

3.5 Radioactive Waste Management System

Radioisotopes are produced during the normal operation of nuclear reactors, through the processes of fission and activation. Fission products may enter the reactor coolant by diffusing from the fuel pellets and then passing through the fuel cladding either through leaks or, again, by diffusion. The reactor coolant may contain dissolved or suspended corrosion products and non-radioactive materials leached from plant components that can be activated by the neutrons in the reactor core as the water passes through the core. These radioisotopes can enter radioactive waste systems either by plant systems designed to remove impurities or by small leaks that occur in the reactor coolant system and auxiliary systems which are captured by collection systems. Therefore, the plant generates radioactive waste that can be liquid, solid, or gaseous. In accordance with General Design Criterion (GDC) 60 of 10 CFR 50, Appendix A, the Radioactive Waste Management Systems design includes means to suitably control the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. These operational occurrences include condenser leakage, maintenance activities, and process equipment downtime.

Radioactive waste management systems are designed to minimize releases from reactor operations to values as low as reasonably achievable (ALARA). These systems are designed and maintained to meet the requirements of 10 CFR 20 and 10 CFR 50, Appendix I. Annual average releases (plant effluents) are provided in Tables 3.5-1 and 3.5-2 for liquid and gaseous effluents, respectively. These effluents were used to determine the maximum individual and population doses from normal plant operations, as reported in Section 5.4. The simplified process flow diagrams of the liquid, gaseous, and solid radioactive waste management systems are presented in Figures 3.5-1 and 3.5-2. The following subsections summarize the radioactive waste management systems and the amount/type of waste produced by each system. Sections 11.1, 11.2, 11.3, and 11.4 of the COLA FSAR should be consulted for additional detailed information regarding the Advanced Boiling Water Reactor (ABWR) source terms, liquid, gaseous, and solid radioactive waste management systems and specific site design features. Site-specific compliance with 10 CFR 50, Appendix I is also addressed in these sections.

3.5.1 Source Terms

The information provided in this subsection defines the radioactive source terms in the reactor water and steam that serve as design bases for the gaseous, liquid, and solid radioactive waste management systems. Radioactive source term data for boiling water reactors (BWRs) have been incorporated in American National Standard ANSI/ANS-18.1, Radioactive Source Term for Normal Operation of Light Water Reactors (Reference 3.5-1). The standard provides bases for estimating typical concentrations of the principal radionuclides that may be anticipated over the lifetime of a BWR plant. The source term data are based on the cumulative industry experience at operating BWR plants. Therefore, they reflect the influence of a number of observations made during the transition period from operation with fuel of older designs to operation with fuel of current improved designs. The source terms specified in this section were obtained by applying the procedures of Reference 3.5-1 for

estimation of typical source terms and adjusting the results upward as appropriate to ensure conservative bases for design. The methodology for adjustment of the source terms is provided in Section 11.1.3 of the DCD. The plant parameters and nuclide removal rate parameters used to perform the adjustment calculations are contained in DCD Tables 11.1-6 and 11.1-7, respectively. A major parameter in the adjustment of the source term is the reactor thermal power. These are 3400 MWt for the reference BWR and 3926 MWt for the ABWR.

The various radionuclides included in the design basis source term have been categorized as fission products or activation products and tabulated in the subsections that follow. The lists do not necessarily include all radionuclides that may be detectable or theoretically predicted to be present. Those that have been included are considered to be potentially significant with respect to one or more of the following considerations:

- Plant equipment design
- Shielding design
- Understanding system operation and performance
- Measurement practicability
- Evaluation of radioactivity in effluents to the environment

3.5.1.1 Fission Products

3.5.1.1.1 Noble Gas Fission Products

Typical concentrations of the 13 principal noble gas fission products as observed in steam flowing from the BWR reactor vessel are provided in the source term standard ANSI/ANS-18.1 (Reference 3.5-1). Concentrations in the reactor water are considered negligible because all of the gases released to the coolant are assumed to be rapidly transported out of the vessel with the steam and removed from the system with the other noncondensables in the main condenser.

The design basis noble gas source term for the ABWR is selected such that the mix is that of Reference 3.5-1 and the total of the release rates of the 13 noble gases from the vessel is 3700 MBq/s (megabecquerels per second) as evaluated after a 30-minute decay. The noble radiogas source term rate after 30-minute decay has been used as a conventional measure of the fuel leakage rate, since it is conveniently measurable and is consistent with the nominal 30-minute offgas holdup system used on a number of early plants. A design basis noble gas release rate of 3700 MBq/s after a 30-minute decay has historically been used for the design of the gaseous waste management systems in BWR plants (Reference 3.5-2) with satisfactory results. It was selected on the basis of operating experience with consideration given to several judgmental factors, including the implications to environmental releases, system contamination, and building air contamination. The design basis value is considered to represent a long-term average value. An instantaneous release rate of several times the annual average permissible release rate limit may be permitted as long as it is consistent with

the limits established by 10 CFR 20 and the annual average is not exceeded. Normal operational noble gas release rates for the ABWR are expected to be approximately 555 MBq/s as evaluated after a 30-minute decay (Reference 3.5-5, Subsection 11.1.1.2). This may be compared with normal release rates on the order of 1850 MBq/s based on fuel experience through the mid 1970s (Reference 3.5-3). Consequently, with continued application of the design basis of 3700 MBq/s, there is increased margin relative to expected release rates when operating with fuel of modern design. The design basis noble gas source terms are presented in Table 3.5-3.

3.5.1.1.2 Radioiodine Fission Products

For many years, design basis radioiodine source terms for BWRs have been specified to be consistent with an I-131 leak rate of 25.9 MBq/s from the fuel (Reference 3.5-2). Experience indicated that I-131 leakage rates this high would be approached only during operation with substantial fuel cladding defects. It would not be anticipated that full power operation would continue for any significant period of time with fuel cladding defects as severe as might be indicated by I-131 leakage as much as of 25.9 MBq/s. The design basis reactor water radioiodine concentrations for the ABWR are based on the relative mix of radioiodines in reactor water predicted by the data of Reference 3.5-1 with magnitudes increased such that the I-131 concentration is consistent with a release rate of 25.9 MBq/s from the fuel. This provides a substantial margin relative to the expected I-131 release rate of approximately 3.7 MBq/s.

Reference 3.5-1 specifies expected concentrations of the five principal radioiodines in reactor water for a reference BWR design and provides the bases for adjusting the concentrations for plants with relevant plant parameters that do not match those of the reference plant. The design basis concentrations in reactor water are presented in Table 3.5-4. The ratio of concentration in reactor steam to concentration in reactor water (carryover ratio) is taken to be 0.015 for radioiodines (Reference 3.5-1). Consequently, the design basis concentrations of radioiodines in steam are determined by multiplying the values of Table 3.5-4 by the factor of 0.015.

3.5.1.1.3 Other Fission Products

This category includes all fission products, other than noble gases and iodines, and also includes transuranic nuclides. Some of the fission products are noble gas daughter products that are produced in the steam and condensate system. The only transuranic nuclide that is detectable in significant concentrations is Np-239. Concentrations of those radionuclides that are typically observable in the coolant are provided in Reference 3.5-1 for a reference BWR plant. The reference plant concentrations were adjusted to obtain estimates for the ABWR plant. The design basis reactor water concentrations are presented in Table 3.5-5. The ratio of concentration in steam to concentration in water (carryover) for these nuclides is expected to be less than 0.001 (Reference 3.5-5, Section 11.1.1.3). The design basis concentrations in steam are obtained by multiplying the values in Table 3.5-5 by 0.001.

3.5.1.2 Activation Products

3.5.1.2.1 Coolant Activation Products

The coolant activation product of primary importance in BWRs is N-16. ANSI-18.1, Radioactive Source Term for Normal Operation of Light Water Reactors (Reference 3.5-1), specifies a concentration of 1.85 MBq/g (megabecquerels per gram) in steam leaving the reactor vessel. This is treated as essentially independent of reactor design because both the production rate of N-16 and the steam flow rate from the vessel are assumed to vary in direct proportion to reactor thermal power. The design basis N-16 concentration in steam for the ABWR is 1.85 MBq/g as shown in Table 3.5-6. This value has, in fact, been used as the design basis concentration for GE BWRs since the early 1970s, and operating experience indicates that it is conservative (Reference 3.5-5, Section 11.1.2.1). Under hydrogen water chemistry conditions, the N-16 source term is 6 times greater due to the more volatile nature of the reduced chemical form of N-16 compound, and is, therefore, regarded as 11.1 MBq/g. It should be noted that a portion of the source term traditionally identified as "N-16" actually represents C-15, which is present to the extent of no more than approximately 0.555MBq/g.

3.5.1.2.2 Non-Coolant Activation Products

Radionuclides are produced in the coolant by neutron activation of circulating impurities and by corrosion of irradiated system materials. Typical reactor water concentrations for the principal activation products are contained in Reference 3.5-1. The values of Reference 3.5-1 were adjusted to ABWR conditions as described in DCD Section 11.1.2.2 to obtain the conservative design basis reactor water concentrations shown in Table 3.5-7. The steam carryover ratio for these isotopes is estimated to be less than 0.001 (Reference 3.5-5, Section 11.1.3). A factor of 0.001 is applied to the Table 3.5-7 values to obtain the design basis concentrations in steam.

3.5.1.2.3 Tritium

Tritium is produced by activation of naturally occurring deuterium in the primary coolant and, to a lesser extent, as a fission product in the fuel (Reference 3.5-2). The tritium is primarily present as tritiated oxide. Since tritium has a long half-life (12 years) and will not be affected by cleanup processes in the system, the concentration will be controlled by the rate of loss of water from the system by evaporation or leakage. All plant process water and steam will have a common tritium concentration. The concentration reached will depend on the actual water loss rate; however, References 3.5-1 and 3.5-3 specify a typical concentration of 3.7E-04 MBq/g, which is stated in Reference 3.5-3 to be based on BWR experience adjusted to account for liquid recycle. This value is taken to be applicable to the ABWR.

3.5.1.2.4 Argon-41

Argon-41 is produced in the reactor coolant as a consequence of neutron activation of naturally occurring Argon-40 in air which is entrained in the feedwater. The Argon-41 gas is carried out of the vessel with the steam and removed from the system with the noncondensable gases in the main condenser. Observed Argon-41 levels are highly variable due to the variability in air in-leakage rates into the system. Reference 3.5-3

specifies an Argon-41 release rate from the vessel of 1.48 MBq/s for a 3400 MWt Reference BWR. Based on adjusting to the ABWR thermal power (3926 MWt), a design basis Argon-41 release rate of 1.70 MBq/s is specified for the ABWR.

3.5.1.3 Process Leakage Sources

Process leakage results in potential release of noble gases and other volatile fission products via ventilation systems. Liquid from process leaks is collected and routed to the liquid-solid radwaste system. With the effective process offgas treatment systems now in use (including the ABWRs for STP), airborne releases have been greatly reduced. Building ventilation releases now account for a higher fraction of the total release than before the deployment of the augmented offgas systems.

Leakage of fluids from the process system results in the release of radionuclides into plant buildings (Reference 3.5-5, Section 11.1.5). In general, the noble gases will remain airborne and will be released to the atmosphere with little delay via the building ventilation exhaust ducts. Other radionuclides will partition between air and water and may plate out (deposit) on metal surfaces, concrete, and paint. Radioiodines are found in ventilation air as methyl iodide and as inorganic iodine (particulate, elemental, and hypoiodous acid forms).

As a consequence of normal steam and water leakage into the drywell, equilibrium drywell concentrations will exist during normal operation (Reference 3.5-5, Section 11.1.5). Purging of this activity from the drywell to the environment will occur via the reactor building ventilation or Standby Gas Treatment System (expected to be an infrequent operation) and will make minor contributions to total plant releases.

Airborne release data from BWR building ventilation systems and the main condenser mechanical vacuum pump have been compiled and evaluated in Reference 3.5-4. Releases due to process leakage are reflected in the airborne release estimates discussed in Subsection 3.5.3.

3.5.2 Liquid Waste Management System

3.5.2.1 Design Objective

The Liquid Waste Management System (LWMS) is designed to segregate, collect, store, and process potentially radioactive liquids generated during various modes of plant operation: startup, normal operation, hot standby, shutdown, and refueling. The system is designed such that it may be operated to maximize the recycling of water within the plant, which would minimize the releases of liquid to the environment. Maximizing recycling serves to minimize the potential for exposure of personnel in unrestricted areas from the liquid release pathway. The Process Diagram for the LWMS is shown on Figure 3.5-1. The basic reference for the LWMS is Reference COLA Section 11.2.

3.5.2.1.1 Design Criteria

- The criteria considered in the design of this system include:
- Minimizing solid waste shipped for burial
- Reducing personnel exposure
- Minimizing offsite releases

Maximizing the quality of water returned to the condensate and feedwater systems. In accordance with GDC 60 of 10 CFR 50, Appendix A, the radioactive waste management systems design includes means to suitably control the release of radioactive materials in gaseous (Figure 3.5-2) and liquid (Figure 3.5-1) effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. These operational occurrences include condenser leakage, maintenance activities, and process equipment downtime. The LWMS provides a discharge to the Main Cooling Reservoir (MCR). Radiation monitoring equipment is placed on this line to measure the activity discharged and to assure that specified limits are not exceeded. A high radiation signal from this monitor will close the discharge valve and terminate the discharge. The discharge from this single discharge line to the MCR is controlled administratively to assure that the discharge will meet the requirements of 10 CFR 20 and Appendix I of 10 CFR 50.

The single discharge line is fed by sample tanks (i.e., tanks that have the necessary connections to allow analysis before discharge). In addition to providing a means for a controlled (i.e., batch) discharge, the sample tanks also function as surge tanks. Liquid discharge to the MCR may be initiated from only one sample tank at a time. The discharge sequence is initiated manually. No single error or failure will result in discharge.

The LWMS is designed to treat process liquids with radionuclide concentrations associated with the design basis fuel leakage and produce water suitable for recycle to condensate storage. When plant water balance considerations require the discharge of processed liquids, the concentrations of radionuclides in the effluent will be controlled to meet the applicable requirements. Radiation exposure to people in unrestricted areas resulting from liquid waste discharged during normal operation and anticipated operational occurrences will be less than the values specified in 10 CFR 50, Appendix I. The design will maintain occupational exposure ALARA in accordance with NRC Regulatory Guide 8.8 (Reference 3.5-6) while operating with the design basis fuel leakage. The average annual liquid releases by nuclide are presented in Table 3.5-1.

3.5.2.2 Occupational Exposure

Design features to minimize occupational exposure include:

- Design of equipment to minimize service time.
- Location of instruments requiring calibration in a central station outside of equipment cells.
- Arrangement of shield wall penetrations to avoid direct exposure to normally occupied areas.
- Piping design to minimize crud traps and plate out (there are no socket welds in contaminated piping systems).
- Provision for remote pipe and equipment flushing.
- Use of remote viewing and handling equipment as appropriate.
- A shielded sampling station to minimize exposure time.
- Controlled tank vents.

3.5.2.3 System Description

The LWMS is composed of four subsystems designed to collect, treat, and recycle or discharge different categories of wastewater. The four subsystems are the Low Conductivity Subsystem, High Conductivity Subsystem, Detergent Waste Subsystem, and the Chemical Drain Subsystem.

The LWMS is designed with adequate margin so that liquid waste should not be discharged except as needed to maintain the plant water balance. Operational flexibility is provided to ensure continued plant operation. Under these conditions, discharge of excess water processed through the LCW and/or the HCW Subsystem may be desirable. The various stream flow rates and the different combinations of events that supply water to the LWMS system for treatment have been tabulated. The LWMS is conservatively designed to handle the largest volume expected to be produced. The liquid waste management subsystems have ample capacity to process the maximum daily generation rate of liquid wastes as shown in Table 3.5-8.

Regeneration of the condensate demineralizers will not be performed. Rather, the resin will be replaced when necessary. The use of titanium-tubed (or stainless steel tubed) condensers has made leaks from condenser tubing virtually non-existent (Reference 3.5-5, Section 11.2.3). Also, the use of condensate high efficiency filters before the condensate demineralizers has reduced the amount of insoluble solids that come into contact with the resin. As a result, it is expected that resin replacement will be less than once per year. Decanting of the cleanup water phase separator will be done as required following cleanup water filter demineralizer backwashes. The LCW can process this volume in addition to the other wastes. The components of the LWMS are sized based on processing the maximum daily volume within 24 hours. This

criterion is more conservative than basing the sizing on normal expected waste volumes.

3.5.2.3.1 Low Conductivity Waste Subsystem

The low conductivity waste (LCW) subsystem collects and processes clean radwaste (i.e., water of relatively low conductivity). Equipment drains and backwash transfer water are typical of wastes found in this subsystem. These wastes are collected, filtered by two stages of filtration for removal of insolubles, processed through reverse osmosis membranes, demineralized on a mixed resin, deep-bed demineralizer for removal of solubles, processed through a second polishing demineralizer, and then routed to condensate storage via sample tanks unless high conductivity requires recycling for further treatment. Mobile process equipment is utilized for the filtration, reverse osmosis, and demineralization unit operations.

The LCW filters and mixed-bed demineralizers are pressure vessels. The collection and sample tanks operate at atmospheric pressure. The LWMS is essentially a manual-start and automatic-stop process. COL Section 11.5 contains detailed information regarding process and radiation instruments. The instrumentation allows for the initiation of processing from the shielded control room area. To ensure that the system performs its intended function in the event of failure of key components, redundancy is provided.

Input to parallel tankage is a feature of the design. Upon high-level signals, inputs are automatically routed to a parallel tank. If input should continue, high-high level results in annunciation in the radwaste control room. The state of system operation such as water level of tanks, position of valves, and pump operating condition are continuously shown on the radwaste system control panels. The operator will be able to see the changes in the system when the automatic transfer has occurred. Individual tanks and process equipment are located in separately shielded rooms where practical. Pumps and valves in general are located in dedicated operating galleries. Piping to and from these pumps and valves penetrate shield walls only to the extent necessary to connect to the process equipment. Runs of piping between process equipment are contained either within the shielded areas or shielded pipe runs so that operating personnel exposure is kept to a minimum and within 10 CFR 20 limits.

The condensate storage tank, which is located outdoors, has liquid level monitoring with alarms in the control room. The tank overflows, drains and sample lines are routed to the radwaste system (Reference 3.5-5, Section 3.5.2.1). A dike is provided around the tank to prevent runoff in the event of a tank overflow. A drain within the dike is routed to the radwaste system.

All radwaste system tanks located outside reactor containment and containing radioactive liquids are indoors and are provided with liquid level monitoring. High liquid level conditions are alarmed locally and in the main control room (Figure 3.5-1 & Reference 3.5-5, Section 3.5.2.1). All radwaste system tank overflows, drains, and sample lines are sent to the radwaste system. All tanks have curbs or elevated thresholds with floor drains routed to the radwaste system. Leakage is prevented from

entering unmonitored and non-radioactive systems and ductwork in the area. Radiation exposures are controlled in accordance with Regulatory Guide 8.8.

3.5.2.3.2 High Conductivity Waste Subsystem

The high conductivity waste (HCW) subsystem collects and processes dirty radwaste (i.e., water of relatively high conductivity and solids content). Floor drains are typical of wastes found in this subsystem. Processing is similar to that for LCW. Mobile process equipment is utilized for the filtration, reverse osmosis, and demineralization unit operations.

3.5.2.3.3 Detergent Waste Subsystem

This subsystem collects and processes detergent wastes from personnel showers and laundry operations. Normally, detergent wastes are collected in the hot shower drain receiver tank, processed through strainers to the hot shower drain sample tank, sampled, and discharged. Most laundry will be sent offsite for processing by a licensed vendor.

3.5.2.3.4 Chemical Waste Subsystem

The chemical waste collected in the chemical drain collection tank consists of laboratory wastes and decontamination solutions. After accumulating in the chemical drain collection tank, chemical drains are recirculated. A sample is taken and if discharge standards are met, the waste is discharged. Off-standard quality waste is recycled to a high conductivity waste collector tank for further processing.

3.5.2.4 Estimated Releases

The estimated annual releases of radionuclides in liquid effluents are provided in Table 3.5-1. Calculated off-site dose rates are provided in Section 5.4.

3.5.2.4.1 Release Points

The release pathway for liquid discharge to the environment is the discharge line from the sample tanks as indicated on the process diagram (Figure 3.5-1).

The radwaste discharges from STP 3 & 4 will be to the MCR at one point, the main circulating water system outfall, similar to STP 1 & 2. Radiation monitoring equipment is placed on each of the STP 3 & 4 discharge lines to measure the activity discharged and to ensure that specified limits are not exceeded. High radiation alarms are provided in the radwaste control room and the main control room. Further discussion regarding details of radiological process monitoring can be found in COLA Section 11.5. Liquid discharge to the MCR may be initiated from only one sample tank at a time, and the discharge sequence is initiated manually.

3.5.2.5 Dilution Factors

Dilution factors used in evaluating the release of liquid effluents are site-specific and are described in detail in FSAR Section 12.2.2.5.1. Liquid discharges are assumed to be to the MCR via the circulating water system having a flow of 272,550 m³/hr. Also, it

is assumed that there will be a dilution factor of ten before subsequent consumption or recreational activity involving liquid effluent. (Note: The MCR, with a volume of approximately 200 million cubic meters, provides for a dilution factor much greater than the design assumption of 10.) Table 3.5-1 discharges are in compliance with 10 CFR 20 and 10 CFR 50, Appendix I.

3.5.2.6 Cost-Benefit Analysis for Liquid Effluent Population Doses

Compliance with Appendix I to 10 CFR 50 numerical guidelines for offsite radiation doses as a result of liquid radioactive effluents during normal plant operations, including anticipated operational occurrences, is provided in FSAR Subsection 12.2.2.2. To demonstrate compliance with 10 CFR 50, Appendix I, Section II, paragraph D, a cost-benefit analysis was performed in accordance with the guidance in Regulatory Guide 1.110. The results of the cost-benefit analysis are documented in FSAR Subsection 11.2.1.2.

3.5.3 Gaseous Waste Management System

Gaseous radioisotopes are produced during the normal operation of STP 3 & 4. They include gaseous fission products and gaseous isotopes formed by neutron reactions with the reactor coolant and contained gases. These gases are retained in the plant systems and removed in a controlled fashion through the gaseous waste management system.

The gaseous waste management system collects waste from multiple sources and delays their release to allow short-lived nuclides to decay. The remaining activity is released in a controlled manner to the environment through the plant stack, a monitored release point.

The objective of the gaseous waste management system (hereafter referred to as the Offgas System) is to process and control the release of gaseous radioactive effluents to the site environs so as to maintain the exposure of people in unrestricted areas to radioactive gaseous effluents ALARA (10 CFR 50, Appendix I). This will be accomplished while maintaining occupational exposure ALARA and without limiting plant operation or availability.

The Offgas System provides for holdup and decay of radioactive gases in the offgas from the main condenser air evacuation system and consists of process equipment along with monitoring instrumentation and control components. The Offgas System minimizes and controls the release of radioactive material into the atmosphere by delaying and filtering the offgas process stream containing the radioactive isotopes of krypton, xenon, iodine, nitrogen, and oxygen sufficiently to achieve adequate decay before discharge from the plant. The Offgas System design minimizes the explosion potential of the gas mixture through recombination of radiolytic hydrogen and oxygen under controlled conditions. The basic reference for the Offgas System is the DCD, Section 11.3 (Reference 3.5-5) as modified in Section 11.3 of the FSAR.

3.5.3.1 Design Criteria

The Offgas System is designed to limit the dose to offsite individuals from routine station releases to significantly less than the limits specified in 10 CFR 20 and to operate within the relevant limits specified in the technical specifications. As a conservative design basis for the Offgas System, an average annual noble radio gas source term (based on 30-minute decay) of 3700 MBq/s is used as discussed in Subsection 3.5.1.1.1. The system is mechanically capable of processing three times the source term without affecting delay time of the noble gases. With a main condenser air in-leakage rate of 51 standard cubic meters per hour or 30 standard cubic feet per minute (10 standard cubic feet per minute per condenser shell), this treatment system results in a delay of 46 hours for krypton and 42 days for xenon. Using the given isotopic activities at the discharge of the Offgas System, the decontamination factor for each noble gas isotope can be determined. Subsection 3.5.1 presents source terms for normal operational and anticipated occurrence releases to the primary coolant. Tables in this section, if not designated otherwise, are based on a design basis offgas release rate of 3700 MBq/s of noble gases and 25.9 MBq/s of I-131. For normal expected operation, the leak rates and doses are expected to be less than one quarter of the design basis numbers.

The average annual exposure at the site boundary during normal operation from all gaseous sources is expected to meet the dose objectives of 10 CFR 50, Appendix I. The radiation dose design basis for the treated offgas is to provide sufficient holdup until the required fraction of the radionuclides has decayed with the daughter products retained by the charcoal and the High Efficiency Particulate Air (HEPA) filter. The Offgas System equipment is selected, arranged, and shielded to maintain occupational exposure ALARA in accordance with NRC Regulatory Guide 8.8.

3.5.3.2 Process Description

3.5.3.2.1 Process Functions

Major process functions of the Offgas System include:

- Dilution of air ejector offgas with steam to less than 4% hydrogen by volume.
- Recombination of radiolytic hydrogen and oxygen into water to reduce the gas volume to be treated and the explosion potential in downstream process components.
- Two-stage condensation of bulk water vapor, first using Turbine Building Cooling Water (TCW) and then chilled water as the coolant, reducing the gaseous waste stream temperature to 10°C or less
- Dynamic adsorption of krypton and xenon isotopes on charcoal at about 25°C
- Filtration of offgas
- Monitoring of offgas radioactivity levels and hydrogen gas concentration

- Release of processed offgas to the atmosphere
- Discharge of liquids to the main condenser and radwaste systems.

Major process functions of the ventilation systems may be found in COLA FSAR Section 9.4.

3.5.3.2.2 Process Equipment

Major process equipment of the Offgas System consists of:

- Process piping starting from the final steam dilution jets steam jet air ejector of the Main Condenser Evacuation System (not a part of the Offgas System)
- Recombiner trains which include a preheater, a recombiner, and a condenser per train
- Cooler-condensers
- Activated charcoal adsorbers
- HEPA filter
- Monitoring instrumentation
- Process instrumentation and controls
- Offgas Evacuation System

3.5.3.2.3 Process Facility

The Offgas System process equipment is housed in a reinforced concrete structure to provide adequate shielding. Charcoal adsorbers are installed in a temperature-monitored and controlled vault. The facility is located in the Turbine Building to minimize piping. TCW is used as the coolant for the offgas condensers.

The gaseous waste stream is then cooled to 10°C or less in the cooler condenser. Chilled water (7°C) is used from the HVAC Normal Cooling Water (HNCW) System. The gaseous waste stream is heated to approximately 25°C by ambient heating in the charcoal vault. Radiation shielding design provides adequate protection of instrumentation and plant personnel required to monitor and operate the system.

3.5.3.3 Offgas System Description

3.5.3.3.1 Main Condenser Offgas

The significant gaseous wastes discharged to the Offgas System during normal plant operation are radiolytic hydrogen and oxygen, main condenser air leakage, and radioactive isotopes of krypton, xenon, nitrogen and oxygen. These non-condensable gases are collected in the main condenser and discharged to the offgas system by the main condenser evacuation system.

Radioactive particles are present as a result of radioactive decay of the noble gas parents. These particulates are removed from the offgas stream by the condensation, adsorption, and filtration equipment. Therefore, effectively no radioactive particulates are released from the Offgas System to the plant stack or vent. Radioiodines (notably I-131) may be present in significant quantities in the reactor steam and to some extent carried over through the condensation stages of the Offgas System. Removal of iodine takes place in the passage of process gas through the activated charcoal adsorbers, so that essentially no iodine is released from the Offgas System to the plant stack or vent.

Releases of gaseous wastes to the atmosphere are controlled to assure the calculated dose to any person in an unrestricted area is maintained below the dose objectives of Appendix I to 10 CFR 50. Every reasonable effort has been made to keep radiation exposures and release of radioactive materials ALARA.

Calculated doses from releases from the offgas system are provided in Section 5.4.

3.5.3.3.2 Process Design

A flow diagram for the Offgas System has been included as Figure 3.5-2. DCD Section 11.3 and FSAR Section 11.3 contain a more detailed discussion of the process design, equipment components, and equipment operation.

3.5.3.4 Other Radioactive Gas Sources

Radioactive gases are present in the power plant buildings as a result of process leakage and steam discharges. The process leakage is the source of the radioactive gases in the air discharged through the ventilation system. Details regarding the design of the ventilation system are provided in COLA Section 9.4, the radiation activity levels from the ventilation systems in COLA Section 12.3, and the ventilation flow rates in COLA Section 9.4.

3.5.3.5 Instrumentation and Control

Control and monitoring of the offgas process equipment is performed both locally and remotely from the main control room. Instrument components are installed, wherever possible, in accessible areas to facilitate operation and maintenance. Only instrument sensing elements are permitted behind shield walls. The temperature of the gaseous waste stream is measured in the preheater and at various locations in the recombiner to ensure that recombination is occurring. The gaseous waste stream temperature is also measured after both the offgas condenser and the cooler condenser to ensure the stream is cooled sufficiently to remove undesired moisture. All of these temperatures are alarmed in the main control room. The flow rate of the air ejector offgases downstream of the recombiner is continuously recorded. This flow rate, in conjunction with activity concentrations in Bq/cc (becquerels per cubic centimeter), as measured by the monitor downstream of the recombiners and the monitor downstream of the charcoal adsorbers, will permit monitoring fission gases from the reactor, calculation of offgas discharge to the vent in MBq and calculation of the charcoal adsorber system performance. Activity release that would exceed the maximum permitted

instantaneous value is alarmed, and causes closure of the final process gas release valve, terminating the release.

3.5.3.6 Radioactive Releases

3.5.3.6.1 Release Points

The primary release point for STP 3 & 4 is the Reactor Building stack (one per unit). This stack serves as the release point for the Reactor Building, Turbine Building, and Radwaste Building. Other exhaust points for clean releases are the roof top vents for the Control and Service Buildings and the Service Building health physics room roof vent. The Reactor Building stack is a roof-mounted steel shell in a steel framework extending to a height of 76 meters above ground level. The closest plant buildings to the stack are the Control Building to a height of 10.7 meters and the Turbine Building to a height of 43 meters above grade.

3.5.3.6.2 Projected Releases

Projected releases from the plant stack have an estimated total flow rate of at least 566,000 m³/h through the 2.4-meter diameter circular stack, which extends 76 meters above ground level. Ventilation releases are assumed to be less than 40°C. Table 3.5-2 lists expected gaseous isotopic releases from one ABWR plant.

3.5.3.7 Cost-Benefit Analysis for Gaseous Effluent Population Doses

Compliance with Appendix I to 10 CFR 50 numerical guidelines for offsite radiation doses as a result of gaseous or airborne radioactive effluents during normal plant operations, including anticipated operational occurrences, is provided in FSAR Subsection 12.2.2.2. To demonstrate compliance with 10 CFR 50, Appendix I, Section II, paragraph D, a cost-benefit analysis was performed in accordance with the guidance in Regulatory Guide 1.110. The results of the cost-benefit analysis are documented in FSAR Subsection 11.3.11.1.

3.5.4 Solid Waste Management System

Solid radioactive wastes are produced by multiple methods in a nuclear power station. The waste can be either dry or wet solids, and the source can be an operational activity, maintenance, or another function.

The Solid Waste Management System (SWMS) will collect, process, and package solid radioactive wastes generated as a result of normal plant operation, including anticipated operational occurrences. The system design ensures that the solid radioactive wastes would be collected, monitored, segregated, stored, and packaged for shipment in a manner that minimizes exposure to plant personnel and the public in accordance with 10 CFR 20 and 10 CFR 50, Appendix I. The basic reference for the SWMS is the COLA, Section 11.4.

3.5.4.1 Design Bases

3.5.4.1.1 Design Objective

The SWMS provides the capability for processing and packaging wastes from the Cleanup Water System, the Fuel Pool Cooling and Cleanup System, the Suppression Pool Cleanup System, the Condensate Polishing System, and the Radwaste System itself. Wastes from these systems will consist of spent resin and backwash slurries.

The SWMS also provides a means of:

- Packaging dry active waste materials, such as HVAC filters and nonflammable organic materials
- Packaging contaminated metallic materials and incompressible solid objects such as small tools and equipment components

The SWMS is designed so that the failure or maintenance of any frequently used component does not impair system or plant operation. Storage is provided ahead of the process equipment to allow holdup for radioactive decay and as required in case of a delay in processing due to maintenance. The operating philosophy of the SWMS is manual start and automatic stop with all functions interlocked to provide a fail-safe mode of SWMS operation.

3.5.4.1.2 Design Criteria

Collection, processing, packaging, and storage of radioactive wastes will be performed to maintain any potential radiation exposure to plant personnel ALARA in accordance with Regulatory Guide 8.8 (Reference 3.5-6) and within the limits of 10 CFR 20. Table 3.5-9 summarizes the radiological activities in the solid waste, based on the sources of waste. Processing of wastes will be done to meet 10 CFR 61 criteria for wastes being shipped to burial, or for wastes being sent to a processor, the waste acceptance criteria of the waste processor. Packaged wastes will be shipped in conformance with the requirements in 49 CFR 173, Subpart I. Sufficient onsite storage is provided to hold at least six months production of radwaste.

The radiation levels of wet solid waste being added to shipping containers are measured to prevent the container from exceeding acceptable radiation levels.

3.5.4.2 System Description

3.5.4.2.1 General Description

The process diagram for the solid radwaste system is shown on Figure 3.5-1. The major SWMS equipment consists of the following:

- Backwash receiving tanks and forwarding pumps
- Spent resin tanks and phase separators
- Decant pumps and transfer pumps

- High integrity container (HIC) dewatering and drying equipment
- Connections and auxiliaries for mobile systems

The following design features are incorporated into the SWMS:

- All components and piping which contain slurries have flushing connections.
- All tanks and equipment that use compressed gases for transport or drying of resins or filter sludges are vented to the plant ventilation exhaust system. The vents are designed to prevent liquids and solids from entering the plant ventilation system.

3.5.4.2.2 General Requirements

The SWMS processes both wet and dry solid wastes in compliance with:

- The releases of radioactive materials to an unrestricted area are within the concentration limits of 10 CFR 20, Appendix B, Table II. All solid wastes are monitored for radiation before either processing or disposal as nonradioactive waste. It is expected that some dry solid waste will be disposable as nonradioactive. All liquids and gases from solid waste processing are treated by the liquid waste management system or by the Radwaste Building ventilation system.
- The SWMS has sufficient storage of both unprocessed and processed wastes to deal with both normal and anticipated operational occurrences. These storage facilities have been designed with adequate shielding to protect the operators from excessive radiation. Wastes will be processed separately by type and source.
- The SWMS is designed in accordance with Regulatory Guide 1.143.

3.5.4.2.3 Spent Resins and Sludges

Wet solid wastes are spent resins and sludges from powdered resins and filter backwashing. The spent resins and sludges are sent to vendor-supplied mobile dewatering systems. See Table 3.5-11 for "Expected Waste Volumes Generated Annually by Each 'Wet' Solid Waste Source and Tank Capacities" (For one ABWR).

3.5.4.2.4 Dry Active Waste

Dry active waste is loaded on marine containers or into shielded containers as required to meet shipping limitations. Estimated volumes of dry active waste generated annually are provided in Table 3.5-10.

3.5.4.2.5 Environmental and Exposure Control

Solid radwaste processing equipment will be designed in accordance with Regulatory Guide 1.143 to minimize the spread of contamination and maintain operator dose rates ALARA.

3.5.4.2.6 Malfunction Analysis

The process system is protected from overfilling the waste container by a high-level trip that closes the waste container fill valve and a camera that enables the operator to view the level of waste in the container and take action as required.

3.5.4.2.7 Shipment

Containers normally can be shipped after processing, provided the proper shielding is available, without exceeding U.S. Department of Transportation radiation limits. If 49 CFR 173 dose limitations cannot be met, the containers are temporarily stored until the appropriate shielding is available.

Normally, high-integrity containers will be shipped after they are filled. While awaiting shipment, the high-integrity containers are temporarily stored with shielding in an appropriate storage location. A barrier to restrict access will be placed around the shielding. The radiation dose rate at the barrier shall be 50 $\mu\text{Gy/hr}$ (micrograys per hour) or less. Storage contingencies in the event that an off-site waste disposal facility becomes unavailable are discussed in Section 4.5. There is space for either box containers or high integrity containers in the solid waste storage area in the radwaste building. Low activity waste collected in buildings outside of the radwaste building will be temporarily stored locally until transferred to the radwaste building for storage/shipment. Uncontaminated shipping containers and vehicles may be stored outside. The annual volumes of solid radwaste to be shipped offsite are estimated in Table 3.5-12 (for one ABWR).

The packaged waste is stored until it is shipped offsite to a licensed radwaste processing facility or burial site. Onsite processing of the waste is in accordance with the existing STP Process Control Program (PCP) to meet transportation and radwaste processing facility or burial site waste acceptance requirements. The existing STP PCP will be modified as required to ensure it addresses the STP 3 & 4 waste streams. Packaging of waste for offsite shipment complies with applicable Department of Transportation and NRC regulations that specify requirements for shipment of radioactive material, including 49 CFR 172, 173 and 178, and 10 CFR 71.

3.5.4.2.8 Contingencies for Class B and C Wastes

It is expected that Class B and C wastes will constitute about 5% of the low level radioactive waste (LLRW) that will be generated by the plant, with the balance being Class A waste (with small amounts of greater than Class C wastes that are subject to separate disposal provisions). As of July 1, 2008, the LLRW disposal facility in Barnwell, South Carolina is no longer accepting Class B and C waste from sources in states such as Texas that are outside of the Atlantic Compact. However, the disposal facility in Clive, Utah, is still accepting Class A waste from out of state.

STP 3 & 4 plans to load fuel in 2015 and begin operation no earlier than 2016 and therefore will not be generating Class B and C waste until then. Typically it takes about a year after fuel load before a sufficient quantity of B/C waste is generated to warrant a shipment for disposal. By 2016, it is likely that a disposal facility will be available that

will accept Class B and C waste from sources in Texas. In particular, in 2004, Waste Control Specialists applied for a license from the Texas Commission on Environmental Quality (TCEQ) to develop a disposal facility in Andrews County, Texas for Class A, B and C waste. As of August 2008, the Executive Director of Texas Commission on Environmental Quality has recommended to the full commission that Waste Control Specialists be granted the license to operate a Class A, B and C waste site for the Texas Compact. Therefore, by the time that STP 3 & 4 begins to operate, it is reasonable to expect that the Andrews County disposal facility will be in operation, and that STP 3 & 4 will be able to ship its Class A, B and C waste offsite to the Andrews County disposal facility.

However, in the event that there are no disposal facilities that will accept Class B and C wastes from sources in Texas at the time the plant begins operation, there are several options available for storage of such waste pending shipment offsite once a disposal facility becomes available:

- As provided in the FSAR Section 11.4.1.2, STP 3 & 4 Radwaste Building is designed to have 6 months of storage capacity for LLRW. Since Class B and C waste constitute only about 5% of the total LLRW, the Radwaste Building has about 10 years of safe storage capacity if it used solely for storage of Class B and C waste and if Class A waste is promptly shipped offsite. Also the waste tables in Section 11.4 are not based on a volume reduction (VR) process. Hence it is possible to extend this storage time frame by utilizing commercially available off-site waste processors. For example a VR of 8 to 10 is presently being achieved for resins.
- As described in FSAR Section 11.4.1.2, STP 1 & 2 have an Onsite Staging Facility (OSF) that could be used to store waste from STP 3 & 4 if that should become necessary. As explained in the Section 11.4.2.3.2 of the STP 1 & 2 UFSAR, the OSF has a 5-year LLRW storage capacity for Units 1 and 2. If that storage capacity were to be devoted to Class B and C waste, the OSF would have approximately 100 years of safe storage capacity for two units or 50 years for four units, assuming that Class A waste is shipped offsite in the normal course of business. Just as explained above, the storage duration could be extended by utilizing VR.
- If still additional storage capacity were eventually to be needed, STP 3 & 4 could construct storage facilities in accordance with applicable NRC guidance, such as Regulatory Issue Summary (RIS) 2008-12, Considerations for Extended Interim Storage of Low-Level Radioactive Waste by Fuel Cycle and Materials Licenses, and NUREG 0800 Section 11.4.

If STP 3 & 4 were to need to store Class B and C waste for an extended period of time, it would implement the provisions of RIS 2008-12 and NUREG 0800 Section 11.4.

The onsite storage facilities are designed to maintain onsite and offsite radiological doses within the limits in 10 CFR Part 20 and to maintain occupational exposures ALARA.

3.5.5 References

- 3.5-1 American National Standard Radioactive Source Term for Normal Operation of Light Water Reactors, ANSI/ANS-18.1.
- 3.5-2 Skarpelos, J.M. and R.S. Gilbert, Technical Derivation of BWR 1971 Design Basis Radioactive Material Source Terms, March 1973 (NEDO-10871).
- 3.5-3 Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors, NUREG-0016, Revision 1, January 1979.
- 3.5-4 Marrero, T.R., Airborne Releases From BWRs for Environmental Impact Evaluations, March 1976 (NEDO-21159).
- 3.5-5 ABWR Design Control Document, Tier 2 Manual, General Electric, Revision 4, March 1997.
- 3.5-6 Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as is Reasonably Achievable, Regulatory Guide 8.8, June 1978, U.S. Nuclear Regulatory Commission, 1978.
- 3.5-7 Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants, Regulatory Guide 1.143., Rev. 2 November 2001.

Table 3.5-1 Annual Average Liquid Releases

Radionuclide	Annual Average Liquid Releases (MBq/yr)
I-131	3.35E+02
I-132	7.15E+01
I-133	1.38E+03
I-134	4.22E+00
I-135	4.03E+02
H-3	2.96E+05
C-14	0.00E+00
Na-24	1.87E+02
P-32	2.10E+01
Cr-51	6.30E+02
Mn-54	1.47E+02
Mn-56	7.55E+01
Co-56	0.00E+00
Co-57	0.00E+00
Co-58	3.10E+02
Co-60	5.69E+02
Fe-55	3.50E+02
Fe-59	8.24E+01
Ni-63	6.29E+01
Cu-64	4.67E+02
Zn-65	1.63E+01
Rb-89	0.00E+00
Sr-89	1.16E+01
Sr-90	9.92E-01
Y-90	0.00E+00
Sr-91	4.64E+01
Y-91	8.70E+00
Sr-92	1.64E+01
Y-92	6.27E+01
Y-93	5.05E+01
Zr-95	4.10E+01

Table 3.5-1 Annual Average Liquid Releases (Continued)

Radionuclide	Annual Average Liquid Releases (MBq/yr)
Nb-95	1.16E+01
Mo-99	9.66E+01
Tc-99m	2.10E+02
Ru-103	1.21E+01
Rh-103m	0.00E+00
Ru-106	3.29E+02
Rh-106	0.00E+00
Ag-110m	4.44E+01
Sb-124	0.00E+00
Te-129m	3.12E+00
Te-131m	3.10E+00
Te-132	5.00E-01
Cs-134	4.18E+02
Cs-136	2.78E+01
Cs-137	6.57E+02
Cs-138	2.96E-02
Ba-140	6.23E+01
La-140	0.00E+00
Ce-141	1.10E+01
Ce-144	1.44E+02
Pr-143	3.00E+00
Nd-147	7.40E-02
W-187	8.24E+00
Np-239	3.51E+02

(FSAR Section 12.2, Table 12.2-22)

Table 3.5-2 Annual Average Gaseous Releases

Radionuclide	Annual Average Gaseous Releases (MBq/yr)
Kr-83m	3.1E+01
Kr-85m	7.8E+05
Kr-85	2.1E+07
Kr-87	9.3E+05
Kr-88	1.4E+06
Kr-89	8.9E+06
Kr-90	1.2E+01
Xe-131m	1.9E+06
Xe-133m	3.2E+03
Xe-133	8.9E+07
Xe-135m	1.5E+07
Xe-135	1.7E+07
Xe-137	1.9E+07
Xe-138	1.6E+07
Xe-139	1.5E+01
I-131	9.6E+03
I-132	8.1E+04
I-133	6.3E+04
I-134	1.4E+05
I-135	8.9E+04
H-3	2.7E+06
C-14	3.4E+05
Na-24	1.5E+02
P-32	3.4E+01
Ar-41	2.5E+05
Cr-51	1.3E+03
Mn-54	2.0E+02
Mn-56	1.3E+02
Fe-55	2.4E+02
Fe-59	3.0E+01
Co-58	8.9E+01
Co-60	4.8E+02
Ni-63	2.4E-01
Cu-64	3.7E+02
Zn-65	4.1E+02

Table 3.5-2 Annual Average Gaseous Releases (Continued)

Radionuclide	Annual Average Gaseous Releases (MBq/yr)
Rb-89	1.6E+00
Sr-89	2.1E+02
Sr-90	2.6E+00
Y-90	1.7E+00
Sr-91	3.7E+01
Sr-92	2.9E+01
Y-91	8.9E+00
Y-92	2.3E+01
Y-93	4.1E+01
Zr-95	5.9E+01
Nb-95	3.1E+02
Mo-99	2.2E+03
Tc-99m	1.1E+01
Ru-103	1.3E+02
Rh-103m	4.1E+00
Ru-106	7.0E-01
Rh-106	7.0E-01
Ag-110m	7.4E-02
Sb-124	6.7E+00
Te-129m	8.1E+00
Te-131m	2.8E+00
Te-132	7.0E-01
Cs-134	2.3E+02
Cs-136	2.2E+01
Cs-137	3.5E+02
Cs-138	6.3E+00
Ba-140	1.0E+03
La-140	6.7E+01
Ce-141	3.4E+02
Ce-144	7.0E-01
Pr-144	7.0E-01
W-187	7.0E+00
Np-239	4.4E+02

(FSAR Section 12.2, Table 12.2-20)

Table 3.5-3 Noble Gas Source Terms in Steam

Isotope	Decay Constant, 1/hours	Source Term t = 30 min (MBq/s)
Kr-83m	3.73 E-01	1.1E+02
Kr-85m	1.55 E-01	2.0E+02
Kr-85	7.37 E-06	8.9E-01
Kr-87	5.47 E-01	5.6E+02
Kr-88	2.48 E-01	6.3E+02
Kr-89	1.32 E+01	6.3E+00
Xe-131m	2.41 E-03	7.4E-01
Xe-133m	1.30 E-02	1.1E+01
Xe-133	5.46 E-03	3.1E+02
Xe-135m	2.72 E+00	2.5E+02
Xe-135	7.56 E-02	8.1E+02
Xe-137	1.08 E+01	2.6E+01
Xe-138	2.93 E+00	7.8E+02
TOTAL		3.7E+03

(Reference 3.5-5, Table 11.1-1)

Table 3.5-4 Iodine Radioisotopes in Reactor Water

Isotope	Decay Constant, 1/hours	Concentration (MBq/g)
I-131	3.59E-03	5.9E-04
I-132	3.03E-01	5.2E-03
I-133	3.33E-02	4.1E-03
I-134	7.91E-01	8.9E-03
I-135	1.05E-01	5.6E-03

(Reference 3.5-5, Table 11.1-2)

Table 3.5-5 Nonvolatile Fission Products in Reactor Water

Isotope	Decay Constant, 1/hours	Concentration (MBq/g)
Rb-89	2.74E+00	7.8E-04
Sr-89	5.55E-04	1.2E-05
Sr-90/Y-90	2.81E-06	8.5E-07
Sr-91	7.31E-02	5.2E-04
Sr-92	2.56E-01	1.4E-03
Y-91	4.93E-04	4.8E-06
Y-92	1.96E-01	8.1E-04
Y-93	6.8E-02	5.2E-04
Zr-95/Nb-95	4.41E-04	9.6E-07
Mo-99/Tc-99m	1.05E-02	2.4E-04
Ru-103/Rh-103m	7.29E-04	2.4E-06
Ru-106/Rh-106	7.83E-05	3.7E-07
Te-129m	8.65E-04	4.8E-06
Te-131m	2.31E-02	1.2E-05
Te-132	8.89E-03	1.2E-06
Cs-134	3.84E-05	3.3E-06
Cs-136	2.22E-03	2.2E-06
Cs-137	2.63E-06	8.9E-06
Cs-138	1.29E+00	1.5E-03
Ba-140/La-140	2.26E-03	4.8E-05
Ce-141	8.88E-04	3.7E-06
Ce-144/Pr-144	1.02E-04	3.7E-07
Np-239	1.24E-02	1.0E-03

(Reference 3.5-5, Table 11-1-3)

NOTE: Nuclides shown as pairs are assumed to be in secular equilibrium. The parent decay constant and concentration are shown.

Table 3.5-6 Coolant Activation Products in Reactor Water and Steam

Isotope	Half-Life	Steam Concentration (MBq/g)	Reactor Water Concentration (MBq/g)
N-16	7.13 s	1.9E+00 [1]	3.6E+00 [2]

[1] Use 11 MBq/g for operation with hydrogen water chemistry

[2] Valid at core exit

(Reference 3.5-5, Table 11.1-4)

Table 3.5-7 Non-Coolant Activation Products in Reactor Water

Isotope	Decay Constant, 1/hours	Concentration (MBq/g)
Na-24	4.63E-02	1.3E-03
P-32	2.02E-03	2.4E-05
Cr-51	1.04E-03	7.4E-04
Mn-54	9.53E-05	8.5E-06
Mn-56	2.69E-01	6.7E-03
Co-58	4.05E-04	2.4E-05
Co-60	1.50E-05	4.8E-05
Fe-55	3.04E-05	1.2E-04
Fe-59	6.33E-04	3.7E-06
Ni-63	7.90E-07	1.2E-04
Cu-64	5.42E-02	3.7E-03
Zn-65	1.18E-04	2.4E-05
Ag-110m	1.16E-04	1.2E-07
W-187	2.90E-02	3.7E-05

(Reference 3.5-5, Table 11.1-5)

Table 3.5-8 Capability of Liquid Radwaste Subsystems to Process Expected Wastes [1]

Subsystem	Process Rate	Maximum Process Capacity [6]	Normal Waste Generation Rate	Maximum Daily Generation Rate
LCW [2]	34 m ³ /h	653 m ³ /day	55 m ³ /day	615 m ³ /day
HCW [3]	34 m ³ /h	653 m ³ /day	15 m ³ /day	65 m ³ /day
DW [4]	34 m ³ /h	653 m ³ /day	4 m ³ /day	12 m ³ /day
CW [5]	4 m ³ /h	192 m ³ /day	2 m ³ /day	2 m ³ /day

[1] FSAR Table 11.2-2

[2] LCW – low conductivity waste

[3] HCW – high conductivity waste

[4] DW – detergent waste

[5] CW – Chemical Waste

[6] For one day only

Table 3.5-9 Solid Radwaste Component Inventories [1]

Halogens		Soluble Fission Products		Insoluble Fission Products		Activation Products	
Isotope	MBq	Isotope	MBq	Isotope	MBq	Isotope	MBq
I-131	7.38E+07	Rb-89	1.37E+05	Y-91	1.78E+06	Na-24	2.44E+07
I-132	1.50E+07	Sr-89	4.59E+06	Y-92	3.58E+06	P-32	6.41E+06
I-133	1.08E+08	Sr-90	3.58E+05	Y-93	6.62E+06	Cr-51	2.44E+08
I-134	9.86E+06	Y-90	3.58E+05	Zr-95	3.60E+05	Mn-54	3.49E+06
I-135	4.67E+07	Sr-91	6.18E+06	Nb-95	3.60E+05	Mn-56	2.16E+07
		Sr-92	4.75E+06	Ru-103	8.42E+05	Co-58	9.14E+06
		Mo-99	1.97E+07	Rh-103M	8.42E+05	Co-60	2.01E+07
		Tc-99M	1.97E+07	Ru-106	1.51E+05	Fe-55	2.63E+07
		Te-129M	1.64E+06	Rh-106	1.51E+05	Fe-59	1.33E+06
		Te-131M	4.51E+05	La-140	1.22E+07	Ni-63	5.06E+07
		Te-132	1.15E+06	Ce-141	1.25E+06	Cu-64	5.93E+07
		Cs-134	7.54E+05	Ce-144	1.50E+05	Zn-65	9.81E+06
		Cs-136	3.13E+05	Pr-143	1.50E+05	Ag-110M	4.91E+04
		Cs-137	2.08E+06			W-187	1.11E+06
		Cs-138	1.01E+06				
		Ba-140	1.22E+07				
		Np-239	6.99E+07				

[1] Sum total of inventories from CUW (Reactor Water Cleanup System) Backwash Receiving Tank, CF (Condensate Filtration) Backwash Receiving Tank, Phase Separator, Spent Resin Storage Tank, Liquid Waste (LW) Backwash Receiving Tank (FSAR Section 12.2, Tables 12.2-15a-d & 12.2-15l)

Table 3.5-10 Estimates of Expected Annual “Dry” Solid Wastes

Dry Waste Source	Volume Generation (m ³ /yr)
Combustible Waste	225
Compactable Waste	38
Other Waste	100

(FSAR Section 11.4, Table 11.4-2)

Table 3.5-11 Expected Waste Volume Generated Annually by Each “Wet” Solid Waste Source and Tank Capacities

Wet Waste Source	Volume Generation (m ³ /yr) [1]
CUW F/D Sludge	4.7
FPC F/ D Sludge	1.8
Condensate Filter sludge	4.6
LCW Filter Sludge	0.6
HCW Filter Sludge	3.8
Condensate Demineralizer resin	18.0
LCW Demineralizer Resin	1.8
HCW Demineralizer Resin	1.8
HCW RO Reject	73.0

CUW F/D – Reactor Water Cleanup filter demineralizer

FPC F/D – Fuel Pool Cooling and Cleanup filter demineralizer

[1] FSAR 11.4, Table 11.4-1

Table 3.5-12 Estimated Shipped Solid Waste Volumes

Waste Type	Shipped Volume (m ³ /yr)
Combustible Waste	225
Compressible Waste	38
Resins and Sludges	110
Other Waste	100
Mixed Waste	0.5

From FSAR Section 11.4 (Table 11.4-3)

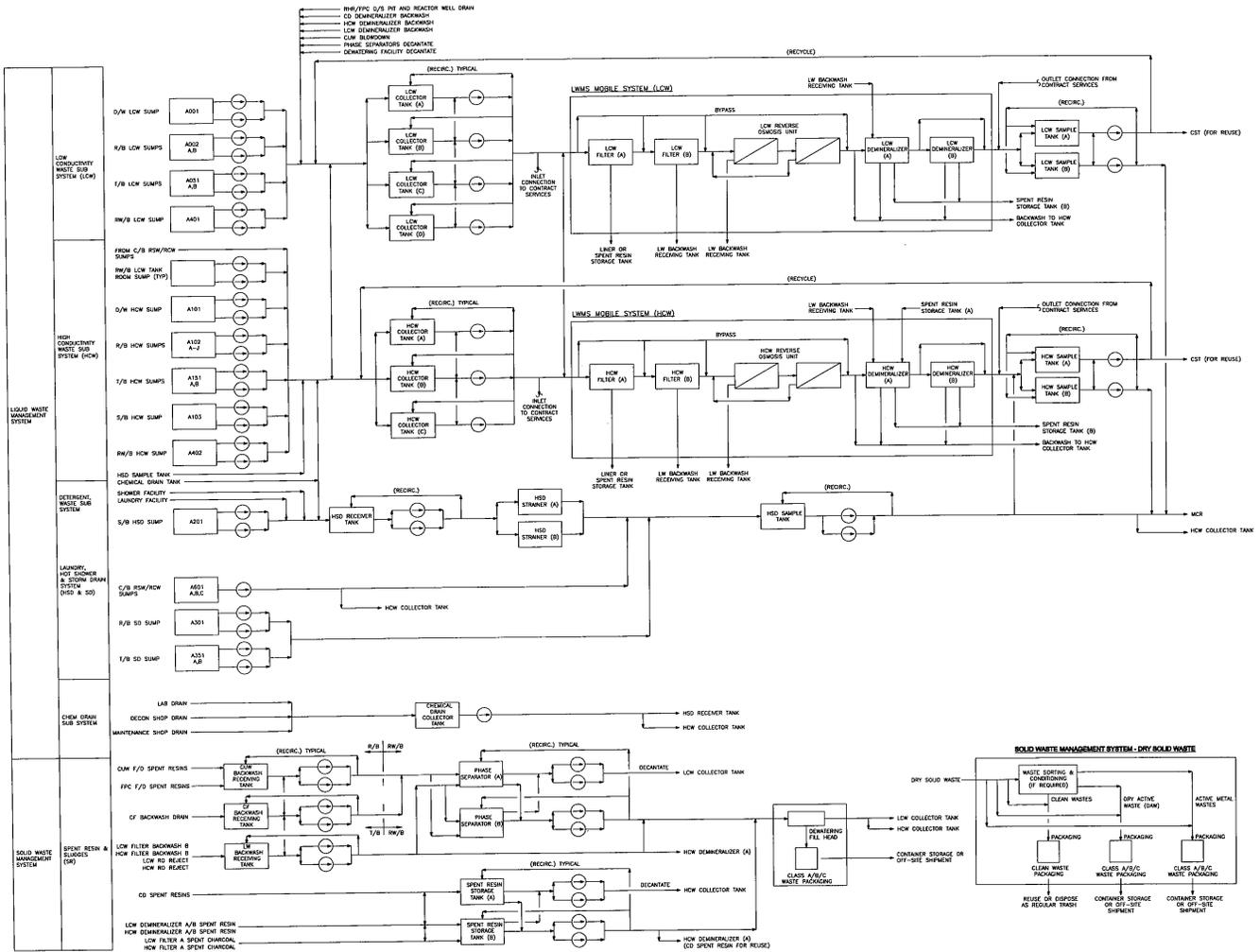


Figure 3.5-1 Liquid & Solid Radioactive Waste Management Systems

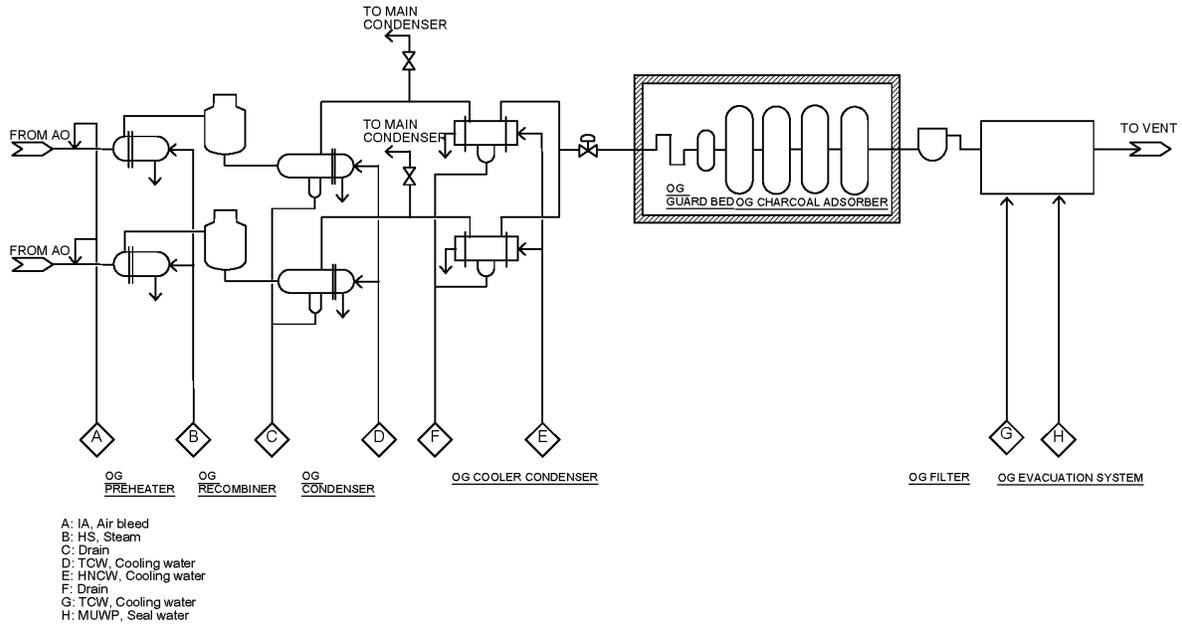


Figure 3.5-2 Gaseous Radioactive Waste Management System