY BLDG	603'-0"	SEE NOTE 2	C	B*	C	C	C	C	A-14	*D IN AREA OF PIPING AB-2 CAN'T CHANGE, E DURING LOCA
412	AUXILIARY BLDG	603'-0"	SEE NOTE 2	C	C	C	C	C	C	A-14	
413	AUXILIARY BLDG	603'-0"	SEE NOTE 2	C	C	C	C	C	C	A-14	
414	AUXILIARY BLDG	603'-0"	SEE NOTE 2	B	C	B	B	B	B	A-14	
415	AUXILIARY BLDG	603'-0"	SEE NOTE 2	B	C	B	B	B	B	A-14	
—	AUXILIARY BLDG	603'-0"	SEE NOTE 2	D	E	D	D	D	D	A-14	
417	AUXILIARY BLDG.	603'-0"	SEE NOTE 2	C	C	C	C	C	C	A-14	
417A	AUXILIARY BLDG.	603'-0"	SEE NOTE 2	C	C	C	C	C	C	A-14	
418	AUXILIARY BLDG	603'-0"	SEE NOTE 2	B	C	B	B	B	B	A-14	
419	AUXILIARY BLDG	603'-0"	SEE NOTE 2	A	C	A	A	A	A	A-14	
420	AUXILIARY BLDG	603'-0"	SEE NOTE 2	A	C	A	A	A	A	A-14	
421	AUXILIARY BLDG	603'-0"	SEE NOTE 2	A	C	A	A	A	A	A-14	
422D	AUXILIARY BLDG	603'-0"	SEE NOTE 2	D	E	D	D	D	D	A-14	
422	AUXILIARY BLDG	603'-0"	SEE NOTE 2	A	C	A	A	A	A	A-14	
423	AUXILIARY BLDG	603'-0"	SEE NOTE 2	A	C	A	A	A	A	A-14	
424	AUXILIARY BLDG.	603'-0"	SEE NOTE 2	C	C	C	C	C	C	A-14	
424A	AUXILIARY BLDG	603'-0"	SEE NOTE 2	B	C	B	B	B	B	A-14	
424B	AUXILIARY BLDG	603'-0"	SEE NOTE 2	B	C	B	B	B	B	A-14	
424C	AUXILIARY BLDG	603'-0"	SEE NOTE 2	B	C	B	B	B	B	A-14	
412C	AUXILIARY BLDG	603'-0"	SEE NOTE 2	C	C	C	C	C	C	A-14	
426	AUXILIARY BLDG	603'-0"	SEE NOTE 2	D	C	D	D	D	D	A-14	
426	AUXILIARY BLDG	603'-0"	SEE NOTE 2	D	E	D	D	D	D	A-14	
427	AUXILIARY BLDG	603'-0"	SEE NOTE 2	C	E4	C	C	C	C	A-14	
428	AUXILIARY BLDG	603'-0"	SEE NOTE 2	B	E3	B	B	B	B	A-14	
429	AUXILIARY BLDG	603'-0"	SEE NOTE 2	A	E1	A	A	A	A	A-14	
430	HEATER BAY	603'-0"	SEE NOTE 2	A	E	A	A	A	A	A-14	
431	TURBINE BLDG	603'-0"	SEE NOTE 2	A*	B*	A*	A*	A*	A*	A-14	*[illegible]
432	TURBINE BLDG	604'-3"	SEE NOTE 2	A	B	A	A	A	A	A-14	
433	OFFICE BLDG	603'-0"	SEE NOTE 2	A	B	A	A	A	A	A-14	
428A	AUXILIARY BLDG	603'-0"	SEE NOTE 2	A	E3	A	A	A	A	A-14	
435	OFFICE BLDG	603'-0"	SEE NOTE 2	A	B	A	A	A	A	A-14	
436	OFFICE BLDG	603'-0"	SEE NOTE 2	A	B	A	A	A	A	A-14	
437	OFFICE BLDG	603'-0"	SEE NOTE 2	A	B	A	A	A	A	A-14	
428B	AUXILIARY BLDG	603'-0"	SEE NOTE 2	D	E3	D	D	D	D	A-14	
438	OFFICE BLDG	603'-0"	SEE NOTE 2	A	B	A	A	A	A	A-14	
436B	OFFICE BLDG	603'-0"	SEE NOTE 2	A	B	A	A	A	A	A-14	
441	OFFICE BLDG	603'-0"	SEE NOTE 2	B	B	B	B	B	B	A-14	[illegible]
—	AUXILIARY BLDG	603'-0"	SEE NOTE 2	A	B	A	A	A	A	A-14	
—	AUXILIARY BLDG	603'-0"	SEE NOTE 2	B	B	B	B	B	B	A-14	
—	AUXILIARY BLDG	603'-0"	SEE NOTE 2	C	B	C	C	C	C	A-14	
—	CTMT. BLDG	603'-0"	SEE NOTE 2	D	E4	D	B	B*	B	A-14	*SEE REMARKS FOR SPACE NO. 317
—	OFFICE BLDG	603'-8"	SEE NOTE 2	A	B	A	A	A	A	A-14	
—	OFFICE BLDG	603'-8"	SEE NOTE 2	A	B	A	A	A	A	A-14	
—	OFFICE BLDG	603'-0"	SEE NOTE 2	A	B	A	A	A	A	A-14	
—	AUXILIARY BLDG	603'-0"	SEE NOTE 2	A	B	A	A	A	A	A-14	
—	AUXILIARY BLDG	603'-0"	SEE NOTE 2	B	B	B	B	B	B	A-14	
—	CTMT. BLDG.	603'-0"	SEE NOTE 2	D	E4	D	B	B*	D	A-14	*SEE REMARKS FOR SPACE NO. 317
—	OFFICE BLDG	603'-8"	SEE NOTE 2	A	B	A	A	A	A	A-14	
422A	AUXILIARY BLDG	612'-8"	SEE NOTE 2	A	E3	A	A	A	A	A-14	
433A	OFFICE BLDG	613'-0"	SEE NOTE 2	A	B	A	A	A	A	A-14	
429A	TURBINE BLDG.	603'-0"	SEE NOTE 2	A	D	A	A	A	A	A-14	
438A	OFFICE BLDG	613'-0"	SEE NOTE 2	A	B	A	A	A	A	A-14	
436A	OFFICE BLDG	613'-0"	SEE NOTE 2	A	B	A	A	A	A	A-14	
437A	OFFICE BLDG	613'-0"	SEE NOTE 2	A	B	A	A	A	A	A-14	
438A	OFFICE BLDG	613'-0"	SEE NOTE 2	A	B	A	A	A	A	A-14	
439A	OFFICE BLDG	613'-0"	SEE NOTE 2	A	B	A	A	A	A	A-14	
440A	OFFICE BLDG	613'-0"	SEE NOTE 2	A	B	A	A	A	A	A-14	
420A	AUXILIARY BLDG	613'-0"	SEE NOTE 2	A	B	A	A	A	A	A-14	
336	TURBINE BLDG	603'-0"	SEE NOTE 2	A	B	A	A	A	A	A-14	
428B	AUXILIARY BLDG	603'-0"	SEE NOTE 2	A	D	A	A	A	A	A-14	
431A	TURBINE BLDG	604'-3"	SEE NOTE 2	C	C	C	C	C	C	A-14	SEE DWG. A-14, NOTE 4.
542	OFFICE BLDG	598'-7½"	SEE NOTE 2	A	A	A	A	A	A	A-14	
494	OFFICE BLDG	608'-0"	SEE NOTE 2	A	A	A	A	A	A	A-14	
—	CTMT BLDG	609'-0"	SEE NOTE 2	E5	E4	E3	E/D	E/D*	E3	A-14	*DURING REFUELING

LEGEND:
N-NORMAL PLANT OPERATION, POWER
LOCA-LOSS OF COOLANT ACCIDENT
HS-HOT STANDBY
R-REFUELING
M-MAINTENANCE OR COMPONENT TESTING
ST-SYSTEMS TESTING

NOTES:
1. FOR RADIATION ZONE DRAWING NOTES SEE DWGS. A-11 AND A-17.
2. FOR SPACE DESIGNATIONS SEE DRAWINGS A-3 THRU A-5.

ZONE	DOSE RATE
A	≤ 0.25 MR/HR.
B	≤ 2.5 MR/HR.
C	≤ 15 MR/HR.
D	≤ 100 MR/HR.
E	> 100 MR/HR
E1	≤ 1000 MR/HR.
E2	≤ 10 R/HR.
E3	≤ 100 R/HR.
E4	> 100 R/HR.

DAVIS-BESSE NUCLEAR POWER STATION

RADIATION ZONES TABULATION SHEET 2

A-18

FIGURE 12.1-8

REVISION 29

DECEMBER 2012

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SPACE AND ACCESS TABULATION OF RADIATION ZONES

SPACE NO.	BUILDING	GENERAL ELEVATION	SPACE DESIGNATION	PLANT CONDITIONS N	LOCA	HS	R	M	ST	SPACE SHOWN ON DWG NO.	REMARKS
300	AUXILIARY BLDG	623'-0"	SEE NOTE 2	B	E	B	B	B	B	A-15	
500	AUXILIARY BLDG	623'-0"	SEE NOTE 2	C	E	C	C	C	C	A-15	
501	AUXILIARY BLDG	623'-0"	SEE NOTE 2	C	E	C	C	C	C	A-15	
502	AUXILIARY BLDG	623'-0"	SEE NOTE 2	A	E3	A	A	A	A	A-15	
503	AUXILIARY BLDG	623'-0"	SEE NOTE 2	A	D	A	A	A	A	A-15	
504	AUXILIARY BLDG	623'-0"	SEE NOTE 2	A	B	A	A	A	A	A-15	
505	AUXILIARY BLDG	623'-0"	SEE NOTE 2	A	D	A	A	A	A	A-15	
506	AUXILIARY BLDG	623'-0"	SEE NOTE 2	A	B	A	A	A	A	A-15	
507	AUXILIARY BLDG	623'-0"	SEE NOTE 2	A	B	A	A	A	A	A-15	
508	AUXILIARY BLDG	623'-0"	SEE NOTE 2	A	B	A	A	A	A	A-15	
509	AUXILIARY BLDG	623'-0"	SEE NOTE 2	A	B	A	A	A	A	A-15	
510	AUXILIARY BLDG	623'-0"	SEE NOTE 2	A	B	A	A	A	A	A-15	
511	AUXILIARY BLDG	623'-0"	SEE NOTE 2	A	B	A	A	A	A	A-15	
512	AUXILIARY BLDG	623'-0"	SEE NOTE 2	A	B	A	A	A	A	A-15	
—	CTMT BLDG	623'-0"	SEE NOTE 2	E2	E4	E2	B/D	B/D*	E2	A-15	*DURING REFUELING
513	TURBINE BLDG	623'-0"	SEE NOTE 2	A	E	A	A	A	A	A-15	
514	HEATER BAY	623'-0"	SEE NOTE 2	A	E3	A	A	A	A	A-15	
407	CTMT BLDG	603'-0"	SEE NOTE 2	E2/E3	E4	E2/E3	B	B*	E2/E3	A-15	*DURING REFUELING
216	CTMT BLDG	565'-0"	SEE NOTE 2	E3	E4	E3	E3	E3	E3	A-15	
218	CTMT BLDG.	565'-0"	SEE NOTE 2	E3	E4	E3	E3	E3	E3	A-15	
515	AUXILIARY BLDG	623'-0"	SEE NOTE 2	B	E4	B	B	B	B	A-15	
518	AUXILIARY BLDG	623'-0"	SEE NOTE 2	A	E2	A	A	A	A	A-15	
517	TURBINE BLDG	623'-0"	SEE NOTE 2	A*	E2	A*	A*	A*	A*	A-16	*COULD BECOME A ZONE "E2" IF THERE IS STEAM GENERATOR TUBE LEAKAGE. SEE ZONE MAP A-15.
519	OFFICE BLDG	623'-0"	SEE NOTE 2	A	B	A	A	A	A	A-16	
519	OFFICE BLDG	623'-0"	SEE NOTE 2	A	B	A	A	A	A	A-15	
520	OFFICE BLDG	623'-0"	SEE NOTE 2	A	B	A	A	A	A	A-15	
521	OFFICE BLDG	623'-0"	SEE NOTE 2	A	B	A	A	A	A	A-15	
522	OFFICE BLDG	623'-0"	SEE NOTE 2	A	B	A	A	A	A	A-15	
523	OFFICE BLDG	623'-0"	SEE NOTE 2	A	B	A	A	A	A	A-15	
517A	TURBINE BLDG.	623'-0"	SEE NOTE 2	A	E	A	A	A	A	A-15	
525	OFFICE BLDG	623'-0"	SEE NOTE 2	A	B	A	A	A	A	A-15	
526	OFFICE BLDG	623'-0"	SEE NOTE 2	A	B	A	A	A	A	A-15	
—	AUXILIARY BLDG	623'-0"	SEE NOTE 2	B	E	B	B	B	B	A-15	
—	AUXILIARY BLDG	623'-0"	SEE NOTE 2	A	B	A	A	A	A	A-15	
—	CTMT BLDG	628'-0"	SEE NOTE 2	E1	E4	E	B	B+D	E1	A-15	*SEE REMARKS FOR SPACE NO. 317
—	OFFICE BLDG	623'-0"	SEE NOTE 2	A	B	A	A	A	A	A-15	
—	OFFICE BLDG	623'-0"	SEE NOTE 2	A	B	A	A	A	A	A-15	
—	CTMT BLDG	628'-0"	SEE NOTE 2	E1	E4	E1	B	B+D	E1	A-15	*SEE REMARKS FOR SPACE NO. 317
—	AUXILIARY BLDG	623'-0"	SEE NOTE 2	A	B	A	A	A	A	A-15	
—	OFFICE BLDG	623'-0"	SEE NOTE 2	A	B	A	A	A	A	A-15	
527	OFFICE BLDG.	623'-0"	SEE NOTE 2	A	B	A	A	A	A	A-15	
517B	TURBINE BLDG.	623'-0"	SEE NOTE 2	A	E	A	A	A	A	A-15	
518A	TURBINE BLDG.	623'-0"	SEE NOTE 2	A	B	A	A	A	A	A-15	
518B	TURBINE BLDG.	623'-0"	SEE NOTE 2	A	B	A	A	A	A	A-15	
300	AUXILIARY BLDG	595'-0"	SEE NOTE 2	B	E	B	B	B	B	A-16	
600	AUXILIARY BLDG	643'-0"	SEE NOTE 2	D	E4	D	D	D	D	A-16	
601	AUXILIARY BLDG	643'-0"	SEE NOTE 2	C/D	E4	C/D	C/D	C/D	C/D	A-16	
601A	AUXILIARY BLDG	643'-0"	SEE NOTE 2	C	E	C	C	C	C	A-16	
—	CTMT BLDG	643'-0"	SEE NOTE 2	E2	E4	E2	B/D	B/D*	B2	A-16	* DURING REFUELING
407	CTMT BLDG	643'-0"	SEE NOTE 2	E3	E4	E3	B	B*	E3	A-16	*DURING REFUELING
602	AUXILIARY BLDG	643'-0"	SEE NOTE 2	C/D	E4	C/D	C/D	C/D	C/D	A-16	
603	AUXILIARY BLDG	638'-0"	SEE NOTE 2	A	C	A	A	A	A	A-16	
603A	AUXILIARY BLDG	638'-0"	SEE NOTE 2	A	E	A	A	A	A	A-16	
604	HEATER BAY	643'-0"	SEE NOTE 2	A	E	A	A	A	A	A-16	
603B	AUXILIARY BLDG	638'-0"	SEE NOTE 2	A	C	A	A	A	A	A-16	
517	TURBINE BLDG	623'-0"	SEE NOTE 2	A	*	A	A	A	A	A-16	*SEE, ZONE MAP A-15

SPACE NO.	BUILDING	GENERAL ELEVATION	SPACE DESIGNATION	PLANT CONDITIONS N	LOCA	HS	R	M	ST	SPACE SHOWN ON DWG NO.	REMARKS
605	OFFICE BLDG	638'-0"	SEE NOTE 2	A	C	A	A	A	A	A-16	
608	OFFICE BLDG	638'-0"	SEE NOTE 2	A	C	A	A	A	A	A-16	
607	OFFICE BLDG	638'-0"	SEE NOTE 2	A	C	A	A	A	A	A-16	
609	OFFICE BLDG	638'-0"	SEE NOTE 2	A	C	A	A	A	A	A-16	
610	OFFICE BLDG	638'-0"	SEE NOTE 2	A	C	A	A	A	A	A-16	
611	OFFICE BLDG	638'-0"	SEE NOTE 2	A	C	A	A	A	A	A-16	
612	OFFICE BLDG	638'-0"	SEE NOTE 2	A	C	A	A	A	A	A-16	
613	OFFICE BLDG	638'-0"	SEE NOTE 2	A	C	A	A	A	A	A-16	
614	OFFICE BLDG	638'-0"	SEE NOTE 2	A	C	A	A	A	A	A-16	
615	OFFICE BLDG	638'-0"	SEE NOTE 2	A	C	A	A	A	A	A-16	
616	OFFICE BLDG	638'-0"	SEE NOTE 2	A	C	A	A	A	A	A-16	
617	OFFICE BLDG	638'-0"	SEE NOTE 2	A	C	A	A	A	A	A-16	
618	OFFICE BLDG	638'-0"	SEE NOTE 2	A	C	A	A	A	A	A-16	
619	OFFICE BLDG	638'-0"	SEE NOTE 2	A	C	A	A	A	A	A-16	
620	OFFICE BLDG	638'-0"	SEE NOTE 2	A	C	A	A	A	A	A-16	
621	OFFICE BLDG	638'-0"	SEE NOTE 2	A	C	A	A	A	A	A-16	
—	AUXILIARY BLDG	638'-0"	SEE NOTE 2	A	B	A	A	A	A	A-16	
—	OFFICE BLDG	638'-0"	SEE NOTE 2	A	B	A	A	A	A	A-16	
—	OFFICE BLDG	638'-0"	SEE NOTE 2	A	B	A	A	A	A	A-16	
—	OFFICE BLDG	638'-0"	SEE NOTE 2	A	B	A	A	A	A	A-16	
—	OFFICE BLDG	638'-0"	SEE NOTE 2	A	B	A	A	A	A	A-16	
—	AUXILIARY BLDG	638'-0"	SEE NOTE 2	A	B	A	A	A	A	A-16	
—	CTMT BLDG	638'-0"	SEE NOTE 2	E1	E4	E1	C	C*	E1	A-16	* DURING REFUELING
—	CTMT BLDG	638'-0"	SEE NOTE 2	E1	E4	E1	C	C*	E1	A-16	* DURING REFUELING
—	OFFICE BLDG	638'-0"	SEE NOTE 2	A	B	A	A	A	A	A-16	
700	CTMT BLDG	653'-0"	SEE NOTE 2	E2	E4	E2	C	C*	E2	A-16	* DURING REFUELING
701	CTMT BLDG	653'-0"	SEE NOTE 2	E2	E4	E2	C	C*	E2	A-16	* DURING REFUELING

LEGEND:

N-NORMAL PLANT OPERATION @POWER
LOCA-LOSS OF COOLANT ACCIDENT
HS-HOT STANDBY
R-REFUELING
M-MAINTENANCE OR COMPONENT TESTING.
ST-SYSTEMS TESTING

NOTES:

1. FOR RADIATION ZONE DRAWING NOTES SEE DWGS. A-11 AND A-17.
2. FOR SPACE DESIGNATIONS SEE DRAWINGS A-3 THRU A-9.

ZONE	DOSE RATE	
A	≤ 0.25	MR/HR
B	≤ 2.5	MR/HR
C	≤ 15	MR/HR
D	≤ 100	MR/HR
E	> 100	MR/HR
E1	≤ 1000	MR/HR
E2	≤ 10	R/HR
E3	≤ 100	R/HR
E4	> 100	R/HR

DAVIS-BESSE NUCLEAR POWER STATION

RADIATION ZONES TABULATION SHEET 3

A-19

FIGURE 12.1-9

REVISION 31

OCTOBER 2016

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Removed in Accordance with RIS 2015-17

DAVIS-BESSE NUCLEAR POWER STATION

EQUIPMENT LOCATION
CONTROL ROOM AND COMPUTER ROOM
M-132
FIGURE 12.1-10

REVISION 30

OCTOBER 2014

DB 07-16-14 DFN=/J/MASDGN/M132.DGN/C1T

Security - Related Information Figure Withheld Under 10 CFR 2.390

DAVIS-BESSE NUCLEAR POWER STATION

ISOMETRIC OF CONTROL ROOM

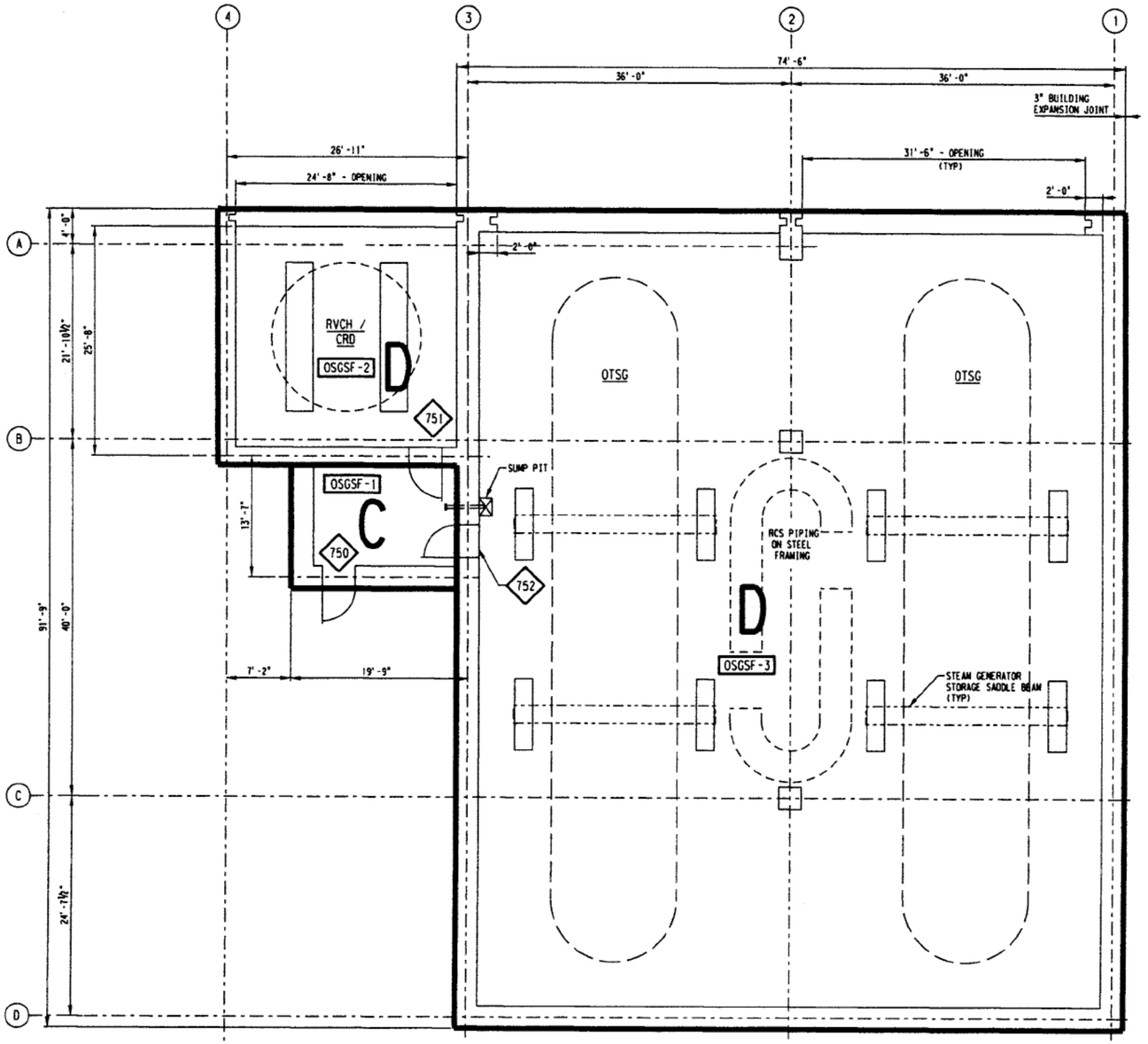
M-133

FIGURE 12.1-11

REVISION 31

OCTOBER 2016

OB 10-13-16 DFN=H:/CONSTS/M133.DGN



PLAN

DESIGNATION	TIME ALLOWED WITHOUT EXCEEDING 5R/YEAR TOTAL HOURS	MAX DOSE RATE BASED ON OPERATION WITH 1% FAILED FUEL MR/HR	DESCRIPTION
A	UNLIMITED	≤0.25	UNCONTROLLED, UNLIMITED ACCESS
B	2000	≤2.5	CONTROLLED, LIMITED ACCESS
C	333 1/3	≤15	CONTROLLED, LIMITED ACCESS FOR ROUTINE TASKS
D	50	≤100	CONTROLLED, LIMITED ACCESS FOR SHORT PERIODS
E1	*	≤1000	CONTROLLED OCCUPANCY FOR VERY SHORT PERIOD AREAS LOCKED AT 1000 MR/HR
E2	*	≤10,000	OCCUPANCY ONLY UNDER CONTROLLED CONDITIONS AREAS LOCKED
E3	*	≤100,000	VERY LIKELY WILL NEVER BE OCCUPIED, AREAS LOCKED
E4	*	>100,000	WILL NEVER BE OCCUPIED, AREAS LOCKED

SPACE NO.	BUILDING	GENERAL ELEVATION	SPACE DESIGNATION
OSGSF-1	OLD STEAM GENERATOR STORAGE FACILITY	584'-1"	VESTIBULE AREA
OSGSF-2	OLD STEAM GENERATOR STORAGE FACILITY	583'-6"	RVCH/CRD AREA
OSGSF-3	OLD STEAM GENERATOR STORAGE FACILITY	583'-6"	OTSG & RCS PIPING AREA

DAVIS BESSE NUCLEAR POWER STATION
RADIATION ZONING FOR NORMAL OPERATION
GENERAL FLOOR PLAN ELEVATION 583'-6"
OLD STEAM GENERATOR STORAGE FACILITY

FIGURE 12.1-12
REVISION 30
OCTOBER 2014

DB 07-17-14 DFN: G/USAR/UFIG12112.DGN

12.2 VENTILATION

12.2.1 Design Objectives

The station is designed to provide maximum safety and convenience for operating personnel with equipment arranged in zones, so that potentially contaminated areas are separated from clean areas. The heating, ventilating, and air conditioning systems for the station are designed to provide a suitable environment for equipment and personnel. The path of ventilating air is from areas of low activity toward areas of higher activity.

The control room emergency ventilation system has redundant prefilters, HEPA filters, and charcoal adsorbers to minimize the dose to operating personnel due to inhalation of fission products after a LOCA. These filters have a total efficiency greater than 95 percent so that the limits of Criterion 19 of 10CFR50, Appendix A are met. The expected dose to operating personnel from inhalation of fission products for the duration of a LOCA is given in Chapter 15. Under normal operating conditions, the control room is free of airborne radioactivity.

The containment vessel purge system has a charcoal filter, the fuel handling and radwaste area exhaust systems are once-through systems, and all systems are provided with prefilter and HEPA filter banks. The Emergency Ventilation System is provided with prefilter HEPA filter and charcoal adsorber banks. The Containment Vessel Purge System and Fuel Handling and Radwaste Exhaust Systems are connected to the Emergency Ventilation System by means of ductwork bypasses, and dampers. In the event radioactivity levels should exceed acceptable limits in these areas, the exhaust air may be passed through the Emergency Ventilation System HEPA filters and charcoal adsorbers, before being discharged through the station vent stack. The Emergency Ventilation System units have a total efficiency greater than 95 percent so that the dose to a member of the public at the site boundary is within the "as low as practicable" guidelines of 10CFR50. The site boundary and low population zone doses will be less than those specified in 10CFR100.

Visitors are not allowed in any areas with airborne radioactivity unless appropriate Radiation Protection procedures are followed. The expected airborne radioactivity levels for postulated accidents are given in Chapter 15.

12.2.2 Design Description

12.2.2.1 Shield Building and Penetration Rooms

There are two ventilation systems provided for the shield building and penetration rooms. These are the Containment Purge System, which can be used during normal operation, and the Emergency Ventilation System, which starts automatically following an SFAS signal. The Containment Purge System is described in Subsection 6.2.3.2. The Emergency Ventilation System is described in Subsection 6.2.3.2.

The Containment Purge System is designed to provide clean fresh air to the containment vessel in Modes 5 and 6 or to the shield building and penetration rooms. Supply air is taken through an outside air intake, roughing filter, heating coil and purge supply fan and discharged into the penetration rooms to provide adequate distribution. The purge air is exhausted by the purge exhaust fan through a roughing filter, a high efficiency particulate filter, and a charcoal adsorber. The air flow rate is 50,000 cfm for this system.

The Emergency Ventilation System is provided with two redundant full capacity fan-filter assemblies. Each filter system is made up of two units. The first contains roughing, HEPA filters, and charcoal adsorbers in series and the second a duplicate set of charcoal adsorbers with an air flow rate of 8,000 cfm.

The charcoal adsorber bank is designed on the basis of 333.3 cfm (nominal flow) per element. The filter beds are 2 inches in depth and have an air flow velocity of approximately 42 ft/min. Impregnated charcoal (activated coconut shell) is used in adsorbers with a total efficiency greater than 95 percent.

The maximum amount of radioactive iodine that the charcoal adsorbers receive is 0.5 percent/day of the amount present in the containment vessel atmosphere. Based upon this leak rate, it is not considered credible that, during any single anticipated running of the Emergency Ventilation System, there could accumulate sufficient radioactive material to raise the charcoal temperature to its desorption limit.

The HEPA filters are of a non-standard manufactured size, approximately 24 inches x 24 inches x 13-3/4 inches with nominal air flow of 1000 cfm. Waterproof and fire-retardant glass fibers are used in these filters.

All fans and filters are located in the auxiliary building. The equipment is designed to operate in maximum ambient conditions of 110°F and 90 percent relative humidity. The total volume of the penetration rooms is 136,450 ft^3.

Following a loss-of-coolant accident, an SFAS signal (refer to Section 6.2.4) starts the Emergency Ventilation System fans. The containment purge equipment, if running, is shut down by the SFAS signal, and the purge isolation valves in each line are closed. Air from the shield building and penetration rooms is drawn through the filter assembly consisting of roughing filters, high efficiency particulate filters and charcoal adsorbers in series and discharged through the station vent. A pressure controller regulates the modulating dampers at the fan discharge to maintain the setpoint negative pressure within the shield building and penetration rooms.

The entire Emergency Ventilation System is designed to operate under negative pressure up to the fan discharge. In all cases, the flow from these regions exceeds the total maximum containment vessel leakage plus the shield building in-leakage.

Temperature indication and radiation monitors are provided for operator information. Differential pressure indicators are provided across the filters.

The system equipment is fully accessible during all normal station operation for maintenance and performance testing, including replacement of filter elements.

During normal operation, the Emergency Ventilation System is held on standby. An SFAS signal actuates the fans and control room instrumentation monitors the operation. The system can be tested during normal operation and the fans can be operated on emergency diesel generator power.

12.2.2.2 Auxiliary Building

The auxiliary building is served by separate ventilation systems for the fuel handling area, the radwaste area, and the non-radioactive area.

The ventilation system for the radwaste area is independent from that used in any other areas and is designed on a "once through" basis to control and direct all potentially contaminated air to the station vent stack via roughing and high efficiency particulate filters. The total volume of building served by these systems is approximately 823,500 ft^3. Exhaust air from the chemistry laboratory work areas is passed through an independent exhaust filtration system consisting of prefilter, HEPA filters, and charcoal adsorber before being exhausted through the station vent filters stack. These systems are further discussed in Section 9.4.3.

The following are the air flow rates for the supply and exhaust systems:

Radwaste area supply system	47,160 cfm
Radwaste area exhaust system	43,850 cfm
Laboratory hood exhaust system	5,300 cfm

The fuel handling area is ventilated by an independent system which is designed on a "once through" basis. This system is further discussed in Subsection 9.4.2. The total air flow for this system is 20,000 cfm and the total volume of building served by this system is approximately 477,900 ft^3.

12.2.2.3 Containment Vessel

The Containment Cooling and Ventilation System, comprised of the Containment Air Cooling System and the Containment Vessel Purge System, accomplishes the following three functions:

1. To remove heat released by equipment and piping in the containment vessel.

2. To purge the containment vessel with clean fresh air.

3. To cool and reduce the pressure of the containment vessel atmosphere.

Functions 1 and 3 are described in Subsection 6.2.2.2, and Function 2 is described in Subsection 6.2.3.2.2.

The following are the air flow rates for the two systems:

Containment vessel purge system	50,000 cfm
Containment air cooling units (normal)	117,000 cfm/unit
Containment air cooling units (LOCA operation)	45,000 cfm/unit

In addition, a containment vessel atmosphere gas monitor with grab sample capability is provided for detecting radioactive particulates, iodine, and noble gases.

12.2.2.4 Control Room

The control room area system employs redundant fans, filters and mechanical refrigeration equipment, plus the necessary dampers and controls for switching to full recirculation for post-accident operation. The control room area system performance is continually monitored

with alarms for high radioactivity and equipment malfunction. The control room operator has a remote manual control for selecting damper position and fan and filter operation in order to ensure satisfactory control room conditions following an accident. All control area ventilation equipment is remote from the control area. The control room is maintained at a pressure somewhat higher than the cable spreading room and the rest of the station in order to prevent in-leakage of airborne radioactivity.

During normal operation, the control room air conditioning system utilizes recirculation of return air with a suitable fresh air makeup. The system employs redundant air handling units and refrigeration equipment to ensure that the failure of any single component does not prevent the fulfilling of the design function. The air flow rate for this system is 21,920 cfm.

The control room Emergency Ventilation System consists of two full capacity, redundant, fan-filter assemblies and is Seismic Class I. The air flow rate for this system is 3,300 cfm.

Smoke detectors and radiation monitors are provided to detect airborne contaminants. A description of the methods utilized for airborne radioactivity monitoring is provided in Subsection 12.2.4.

The control room air conditioning system and Emergency Ventilation System are fully described in Section 9.4.1.

An analysis of expected radiation and airborne radioactivity levels in the control room under accident conditions is included in Chapter 15, Accident Analyses.

The outside air required for the control room air conditioning system and Emergency Ventilation System is taken from the roof at elevation 654.

The efficiency of the filters is sufficient to prevent the members of the operating crew from receiving a dose in excess of Criterion 19 of 10CFR.50, Appendix A.

12.2.2.5 System Reliability

Ventilation systems and equipment are designed in accordance with the recommended practices of the American Society of Heating, Refrigerating and Air Conditioning Engineers Guide, the Air Moving and Conditioning Association and National Fire Protection Association. Redundant exhaust fans are provided for the potentially contaminated areas. The components of the ventilation systems are accessible for periodic inspection and maintenance.

12.2.3 Source Terms

The estimates of equipment leakages resulting in airborne radioactivity within the station buildings are given in Section 11.1.3. The resultant dosages at the site boundary of the various operational occurrences resulting in an airborne activity are discussed in Section 15.3.

12.2.4 Airborne Radioactivity Monitoring

Airborne radioactivity levels are monitored in the ventilation systems for the fuel handling area, the radwaste area, the penetration rooms and the main control room. Each off-line monitor has three detector channels which continuously perform the following measurements selective to (a) particulate activity (b) the isotope I-131 and (c) the isotope Xe-133.

See Subsection 11.4.2.1 and the description of similar systems in Subsections 11.4.2.2.4 and 11.4.2.2.5.

A readout of all measurements is provided in the control cabinet room (on respective control units). Alarm indication is provided in the main control room.

The penetration rooms and radwaste area airborne radioactivity monitors initiate engineering functions in addition to alarms. Each monitor, upon detecting a high activity level, redirects the respective exhaust air flow through the Emergency Ventilation System before it discharges to the station vent.

In addition to the airborne radioactivity monitors, inplant airborne radioactivity level surveys are performed on a routine monthly basis at designated locations in the auxiliary building and containment vessel. If these routine surveys indicate that airborne activity levels are higher than expected for normal conditions or if operating conditions indicate that higher than normal levels could be expected, the frequency of surveys are increased and investigations are made to identify the airborne activity source.

During special operations such as fuel handling, samples are taken on a frequency necessary to determine the levels of airborne contamination. This frequency varies, depending on the operation being performed and on the prevailing conditions.

Records of these airborne radioactivity surveys are retained for a minimum of six years.

12.2.5 Operating Procedures

12.2.5.1 Introduction

On-site inhalation exposures are kept as low as is reasonably achievable during operation and maintenance. This is accomplished by training of personnel, area surveillance, contamination control, and proper work procedures.

The Radiation Protection procedures used at the Davis-Besse Nuclear Power Station were based on USNRC documents including standards, regulatory guides, reports, NCRP and ICRP recommendations, and selected documentation relating to health physics. The present procedures reflect the experience gained by the operation of the Davis-Besse Generating Station.

12.2.5.2 Training

Radiological control training classes are given to all workers requiring unescorted access to Radiologically Controlled Areas. Other workers are given a radiological control orientation session or remain in the company of qualified radiation workers. Radiation Protection procedures are available to all personnel. As part of radiological controls training, workers are instructed in the use and applications of protective clothing.

12.2.5.3 Area Surveillance

Area surveillance for contamination is accomplished with airborne particulate surveys and smear surveys.

Low volume and lapel air samplers are available for collection of samples for subsequent alpha, beta or gamma activity analysis.

Survey records of airborne radioactivity measurements for particulates, noble gases, radioiodines and tritium are kept for a minimum period of three years in accordance with the retention periods stated in 10CFR20 subpart L. Smear surveys are performed to establish the levels of removable surface contamination. These surveys are used to assess the levels of contamination so that appropriate decontamination and personnel protective measures can be taken.

12.2.5.4 Contamination Control

The Radiation Protection Section is responsible for contamination control and decontamination procedures.

Routine surveys, selected to represent all areas within the controlled area are made to ensure that contamination does not exist outside of restricted areas. Surveys are made within the restricted areas so that the contamination status of each location is known at all times with reasonable accuracy.

The frequency of surveying a particular location depends on:

1. the probability of contamination spread to the location,
2. the probability of personnel contamination, and
3. the probability of significant dose rates.

Special smear tests and radiation surveys are conducted whenever it is suspected that contamination has been spread or limits have been exceeded. Transfer of radioactive materials within the station area is carefully controlled to safeguard personnel and prevent the spread of radioactive contamination to roadways, vehicles, personnel and clean areas. Employees engaged in the preparation of material for transfers or in the actual movement of radioactive material are responsible to see that necessary precautions are taken. These precautions include adequate provisions such as wrapping to prevent the spread of contamination to clean areas. In case of a spill or accident, personnel shall move to a safe distance but remain close enough to warn others who may approach. The Radiation Protection Section is notified as soon as possible.

Particles of radioactive dust may become airborne inside the containment vessel, in the auxiliary building, and at other locations around the station. Respiratory protective equipment is available to protect personnel from these materials. When the concentration of this airborne activity is very low, no respiratory protection is required to keep the amount of radioactive material taken up by the body within prescribed limits. For higher concentrations of particulate matter, various types of respiratory protective devices may be required to maintain Total Effective Dose Equivalent (TEDE) ALARA.

The type of material used for protective clothing and the combination of clothing, is dependent upon the work assignment. Coveralls are available in cotton, plastic, and paper. Shoe covers and gloves are available in cloth or rubber. Cloth head covers are normally worn at all times and are to be worn under air supplied hoods.

Respiratory equipment includes full-face air purifying mask, airline supplied masks and hoods, and Self Contained Breathing Apparatus (SCBA). Radioiodines and radioactive particulates are the principal airborne contributors to inhalation doses.

In a radioactive noble gas cloud, which is free of radioactive particulate material, the dose to the whole body is the limiting factor.

Continuous monitoring techniques for tritium detection are not used since the tritium concentration is factors of 10-100 less than the Derived Air Concentration (DAC) for tritium in air for restricted areas .

Areas monitored for tritium during normal operation are selected areas in the Auxiliary Building. During refueling periods, the Containment Vessel is also monitored.

If a significant intake of a radionuclide(s) (excluding tritium) is suspected, an analysis for comparison to Annual Levels of Intake (ALI) will be performed by whole body counting. A bioassay program for tritium uptake is conducted on a select group of individuals who are representative of the station personnel that work in areas where there is a possibility for tritium inhalation or absorption. The method to be used for monitoring tritium uptakes is based on urinalysis.

12.2.6 Estimation of Inhalation Doses

Peak airborne concentrations are determined by grab sampling; the original evaluation assumptions for airborne concentrations are given in Table 12.2-1. Grab samples are counted in the laboratory for gross activity determination and/or with a gamma spectroscopy system for specific isotopic analysis. Containment vessel gross particulates during plant operation are estimated to reach a peak of 1.5×10^{-6} µCi/cc with auxiliary building concentrations being insignificant. These particulates are primarily the short lived Rubidium-88. Airborne iodine samples are collected with a low volume pump and charcoal filter. Essentially no iodine is expected in the auxiliary building. In the containment, the peak Iodine-131 concentration during operation could reach 3×10^{-7} µCi/cc. Noble gas airborne activity is primarily Xenon-133. This is determined by collecting an air sample and counting it on the gamma spectroscopy system. Peak Xe-133 concentrations in the containment during reactor operation are estimated at 10^{-2} µCi/cc.

The peak tritium concentration in the containment during reactor operation is expected to be approximately 8×10^{-4} µCi/cc.

Inhalation doses are only a very small fraction of 10CFR20 limits. The noble gas Xe-133 is considered as a whole body immersion dose. The dose from airborne particulates and the tritium occurs primarily during maintenance outages and is estimated to be much less than 10% of the 10CFR20 limit for all workers.

TABLE 12.2-1

Original Assumptions for Airborne Concentrations

Auxiliary building 10,000 $ft.^3$ room (typical)	10 GPD of 1% of coolant concentration	570 cfm	1% of Iodine released 100% of noble gases released
Turbine building 4.66×10^6 $ft.^3$	10 GPM of secondary coolant	300,000 cfm	1% of Iodine released All noble gases exhausted through Condenser Air Ejector Exhaust
Containment Vessel 2.866×10^6 $ft.^3$	150 GPD of Reactor Coolant	Purge at 50,000 cfm	1% of Iodine released 100% of noble gases released

12.3 HEALTH PHYSICS

12.3.1 Program Objectives

The Radiation Protection Program for the Davis-Besse Nuclear Power Station is being conducted in a manner which ensures compliance with the U.S. Nuclear Regulatory Commission's regulations as covered in 10CFR20, Standards for Protection Against Radiation.

In order to minimize dose to personnel on the site, proper signs, roping, tagging, and other methods to reduce exposure to radiation and radioactive material are used.

Radiological control training classes are given to all workers assigned to designated restricted areas of the station. Other workers are given a radiological control orientation session or remain in the company of qualified workers.

A radiation work permit procedure is used to minimize an individual's dose when working in designated restricted areas.

A thorough radiation surveillance program is carried out within the restricted and controlled areas such that the levels of radiation are known for areas where personnel are working.

A radiological environmental monitoring program is conducted to ensure that radioactivity released to the environment is well within regulatory guidelines.

For further information, refer to Davis-Besse policies and nuclear group and administrative procedures.

The Radiation Protection Program and responsibilities are carried out through the Radiation Protection Section described in Subsection 13.1.1.1.b. The duties and responsibilities of the Manager-Radiation Protection are also given in this Subsection.

The Manager-Radiation Protection is responsible for seeing that procedures that implement the Radiation Protection Program are written, approved and implemented. The Plant Operations Review Committee (PORC) and the Company Nuclear Review Board (CNRB) may review the Radiation Protection Program in accordance with Section 13.4.

Reporting to the Manager - Radiation Protection are supervisory personnel who are responsible for directing the work of Radiation Protection testers, servicemen and professionals.

It is the responsibility of Radiation Protection to monitor and control the dose to personnel, to continuously evaluate and review the radiological status of the station, to make recommendations for control or elimination of radiation hazards, to support the training of personnel in radiological controls, and to protect the health and safety of the public, both on and off site. Radiation Protection also functions in an advisory capacity to assist all personnel in carrying out their radiological control responsibilities.

Radiation Protection personnel are present on all shifts.

The personnel monitoring program is administered by Radiation Protection to determine dose to all individuals in the restricted and controlled areas. Dosimetry is available for all individuals who have to enter restricted areas.

Radiation Protection is responsible for contamination control and decontamination procedures.

Radioactive materials received on the site are monitored and recorded by Radiation Protection. Shipments of all radioactive materials comply with Department of Transportation and NRC regulations.

12.3.2 Facilities and Equipment

12.3.2.1 Radiation Protection

The Auxiliary Building, the Fuel Handling Building, the Low-Level Radwaste Storage Facility, the Shield Building, Containment Vessel and Old Steam Generator Storage Facility, and portions of the Containment Access Facility (CAF) are Radiologically Controlled Areas (RCAs) which are designated as requiring a Radiation Work Permit (RWP) for entry. The Radiation Protection Manager (RPM) may designate other RCAs as requiring an RWP for access.

In designated RCAs, a worker signs in under a RWP which details the radiation protection and dosimetry requirements for the work assignment. When required by the RWP, self-reading dosimetry is normally issued at the entrance to the RCAs. When required by the RWP, protective clothing is picked up and put on in the protective clothing storage area.

Radiation Work Permits (RWPs) to allow entry to designated RCAs are approved by Radiation Protection supervisory personnel.

Entered on the RWP are location and description of work, radiation levels, dosimetry requirements, protection requirements, radiological control limitations, approvals and the name and approval for each individual permitted to enter the RCA under that RWP.

When returning from a work assignment, which required protective clothing, the worker removes the protective clothing and checks for personal contamination using friskers and/or a personnel contamination monitor. Should contamination exceeding specified limits be present, decontamination is performed until the level of contamination has been adequately reduced.

Non-disposable protective clothing is cleaned, monitored, and where possible, returned to service.

The Chemistry radiochemical laboratory is equipped with exhaust hoods which are vented through absolute and charcoal filters.

For counting samples which have been prepared in the radiochemical laboratory, a counting room with two foot thick concrete shield walls is used. A labyrinth is provided into the counting room to further shield the counting area from radiation in the radiochemical laboratory. Samples may also be counted in the Radiation Protection counting area.

There is an instrument calibration room with a gamma source of sufficient strength to check all ranges of survey instruments.

12.3.2.2 Equipment

12.3.2.2.1 Portable Survey Instruments

For beta-gamma radiation measurements, both high and low range instruments are used. High range instruments are capable of detecting radiation rates up to 1,000 rad/hour. These instruments have ion chamber or GM type detectors.

Alpha surveys are conducted using portable instruments with gas proportional or alpha scintillation detectors. A plutonium-239 standard source and a pulse generator are used for calibrating the alpha detectors.

Neutron radiation measurements are performed using a portable instrument. The detector is capable of measuring the dose equivalent, in rems, of neutrons of varying energies. The calibration of the detector is accomplished using an americium-241 beryllium neutron source and a pulse generator. For calibrating gamma detecting instruments, a pulse generator may be used and/or a source traceable to nationally recognized standards.

A depleted uranium source is used for determining beta corrections factors for beta detectors.

Low volume and lapel samplers provide general area and individual air sampling.

Portable survey instruments are normally stored at the entrance to the RCA. Survey instruments in use are calibrated semi-annually.

12.3.2.2.2 Counting Equipment for Radioactivity Measurements

There are gas flow proportional counters for low-level alpha and beta/gamma counting. These counters are shielded to reduce background. These counters are used for routine smear and air sample counting.

Counters using Geiger-Mueller detectors are utilized for high activity smear and air sample counting.

A scintillation detector is available for alpha counting.

Two gamma spectroscopy systems exist, each with at least one high purity germanium detector and a multichannel analyzer which is interfaced to a computer. The system may be operated and data may be reduced with or without use of the computer. Although data may be read directly from the analyzer, each system has optional outputs to printers or plotters. Each detector is mounted in a background reducing shield.

There is a liquid scintillation counter for beta analysis.

There is a wide range in the types of samples which can be analyzed for alpha, beta, and gamma activity. The samples can be particulate matter, gases, smears, liquids, evaporated samples, and other samples collected for radiological control purposes.

All counting instrumentation is calibrated on a regular basis with sources traceable to nationally recognized standards.

Electronic calibration, alignment, and repair of portable radiation survey instruments is performed by Instrument and Control or Radiation Protection personnel. Source calibration and battery checks are performed by Radiation Protection. Records of calibration and maintenance are maintained.

12.3.2.2.3 Protective Clothing and Respiratory Equipment

The type of material required and the combination of clothing will depend upon the work assignment. Cotton, plastic and paper coveralls are available. Shoe covers and gloves are available in cloth, rubber or plastic. Cloth head covers or hoods are available for use.

Respiratory equipment, including full-face air purifying masks, self-contained breathing air packs, and air supplied masks and hoods, are available.

The Training Section trains personnel on proper fitting and use of each type of respiratory protective equipment prior to their being authorized entry into an area requiring its use. Fit tests for negative pressure air purifying equipment include both a quantitative fit test annually and a negative pressure test each time the device is worn.

The respiratory equipment is routinely checked and tested This includes inspection of regulators, hydrostatic tests of the tanks, ensuring adequate supply of air, and inspection of all metal-to-rubber seals. Following the use of respiratory equipment, each piece is decontaminated (general cleaning), surveyed for fixed and smearable contamination, tested, inspected, and properly stored for reuse.

12.3.3 Personnel Dosimetry

The permissible doses for occupationally-exposed people as given in 10CFR20 are the operating limits for the station.

Administratively controlled dose levels are established to prevent personnel from exceeding the dose limits of 10CFR20. The Radiation Protection Manager may approve exceeding these administrative dose levels after ensuring compliance with all other applicable requirements of 10CFR20. In all cases personnel dose shall be controlled to less than 10CFR20 limits.

Thermoluminescent dosimeters (TLDs), or an equivalent, are used as the primary personnel monitor for external beta-gamma dose. Doses of 10 millirem or more are reported. Neutron dosimetry monitors for neutron dose. Calibration and processing of this dosimetry are performed by an outside vendor. In certain types of work, additional TLDs are attached to the whole body and the extremities. The issuance of multiple whole body and extremity dosimetry is controlled by procedure.

A self-reading dosimeter is used for recording daily exposure. A self reading dosimeter capable of reading high dose values is used when an individual's work assignment is in a high radiation area. A radiation monitor with an audible warning is also used for working in high radiation areas. These functions may be combined in a single self-reading dosimeter.

An in vivo bioassay system has been installed at the station and uptake analyses are performed by Radiation Protection personnel. Annual in vivo bioassay or passive monitoring is performed on individuals who work in designated Radiologically Controlled Areas.

The requirements of NRC Form 4 and NRC Form 5 are followed.

Daily dose and the accumulated dose are recorded for each occupational worker in accordance with the regulations of 10 CFR 20.1502, “Conditions requiring individual monitoring of external and internal occupational dose”. This data is obtained from self-reading dosimeters and TLDs. Radiation Protection verifies that the entries in computer based dose records are correct.

12.4 RADIOACTIVE MATERIALS SAFETY

Davis-Besse commits to the regulatory position of Regulatory Guide 4.15 (Revision 1, 1979).

12.4.1 Materials Safety Program

Fuel storage and handling is described in Section 9.1. The Radwaste Management Program is described in Chapter 11. This section describes the Materials Safety Program for the remainder of the radioactive materials to be used at the site.

All licensed source and byproduct materials used for calibration and sample analysis are kept in a shielded container when not in use. These containers are posted or labeled in accordance with 10CFR20 regulations and are kept locked when not in use to prevent removal of sources by unauthorized persons. When not in use, the sources in their containers are stored in a locked designated storage area.

Any Non-Destructive Test (NDT) sources which are brought on site are controlled by the contractor who will do the testing. The contractor is then required to comply with those regulations which are applicable to him to assure adequate materials safety.

12.4.2 Facilities and Equipment

The hot laboratory (Room 424) is designed for handling radioactive materials and is maintained at a negative pressure to prevent airborne particles from leaving the area. The fume hoods in the hot laboratory, as well as the ventilation for the room, vent through a prefilter/absolute-filter/charcoal filtering system. There is an area monitor located in the hot laboratory, and its output is continuously recorded. Equipment, glassware, and tools used in working with radioactive materials are suitably controlled and are not used in clean areas unless the item has been surveyed and meets the limits for uncontrolled areas.

Sealed neutron and gamma sources used for calibration are stored in designated storage locations controlled by Radiation Protection personnel. Portable instrumentation is calibrated in the Instrument Calibration Room or other designated low background radiation areas. Radiation sources used at the location of permanently installed monitoring equipment, such as area radiation monitors and reactor instrumentation, will be manipulated with remote handling devices and shields to maintain personnel exposures as low as reasonably achievable, when necessarily.

Remote handling devices and shields are available to be used when handling radiation sources. Licensed gamma sources are stored in lead or steel containers. Licensed neutron sources are contained in a hydrogenous shield when being transported or stored.

12.4.3 Personnel and Procedures

Section 13.2 describes the training program implemented to ensure that only well-trained personnel handle licensed radioactive sources and byproduct material. These persons have the proper personnel dosimetry and survey the source upon removal from its container. Sealed sources are leak-tested semi-annually. When unsealed sources are used, the work is performed when possible in a fume hood in the hot laboratory or in other radiologically controlled areas with suitable ventilation controls. Sources are handled with remote handling equipment, as necessary, and shielding is used whenever possible and exposure rates necessitate to keep personnel exposures as low as practicable.

12.4.4 Required Materials

Table 12.4-1 lists the isotopes, quantity, form, and use for typical byproduct, source, and special nuclear materials which equal or exceed 100 millicuries and are not required for reactor operation.

TABLE 12.4-1

Typical Radiation Sources

Material	Form and Use	Quantity
$^{241}AmBe^{242}Cm$	Sealed Source, Startup Sources	(2) - 500 curie
$^{241}AmBe$	Sealed Source, Automatic Boron Analyzer	1 curie
$^{241}AmBe$	Sealed Source, Calibration of Reactor and Survey Instrumentation	5 curies
Cs-137	Sealed Source, Calibration of Area Monitors	(3) 100 millicuries
Cs-137	Sealed Source, Calibration of Survey Instruments	245 curies
Cs-137	Sealed Source, Calibration of Survey Instruments	400 curie
Cs-137	Sealed Source, Calibration of self-reading Dosimetry	1.2 curie