12.3 Radiation Protection Design Features

The information in this section of the reference ABWR DCD, including all subsections, tables, and figures, is incorporated by reference with the following departures and supplements.

STD DEP T1 2.14-1 (Figures 12.3-5, 12.3-10, 12.3-16 and 12.3-21)

STD DEP T1 3.4-1

STD DEP 1.2-2 (Table 12.3-7, and Figures 12.3-49 thru 12.3-53, 12.3-55, 12.3-68 thru 12.3-73, and 12.3-75 thru 12.3-77)

STD DEP 1.8-1

STP DEP T1 2.5-1

STD DEP 3.8-1 (Table 12.3-7, and Figures 12.3-37 thru 12.3-41, 12.3-65 thru 12.3-68)

STP DEP 9.4-1

STD DEP 11.5-1

STD DEP 12.3-1

STD DEP 12.3-3

STD DEP 12.3-4 (Tables 12.3-3, 12.3-6 and 12.3-7, and Figures 12.3-56 thru 12.3-58, 12.3-60 and 12.3-62)

STD DEP Admin (Figures 12.3-1, 12.3-3, 12.3-6 and 12.3-74)

12.3.1 Facility Design Features

STD DEP Admin

The ABWR Standard Plant is designed to meet the intent of Regulatory Guide 8.8 (i.e., to keep radiation exposures to plant personnel as low as reasonably achievable (ALARA)). This section describes the component and system designs, in addition to the equipment layout, employed to maintain radiation exposures ALARA. Where possible, consideration of individual systems is provided to illustrate the application of these principles. ~~Owing to the ABWR being a standard plant, specific details as to precise equipment definition are not available and are to be provided by the COL applicant during the final design detail stage.~~ To insure that the plant as designed meets all applicable radiation criteria, a two-step process is then applied where design details not included in this document are then subject to review and confirmation in accordance with radiation protection criteria. Therefore, the details in this section serve as input to the final design configuration and serve to determine the adequacy of the design with respect to radiation protection.

12.3.1.1.2 Material Selection

STD DEP 12.3-1

In the ABWR design maintaining radiation exposure ALARA has been considered in the material selection of systems and components exposed to reactor coolant. For example, radiation exposure potential has been reduced appreciably through the removal or reduction of cobalt from many components as compared to current BWR fleet. Much of the cobalt is removed from contact with reactor coolant by eliminating Stellite where practical and reducing cobalt in the core stainless steel components. ~~*The cost of using very low cobalt materials through out the plant is prohibitive with the cost of 0.02 wt percent cobalt stainless steel approximately 8 times that of 0.05 wt percent stainless steel. Therefore, the*~~ The *ABWR design has taken a graded approach* to using various grades of low cobalt stainless steel, *by using the* ~~*most expensive though*~~ *lowest cobalt bearing materials in the most radiologically significant areas with increasing cobalt content in less sensitive areas. The ABWR standards for cobalt are: 0.02 wt percent for those items in the core; 0.03 wt percent for those items in the vessel internals; and 0.05 wt percent for all other components. Also, with the current materials, there are no proven substitutes for Stellite for many hard surface applications shuch as MSIV seats. Current efforts by the nuclear and metallurgical industry indicate that in the future, practical alternatives to Stellite maybe feasible and are being researched.*

~~*The COL applicant shall address material selection of systems and components exposed to reactor coolant to maintain radiation exposures ALARA. See Subsection 12.3.7.4 for COL license information requirements.*~~ These cobalt contents are target values for reduced occupational exposure per ALARA principles and are not specifications.

The estimation of occupational exposure for the ABWR, was generated by reviewing current plant work records and practices at operating BWRs, and taking into account distinct plant features in the ABWR. An estimate of the average annual occupational exposure was made during the US Standard ABWR certification work. It is noted that the reduced cobalt loadings (i.e. target values) were not considered in the estimation. Therefore, based upon the methods used and assumptions made in evaluating occupational exposure, the materials procured with 0.05 wt percent maximum cobalt have no adverse effect on the estimated occupational exposure.

12.3.1.4.1 Reactor Water Cleanup (CUW) System

The CUW System is provided with chemical cleaning connections which can utilize the condensate system to flush piping and equipment prior to maintenance. The CUW filter/demineralizers can be remotely backflushed to remove spent resins and filter aid material. If additional decontamination is required, chemical addition connections are provided in the piping to clean piping as well as equipment prior to maintenance. The backwash tank employs an arrangement to agitate resins prior to discharge. The tank vent is fitted with a charcoal filter canister to reduce emission of radioiodines into the plant atmosphere. The HVAC System is designed to limit the spread of contaminants *from these shielded cubicles by maintaining a negative pressure in the cubicles relative to the surrounding areas.*

12.3.1.4.4 Main Steam System

STD DEP 12.3-3

Penetrations through the steam tunnel walls are minimized to reduce the streaming paths made available by these penetrations. ~~*The blowout panels for the steam tunnel are located in the relatively inaccessible upper section of the RHR heat exchanger shielded cubicles which are controlled access areas.*~~ *Penetrations through the steam tunnel walls, when they are required, are located so as to exit in controlled access areas or in areas that are not aligned with the steamlines. A lead-loaded silicone foam* <u>or equivalent</u> *is employed whenever possible for these penetrations to reduce the available streaming area presented.*

12.3.2.2.2 Method of Shielding Design

The following site specific supplement provides information to address the methods used to determine shield parameters:

As provided in Tier 1 ITAAC Table 3.2a, commonly accepted shielding codes using nuclear properties derived from well known references shall be used to model and evaluate plant radiation environments.

The DIJESTER computer code is used to assist in determination of the activity in the components of the radwaste system. It is basically a bookkeeping program, which keeps track of each nuclide as it is processed through the pipes, tanks, filters, demineralizers, etc., that make up the radwaste system. It couples the rate equations that govern the operation of the various components of the radwaste system with radionuclide decay chains to model the buildup and decay of the radionuclide activity in each component. This is not a shielding design code of the types described in the DCD and is not used to calculate dose rates. Its purpose is limited to determining the source in the components of the radwaste system, and the use of the code is consistent with the description of the source term in the DCD.

Per Section 12.3.2.2.2.2 of the reference ABWR DCD, pure gamma dose rate calculations are conducted using point kernel codes. The point kernel codes used are those that are presently widely used in the nuclear industry and include codes such as QAD-CGGP, QAD-CGGP-A, MicroShield, ISOSHLD and G33-GP. For combined gamma and neutron shielding situations, discrete ordinates or Monte Carlo techniques are applied. Typical codes used for this application include the discrete ordinates code DORT and the Monte Carlo code MCNP. Where shielded entries to high radiation areas such as labyrinths are required, a gamma ray scattering code such as G33-GP or a Monte Carlo code such as MCNP is used to confirm the adequacy of the labyrinth design. These computer codes, along with the computer codes identified in the DCD and, when they become available, updated computer codes using similar techniques, are used to design the shielding for STP 3 & 4.

12.3.2.3 Plant Shielding Description

STP DEP 1.2-2

STD DEP 1.8-1

STD DEP 12.3-3

(6) The main steam tunnel extends from the primary containment boundary in the Reactor Building through the Control Building up to the turbine stop valves. The primary purpose of the steam tunnel is to shield the plant complex from N-16 gamma shine in the main steamlines. A minimum of 1.6 meters of concrete or its equivalent (other material or distance) is required on any ray pathway from the main steamlines to any point which may be inhabited during normal operations. The design of the steam tunnel is shown on Figures 1.2-14, 1.2-15, 1.2-20, 1.2-21, and 1.2-33~~*1.2-28*~~*. The tunnel is classified as Seismic Category I in the Reactor Building and in the Control Building and is designed to* IBC~~*UBC*~~ *Seismic Standards in the Turbine Building. The interface between the buildings provides for bayonet connection to permit differential building motion during seismic events and shielding in the areas between buildings. The exact details on the bayonet design are not shown on the referenced arrangement drawings but requires complete shielding in the building interface area.*~~*The tunnel also serves a secondary purpose as a relief and release pathway for high energy events in the Reactor Building. Any high energy event (line break) in the Reactor Building will, through a series of blow-out panels, vent into the steam tunnel and from the steam tunnel through the tunnel vent shaft to the Turbine Building (Figure 1.2-28) for processing to the plant stack. See Subsection 6.2.3.3.1 for more complete description of this function.*~~

12.3.3.2.1 Control Room Ventilation

STD DEP Admin

Outside air coming into the intakes is normally filtered by a particulate filter. If a high radiation level in the air is detected by the ~~*Airborne*~~ Process *Radiation Monitoring System, flow is automatically diverted to another filter train (an outdoor air cleanup unit) that has:*

(1) A ~~*particular*~~ particulate *filter*

(2) A HEPA filter

(3) A charcoal filter

(4) Another HEPA filter

The outdoor cleanup units are located in individual, closed rooms that help prevent the spread of any radiation during maintenance. Adequate space is provided for maintenance activities. The particulate and HEPA filters can be bagged when being *removed from the unit. Before removing the charcoal, any radioactivity is allowed to decay to minimal levels, and is then removed through a connection in the bottom of the filter by a pneumatic transfer system. Air used in the transfer system goes through a HEPA filter before being exhausted, or equivalent. Face masks can be worn during maintenance activities, if desired.*

12.3.3.2.4 Radwaste Building

Subsection 12.3.3.2.4 has been replaced in its entirety with the following standard departure.

STP DEP 9.4-1

The Radwaste Building HVAC is described in detail in subsection 9.4.6.

The radwaste building ventilation systems are engineered and designed to provide the proper environmental conditions within all areas of the radwaste building during normal plant operation. The radwaste building ventilation systems include:

- Radwaste building process area HVAC system.
- Radwaste control room HVAC system.
- Non-Class IE electrical, and HVAC equipment rooms ventilation system.

From a radiological perspective the system is designed to:

- Provide an environment with controlled temperature and airflow patterns to ensure the comfort and safety of plant personnel and to allow for the continuous operation of the equipment and components.
- Maintain positive pressure within the radwaste control room, electrical room and other areas not containing radioactive materials.
- Limit exfiltration from the radwaste areas with potential airborne radioactive contaminants by maintaining sub atmospheric pressure during the normal plant operation.
- Maintain airflow from areas of low potential radioactivity to areas of progressively higher potential radioactivity.
- Limit airborne fission product release to the atmosphere from the ventilation system exhaust during normal plant operation.
- Limit concentration of airborne radioactivity to levels below the values specified in Appendix B to 10CFR20.

Exhaust air from the Radwaste Building is routed to and exhausted through the plant stack. Upon radiation detection in the main exhaust duct, the exhaust air is automatically routed to the air filtration equipment to be filtered through a prefilter and a HEPA filter before being released through the plant stack. A high level of radioactivity detected by a radiation monitor downstream of the HEPA filter also activates an alarm in the radwaste and the main control rooms.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

STD DEP 11.5-1

(2) The Containment Atmospheric Monitoring System (D23/CAM) continuously measures, indicates, and records the gamma radiation levels within the primary containment (drywell and suppression chamber), and activates alarms in the main control room on high radiation levels. As described in Subsection 7.6.2, four gamma sensitive ion chamber channels are provided to monitor gamma radioactivity in the primary containment during normal, abnormal and accident conditions. Each of the four monitoring channels covers the range from 10^{-2} Sv/h~~Gy/h~~ *to* 10^{5} Sv/h~~Gy/h~~*. The CAM System is classified as safety-related.*

12.3.4.1 ARM System Description

STD DEP T1 3.4-1

The Area Radiation Monitoring (ARM) System consists of gamma sensitive detectors, digital area radiation monitors, local auxiliary units with indicators and local audible warning alarms, and recording devices. The detector signals are ~~digitized and optically multiplexed for transmission~~ transmitted *to the radiation monitors in the main control room. Each ARM radiation channel has two independently adjustable trip alarm circuits, one is set to trip on high radiation and the other is set to trip on downscale indication (loss of sensor input). Also, each ARM monitor is equipped with self-test feature that monitors for gross failures and will activate an alarm on loss of power or when a failure is detected. Auxiliary units with local alarms are provided in selected local areas for radiation indication and for activating the local audible alarms on abnormal levels. Each area radiation channel is powered from the non-Class 1E vital 120 VAC source, which is continuously available during loss of offsite power. The recording devices are powered from the 120 VAC instrument bus.* The ARMs are calibrated in accordance with procedures developed from calibration instructions provided by the manufacturer. Periodic calibration verification and channel functional tests are performed with procedures based on pre-operational acceptance testing to verify operability, including alarm functions.

12.3.4.2 ARM Detector Location and Sensitivity

STD DEP 11.5-1

The location of each area detector is shown on the plant layout drawings for each building (Figures 12.3-56 through 12.3-73). The specific area radiation channels for each building are listed in Tables 12.3-3 through 12.3-7, along with reference to map location of the detector, the channel sensitivity range, and the areas for the local alarms. The range and sensitivity of each area radiation channel is classified as follows:

(1) *Range 0.10* µSv/h~~*µGy/h*~~ *to 1* mSv/h~~*mGy/h*~~ *-H (High Sensitivity)*

(2) *Range 1* µSv/h~~*µGy/h*~~ *to 10* mSv/h~~*mGy/h*~~ *-M (Medium Sensitivity)*

(3) *Range 10* µSv/h~~*µGy/h*~~ *to* 10^2 mSv/h ~~*mGy/h*~~ *-L (Low Sensitivity)*

(4) *Range 1* mSv/h~~*mGy/h*~~ *to 10* Sv/h~~*Gy/h*~~ *-LL (Low Low Sensitivity)*

(5) *Range 1* mSv/h~~*mGy/h*~~ *to* 10^2 Sv/h~~*Gy/h*~~ *-VL (Very Low Sensitivity)*

12.3.4.3 Pertinent Design Parameters and Requirements

STP DEP T1 2.5-1

Two high-range radiation channels are provided to monitor radiation from accidental fuel handling. One detector is positioned near the fuel pool and the other located in the fuel handling area. Criticality detection monitors are not needed to satisfy the criticality accident requirements of 10CFR70.24, when specialized high density fuel storage racks preclude the possibility of criticality accident under normal and abnormal conditions. The new and ~~*fuel bundles are stored in racks that are located in the fuel vault while the*~~ *spent fuel bundles are stored in racks that are placed at the bottom of the fuel storage pool. A full array of loaded fuel storage racks are designed to be subcritical, as defined in Sections 9.1 and 9.2. The COL applicant must verify and certify that the design meets the criteria specified in Subsection 12.3.7.3.*

12.3.7 COL License Information

12.3.7.1 Airborne Radionuclide Concentration Calculation

The following site specific supplement addresses COL License Information Item 12.6.

Calculations of the expected airborne radionuclide concentrations are performed, as part of the plant inspections, tests, analyses and acceptance criteria (ITAAC Tier 1 Table 3.2b), to verify the adequacy of the ventilation system prior to fuel load.

12.3.7.2 Operational Considerations

The following site specific supplement addresses COL License Information Item 12.7.

Alarm setpoints are established based on design background radiation levels, which are then, confirmed during the Startup Test Program. The Preoperational Test Program will check for proper calibration of the detectors, and then check the proper functioning of alarms (local and remote, audible and visual) and protective features including alarm setpoints. The Preoperational Test Program will also check for proper response to various loss of power conditions.

In addition to the Area Radiation Monitoring system, radiation monitoring operational considerations, such as procedures for placement, operation and calibration of portable monitors, are established in accordance with the Operational Radiation Protection Program described in Section 12.5S.

The ARMs and airborne radioactivity monitors are calibrated using one or more reference standards, such as ANSI/ANS 6.8.1 with radioactive sources traceable to the National Institute for Standards and Technology (NIST). The instruments are calibrated at one or more points within the response range. A channel calibration that includes a channel functional test is performed periodically on the non safety-related area monitors. A channel calibration that includes a channel functional test is performed on the safety-related area monitors at least once every 18 months or during the refueling outage if the detector is not readily accessible. In the event a calibration is questionable, the channel can be isolated and a more thorough calibration performed. Calibration is also performed after maintenance or replacement of any components that could affect calibration.

12.3.7.3 Requirements of 10CFR70.24

The following site specific supplement addresses COL License Information Item 12.8.

The information demonstrating that the plant meets the criticality accident monitoring requirements of 10CFR70.24 will be provided by meeting the requirements of 10 CFR 50.68(b), as provided for in 10 CFR 70.24(d)(1).(COM 12.3-1).

12.3.7.4 Material Selection

The following site specific supplement addresses the unnumbered COL License Information Item contained in this section of the reference ABWR DCD.

STPNOC continues to monitor industry state-of-the-art developments in material selection options for maintaining exposure ALARA, including Stellite reduction efforts. A graded approach to using the various levels of cobalt in the primary systems has been undertaken as discussed in Subsection 12.3.1.1.2.

12.3.8 Radiation Exposure to Construction Workers During Plant Construction

The following site specific supplement provides information to address RG 1.206, CIII Subsection 12.3.5, dealing with dose to construction workers for multi-unit sites.

Regulatory Guide 1.206, Section C.III.12.3.5, states in part, for multi unit sites, the COL applicant will provide estimated annual dose to construction workers in a new construction area, as a result of radiation from on-site radiation sources from the existing operating plant(s).

During the construction of STP 3 & 4, workers will be exposed to several potential sources of radiation. This section identifies the potential sources of radiation and estimates the doses that workers would receive during the construction of STP 3 & 4 due to the operation of STP 1 & 2. In addition, with STP 3 scheduled to be operational one year earlier than STP 4, STP 3 will be a source of radiation for STP 4 construction workers during that year. Thus, the dose contribution from STP 3 sources of radiation is also evaluated.

Three types of sources are considered: direct radiation, gaseous effluents, and liquid effluents. The maximum annual doses from all three pathways during any year of the construction of STP 3 & 4 occur during the year that STP 3 is operational and STP 4 is under construction. This is further discussed later in this subsection. A comparison of these calculated doses for this time period shows that the limits in 10 CFR 20.1301 and 40 CFR 190.10 for members of the public are satisfied. For 10 CFR 20.1301 the calculated annual dose is 18 mrem TEDE and the limit is 100 mrem TEDE.

Annual Doses for Individuals Working on Unit 4

	Worker Annual Dose (mrem)	
	From Unit 3	**From Units 1, 2 & 3**
Whole body dose from liquid effluents	0.00026	0.032
Organ dose from liquid effluentst	0.00043	0.032
Whole body dose from gaseous effluents	6.6	8.3
Skin dose from gaseous effluents	16	17
Organ dose from radioactive iodine and radioactive material in particulate form from gaseous effluents	12	18

These calculated doses assume a full power equilibrium core with power history for the entire year. It is not expected that Unit 3 will be at 100% power during the full year that STP 4 is still under construction. During this period, STP 3 will be undergoing startup testing. Full power operation is likely to occur only for about 25% of this first year, resulting in decreased annual doses from those presented in the table.

The STP 3 & 4 site will be continually monitored during the construction period and appropriate actions taken to ensure that doses to the construction workers remain ALARA. In addition, the Operational Radiation Protection Program described in Section 12.5S will be in place while Unit 3 is operating with Unit 4 still under construction. Thus, there will be ample oversight to ensure that doses to construction workers remain ALARA during the construction period.

The bases, assumptions, and methods used to calculate the construction worker dose are given below with the maximum annual dose (person-Sieverts) shown in Table 12.3-8.

Dose rates at the construction site are estimated based on dose rate measurements and calculations. Although the construction workers will occupy a large area over the course of the construction period, dose rates are estimated based on average distances from radiation sources.

- Direct radiation: The direct radiation dose rates from STP 1 & 2 sources are based on TLD measurements taken at various onsite locations from 2002 through 2006. This 5-year period provides sufficient data to be representative of plant conditions. Since the construction location for STP 3 & 4 is farther away from STP 1 & 2 than are the respective TLD stations where dose rates are measured from each source, the STP 1 & 2 Offsite Dose Calculation Manual (ODCM) is used to extrapolate the dose rates from the TLD locations to the STP 3&4 location. In determining direct radiation dose rates, it is assumed that the worker is located in the center of the construction area of the unit (either STP 3 or 4) nearest to the source. Given that workers will move about the construction area over the course of a year, it is reasonable to select the center of the area as a representative location for occupancy. No credit is taken for any shielding provided by structures under construction. The estimated total body dose rate to Units 3 and 4 construction workers due to operation of Units 1 and 2 is 2.4 mrem/yr. The estimated total body dose rate to Unit 4 construction workers due to operation of Units 1, 2 and 3 is 9.3 mrem/yr.

- Gaseous effluents: The annual dose rates from the release of gaseous effluents to the maximally exposed member of the public are based on the STP 1 & 2 REMPs for 2002 to 2006. The composite maximum annual dose rate for each organ over these 5 years was calculated using the methodology found in the STP 1 & 2 ODCM. These offsite dose rates are used to estimate construction worker doses. The ratio of the total body dose onsite to that offsite was used to estimate the organ doses onsite for the years 2002 through 2006, yielding the maximum annual onsite doses to construction workers from STP 1&2 over the five-year period. This maximum dose was doubled to address measurement uncertainty. Using the atmospheric dispersion factors in FSAR Section 2.3, the estimated total body dose rate to construction workers from operation of Units 1&2 is 1.7 mrem/yr and 6 mrem/yr to the critical organ and operation of Units 1,2 and 3 is 8.3 mrem/yr total body dose rate and 18 mrem/yr to the critical organ.

- Liquid effluents: The annual dose rates from release of liquid effluents to the maximally exposed member of the public are due to sport fish ingestion and shoreline exposure. Although construction workers would not be exposed to these pathways at the construction site, it was conservatively assumed that the construction workers receive the same doses as the maximally exposed member of the public. Furthermore, the doses are doubled to address measurememt uncertainty. These liquid effluents are based on the STP 1 & 2 REMPs for 2002 to 2006. The composite maximum annual dose rate for each organ over these 5 years was calculated using the methodology found in the STP 1 & 2 ODCM. The offsite dose rates from STP 1, 2, and 3 are calculated at the Little Robbins Slough area due to sport fish ingestion and shoreline exposure. These dose rates are used to estimate construction location doses. The estimated total dose rate to construction workers from operation of Units 1,2 and 3 is 0.032 mrem/yr to both total body and the critical organ.

The calculated annual person-Sievert construction worker doses for total body and critical organ are provided in Table 12.3-8. For the calculation, the manpower estimates are for the timeframe when construction on both Units 3 and 4 is in progress for both units, as it is not feasible to break down the workforce estimates by unit. The estimated doses for each of the three construction phases shown in the table are based on the maximum average annual workforce during that phase.

12.3.9 Minimization of Contamination

The following site-specific supplement provides information to address 10 CFR 20.1406, as implemented by NEI 08-08A.

As stated in Section 12.3 of the Design Control Document (DCD), The Advanced Boiling Water Reactor (ABWR) incorporates many Radiation Protection design features to limit contamination. Section 12.3 of the DCD is incorporated by reference and provides the features summarized below, among others:

- Pumps located in radiation areas are provided with flush lines and in certain cases chemical cleaning capabilities for use prior to maintenance. Pump casing drains provide a means for draining pumps to the sump prior to disassembly, thus reducing the exposure of personnel and decreasing the potential for contamination.
- Instrumentation lines in liquid service for systems containing radioactive fluids are provided with vent and backflush provisions. Reactor vessel sensing lines may be flushed with condensate following reactor blowdown.
- Heat exchangers are constructed of stainless steel or Cu/Ni tubes to minimize the possibility of failure. The heat exchanger design allows for complete draining of fluids from the exchanger, and connections are available for condensate or demineralized water flushing.
- Valves have back seats to minimize leakage through the packing. Teflon gaskets are not used.
- Piping was selected to provide a service life equivalent to the design life of the plant, with consideration given to corrosion allowances and environmental conditions. Piping for systems containing radioactive fluids is welded to the most practical extent to reduce leakage through flanged or screwed connections.
- Floor drains with appropriately sloped floors are provided in shielded cubicles where the potential for spills exist. Smooth, epoxy-type coatings are employed to facilitate decontamination when a spill does occur. Curbs are provided to limit contamination and simplify washdown operations, and expanded metal-type floor gratings are minimized in favor of smooth surfaces in areas where radioactive spills could occur. Equipment and floor drain sumps are stainless steel lined to preclude leakage.
- Material selection consideration is used for systems and components exposed to reactor coolant. Specifically, a graded approach to the use of cobalt lowers the potential for the spread of contamination. Much of the cobalt is removed from

contact with reactor coolant by eliminating Stellite where practical and reducing cobalt in the core stainless steel components.

- Sample stations in the plant contain flushing provisions using demineralized water, and sample station piping drains to plant sumps minimize the possibility of spills. Fume hoods are employed for airborne contamination control. Working areas and fume hoods are stainless steel to ease decontamination should a spill occur, and sample spouts are located above the sink to reduce the possibility of contaminating surrounding areas during the sampling process.
- HVAC systems are designed to limit the extent of airborne contamination by providing air flow patterns from areas of low contamination to more contaminated areas. HVAC Equipment drain sump vents are fitted with charcoal canisters or are piped directly to the Radwaste HVAC System to remove airborne contaminants evolved from discharges to the sump. HVAC penetrations through outer walls of buildings containing radioactive sources are sealed to prevent miscellaneous leaks into the environment.

Additionally, all below grade piping carrying radioactive fluids is located in tunnels. No direct buried piping containing radioactive fluids is incorporated in the STP 3 & 4 design.

Nuclear Energy Institute Report 08-08A, “Generic FSAR Template Guidance for Life Cycle Minimization of Contamination” provides the Minimization of Contamination Program for STP 3 & 4. This NEI template is incorporated by reference with the clarification that design changes for certified design materials are not required by implementation of this program. The evaluations, programs, and procedures required by NEI 08-08A will be issued six months prior to commencement of the Preoperational Test Program.

12.3.9.1 Operational Programs and Operating Procedures

Operational programs and operating procedures will be developed to address 10 CFR 20.1406, and will be issued six months prior to commencement of the Preoperational Test Program. These programs and procedures will include:

- Work practices, preventive maintenance, and procedures to minimize leaks and spills and provide containment and early and adequate detection including instruments for detection. This includes surveillance and monitoring.
- Surveillance and maintenance is performed to mitigate the consequences of undetected leakage over a long period of time.
- Operational practices will be documented such that they are subject to audit and inspection.
- Following construction, establishment of an onsite monitoring program as a part of the environmental monitoring program to prevent offsite migration of radionuclides via an unmonitored pathway.

- To facilitate decommissioning, maintain a system of records detailing contamination events and residual levels of environmental contaminates for the life of the facility, readily accessible to facilitate cleanup.
- Minimizing the generation of radioactive waste as a major operational consideration that will be addressed through careful work planning. Plant procedures will include provisions for proper packaging of wastes for transportation and acceptance by disposal or treatment facilities. Onsite storage is considered in certain circumstances as necessary.

Maintenance and other operating procedures are provided in Subsection 13.5.3.4.2, which includes plant radiation protection, chemical-radiochemical control, and radioactive waste management procedures. Subsection 13.5.3.4.3 lists radiation control procedures, including area radiation monitoring, process radiation monitoring, and meteorological monitoring procedures as well as procedures for discharge of effluents. Operational programs are listed in Table 13.4S-1, and included the Process and Effluent Monitoring and Sampling Program and Radiation Protection Program. 10 CFR 20.1406 requirements are considered in the development of these programs.

12.3.9.2 Design Provisions and Design Features

Design of structures, systems, and components, such as the following, outside the scope of ABWR DCD or departures from the DCD in COLA Part 2, Tier 2, are applicable to 10CFR20.1406 and RG 4.21.

- Section 12.3 describes facility radiation protection design features with STD DEP 12.3-4 adding ARMs and alarm capability.
- Section 12.5 provides supplemental information related to radiation protection facilities and equipment.
- Sections 11.5 and 12.0 describe process and effluent radiological monitoring systems with STD DEP 11.5-1 providing for equipment and instrumentation reliability improvements.
- Section 11.4 describes the solid waste management system and processes with STD DEP 11.4-1 to minimize the generation of waste.
- Section 11.3 describes the gaseous waste management system with STD DEP 11.3-1 providing for optimization.
- Section 11.2 describes the liquid radwaste management system with STD DEP 11.2-1 providing for equipment modernization.

12.3.10 References

STD DEP Admin

12.3-1 N. M. Schaeffer, “Reactor Shielding for Nuclear Engineers”, TID-25951, U.S. Atomic Energy Commission (1973).

12.3-2 J. H. Hubbell, “Photon Cross Sections, Attenuation Coefficients, and Energy Absorption Coefficients from 10 KeV to 100 GeV”, NSRDS-NBS 29, U.S. Department of Commerce, August 1969.

12.3-3 “Radiological Health Handbook”, U.S. Department of Health, Education, and Welfare, Revised Edition, January 1970.

12.3-4 “Reactor Handbook”, Volume III, Part B, E.P. Blizzard, U.S. Atomic Energy Commission (1962).

12.3-5 Lederer, Hollander, and Perlman, “Table of Isotopes”, Sixth Edition (1968).

12.3-6 M.A. Capo, “Polynomial Approximation of Gamma Ray Buildup Factors for a Point Isotropic Source”, APEX-510, November 1958.

12.3-7 Reactor Physics Constants, Second Edition, ANL-5800, U.S. Atomic Energy Commission, July 1963.

12.3-8 ENDF/B-III and ENDF/B-IV Cross Section Libraries, Brookhaven National Laboratory.

12.3-9 PDS-31 Cross Section Library, Oak Ridge National Laboratory.

12.3-10 DLC-7, ENDF/B Photo Interaction Library.

Table 12.3-3 Area Radiation Monitors Reactor Building

No.	Location & Description	Figure #	Sensitivity Range	Local Alarms
2	Reactor area (B)-4F	12.3-62	LL	X
4	Fuel storage pool area (B)-4F	12.3-62	LL	X
5	R/B 4F south area	12.3-62	H	X
7	R/B 3F NW area	12.3-60	H	X
9	CUW control panel area-B3F	12.3-56	H	X
14	R/B 1F SE hatch area	12.3-59	H	X
17	R/B B1F SE hatch area	12.3-58	H	X
26	R/B B3F SW area-RHR “C” equip area	12.3-56	H	X
27	R/B Operating Deck C	12.3-62	H	X
28	R/B Corridor D	12.3-57	M	X
29	R/B Cask Pil	12.3-60	M	X
30	R/B Sampling Room	12.3-58	M	X

Table 12.3-6 Area Radiation Monitors Radwaste Building

No.	Location and Description	Figure #	Sensitivity Range	Local Alarms
1	Electrical Equipment Room El 12300	12.3-67	H	X
2	Control Room El 12300	12.3-67	H	X
3	High Activity Spent Resin Tank Room Tank A El 5300	12.3-66	H	X
4	High Activity Spent Resin Tank Room Tank B El 5300	12.3-66	H	X
5	Trailer Access Area El 12300	12.3-67	H	X
6	LRW Mobile Skid Area El 12300	12.3-67	H	X
7	DAW & Wet Solid Waste Accumulation Area El 12300	12.3-67	H	X
8	High Activity Waste Storage Area El 12300	12.3-67	H	X
9	Waste Sorting Area El 12300	12.3-67	H	X
10	Phase Separator Tank A El 5300	12.3-66	H	X
11	Phase Separator Tank B El 5300	12.3-66	H	X

Table 12.3-7 Area Radiation Monitors Turbine Building

No.	Location & Description	Figure #	Sensitivity Range	Local Alarms
1	~~Condensate Pump Maintenance Area~~ Operation Area (Laydown Space)	12.3-72~~70~~	M	X
2	Corridor (Condensate Sampling & Control Area)	12.3-69~~70~~	M	X
3	~~Offgas Sample & Control Area~~ Rack Room	12.3-68~~70~~	M	X
4	~~RFP 1A. 1B& 1C Area~~ MD-RFP Area	12.3-70	H	X
5	~~Filter Maintenace Area~~ CF Maintenance Area	12.3-~~70~~71	M	X
6	~~Demineralizer Area~~ CD Resin Strainer Room	12.3-~~71~~70	H	X
7	~~SJAE A & Recombiner Area~~ Steam Ejector Units Room	12.3-70~~71~~	H	X
8	~~SJAE B & Recombiner Area~~ OG Recombiner (A) Room	12.3-70~~71~~	H	X
9	~~HP Heaters & Drain Tank Area 1~~ OG Recombiner (B) Room	12.3-70~~71~~	H	X
10	~~HP Heaters & Drain Tank Area 2~~ High Pressure Drain Room	12.3-70~~71~~	H	X
11	~~MSR 1A & 1C Area~~ Moisture Separator and Reheater (A) Room	12.3-72	H	X
12	~~MSR 1B & 1D Area~~ Moisture Separator and Reheater (B) Room	12.3-72	H	X
13	Turbine Building Operating Floor	12.3-73	H	X
14	~~Equipment Main Access Area~~ Corridor (Unloading Bay)	12.3-70~~73~~	H	X

Table 12.3-8 Maximum Annual Dose (Person-Sieverts) to Construction Workers

	Unit 3 Construction Only[1]		Unit 3 & 4 Construction[2]		Unit 4 Construction Only[1]	
	Total Body	**Critical Organ**	**Total Body**	**Critical Organ**	**Total Body**	**Critical Organ**
Direct Radiation	0.076	–	0.142	–	0.175	–
Gaseous Effluents	0.054	0.190	0.101	0.356	0.156	0.339
Liquid Effluents	0.001	0.001	0.002	0.002	0.001	0.001
Total	0.131	0.191	0.245	0.358	0.332	0.340

[1] Dose for construction of one unit.

[2] Dose for construction of two units.

The following figures are located in Chapter 21:

- *Figure 12.3-1* *Reactor Building Radiation Zone Map for Full Power and Shutdown Operation at Elevation – 8200 mm*
- *Figure 12.3-3* *Reactor Building Radiation Zone Map for Full Power and Shutdown Operation at Elevation 4800/8500 mm*
- *Figure 12.3-5* *Reactor Building Radiation Zone Map for Full Power and Shutdown Operation at Elevation 12300 mm*
- *Figure 12.3-6* *Reactor Building Radiation Zone Map for Full Power and Shutdown Operation at Elevation 18100 mm*
- *Figure 12.3-10* *Reactor Building Radiation Zone Map for Full Power and Shutdown Operation at Cross Section View A–A*
- *Figure 12.3-16* *Reactor Building Radiation Zone Map Post-LOCA at Elevation 12300 mm (1F)*
- *Figure 12.3-21* *Reactor Building Radiation Zone Map Post-LOCA at Cross Section A–A*
- *Figure 12.3-37* *Radwaste Building, Radiation Zone Map, Normal Operation at Elevation – 1700 mm*
- *Figure 12.3-38* *Radwaste Building, Radiation Zone Map, Normal Operation at Elevation 5300 mm*
- *Figure 12.3-39* *Radwaste Building, Radiation Zone Map, Normal Operation at Elevation 12300 mm*
- *Figure 12.3-40* *Radwaste Building, Radiation Zone Map, Normal Operation at Elevation ~~18300~~*19100 *mm*
- *Figure 12.3-41* *Radwaste Building, Radiation Zone Map, Normal Operation at ~~Cross~~ Section A-A*
- *Figure 12.3-49* *Turbine Building, Radiation Zone Map, at Elevation 2300 mm~~5300 mm~~*
- *Figure 12.3-50* *Turbine Building, Radiation Zone Map, at Elevation 6300 mm~~12300 mm~~*
- *Figure 12.3-51* *Turbine Building, Radiation Zone Map, at Elevation 12300 mm~~20300 mm~~*
- *Figure 12.3-52* *Turbine Building, Radiation Zone Map, at Elevation 19700 mm~~30300 mm~~*

The following figures are located in Chapter 21 (continued):

The following figures are located in Chapter 21 (continued):

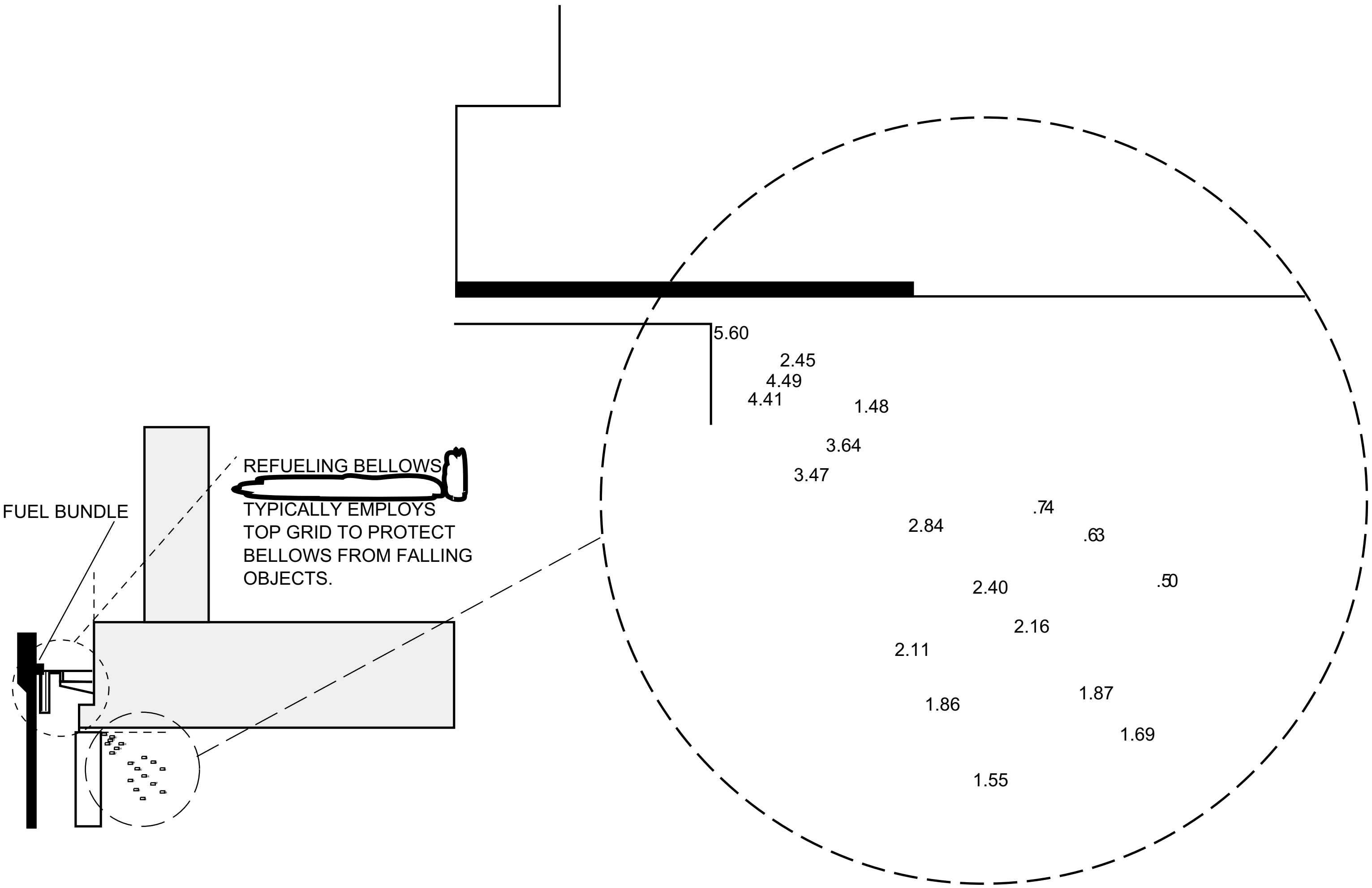

Figure 12.3-74 Upper Drywell Shielding Radiation Dose Rates with Fuel Bundle on Refueling Bellows (Gy/h)