11 RADIOACTIVE WASTE MANAGEMENT

11.1 DESIGN CRITERIA AND SOURCE TERMS

11.1.1 GENERAL DESIGN CRITERIA

11.1.1.1 AIF General Design Criterion 70 (1967)

The following design criterion was used during the licensing of Ginna Station. It was included in the Atomic Industrial Forum version of proposed criteria issued by the AEC for comment on July 10, 1967. These criteria are discussed in Section 3.1.1.

CRITERION: The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control must be justified
(a) on the basis of 10 CFR 20 requirements, for normal operation and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10 CFR 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence (AIF-GDC 70).

With respect to the above criterion, liquid, gaseous, and solid waste disposal facilities are designed so that discharge of effluents and offsite shipments are in accordance with applicable NRC regulations and guidelines.

Radioactive fluids entering the waste disposal system are collected in sumps and tanks until determination of subsequent treatment can be made. They are sampled and analyzed to determine the quantity of radioactivity, with an isotopic breakdown if necessary. Before any attempt is made to discharge, they are processed as required and then released under controlled conditions. The system design and operation are characteristically directed toward minimizing releases to unrestricted areas. Discharge streams are appropriately monitored and safety features are incorporated to preclude excessive releases in accordance with the Offsite Dose Calculation Manual (ODCM).

The bulk of the radioactive liquids discharged from the reactor coolant system are processed and retained inside the plant by the chemical and volume control system recycle train. This minimizes liquid input to the waste disposal system which processes relatively small quantities of generally low-activity level wastes. The processed water from waste disposal, from which most of the radioactive material has been removed, is discharged through a monitored line into the circulating water discharge.

Radioactive gases are pumped by compressors through a manifold to one of the gas decay tanks where they are held a suitable period of time for decay. Cover gases in the nitrogen blanketing system are reused to minimize gaseous wastes. During normal operation, gases are discharged intermittently at a controlled rate from these tanks through the monitored plant vent. The system is provided with discharge controls so that environmental conditions do not restrict the release of radioactive effluents to the atmosphere.

Liquid wastes are processed to remove most of the radioactive materials. Spent resins from the demineralizers and filter cartridges are packaged and shipped to an appropriate licensed facility. This facility may volume reduce prior to disposal or may be a direct to ground disposal facility.

All solid waste is placed in suitable containers and stored onsite until shipment offsite is made for processing and disposal.

The design of the radioactive waste management systems was reviewed in 1972 (*Reference 1*) on the bases of the General Design Criteria contained in Appendix A to 10 CFR 50, which were promulgated after the licensing of Ginna Station, and the then proposed Appendix I to 10 CFR 50.

11.1.1.2 Appendix A General Design Criteria (1972)

Compliance of the design with the 1972 version of the General Design Criteria of Appendix A to 10 CFR 50 is discussed in Section 3.1.2. Specifically, compliance of the original design with General Design Criteria 60, 63, and 64 as they relate to the radioactive waste management systems was either shown or achieved by design modifications.

Compliance with the requirements of the then proposed Appendix I to 10 CFR 50 is documented in *Reference 1*. On May 5, 1975, the NRC published Appendix I to 10 CFR 50 which finalized the numerical guides for design objectives and limiting conditions for operation to meet the criterion "as low as practicable." Rochester Gas and Electric Corporation (RG&E) responded to these requirements in submittals to the NRC in June and October 1976, (*References 2* and *3*).

Implementation of the overall requirements of 10 CFR 50, Appendix I, as to the utilization of radwaste treatment equipment to ensure that radioactive discharges are as low as reasonably achievable (ALARA), has been formalized in the Technical Specifications requirements for the Radioactive Effluent Controls Program and the Offsite Dose Calculation Manual (ODCM).

11.1.2 SOURCE TERMS

The waste disposal system collects and processes all potentially radioactive plant wastes for removal from the plant site within limitations established by applicable governmental regulations. Fluid wastes are sampled and analyzed to determine the quantity of radioactivity, with an isotopic breakdown if necessary, before any attempt is made to discharge. These wastes are then released under controlled conditions. The system is capable of processing all wastes generated during continuous operation of the primary system, assuming that fission products escape from 1% of the fuel pellets into the reactor coolant by diffusion through defects in the cladding.

As secondary functions, system components supply hydrogen and nitrogen to primary system components, as required during MODES 1 and 2; provide facilities to transfer fluids from inside the containment to other systems outside the containment; and act as backup to engineered safety features systems components during the long-term recirculation phase of postaccident operation (see Section 6.3.3).

11.1.2.1 Liquid Sources

During MODES 1 and 2 the waste disposal system processes liquids from the following sources:

- Equipment drains and leaks.
- Radioactive chemical laboratory drains.
- Hot shower drains.
- Decontamination area drains.

Secondary regenerations and secondary system drains are released directly to the discharge canal via radioactivity-monitored paths.

The system also collects and transfers liquids from the following sources directly to the chemical and volume control system for processing:

- Reactor coolant loops.
- Pressurizer relief tank.
- Reactor coolant pump secondary seals.
- Excess letdown during startup.
- Accumulators.
- Valve and reactor vessel flange leakoffs.

Liquid wastes are generated primarily by plant maintenance and service operations.

Source term influents to the waste disposal system have changed considerably since the original design of the system. However, the current influent quantities into the system are smaller than the quantities for which the system was originally designed. Actual liquid waste discharge quantity figures are provided in the Radioactive Effluent Release Reports required by the Technical Specifications.

11.1.2.2 Gaseous Sources

The primary source of gas received by the waste disposal system during MODES 1 and 2 is cover gas displaced from the chemical and volume control system holdup tanks as they fill with liquid. Gaseous wastes consist primarily of hydrogen stripped from coolant discharged to the chemical and volume control system holdup tanks during boron dilution; nitrogen and hydrogen gases purged from the chemical and volume control system volume control tank when degassing the reactor coolant; and nitrogen from the closed gas blanketing system. The gas decay tank capacity will permit 45 days decay of waste gas before discharge. Gaseous activity concentrations stated in terms of annual release values are given in Section 11.3.

11.1.2.3 Radioactivity Inputs

Radioactivity inputs into the waste disposal system are based on primary coolant equilibrium activity. Reactor coolant activity concentrations, based on a design with 1% fuel defects, are

discussed in Section 9.3.4.4.9 and listed in Tables 9.3-9 and 9.3-10. Tritium production, control, and discharge is discussed in Sections 9.3.4.4.8, 9.3.4.4.9, and Table 9.3-11a.

REFERENCES FOR SECTION 11.1

1. Rochester Gas and Electric Corporation, Technical Supplement Accompanying Application for Full-Term Operating License, August 1972.

2. Letter from L. D. White, Jr., RG&E, to R. A. Purple, NRC, Subject: 10 CFR 50, Appendix I, dated June 3, 1976.

3. Letter from L. D. White, Jr., RG&E, to A. Schwencer, NRC, Subject: Supplemental Information, 10 CFR 50, Appendix I, dated October 25, 1976.

11.2 LIQUID WASTE MANAGEMENT SYSTEM

11.2.1 DESIGN BASES

Radioactive fluids entering the waste disposal system are collected in sumps and tanks until determination of subsequent treatment can be made. They are sampled and analyzed to determine the quantity of radioactivity, with an isotopic breakdown if necessary. Before any attempt is made to discharge, they are processed as required and then released under controlled conditions. The system design and operation are characteristically directed toward minimizing releases to unrestricted areas. Discharge streams are appropriately monitored and safety features are incorporated to preclude releases in excess of the limits of 10 CFR 20 and to maintain radioactive discharges to levels as low as reasonably achievable (ALARA) according to the requirements of 10 CFR 50, Appendix I.

The bulk of the radioactive liquids discharged from the reactor coolant system are processed and retained inside the plant by the chemical and volume control system recycle train. This minimizes liquid input to the waste disposal system which processes relatively small quantities of generally low activity level wastes. The processed water from the waste disposal system, from which most of the radioactive material has been removed, is discharged through a monitored line into the circulating water discharge.

Codes applying to components of the liquid waste management system are shown in Table 11.2-1.

11.2.2 SYSTEM DESCRIPTION

The liquid waste disposal system process flow diagrams are shown in Drawings 33013-1259 and 33013-1270 through 33013-1272. Summaries of design data and performance data for the liquid waste management system are shown in Tables 11.2-2 and 11.2-3, respectively. A description of the liquid waste processing system components follows.

11.2.2.1 Laundry and Hot Shower Tanks

Two 600-gal tanks (Drawing 33013-1259) retain liquid wastes containing detergents originating from the hot shower and from the laundry, prior to its dismantling in 1994. The tanks are constructed of stainless steel. When a tank has been filled, its contents are pumped to the waste holdup tank.

11.2.2.2 Chemical Drain Tank

The 375-gal chemical drain tank (Drawing 33013-1259) is austenitic stainless steel and collects drainage from the hot section of the chemistry laboratory. After analysis, the tank contents are pumped to the waste holdup tanks or to the waste condensate tanks.

11.2.2.3 Reactor Coolant Drain Tank and Pumps

The reactor coolant drain tank (Drawing 33013-1272, Sheets 1 and 2) collects all water sources that are potentially tritiated. This water is recycled as much as practicable to minimize tritium release to the environment. Its sources include the following:

- Reactor coolant system loop drains.
- Reactor coolant pump No. 2 seal leakoff.
- Excess letdown.
- Valve and reactor vessel flange leakoffs.
- Safety injection accumulator drains.
- Pressurizer relief tank drain.
- Refueling canal drains.

Two reactor coolant drain tank pumps (Drawing 33013-1272, Sheets 1 and 2) located in the residual heat removal pit transfer the waste water to the chemical and volume control system holdup tanks. From there it is sent to the boric acid evaporator system to be reprocessed. One reactor coolant drain tank pump is a 150-gpm centrifugal pump, and the other is a 50-gpm centrifugal pump powered respectively from Class 1E motor control centers 1C and 1D. Both pumps are operated automatically by a reactor coolant drain tank level controller. These pumps can also be used to pump the water from the refueling canal and cavity and the fuel transfer canal to the refueling water storage tank (RWST) if needed. The reactor coolant drain tank is a 350-gal tank with pressure, temperature, and level indications on the waste disposal panel in the auxiliary building.

Additionally, two air-operated valves in the reactor coolant drain tank pump suction lines will shut on a containment isolation signal.

The reactor coolant drain tank is normally kept pressurized between 0.5 psig and 2 psig with nitrogen to minimize air in-leakage. The tank can be vented to the radwaste vent header via two air-operated valves, which will shut on a containment isolation signal.

11.2.2.4 Waste Holdup Tank

11.2.2.4.1 Liquid Waste Sources

The waste holdup tank (21,000 gal) shown in Drawing 33013-1270, Sheets 1 and 2 is the collection point for most primary liquid wastes, via gravity drain where possible. Other drains, such as basement level drains, drain to a 375-gal capacity sump tank that is then pumped to the waste holdup tank by two 20-gpm sump level-controlled centrifugal pumps. The waste holdup tank can also receive pump discharges from the reactor coolant drain tank, the chemical drain tank, the laundry and hot shower tanks, the intermediate and auxiliary building sumps, and the steam-generator blowdown tank. Some sources of liquid waste to the waste holdup tank from the auxiliary building are:

A. Equipment and floor drains for the operating level.

B. Equipment and floor drains for the intermediate level drain to the waste holdup tank.

C. Equipment and floor drains for the basement level drain to the auxiliary building sump to be pumped to the waste holdup tank.

D. Equipment and drains in residual heat removal sump drain to the auxiliary building sump and are pumped to the waste holdup tank.

E. Water used in resin replacement of primary plant demineralizers to the waste holdup tank.

The sources of liquid waste to the waste holdup tank from the intermediate building are:

AA. Intermediate building (restricted area side) floor and equipment drains to intermediate building sump to be pumped to waste holdup tank.

BB. Hot shower drains.

CC. Decontamination area drains to the laundry and hot shower tanks to be pumped to the waste holdup tank.

DD. Radioactive chem lab drains.

EE. Steam-generator blowdown tank to be pumped to the waste holdup tank, if necessary.

Containment floor drains are collected in containment sump A and pumped via two sump pumps to the waste holdup tank. On a containment isolation signal the containment sump pumps would trip and two series air-operated valves would shut isolating their combined discharge line. Containment sump A level transmitters (LT-2039 and LT-2044) provide input for level indication on the main control board.

11.2.2.4.2 Waste Holdup Tank Discharge

From the waste holdup tank, the waste water can be processed through the vendor supplied demineralization system to monitor tank A or B (Section 11.2.2.17) and ultimately released to the circulating water discharge canal or recycled to the reactor makeup water tank.

The waste holdup tank vent line is routed through the auxiliary building charcoal filters.

11.2.2.5 Auxiliary Building Sump Tank, Sump Tank Pumps, and Sump Pumps

The auxiliary building sump tank and sump tank pumps are shown in Drawing 33013-1270, Sheet 1. The auxiliary building sump pumps are shown in Drawing 33013-1272, Sheet 2.

The 375-gal capacity auxiliary building sump tank serves as a collecting point for equipment drain water discharged to the basement level drain header. It is located at the lowest point in the auxiliary building in the residual heat removal pit. The drain header receives equipment drains from the refueling water storage tank (RWST), residual heat exchangers, chemical and volume control system holdup tanks and recirculation pump, gas stripper feed pumps, boric acid evaporator, spent resin storage tanks, seal water filter, charging pump seal leakoff tank, charging pumps, spray additive tank, seal water heat exchanger, and nonregenerative heat exchanger. All equipment drains entering this tank contain loop seals to prevent gas from leaving the pressure vent system. Two horizontal 20-gpm level-controlled centrifugal sump tank pumps take suction from this tank and pump to the waste holdup tank in the basement level of the auxiliary building. All welded parts of the pumps are stainless steel. The tank is all-welded austenitic stainless steel.

The auxiliary building sump tank pumps may also be used to collect reactor cavity liner leaks during shutdown and discharge through cartridge demineralizers via a remote hose connection. By using the sump tank pump for this function, overloading of the liquid waste processing system is avoided.

The auxiliary building sump, Drawing 33013-1272, Sheet 2, located below the residual heat removal pit, receives equipment drains from the residual heat removal pumps, safety injection pumps, containment spray pumps, reactor coolant drain tank pumps, and reactor coolant drain tank system relief valve 1814 to the residual heat removal system. Basement level floor drains and residual heat removal pit floor drains also lead to the sump. Two 50-gpm auxiliary building sump pumps take suction from this sump and discharge to the waste holdup tank. (See Section 5.4.5.3.5.)

11.2.2.6 Waste Evaporator

The following section described the waste evaporator package, which was part of the original plant design. The waste evaporator processed liquid waste from the waste holdup tank. Use of the waste evaporator system was discontinued in 1990 and the system was physically removed from the plant in 1999. The remaining description is retained for historical purposes only.

Liquids requiring cleanup before release are processed in batches by the waste evaporator. The concentrated bottoms are discharged to the drumming facility where they are packaged for removal to a burial facility. The condensate is routed to one of two waste condensate tanks.

The waste evaporator unit is a skid-mounted self-contained unit designed to process waste fluids for eventual removal from the plant site by dilution of the distillate with the circulating water discharge to the lake or by drumming and eventual offsite disposal in solid form. The waste evaporator unit is designed to evaporate at a 2-gpm rate. It consists of a feed tank, concentrator, distillate tank, hot water converter, several pumps, heating coils, cooling coils, and necessary piping and instrumentation.

The length of an evaporator operating cycle is determined by activity of the concentrate or the boron concentration. The entire evaporator is austenitic stainless steel of welded construction, except for the heat transfer surfaces, which are admiralty metal.

The other evaporator system, which can be used as a backup system to process waste from the waste holdup tank or normal clean chemical and volume control system drains, is the boric acid evaporator. The concentrates from this evaporator can be transferred to the waste evaporator feed tanks for disposal or to the concentrates holding tank for reuse. The parameters used to control the batch operations are boric acid concentration and gross degassed activity. These concentrations are limited by procedure although activity may be further limited by burial ground dose rate limits.

11.2.2.7 Evaporator Feed Tank

The following section described the waste evaporator feed tank, which was part of the original plant design. The waste evaporator processed liquid waste from the waste holdup tank. Use of the waste evaporator system was discontinued in 1990 and the system was physically removed from the plant in 1999. The remaining description is retained for historical purposes only.

The evaporator feed tank was a 500-gal tank whose level is controlled around 400 gal using makeup from the waste holdup tank via a cycling waste evaporator feed pump and a 5-micron filter. The feed tank also receives the following:

A. Some concentrator bottoms via an eductor.

B. Steam air ejector discharge from the concentrator.

C. Chemical addition for pH and antifoaming control.

The pH is normally maintained between 5.0 and 6.0 to minimize carryover. Foaming is minimized by adding very small quantities of an antifoam solution. The tank has an electrical immersion heater to maintain the tank solution temperature above the saturation point at which crystallization would occur.

The waste evaporator feed tank vent line was routed through the auxiliary building charcoal filters.

11.2.2.8 Evaporator Feed Tank Pumps

The following section described the waste evaporator feed tank pumps, which were part of the original plant design. The waste evaporator processed liquid waste from the waste holdup tank. Use of the waste evaporator system was discontinued in 1990 and the system was physically removed from the plant in 1999. The remaining description is retained for historical purposes only.

The purpose of the feed tank pump was to pump the feed tank water to the evaporator concentrator section. A portion of the discharge passes through an eductor which serves to remove the concentrator bottoms back into the feed tank. When the feed tank activity and/or boric acid concentrations get too high, valving is realigned to pump the feed tank to the drumming station. A sample line is provided for checking the concentration. The waste evaporator recirculation line provides a path from the drumming station back to the feed tank for that portion of the feed pumped to but not released to the drumming station. The feed tank pump has a capacity of 20 gpm at a 100 ft head at design pressure and temperature of 100 psig and 180°F.

When the waste in the evaporator feed tank reaches a low level, the evaporator feed tank pump is tripped to prevent possible damage to the pump.

11.2.2.9 Concentrator

The following section described the waste evaporator concentrator, which was part of the original plant design. The waste evaporator processed liquid waste from the waste holdup tank. Use of the waste evaporator system was discontinued in 1990 and the system was physically removed from the plant in 1999. The remaining description is retained for historical purposes only.

The concentrator separated the impurities from the distillate, under vacuum, by means of a closed loop hot water heating section for boiling the feedwater and a component cooling water cooled condenser section for condensing the evaporated distillate. Vacuum is

maintained by an air ejector that uses 100 psig house heating steam. This also serves to prevent any buildup of noncondensible gases in the concentrator. Air ejector discharge is condensed in a component cooling water cooled heat exchanger. The closed loop hot water heating section uses 20 psig house heating steam via a hot water converter and a closed loop water system consisting of a circulating pump and expansion tank with associated piping.

The distillate steam vapor rises across screens, a demister, and a sieve to remove impurities from the vapor. The impurities fall to the bottom of the concentrator and are removed via the eductor to the feed tank. The pure distillate steam vapor rises to the condenser section where it is condensed and collected on a tray. The condensate is then vacuum-dragged through another eductor to the distillate tank.

11.2.2.10 Distillate Tank

The following section described the waste evaporator distillate tank, which was part of the original plant design. The waste evaporator processed liquid waste from the waste holdup tank. Use of the waste evaporator system was discontinued in 1990 and the system was physically removed from the plant in 1999. The remaining description is retained for historical purposes only.

The 50-gal distillate tank collected the condensate from the evaporator. A distillate pump recirculates the distillate through an eductor. A line connecting the condensate collection tray to the low-pressure area of the eductor removes condensate from the evaporator. As the distillate tank fills, the pump discharge will divert to the waste condensate tanks via a distillate cooler which uses component cooling water as its cooling medium. From here, the distillate is sent to the two waste condensate demineralizers to remove any fission products that may have been carried over in the distillate.

11.2.2.11 Waste Condensate Demineralizers

The waste condensate demineralizers (Drawing 33013-1276) are 30 ft^3 in size and contain nonregenerative mixed-bed resin. At their outlet is a 25-micron filter to retain any resin fines.

11.2.2.12 Waste Condensate Tanks

The following section described the waste condensate tanks, which were part of the original plant design. the waste evaporator processed liquid waste form the waste holdup tank. Use of the waste evaporator system was discontinued in 1990 and the system was physically removed from the plant in 1999. The remaining description is retained for historical purposes only.

Two 600-gal tanks collected evaporator condensate. The contents were sampled and analyzed for radioactivity and purity before discharge. The condensate was transferred by one of two waste condensate pumps to the waste holdup tank if the activity was high or to the condenser circulating water if the activity was sufficiently low. These tanks are constructed of all-welded stainless steel.

Protection against an inadvertent release of high activity from the waste condensate tank was provided by an interlock which prevented the discharge valve from opening if the measured activity was above specified limits. The waste condensate tanks were retired in place in 1999.

11.2.2.13 Vendor Supplied Demineralization System

A vendor supplied demineralization system is used for processing liquid radwaste. This system uses mixed media filtration and anion, cation, and mixed-bed resin to process water from the waste holdup tank. The system consists of five to six resin vessels, a booster pump, mechanical filtration, dewatering pump, and process control unit. After processing, waste water is collected in monitor tank A (Section 11.2.2.17). If the processed water is unacceptable for release because of radioactivity, it is recycled to the waste holdup tank. When chemistry and activity release parameters are met, the waste water is released from the monitor tank into the circulating water discharge canal. When the resin media is spent, it is sluiced to a shipping container for disposal.

11.2.2.14 High Conductivity Waste Tank

The high conductivity waste tank is the collection point for condensate polisher regenerant wastes and high conductivity waste effluent. It retains those effluents prior to release into the circulating water system. It is a 50,000-gal carbon steel tank.

11.2.2.15 Retention Tank

The retention tank is the collection point for the various building floor and equipment drains. It retains these effluents prior to discharging into the circulating water discharge. It is continuously monitored for pH and radioactivity (see Section 11.5.2.2.14).

11.2.2.16 Neutralizing Tank

The neutralizing tank was permanently removed from the primary water treatment system per PCR-2006-0017.

11.2.2.17 Monitor Tanks

The monitor tanks are part of the chemical and volume control system. The B monitor tank accepts effluents from the boric acid evaporator and the A monitor tank accepts effluents from the vendor supplied demineralization system (Section 11.2.2.13). Discharge from the monitor tanks may be pumped to the reactor makeup water storage tank, aligned to the suction of the reactor makeup water pumps (when the reactor makeup water tank is out of service), recycled through the evaporator condensate demineralizers, returned to the waste holdup tank for reprocessing, or discharged to the environment with the condenser circulating water within the allowable activity concentration.

11.2.2.18 Radwaste Control System

Control of radwaste operations was aided by monitoring provided by a process computer system tied to various radwaste panels. In 1995, this computer system was removed after an evaluation determined that it provided no value to Ginna Station nor did plant procedures require the use of the computer for accident monitoring or recovery.

11.2.2.19 Dry Cleaning Unit

Contaminated clothing was cleaned using a dry cleaning unit, prior to its dismantling in 1994. All laundry is currently sent offsite for cleaning.

11.2.2.20 Piping and Valves

Liquid waste piping is stainless steel. Piping connections are welded except (1) where flanged connections are necessary to facilitate equipment maintenance or (2) where Cam-Lok quick connect fittings are necessary for operation of the vendor supplied demineralization system (Section 11.2.2.13). No threaded fittings are used in waste piping.

The laundry and hot shower tank valves and the compressor seal-water supply and cooling valves are carbon steel. All other valves are stainless steel. All valves have stem leakage control. Globe valves are installed with flow over the seats when such an arrangement reduces the possibility of leakage. All relief valves handling radioactive gases are of the closed bonnet design and contain bellows seals. Relief valves in the systems handling radioactive fluids are of the closed bonnet design and are constructed of stainless steel.

Stop valves are provided to isolate each piece of equipment for maintenance, to direct the flow of waste through the system, and to isolate storage tanks for radioactive decay.

Relief valves are provided for tanks containing radioactive wastes if the tanks might be overpressurized by improper operation or component malfunction. Tanks containing wastes which are normally of low radioactivity are vented locally.

11.2.3 LIQUID EFFLUENT RELEASE CONCENTRATIONS AND DOSES

11.2.3.1 Liquid Release Control

Liquid batch releases are controlled individually and each batch release is authorized based upon sample analysis and the existing dilution flow in the discharge canal. Plant procedures establish the methods for sampling and analysis of each batch prior to release. A release rate limit is calculated for each batch based upon analysis, dilution flow, and all procedural conditions being met before it is authorized for release. The waste effluent stream entering the discharge canal is continuously monitored and the release will be automatically terminated if the preselected monitor setpoint is exceeded.

If gross beta analysis is performed for each batch release in lieu of gamma isotopic analysis, then a weekly composite for principal gamma emitters and Iodine-131 is performed. Additional monthly and quarterly composite analyses are performed as specified, and the methodology and equations used to calculate activity are included in the Ginna Station Offsite Dose Calculation Manual (ODCM).

11.2.3.2 Dose Calculations

The dose contribution received by the maximally exposed individual from the ingestion of Lake Ontario fish and drinking water is determined using the methodology described in the Offsite Dose Calculation Manual. The calculations assume a near field dilution factor of 1.0 in evaluating the fish pathway dose and a dilution factor of 200 between the plant discharge

and the Ontario Water District drinking water intake located 1.1 miles away. The dilution factor of 20 was derived from drift and dispersion studies documented in Appendix 2B. The Monroe County Water Authority Webster plant is located 4.1 miles upstream and due to this location, this facility is not considered an uptake pathway. The Webster plant will begin water production in 2014. Dose contributions from shoreline recreation, boating, and swimming have been shown to be negligible in the Appendix I dose analysis (*Reference 1*) and do not need to be routinely evaluated. Also, there is no known human consumption of shellfish from Lake Ontario.

The dose contribution to an individual is determined to ensure that it complies with the Off-site Dose Calculation Manual (ODCM) requirements. The dose or dose commitment to an individual from radioactive materials in liquid effluents released to unrestricted areas shall be limited: (1) during any calendar quarter to less than or equal to 1.5 mrem to the total body and less than or equal to 5 mrem to any organ, and (2) during any calendar year to less than or equal to 3 mrem to the total body and to less than or equal to 10 mrem to any organ. Offsite receptor doses will be determined for the limiting age group and organ, unless census data show that actual offsite individuals are of a less limiting age group.

Calculations were performed in 1976 (*References 1* and *2*) to demonstrate conformity with numerical guides on design objectives presented in Appendix I to 10 CFR 50 for liquid effluents. Tables 11.2-4 and 11.2-5 present the assumptions and the annual liquid effluent releases, respectively. The maximum individual doses are shown in Table 11.2-6.

In 2005, scaling techniques, based on NUREG-0017, Revision 1 methodology, were utilized to assess the impact of core power uprate on radioactive liquid effluents at Ginna.

As described in *Reference 3*, the conservatively performed power uprate analysis utilized the plant core power operating history during the years 1999 to 2003, the reported liquid effluent and dose data during that period, NUREG-0017 equations and assumptions. Also utilized was conservative methodology, to estimate the impact of operation at the analyzed uprate core power level of 1811 MWt, over that of operation at the previously licensed power level, on radioactive liquid effluents and consequent normal operation off-site doses.

The licensed reactor core power level during the 1999 to 2003 time frame was 1520 MWt. For the uprate condition, the system parameters utilized in the power uprate analysis reflected the flow rates and coolant masses at an analyzed NSSS power level of 1817 MWt and a core power level of 1811 MWt. For the pre-uprate condition, the evaluation utilized offsite doses based on an average 5 year set of organ and whole body doses calculated using data presented in the Ginna Annual Radioactive Effluent Release Reports for the years 1999 through 2003, taking into consideration the associated average annual core power level, extrapolated to 100 percent availability at the licensed power level.

Using the methodology and equations found in NUREG-0017, Revision 1, and based on a comparison of the change in power level and in plant coolant system parameters (e.g., reactor coolant mass, steam generator liquid mass, steam flow rate, reactor coolant letdown flow rate, flow rate to the cation demineralizer, letdown flow rate for boron control, steam generator blowdown flow rate, steam generator moisture carryover, etc.) for both pre-uprate and uprate

conditions, the maximum potential percentage increase in coolant activity levels due to the uprate, for each chemical group identified in NUREG-0017, was estimated.

To estimate an upper bound impact on off-site doses, the highest factor found for any chemical group pertinent to the release pathway was applied to the average doses previously determined as representative of operation at pre-uprate conditions. This approach was utilized to estimate the maximum potential increase in effluent doses due to the uprate, and demonstrate that the estimated off-site doses following the uprate, although increased, will continue to remain below the regulatory limits set by 10CFR50, Appendix I. Table 11.2-7 shows that based on operating history, the maximum estimated dose due to liquid radwaste effluents, following power uprate, will continue to remain significantly below the annual design objectives for liquid radwaste effluents set by 10CFR50, Appendix I.

It is noted that actual liquid effluent isotopic release curie and dose information are provided in the Annual Radioactive Effluent Release Reports required by the Technical Specifications.

Actual liquid effluent isotopic release curie figures are provided in the Radioactive Effluent Release Reports required by the Technical Specifications.

11.2.3.3 Accidental Spill of Liquid Radwastes

An accidental spill of liquid radwastes into Lake Ontario cannot result from any single failure of equipment, either an active or passive failure. For example, all of the tanks and pipes outside of the containment that can potentially contain significant radioactivity are located within the auxiliary building. A break in any tank or pipe would drain to the auxiliary building sump. The sump is well below lake level so that gravity spill to the lake is not possible. Therefore, the worst "spill" accident is to pump out a monitor tank of maximum possible activity. Because of the low radioactivity levels in the circulating water discharge, the concentration of liquid radioactive effluents at this point is not measured directly. The concentrations in the circulating water discharge are calculated from the measured concentration in the waste condensate tank, the flow rate of the waste condensate pumps, and the flow in the circulating water system.

REFERENCES FOR SECTION 11.2

1. Letter from L. D. White, Jr., RG&E, to R. A. Purple, NRC, Subject: Calculations to Demonstrate Compliance with the Design Objectives of 10 CFR 50, Appendix I, dated June 3, 1976.

2. Letter from L. D. White, Jr., RG&E, to A. Schwencer, NRC, Subject: Response to NRC Additional Information Requests, Appendix I, dated October 25, 1976.

3. Letter, M.G. Korsnick, Ginna LLC, to Document Control Desk, NRC, Subject: License Amendment Request Regarding Extended Power Uprate, dated July 7, 2005.

Table 11.2-1
WASTE DISPOSAL COMPONENTS CODE REQUIREMENTS

Component	Code
Chemical drain tank	No code
Reactor coolant drain tank	ASME III[a] Class C
Sump tank	No code
Spent resin storage tanks	ASME III[a] Class C
Waste holdup tank	No code
Waste condensate tank	No code
Waste retention tank	No code
High conductivity waste tank	No code
Laundry and hot shower tank	No code
Waste filter	ASME III[a] Class C
Leakoff collection tank	No code
Piping and valves	ASA-B31.1 Section 1[b]

a. ASME III is the American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section IV, Nuclear Vessel.

b. ASA-B31.1 is the code for pressure piping American Standards Association and special nuclear cases where applicable.

Table 11.2-2a
LIQUID WASTE SYSTEM COMPONENT SUMMARY DATA - TANKS

Tanks	Qty	Type	Volume	Design Pressure	Design Temperature	Material
Reactor coolant drain	1	H[a]	350 gal	25 psig	267°F	S[b]
Laundry and hot shower	2	V[c]	600 gal	Atmosphere	180°F	S[b]
Chemical drain	1	V[c]	375 gal	Atmosphere	180°F	S[b]
Sump tank	1	H[a]	375 gal	Atmosphere	180°F	S[b]
Waste holdup	1	H[a]	21,000 gal	Atmosphere	150°F	S[b]
Spent resin storage	2	V[c]	1,000 gal	100 psig	150°F	S[b]
Waste condensate (retired in place in 1999)	2	V[c]	600 gal	Atmosphere	180°F	S[b]
Gas decay	4	V[c]	470 ft^3	150 psig	150°F	C[d]
Leakoff collection	1	H[a]	50 gal	5 ft H_2O vacuum	120°F	S[b]
High conductivity waste	1	V[c]	50,000 gal	Atmosphere	Ambient	C[d]
Waste retention	1	H[a]	25,000 gal	Atmosphere	Ambient	C[d]

a. Horizontal
b. Stainless steel - s/s
c. Vertical
d. Carbon steel - c/s

Table 11.2-2b
LIQUID WASTE SYSTEM COMPONENT SUMMARY DATA - PUMPS

Pumps	Qty	Type	Flow (gpm)	Head (ft)	Design Pressure (psig)	Design Temperature (°F)	Material[a]
Reactor coolant drain (A)	1	Horizontal centrifugal canned	50	175	100	267	Stainless steel
Reactor coolant drain (B)	1	Horizontal centrifugal canned	150	175	100	267	Stainless steel
Chemical drain	1	Horizontal centrifu-gal[b]	20	100	100	180	Stainless steel
Laundry	1	Horizontal centrifu-gal[b]	20	100	100	180	Stainless steel
Sump tank	2	Horizontal centrifu-gal[b]	20	100	100	180	Stainless steel
Sump pit	2	Vertical centrifugal	50	55	---	---	Stainless steel
Waste condensate	2	Horizontal centrifu-gal[cb]	20	100	100	180	Stainless steel
Waste retention tank discharge	1	Vertical centrifugal	500	20	---	---	Cast iron
High conductivity waste tank discharge	1	Horizontal centrifugal	300	100	---	---	Stainless steel
Waste holdup tank	1	Horizontal centrifugal	20	100	100	180	Stainless steel

a. Wetted surfaces only.

b. Mechanical seal provided.

c. 1 Waste condensate pump abandoned in place. 1 removed from plant.

Table 11.2-2c
LIQUID WASTE SYSTEM COMPONENT SUMMARY DATA - COMPRESSORS

Compressors	Quantity	Type	Capacity	Model
Waste Gas Compressor 'A'	1	Horizontal centrifugal [a]	48 cfm	Nash AL-623C
Waste Gas Compressor 'B'	1	Horizontal centrifugal [a]	48 cfm	Nash AL-623C

a. Mechanical seal provided.

Table 11.2-3
LIQUID WASTE DISPOSAL SYSTEM PERFORMANCE DATA

(System physically removed in 1999. This information is for historical purposes only.)

Plant design life	40 years
Normal process capacity, liquids (evaporator)	2 gpm
Load factor[a]	26%
Annual processed liquid discharge[b]	
Volume	3.07×10^5 gal
Activity	5.9 mCi
Annual total liquid discharge[b]	
Volume	2.29×10^7 gal
Activity	115 mCi

a. Batch operation.
b. Actual 1983 operation.

Table 11.2-4a
LIQUID EFFLUENTS, 10 CFR 50, APPENDIX I CALCULATIONS, ASSUMPTIONS (HISTORICAL)

Input information to the computer program GALE.

Thermal power level, Mwt	1520
Plant capacity factor	0.80
Mass of primary coolant, lb	2.82×10^5
Primary system letdown rate, gpm	40
Letdown cation demineralizer flow, gpm	3.60
Number of steam generators	2
Total steam flow, lb/hr	6.58×10^6
Mass of steam in each steam generator, lb	4.622×10^3
Mass of liquid in each steam generator, lb	8.5×10^4
Total mass of secondary coolant, lb	8.15×10^5
Blowdown rate, lb/hr	1.1×10^4
Radwaste dilution flow, gpm	4.0×10^5

Note: The information presented in the above table represents the inputs utilized in the original 10CFR50 Appendix I analysis performed for Ginna and is retained herein for historical purposes only.

Table 11.2-4b

LIQUID EFFLUENTS, 10 CFR 50, APPENDIX I CALCULATIONS, ASSUMPTIONS[a] (HISTORICAL)

LIQUID WASTE INPUTS						Decontamination Factors		
Stream	Flow Rate (gal/ day)	Fraction of PCA [b]	Fraction Discharged	Collection Time (days)	Decay Time (days)	Iodine	Cesium	Others
Shim bleed	7.0×10^2	1.000	0.100	52.000	1.400	1.0×10^4	2.0×10^4	1.0×10^5
Equipment drains	2.5×10^2	1.000	0.100	52.000	1.400	1.0×10^4	2.0×10^4	1.0×10^5
Dirty waste	6.2×10^2	0.060	1.000	11.500	6.000	1.0×10^4	1.0×10^5	1.0×10^5
Clean waste	1.3×10^2	0.160	1.000	11.500	6.000	1.0×10^4	1.0×10^5	1.0×10^5
Blowdown	3.16×10^4	---	0.0	0.0	0.010	1.0×10^4	1.0×10^3	1.0×10^3
Untreated blowdown	0.0	---	1.000	0.0	0.0	1.0	1.0	1.0

Note: The information presented in the above table represents the assumptions utilized in the original 10CFR50 Appendix I analysis performed for Ginna and is retained herein for historical purposes only.

a. Input information to the computer program GALE.

b. Primary coolant activity.

Table 11.2-5
LIQUID EFFLUENTS, 10 CFR 50, APPENDIX I CALCULATIONS, RESULTS[a.] (HISTORICAL)

		Coolant Concentrations		Annual Releases to Discharge Canal							
	Nuclide Half-Life (days)	Primary (μ Ci/m1)	Secondary (μ Ci/ml)	Boron Recycle System (Ci)	Misc. Wastes (Ci)	Secondary (Ci)	Turbine Bldg (Ci)	Total Liquid Waste System (Ci)	Adjusted Total (Ci/yr)	Detergent Wastes (Ci/yr)	Total (Ci/yr)
Corrosion and Activation Products											
^{51}Cr	2.78×10^{1}	1.57×10^{-3}	5.61×10^{-7}	0.0	0.0	0.0	0.00001	0.00001	0.00006	0.0	0.00006
^{54}Mn	3.03×10^{2}	2.56×10^{-4}	1.36×10^{-7}	0.0	0.0	0.0	0.0	0.0	0.00002	0.00003	0.00005
^{55}Fe	9.50×10^{2}	1.32×10^{-3}	4.74×10^{-7}	0.0	0.0	0.0	0.0	0.00001	0.00006	0.0	0.00006
^{59}Fe	4.50×10^{1}	8.27×10^{-4}	3.46×10^{-7}	0.0	0.0	0.0	0.0	0.0	0.00004	0.0	0.00004
^{58}Co	7.13×10^{1}	1.32×10^{-2}	4.81×10^{-6}	0.0	0.00001	0.0	0.00005	0.00006	0.00056	0.00013	0.00070
^{60}Co	1.92×10^{3}	1.65×10^{-3}	6.10×10^{-7}	0.0	0.0	0.0	0.00001	0.00001	0.00007	0.00029	0.00036
^{95}Zr	6.50×10^{1}	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.00005	0.00005
^{95}Nb	3.50×10^{1}	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.00007	0.00007
^{239}Np	2.35×10^{0}	1.00×10^{-3}	2.78×10^{-7}	0.0	0.0	0.0	0.0	0.0	0.00002	0.0	0.00002
Fission Products											
^{83}Br	1.00×10^{-1}	4.14×10^{-3}	2.99×10^{-7}	0.0	0.0	0.0	0.00001	0.00001	0.00005	0.0	0.00005
^{89}Sr	5.20×10^{1}	2.89×10^{-4}	1.38×10^{-7}	0.0	0.0	0.0	0.0	0.0	0.00002	0.0	0.00001
^{99}Mo	2.79×10^{0}	7.00×10^{-2}	2.68×10^{-5}	0.0	0.0	0.0	0.00025	0.00026	0.00234	0.0	0.00230
^{99m}Tc	2.50×10^{-1}	4.11×10^{-2}	5.11×10^{-5}	0.0	0.0	0.0	0.00037	0.00037	0.00340	0.0	0.00340

		Coolant Concentrations		Annual Releases to Discharge Canal							
	Nuclide Half-Life (days)	Primary (μ Ci/mL)	Secondary (μ Ci/ml)	Boron Recycle System (Ci)	Misc. Wastes (Ci)	Secondary (Ci)	Turbine Bldg (Ci)	Total Liquid Waste System (Ci)	Adjusted Total (Ci/yr)	Detergent Wastes (Ci/yr)	Total (Ci/yr)
^{106}Ru	3.67×10^{2}	8.27×10^{-6}	3.39×10^{-9}	0.0	0.0	0.0	0.0	0.0	0.0	0.00008	0.00008
^{110m}Ag	2.53×10^{2}	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.00001	0.00001
^{127}Te	3.92×10^{-1}	7.23×10^{-4}	3.05×10^{-7}	0.0	0.0	0.0	0.0	0.00000	0.00002	0.0	0.00002
^{129m}Te	3.40×10^{1}	1.16×10^{-3}	4.18×10^{-7}	0.0	0.0	0.0	0.0	0.00001	0.00005	0.0	0.00005
^{129}Te	4.79×10^{-2}	1.39×10^{-3}	1.29×10^{-6}	0.0	0.0	0.0	0.0	0.0	0.00003	0.0	0.00003
^{130}I	5.17×10^{-1}	1.78×10^{-3}	3.53×10^{-7}	0.0	0.0	0.0	0.00003	0.00003	0.00023	0.0	0.00023
^{131m}Te	1.25×10^{0}	2.10×10^{-3}	5.45×10^{-7}	0.0	0.0	0.0	0.0	0.0	0.00004	0.0	0.00004
^{131}I	3.05×10^{0}	2.24×10^{-1}	8.33×10^{-5}	0.00019	0.00068	0.0	0.00812	0.00899	0.08223	0.0	0.08200
^{132}Te	3.25×10^{0}	2.25×10^{-2}	6.92×10^{-6}	0.0	0.0	0.0	0.00007	0.00007	0.00062	0.0	0.00062
^{132}I	9.58×10^{-2}	8.64×10^{-2}	2.21×10^{-5}	0.0	0.0	0.0	0.00042	0.00042	0.00383	0.0	0.00380
^{133}I	8.76×10^{-1}	3.20×10^{-1}	7.85×10^{-5}	0.00001	0.0	0.0	0.00641	0.00643	0.05880	0.0	0.05900
^{134}Cs	7.49×10^{2}	2.08×10^{-2}	8.20×10^{-6}	0.00008	0.00002	0.0	0.00008	0.00018	0.00167	0.00043	0.00210
^{135}I	2.79×10^{-1}	1.62×10^{-1}	2.32×10^{-5}	0.0	0.0	0.0	0.00124	0.00124	0.01138	0.0	0.01100
^{136}Cs	1.30×10^{1}	1.09×10^{-2}	3.68×10^{-6}	0.00001	0.0	0.0	0.00004	0.00005	0.00050	0.0	0.00050
^{137}Cs	1.10×10^{4}	1.50×10^{-2}	5.46×10^{-6}	0.00006	0.00001	0.0	0.00005	0.00013	0.00118	0.00079	0.00200
^{137m}Ba	1.77×10^{-3}	1.39×10^{-2}	1.84×10^{-5}	0.00006	0.00001	0.0	0.00005	0.00012	0.00110	0.0	0.00110
^{144}Ce	2.34×10^{2}	2.73×10^{-5}	1.36×10^{-8}	0.0	0.0	0.0	0.0	0.0	0.0	0.00017	0.00017
All others		2.20×10^{-1}	4.85×10^{-6}	0.0	0.0	0.0	0.00001	0.00001	0.00007	0.0	0.00007

		Coolant Concentrations		Annual Releases to Discharge Canal							
	Nuclide Half-Life (days)	Primary (μ Ci/mL)	Secondary (μ Ci/ml)	Boron Recycle System (Ci)	Misc. Wastes (Ci)	Secondary (Ci)	Turbine Bldg (Ci)	Total Liquid Waste System (Ci)	Adjusted Total (Ci/yr)	Detergent Wastes (Ci/yr)	Total (Ci/yr)
Total (except tritium)		1.24	3.43×10^{-4}	0.00044	0.00075	0.0	0.01722	0.01841	0.16341	0.00206	0.17000
Tritium release		210 Ci/yr									

a. Computer program GALE

Note: The information presented in the above table represents the results of the original 10CFR50 Appendix I analysis performed for Ginna and is retained herein for historical purposes only.

Table 11.2-6
MAXIMUM INDIVIDUAL DOSES FROM LIQUID EFFLUENTS (MREM/YEAR) (HISTORICAL)

Pathway	Age Group	Total Body	GI Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
Drinking water	Adult	0.0013	0.0013	a	0.0013	0.0013	0.00900	0.0013	a
	Teen	a	a	a	a	a	0.0072	a	a
	Child	0.0014	0.0014	a	0.0015	a	0.017	0.0014	a
	Infant	0.0020	0.0020	a	0.0023	a	0.040	0.0020	a
Fish ingestion	Adult	0.022	0.0030	0.016	0.029	0.011	0.063	0.0038	a
	Teen	0.013	0.0020	0.016	0.029	0.0082	0.057	0.0040	a
	Child	0.0051	0.0010	0.019	0.025	0.0035	0.059	0.0031	a
	Infant	0.	0.	0.	0.	0.	0.	0.	0.
Shoreline recreation	Adult	a	a	a	a	a	a	a	a
	Teen	a	a	a	a	a	a	a	a
	Child	a	a	a	a	a	a	a	a
	Infant	0.	0.	0.	0.	0.	0.	0.	0.
Boating	Adult	a	a	a	a	a	a	a	a
	Teen	a	a	a	a	a	a	a	a
	Child	a	a	a	a	a	a	a	a
	Infant	0.	0.	0.	0.	0.	0.	0.	0.

Pathway	Age Group	Total Body	GI Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
Swimming	Adult	a	a	a	a	a	a	a	a
	Teen	a	a	a	a	a	a	a	a
	Child	a	a	a	a	a	a	a	a
	Infant	0.	0.	0.	0.	0.	0.	0.	0.
Totals	Adult	0.023	0.0040	0.016	0.030	0.011	0.073	0.0051	a
	Teen	0.014	0.0030	0.016	0.029	0.0093	0.064	0.0048	a
	Child	0.0065	0.0024	0.019	0.026	0.0045	0.076	0.0045	a
	Infant	0.0021	0.0021	a	0.0023	a	0.040	0.0021	a

Note: The information presented in the above table represents the results of the original 10CFR50 Appendix I analysis performed for Ginna and is retained herein for historical purposes only.

a. Indicates dose less than 0.001 mrem/year.

Table 11.2-7
ESTIMATED ANNUAL DOSES TO THE PUBLIC DUE TO NORMAL OPERATION LIQUID RADWASTE EFFLUENTS - CORE POWER LEVEL 1811 MWt

Type or Dose	Appendix I Design Objectives	Pre-Uprate[a]	Uprate[b]	Percentage of Appendix I Design Objectives
Dose to total body from all pathways	3 mrem/yr	3.16E-03 mrem/yr	3.77E-3 mrem/yr	0.126%
Dose to any organ from all pathways	10 mrem/yr	3.37E-3 mrem/yr	4.01E-3 mrem/yr	0.040%

a. Base Case - doses based on an average 5 year set of organ and whole body doses calculated using data presented in the Ginna Annual Radioactive Effluent Release Reports for the years 1999 through 2003, taking into consideration the associated average annual core power level, extrapolated to 100 persent availability at the licensed power level of 1520 MWt.

b. Estimated upper bound off-site doses developed as discussed in Section 11.2.3.2 by using scaling techniques that are applied to the Base Case.

11.3 GASEOUS WASTE MANAGEMENT SYSTEM

11.3.1 DESIGN BASES

The gaseous waste management system is designed to collect waste gases from various tanks and sampling systems throughout the plant. A number of tanks in the plant are maintained with a low-pressure hydrogen or nitrogen gas blanket to prevent air introduction into their systems. As these tanks fill with liquid, the displaced gas volume is collected by the gaseous waste management system via a vent header. Most of the gas received by the system during normal operation is due to the cover gases displaced from the chemical and volume control system holdup tanks as they fill with liquid. These gases are collected and stored in gas decay tanks until reuse or discharge to the environment. Since the chemical and volume control system holdup tanks cover gases must be replaced when they are emptied during processing, provisions are made to return the gas from the gas decay tanks to the chemical and volume control system holdup tanks via a reuse header.

11.3.2 SYSTEM DESCRIPTION

11.3.2.1 Operation

The flow diagrams for the gaseous waste management system are shown in Drawings 33013-1274 and 33013-1273.

During MODES 1 and 2, the gaseous waste management system supplies nitrogen and hydrogen from standard cylinders to primary plant components. Two headers are provided, one for operation and one for backup. The nitrogen supply pressure regulator in the operating header is set for 200 psig discharge. When the operating header is exhausted, its discharge pressure will fall below 150 psig and a warning light will alert the operator that automatic alignment to the backup source has occurred. The operator can also manually align nitrogen to its backup source. Hydrogen cylinders have to be manually aligned to the backup source, upon receiving a low pressure alarm.

Most of the gas received by the gaseous waste management system during MODES 1 and 2 is cover gas displaced from the chemical and volume control system holdup tanks. Since this gas must be replaced when the tanks are emptied during processing, facilities are provided to return gas from the decay tanks to the holdup tanks. A backup supply from the nitrogen header is provided for makeup if return flow from the gas decay tanks is not available. Since the hydrogen concentration may exceed the combustible limit during this type of operation, components discharging to the vent header system are restricted to those containing no air or aerated liquids and the vent header itself is designed to operate at a slight positive pressure (0.5 psig minimum to 2.0 psig maximum) to prevent inleakage. Outleakage from the system is minimized by using Saunders patent diaphragm valves, bellows seals, self-contained pressure regulators, and soft-seated packless valves throughout the radioactive portions of the system.

11.3.2.2 Components

The gaseous waste management system consists of a waste disposal panel, a gas analyzer, two waste gas compressors with moisture separators, four gas decay tanks, the associated piping

for routing the waste gases, and various monitoring points as well as provisions for isolating an environmental discharge on high activity level.

The vent header receives gases from the following sources:

- Gas stripper operation.
- Volume control tank purges.
- Sampling system discharges.
- Chemical and volume control system holdup tank cover gas.
- Spent resin storage tank venting.
- Gas analyzer discharge.
- Gas decay tank rupture disk discharge.
- Pressurizer relief tank venting.
- Reactor coolant drain tank venting.
- Charging pump leakoff collection tank venting.

11.3.2.2.1 Waste Gas Compressors

Once waste gases enter the vent header, they flow to the suction of two centrifugal displacement type waste gas compressors. The compressors are initially filled manually with component cooling water for sealing purposes, and additional makeup of sealing water is automatic. A small amount of the seal-water is continuously supplied to the pumps to remove compression heat, and the excess water from the pumps is discharged with the gases. This small amount of water is removed from the gases in a baffle separator before the gases are discharged to one of the gas decay tanks. The seal-water heat exchangers for the compressors are cooled by component cooling water.

One of the two compressors is in continuous operation with the second unit instrumented to act as backup for peak load conditions or failure of the first unit. The standby compressor will start at a pressure of 1.8 psig increasing in the vent header and can be secured at 1.7 psig decreasing. The compressors are powered from motor control center 1E on the auxiliary building main floor.

From the compressors, gas flows to one of four gas decay tanks. The control arrangement on the gas decay tank inlet header allows the operator to place one tank in service and to select one tank for backup if the tank in operation becomes fully pressurized. When the tank in service becomes pressurized to 110 psig, a pressure transmitter automatically closes the inlet valve to that tank, opens the inlet valve to the backup tank, and sounds an alarm to alert the operator of this event so that a new backup tank may be selected. Pressure indicators are supplied to aid the operator in selecting the backup tank.

11.3.2.2.2 Gas Decay Tanks

The gas decay tanks are 470 ft^3 each, having a design pressure of 150 psig and normally operating between 0 to 110 psig. Their normal operating temperature is 50°F to 140°F. They

are protected from overpressure by 150-psig relief valves and by rupture disks and are equipped with high-pressure alarms. The gas decay tanks are designed to ASME III, Section IV, Class C, code requirements. They can be lined up for draining, gas analyzer sampling, or for being pressurized with nitrogen. In addition, gas held in the decay tanks can either be returned to the chemical and volume control system holdup tanks via the reuse header, or discharged to the atmosphere if it has decayed sufficiently for release.

Before a tank can be emptied to the environment, it is sampled and analyzed to determine and record the activity to be released, and only then discharged to the plant vent at a controlled rate through a radiation monitor. Samples are taken manually from the gas analyzers. During release (through charcoal filters), a trip valve in the discharge line is closed automatically by a high activity level indication in the plant vent.

During operation, gas samples are drawn periodically from tanks discharging to the waste gas vent header as well as from the particular gas decay tank being filled at the time, and automatically analyzed to determine their hydrogen and oxygen content.

11.3.2.2.3 Waste Disposal Panel

The waste disposal panel contains pressure gauges for the tanks using cover gas and also for the gas decay tanks, and the vent header. There is also a local plant stack radiation monitor (R-14) to be used by the operator during releases. The monitor gauge is located beside the control for the gas release valve. All gas system manual operations and releases are controlled locally at the waste disposal panel by the operator. The ac power is supplied to the waste disposal panel for switch and pump indication lights, and dc power is supplied for the annunciator section of the panel. The alarm conditions that are associated with the gaseous waste management system are

- Moisture separator No. 1 and No. 2 level, high-low.
- Vent header pressure, high.
- Gas analyzer control panel alarm or trouble.
- Plant stack monitor radiation, high.
- Gas decay tank No. 1, 2, 3, and 4, pressure, high.
- Gas decay tank new standby selection, when required.

There are also high-pressure alarms on the tanks that vent to the vent header. An alarm on the waste disposal panel will light an annunciator on the main control board.

11.3.2.2.4 Gas Analyzer

A gas analyzer is provided to monitor the concentrations of oxygen and hydrogen in the cover gas of the waste disposal system and chemical and volume control system tanks. The gas analyzer is manually aligned to take samples from vessels of the waste disposal system, analyzes the samples for oxygen and hydrogen, records the results of the analysis, and provides alarms when a hazardous operating condition exists. Upon indication of a high oxygen level, provisions are made to purge the systems to the gaseous waste system with an inert gas. The gas analyzer is normally aligned to continually sample the in-service Gas

Decay Tank. The analyzer system is capable of analyzing oxygen concentrations of 0-10% accurately to ±2% full scale linearity and ±1% full scale repeatability and hydrogen concentrations of 0.5-100% accurately to ±0.3% absolute (excluding allowable drift) for 0.5-10% hydrogen and ±1.0% absolute (excluding allowable drift) for 10-100% hydrogen.

11.3.2.2.5 Nitrogen Manifold

Nitrogen, used to purge the vapor space of various components to reduce the hydrogen concentration or to replace fluid that has been removed, is supplied from a dual manifold. The dual manifold arrangement ensures a continuous supply of gas.

11.3.2.2.6 Hydrogen Manifold

Hydrogen is supplied to the volume control tank to maintain the hydrogen partial pressure as hydrogen dissolves in the reactor coolant. The hydrogen is supplied from a dual manifold. The dual manifold arrangement ensures a continuous supply of gas.

11.3.2.2.7 Valves

All valves exposed to gases are carbon steel. All valves have stem leakage control; all relief valves handling radioactive gases are of the closed bonnet design and contain bellows seals.

11.3.3 GASEOUS RADIOACTIVE RELEASES

Gaseous effluent monitor setpoints as described in Section 11.5.2 are established at concentrations that permit some margin for corrective action to be taken before exceeding offsite dose rates corresponding to 10 CFR 20 limitations. Plant procedures establish the methods for sampling and analysis for continuous ventilation releases and for containment purge releases, as well as the methods for sampling and analysis prior to gas decay tank releases. The dose rates are determined using methodology included in the Offsite Dose Calculation Manual.

Calculations were performed in 1976 (*References 1* and *2*) to demonstrate conformity with numerical guides on design objectives presented in Appendix I to 10 CFR 50 for gaseous effluents. Table 11.3-1 and Table 11.3-2 show the assumptions and the annual gaseous releases calculated. Table 11.3-3 shows the maximum individual doses resulting from gaseous effluents. Atmospheric dispersion factors are discussed in Section 2.3.4.

In 2005, scaling techniques, based on NUREG-0017, Revision 1 methodology, were utilized to assess the impact of core power uprate on radioactive gaseous effluents at Ginna.

As described in *Reference 3*, the conservatively performed power uprate analysis utilized the plant core power operating history during the years 1999 to 2003, the reported gaseous effluent and dose data during that period, NUREG-0017 equations and assumptions. Also utilized was conservative methodology, to estimate the impact of operation at the analyzed uprate core power level of 1811 MWt, over that of operation at the previously licensed power level, on radioactive gaseous effluents and consequent normal operation off-site doses.

The licensed reactor core power level during the 1999 to 2003 time frame was 1520 MWt. For the uprate condition, the system parameters utilized in the power uprate analysis reflected

the flow rates and coolant masses at an analyzed NSSS power level of 1817 MWt and a core power level of 1811 MWt. For the pre-uprate condition, the evaluation utilized offsite doses based on an average 5 year set of organ and whole body doses calculated using data presented in the Ginna Annual Radioactive Effluent Release Reports for the years 1999 through 2003, taking into consideration the associated average annual core power level, extrapolated to 100 percent availability at the licensed power level.

Using the methodology and equations found in NUREG-0017, Revision 1, and based on a comparison of the change in power level and in plant coolant system parameters (e.g., reactor coolant mass, steam generator liquid mass, steam flow rate, reactor coolant letdown flow rate, flow rate to the cation demineralizer, letdown flow rate for boron control, steam generator blowdown flow rate, steam generator moisture carryover, etc.) for both pre-uprate and uprate conditions, the maximum potential percentage increase in coolant activity levels due to the uprate, for each chemical group identified in NUREG-0017, was estimated.

To estimate an upper bound impact on off-site doses, the highest factor found for any chemical group pertinent to the release pathway was applied to the average doses previously determined as representative of operation at pre-uprate conditions. This approach was utilized to estimate the maximum potential increase in effluent doses due to the uprate, and demonstrate that the estimated off-site doses following the uprate, although increased, will continue to remain below the regulatory limits set by 10CFR50, Appendix I. Table 11.3-4 shows that based on operating history, the maximum estimated dose due to gaseous radwaste effluents, following power uprate, will continue to remain significantly below the annual design objectives for gaseous radwaste effluents set by 10CFR50, Appendix I.

It is noted that actual gaseous effluent isotopic release curie and dose information are provided in the Annual Radioactive Effluent Release Reports required by the Technical Specifications.

The evaluation of an accidental decay tank rupture is presented in Section 15.7.

REFERENCES FOR SECTION 11.3

1. Letter from L. D. White, Jr., RG&E, to R. A. Purple, NRC, Subject: Calculations to Demonstrate Compliance with the Design Objectives of 10 CFR 50, Appendix I, dated June 1976.

2. Letter from L. D. White, Jr., RG&E, to A. Schwencer, NRC, Subject: Response to NRC Additional Information Requests, Appendix I, dated October 25, 1976.

3. Letter, M.G. Korsnick, Ginna LLC, to Document Control Desk, NRC, Subject: License Amendment Request Regarding Extended Power Uprate, dated July 7, 2005.

Table 11.3-1

GASEOUS EFFLUENTS, 10 CFR 50, APPENDIX I CALCULATIONS, ASSUMPTIONS[a] (HISTORICAL)

Thermal power level, Mwt	1520
Plant capacity factor	0.80
Mass of primary coolant, lb	2.82×10^5
Percent fuel with cladding defects	0.12
Primary system letdown rule, gpm	40
Letdown cation demineralizer flow, gpm	3.6
Number of steam generators	2
Total steam flow, lb/hr	6.58×10^6
Mass of steam in each steam generator, lb	4.622×10^3
Mass of liquid in each steam generator, lb	8.5×10^4
Mass of water in steam generators, lb	1.7×10^5
Total mass of secondary coolant, lb	8.15×10^5
Blowdown rate, lb/hr	1.1×10^4
Primary to secondary leak rate, lb/day	100
Fission product carryover fraction	0.0010
Halogen carryover fraction	0.0100
Radwaste dilution flow, gpm	4.0×10^5

Gaseous Waste Inputs

There is continuous low volume purge of volume control tank.	
Holdup time for xenon, days	90
Holdup time for krypton, days	90
Fill time of decay tanks for the gas stripper, days	45
Gas waste system particulate release fraction	0.01000
Auxiliary building iodine release fraction	0.10000
Auxiliary building particulate release fraction	0.01000
Containment volume, ft^3	1.0×10^6

GASEOUS EFFLUENTS, 10 CFR 50, APPENDIX I CALCULATIONS, ASSUMPTIONS[a] (HISTORICAL)

Containment atmosphere cleanup rate, cfm	1.02×10^4
Frequency of containment high volume purge, times/year	28
Containment shutdown purge, iodine release fraction	1.00000
Containment shutdown purge, particulate release fraction	0.01000
There is not a containment low volume purge[b]	
Steam leak to turbine building, lb/hr	1.7×10^3
Fraction iodine released from blowdown tank vent	0.0
Fraction iodine released from main condenser air eject	1.0000

Note: The information presented in the above Table represents the inputs and assumptions utilized in the original 10CFR50 Appendix I analysis performed for Ginna and is retained herein for historical purposes only.

a. Input to computer program GALE
b. See Section 6.2.4.4.9 for current containment purging methodology.

Table 11.3-2
GASEOUS EFFLUENTS, 10 CFR 50, APPENDIX I CALCULATIONS, RESULTS[a] (HISTORICAL)

			Gas Stripping		Building Ventilation					
	Primary Coolant (µCi/g)	Secondary Coolant (µCi/g)	Shutdown	Continuous	Reactor (ci/yr)	Auxiliary (Ci/yr)	Turbine (Ci/yr)	Blowdown Vent Offgas (ci/yr)	Air Ejector Exhaust (Ci/yr)	Total
^{83m}Kr	1.749×10^{-2}	1.101×10^{-8}	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
^{85m}Kr	8.637×10^{-2}	5.549×10^{-8}	0.0	0.0	0.0	2.0	0.0	0.0	1.0	3.0
^{85}Kr	6.519×10^{-3}	4.161×10^{-9}	2.0	1.1×10^{2}	2.0	0.0	0.0	0.0	0.0	1.1×10^{2}
^{87}Kr	5.070×10^{-2}	3.082×10^{-8}	0.0	0.0	0.0	1.0	0.0	0.0	0.0	1.0
^{88}Kr	1.629×10^{-1}	1.021×10^{-7}	0.0	0.0	0.0	3.0	0.0	0.0	2.0	5.0
^{89}Kr	4.354×10^{-3}	2.779×10^{-9}	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
^{131m}Xe	1.545×10^{-2}	9.926×10^{-9}	0.0	0.0	4.0	0.0	0.0	0.0	0.0	4.0
^{133m}Xe	8.420×10^{-2}	5.410×10^{-8}	0.0	0.0	9.0	2.0	0.0	0.0	1.0	1.2×10^{1}
^{133}Xe	4.196×10^{0}	2.657×10^{-6}	0.0	0.0	8.5×10^{2}	8.9×10^{1}	0.0	0.0	5.6×10^{1}	9.9×10^{2}
^{135m}Xe	1.126×10^{-2}	7.108×10^{-9}	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
^{135}Xe	2.482×10^{-1}	1.568×10^{-7}	0.0	0.0	5.0	5.0	0.0	0.0	3.0	1.3×10^{1}
^{137}Xe	7.834×10^{-3}	4.961×10^{-9}	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
^{138}Xe	3.809×10^{-2}	2.368×10^{-8}	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Total N ble Gases										**1.1×10^{3}**

			Gas Stripping		Building Ventilation					
	Primary Coolant (μCi/g)	Secondary Coolant (μCi/g)	Shutdown	Continuous	Reactor (ci/yr)	Auxiliary (Ci/yr)	Turbine (Ci/yr)	Blowdown Vent Offgas (ci/yr)	Air Ejector Exhaust (Ci/yr)	Total
^{131}I	2.236×10^{-1}	9.207×10^{-5}	0.0	0.0	1.0×10^{-3}	3.6×10^{-3}	5.0×10^{-3}	0.0	2.2×10^{-2}	3.2×10^{-2}
^{133}I	3.199×10^{-1}	8.415×10^{-5}	0.0	0.0	1.2×10^{-3}	5.1×10^{-3}	4.6×10^{-3}	0.0	3.2×10^{-2}	4.3×10^{-2}

Tritium Gaseous Release **400 Ci/yr**

Note:
1. 0.0 appearing in the table indicates release is less than 1.0 Ci/yr for noble gas, 0.0001 Ci/yr for Iodine.
2. The information presented in the above table represents the results of the original 10CFR50 Appendix I analysis performed for Ginna and is retained herein for historical purposes only.

a. Calculated by computer program GALE.

Table 11.3-3

MAXIMUM INDIVIDUAL DOSES FROM GASEOUS EFFLUENTS (HISTORICAL)[a]

(mrem/year)

Pathway	Age Group	Total Body	GI Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
Plume	All	0.022	0.022	0.022	0.022	0.022	0.022	0.022	0.051
Ground shine and inhalation	Adult	0.046	0.046	0.018	0.046	0.046	0.075	0.046	0.046
	Teen	0.034	0.034	0.018	0.034	0.038	0.057	0.034	0.034
	Child	0.034	0.034	0.018	0.034	0.028	0.066	0.034	0.034
	Infant	0.035	0.035	0.018	0.035	0.025	0.089	0.035	0.035
Vegetables	Adult	0.229	0.22	0.62	0.22	0.23	0.73	0.22	0.22
	Teen	0.290	0.29	0.20	0.29	0.26	0.70	0.29	0.28
	Child	0.620	0.62	0.49	0.62	0.22	1.26	0.62	0.62
	Infant	b	b	b	b	b	b	b	b
Cow or goat	Adult	0.0014	0.0014	0.0042	0.0014	0.0015	0.017	0.0014	0.0014
	Teen	0.0020	0.002	0.0015	0.0021	0.0019	0.025	0.002	0.002
	Child	0.0045	0.0045	0.0037	0.0045	0.0016	0.051	0.0045	0.0045
	Infant	0.0091	0.0089	0.0078	0.0092	0.0016	0.120	0.009	0.009
Meat	Adult	0.058	0.058	0.221	0.058	0.058	0.11	0.058	0.058
	Teen	0.041	0.041	0.035	0.042	0.034	0.08	0.041	0.041
	Child	0.074	0.074	0.066	0.074	0.022	0.13	0.074	0.074
	Infant	b	b	b	b	b	b	b	b

MAXIMUM INDIVIDUAL DOSES FROM GASEOUS EFFLUENTS (HISTORICAL)[a]

(mrem/year)

Pathway	Age Group	Total Body	GI Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
Total (not including plume)	Adult	0.334	0.325	0.863	0.325	0.336	0.932	0.325	0.325
	Teen	0.367	0.367	0.254	0.368	0.334	0.862	0.367	0.357
	Child	0.732	0.732	0.577	0.733	0.272	1.507	0.732	0.732
	Infant	0.044	0.044	0.026	0.044	0.026	0.209	0.040	0.044

Note: The information presented in the above table represents the results of the original 10CFR50 Appendix I analysis performed for Ginna and is retained herein for historical purposes only.

a. Highest offsite annual beta air dose = 0.14 mrad/year, 0.3 miles east. Highest offsite annual gamma air dose = 0.10 mrad/year, 0.3 miles east.

b. Indicates dose less than 0.001 mrem/year.

Table 11.3-4
ESTIMATED ANNUAL DOSES TO THE PUBLIC DUE TO NORMAL OPERATION GASEOUS RADWASTE EFFLUENTS - CORE POWER LEVEL 1811 MWt

Type or Dose	Appendix I Design Objectives	Pre-Uprate[a]	Uprate[b]	Percentage of Appendix I Design Objectives for EPU Case
Gamma Dose in Air	10 mrad/yr	Not Reported Separately in Annual Radioactive Release Report[c]	19% increase	As other doses are a small fraction of Appendix I Limits, it is assumed that this dose and consequent increase is also a small fraction of Appendix I.
Beta Dose in Air	20 mrad/yr>	Not Reported Separately in Annual Radioactive Release Report	19% increase	As other doses are a small fraction of Appendix I Limits, it is assumed that this dose and consequent increase is also a small fraction of Appendix I.
Dose to total body of an individual	5 mrem/yr	7.06E-03 mrem/yr	8.41E-03 mrem/yr	0.168%
Dose to skin of an individual	15 mrem/yr	9.25E-03 mrem/yr	1.10E-02 mrem/yr	0.073%
	Radioiodines and Particulates Released to the Atmosphere			
Dose to any organ from all pathways	15 mrem/yr	1.76E-02 mrem/yr	2.27E-02 mrem/yr	0.151%

a. Base Case - doses based on an average 5 year set of organ and whole body doses calculated using data presented in the Ginna Annual Radioactive Effluent Release Reports for the years 1999 through 2003, taking into consideration the associated average annual core power level, extrapolated to 100 persent availability at the licensed power level of 1520 MWt.

b. Estimated upper bound off-site doses developed as discussed in Section 11.3.3 by using scaling techniques that are applied to the Base Case.

c. Ginna historically included Gamma Air Dose in total body dose and Beta Air Dose in skin dose.

11.4 SOLID WASTE MANAGEMENT SYSTEM

11.4.1 DESCRIPTION

11.4.1.1 General

11.4.1.1.1 Types of Solid Waste

The waste disposal system is designed to package all solid waste in standard liners and other approved packages for removal to burial or processing facilities. The types of solid waste that are produced at Ginna Station in addition to dry active waste are:

I. Sludge.

II. Oily waste.

III. Bead resin.

IV. Filters.

11.4.1.1.2 Sludge

Suspended solids and other sludges occasionally require processing. This material is processed using a vendor-supplied system. A topical report demonstrating satisfactory processing by a vendor is required. The vendor's procedures must be approved in accordance with applicable station administrative procedures. Lab samples are created and tested and following quality control review, full-scale solidification is performed.

11.4.1.1.3 Oily Waste

Oily waste is processed by incineration at a central processing facility. An alternative method is to solidify and bury the waste at a licensed burial site. An approved method of solidification is to add an emulsifier to the oily waste and water at a neutral pH. The mixture is solidified by adding an approved solidification matrix. Other methods that may be employed utilize filtration. This would be a vendor-supplied system and would require the appropriate review and approval.

11.4.1.1.4 Bead Resin

Bead resin is used to remove chemical impurities and radioactive contamination from the reactor coolant, the chemical and volume control system, the spent fuel pool (SFP), and the liquid waste processing system. When the resin is exhausted or reaches a radiation limit, the spent resin is sluiced to one of two 150 ft^3 (1122-gal) spent resin storage tanks. After sufficient resin has been collected in one of the storage tanks, a transport cask sufficient for the radioactivity present is ordered. The transport cask is inspected using a quality control inspection procedure specific for each type of cask to ensure that the cask meets all the requirements of the Certificate of Compliance. A liner, which contains internal piping to completely dewater the resin, is installed in the cask. The cask is handled, loaded, and unloaded using a procedure specific for the model cask used. Piping is run from the drumming station to the manway in the top of the liner. Spent resin is then slurried from the spent resin storage tank into the liner with water used for sparging and mixing the resin, and

nitrogen gas pressure used to move the resin. A representative sample of the resin is obtained and the concentration of each radioisotope is calculated. After the resin is dewatered, the liner is capped and sealed and the top is put back on the cask. The cask is surveyed for radiation and contamination and properly labeled and marked. The resin is then transported to the licensed facility.

11.4.1.1.5 Spent Filters

When filters become saturated or have a high dose rate, they are dewatered and then replaced. The spent filters are placed in a high integrity container or solidified in an approved media and shipped in accordance with 10 CFR 71, 10 CFR 61, and burial site licenses. The maximum dose rate allowed on the surface of the container is determined by the shielding of the package in which the container is shipped.

11.4.1.1.6 Dry Active Waste

Dry active waste is shipped in bulk form to a vendor for volume reduction and packaging for delivery to the disposal site.

Components associated with the solid waste processing system are described in the following sections.

11.4.1.2 Spent Resin Storage Tanks

The spent resin storage tanks (Drawing 33013-1270, Sheets 1 and 2) retain spent resin normally discharged from the mixed-bed, spent fuel pool (SFP), base removal, and cation demineralizers. Normally, the tank is filled over a long period of time, the contents are allowed to decay, and are then emptied prior to receiving any additional resin. However, the contents can be removed at any time, if sufficient shielding is provided for the spent resin shipping vessel. A layer of water is maintained over the resin surface to prevent resin degradation due to heat generation from decaying fission products. Resin is removed from the tank by first backflushing with nitrogen to loosen the resin bed and then flushing the resin out with nitrogen entering the top of the tank. The tanks are all-welded austenitic stainless steel.

11.4.1.3 Storage Facilities

The upper radwaste storage facility (URSF) is located northeast of the plant within the security fence and is used for temporary storage of radioactive waste. The URSF has a bridge crane with a 3-ton capacity that provides versatility for moving radwaste in and out. This facility is large enough to accommodate a 45-ft box van. Additionally, shielded storage containers are used to provide supplemental storage of radioactive waste. These concrete casks are designed for outside storage and may be used for temporary storage of resins or other materials that require additional shielding.

Additionally, there is a reinforced concrete structure located northwest of the plant outside the security fence. This facility houses the old steam generators and is designed for long-term storage.

11.4.2 SOLID WASTE ESTIMATES

Ginna Station generated and shipped offsite approximately 12,000 ft^3 of solid radwaste in

1983 and approximately 9000 ft^3 in 1984. Average values in 1986-1988 were approximately 6000 ft^3. The March 1991 RG&E update to the Final Environmental Statement related to the operation of Ginna Station states that in total, Ginna delivers an average of 5000 ft^3 with a content of 200 curies of waste to disposal sites each year (*Reference 1*). A breakdown of quantities and radioactive content according to waste classification for the period July 1990 to June 1991 is shown in Table 11.4-1. Reference 2 indicates that the extended power uprate (EPU) will have minimal impact on the volume of solid radwaste generated at Ginna Station. The quantities shipped offsite for processing and burial are reported to the NRC in the Radioactive Effluent Release Report.

R.E. Ginna shipped approximately 60,000 ft^3 of Radioactive Waste during the period of 2013-2018. Approximately 5000 ft3 of waste shipped during this period was stored legacy waste. This results is an average generation of approximately 11,000 ft^3 of Radioactive Waste per year during that time frame. The total activity for the radioactive waste shipped during this period was 222 Curies. The volume and activity for various waste types is summarized in Table 11.4-2.

11.4.3 PROCESS CONTROL PROGRAM

The Offsite Dose Calculation Manual (ODCM) controls the establishment of a Process Control Program. The Process Control Program outlines the method for processing wet solid wastes and for solidification of liquid wastes. It includes applicable process parameters and evaluation methods used at Ginna Station to ensure compliance with the requirements of 10 CFR 71 prior to shipment of containers of radioactive waste from the site.

A radwaste sampling and analysis program has been instituted to ensure compliance with 10 CFR 61. Scaling factors have been developed to calculate concentrations of hard-to-measure isotopes from more easily determined isotopes. The scaling factors will enable concentrations of all required isotopes to be determined for each radwaste shipment.

All radioactive waste is shipped to a licensed burial site in accordance with applicable NRC, Department of Transportation, and state regulations, including burial site regulation requirements.

To ensure that personnel exposure is minimized, "as low as reasonably achievable" considerations are addressed in all phases of the solidification process.

REFERENCES FOR SECTION 11.4

1. Letter from R. C. Mecredy, RG&E, to A. R. Johnson, NRC, Subject: Environmental Issues Related to CP-0L R. E. Ginna Nuclear Power Plant, dated March 8, 1991.

2. Letter, M.G. Korsnick, Ginna LLC, to Document Control Desk, NRC, Subject: License Amendment Request Regarding Extended Power Uprate, dated July 7, 2005.

Table 11.4-1
ANNUAL SHIPMENT OF SOLID WASTE (JULY 1990-JUNE 1991)

Type of Waste	
Spent resins, filter sludges, evaporator bottoms, etc.	
Volume, m^3	89.9
Activity, Ci	207.3
Dry compressible waste, contaminated equipment, etc.	
Volume, m^3	63.2
Activity, Ci	4.2
Irradiated components, control rods, etc.	
Volume, m^3	0
Activity, Ci	0

NOTE: Total volume = 153.1 m^3 = 5405 ft^3

Table 11.4-2
Annual Shipment of Solid Waste (January 2013 – December 2018)

Type of Waste

Resins, Filters, And Evaporator Bottoms		
	Total	Average (Year)
Volume, m^3	7.32E+01	1.46E+01
Activity, Ci	2.17E+02	4.34E+01
Dry Active Waste (DAW)		
	Total	Average (Year)
Volume, m^3	1.58E+03	3.16E+02
Activity, Ci	5.35E+00	1.07E+00
Irradiated Components		
	Total	Average (Year)
Volume, m^3	0.00E+00	0.00E+00
Activity, Ci	0.00E+00	0.00E+00
Other Waste		
	Total	Average (Year)
Volume, m^3	3.77E+01	7.54E+00
Activity, Ci	3.85E-02	7.70E-03

Note: Total Volume = 1.69E+03 m^3 = 5.97E+04 ft^3

11.5 PROCESS AND EFFLUENT RADIATION MONITORING AND SAMPLING SYSTEMS

11.5.1 DESIGN BASES

All liquid and gaseous radioactive releases are continuously monitored for gross activity during discharge to ensure that the activity limits specified in 10 CFR 20 for unrestricted areas are not exceeded. The Offsite Dose Calculation Manual (ODCM) include limits applicable to release of radioactive material in liquid and gaseous effluents.

The radioactive liquid effluent instrumentation is provided to monitor and/or control, as applicable, the releases of radioactive materials in liquid effluents. The alarm and/or trip setpoints for these instruments are calculated in accordance with the Offsite Dose Calculation Manual to ensure that alarm and/or trip will occur prior to exceeding the limits of 10 CFR 20. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR 50.

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents. The alarm and/or trip setpoints for these instruments are calculated in accordance with the Offsite Dose Calculation Manual to ensure that alarm and/or trip will occur prior to exceeding the limits of 10 CFR 20. This instrumentation also includes provisions for monitoring the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The operability and use of this instrumentation is consistent with the requirements of General Design Criterion 64 of Appendix A to 10 CFR 50.

11.5.2 SYSTEM DESCRIPTION

11.5.2.1 General

The radiation monitoring system detects, computes, indicates, annunciates, and records the radiation level at selected locations inside and outside the reactor plant and is divided into the following subsystems.

A. The process radiation monitoring system consists of channels that primarily give early warning of a plant malfunction and secondarily warn personnel of increasing radiation which might result in exceeding release rate limits.

B. The area radiation monitoring system consists of channels which primarily warn personnel of increasing radiation that might result in an unnecessary exposure.

The radiation monitoring system continuously monitors plant effluents and various in-plant points selected to provide indication and warning in areas where radioactive sources exist and operating personnel are required to be present. Laboratory analysis equipment provides analytical information on the chemical and radiochemical contents from the many samples taken throughout and adjacent to the plant. Personnel monitors are provided to record integrated exposure for all site personnel. The above provides adequate information and warning for the continued safe operation of the plant.

The area radiation monitoring system is described in detail in Section 12.3.4. The process radiation monitoring system is discussed below. The postaccident sampling system is described in Section 9.3.2.3.

11.5.2.2 Process Radiation Monitoring System

11.5.2.2.1 General Description

The process radiation monitoring system continuously monitors various fluid and air streams for indication of increasing radiation levels or the presence of radioactivity in the selected process systems. This system also provides visual and audible alarms in the control room in order to alert the operators to any significant increases in activity. The system consists of liquid and airborne radioactivity monitors that will warn of hazardous or potentially hazardous contamination of effluent waters and gases. The system is also designed to provide a visual alarm in the control room if there is a malfunction of the meter or detector circuitry. Liquid process radiation monitoring channels R-17, R-18, R-19, R-21, and R-22 also provide automatic control functions as a result of a high alarm. The majority of the channels within the process radiation monitoring system utilizes a scintillation type detector described in Section 12.3.4.3. The control room operators have access to data from the effluent monitors via the plant process computer system.

The process radiation monitoring points and the associated release rate limits and alarm setpoints (calculated by methods described in the Offsite Dose Calculation Manual) are listed in plant procedures. The alarm setpoints are set at a fraction of the release limit value and are controlled by plant procedures. The process radiation monitoring locations in relation to the gaseous and liquid waste effluent paths are shown schematically in Figures 11.5-1 and 11.5-2.

Periodic testing of the process radiation monitors consists of channel checks, source checks, functional tests, and calibrations as specified in the Offsite Dose Calculation Manual (ODCM).

11.5.2.2.2 Containment Iodine Monitor

The containment vent or containment atmosphere iodine monitor (R-10A) collects radioiodine on a cartridge. A detector readout in counts per minute (cpm) indicates the quantity (microcuries) of Iodine-131 on the cartridge. The change in the counts per minute reading over a time period (delta cpm/hour) indicates the concentration of Iodine-131 in the air being sampled.

11.5.2.2.3 Plant Vent Iodine Monitor

The plant vent iodine monitor (R-10B) collects radioiodine on a cartridge. A detector readout in counts per minute (cpm) indicates the quantity (microcuries) of Iodine-131 on the cartridge. The change in the counts per minute reading over a time period (delta cpm/hour) indicates the concentration of Iodine-131 in the plant vent air.

11.5.2.2.4 Containment Particulate and Noble Gas Monitors

The containment vent or containment atmosphere particulate monitor (R-11) normally measures short-lived particulate daughters of noble gas. The usual isotope seen is Rubidium-88 which has an 18-minute half-life and is the daughter of Krypton-88 gas. The radioactive particulates of concern are those with half-lives greater than 8 days, such as Cesium-137. The release rate limit for Cesium-137 is calculated by methods described in the Offsite Dose Calculation Manual (ODCM). The alarm setpoint is a fraction of the release limit value and is sufficient during containment venting. This setpoint is also sufficient to warn of coolant leaks when the containment is isolated during reactor operation. When the containment is isolated, and the monitor is not monitoring an effluent release, the alarm setpoint can be changed as necessary to sense coolant leaks.

The containment vent or containment atmosphere noble gas monitor (R-12) measures the noble gas concentration in which Xenon-133 is the major isotope present. The release rate limit for Xenon-133 is calculated by methods described in the Offsite Dose Calculation Manual (ODCM). The alarm setpoint is a fraction of the release limit value and is sufficient for sampling both the containment vent and the containment atmosphere when the containment is isolated.

Monitor R-11 or R-12 is required to sample the containment atmosphere during mini-purge operation and will automatically isolate the mini-purge system on high radiation levels in containment (see section 7.3.1.1). Monitor R-11 or R-12 is required to be operable for reactor coolant system leakage detection as described in section 5.2.5.1.

11.5.2.2.5 Containment Vent High-Range Effluent Monitor

The containment vent high-range effluent monitor (RM-12A) installed in accordance with NUREG 0737, Item II.F.1, monitors particulate, Iodine-131, and the noble gas in the containment vent. The alarm setpoints are set to correspond to certain fractions of the release rate limits given in the Offsite Dose Calculation Manual (ODCM). For the particulate alarms, Cesium-137 is used as representative of isotopes with greater than 8-day half-lives. The release limit values and alarm setpoints are listed in plant procedures.

11.5.2.2.6 Plant Vent Particulate Monitors

The plant vent particulate monitor (R-13) normally is measuring short-lived particulate daughters of noble gas. The usual isotope seen is Rubidium-88 which has an 18-minute half-life and is the daughter of Krypton-88 gas. The radioactive particulates of concern are those with half-lives of greater than 8 days, such as Cesium-137. The release rate limit for Cesium-137 is calculated by methods described in the Offsite Dose Calculation Manual (ODCM). The alarm setpoint is a fraction of the release limit value and is sufficient to provide warning of a problem some time before the release rate limit is reached. The release limit values and alarm setpoints are listed in plant procedures.

11.5.2.2.7 Plant Vent Noble Gas and High-Range Effluent Monitor

The plant vent noble gas monitor (R-14) normally measures low concentrations of Xenon-133 from reactor coolant leaks in the auxiliary building, gas decay tank releases, or from

taking primary system samples in the nuclear sample room. The release rate limit for Xenon-133 is calculated by methods described in the Offsite Dose Calculation Manual (ODCM). The alarm setpoint is a fraction of the release limit value and terminates gas decay tank releases, initiates plant vent isolation, and provides a warning of unusual conditions before the release rate limit is reached.

The plant vent high-range effluent monitor (RM-14A) installed in accordance with NUREG 0737, Item II.F.1, monitors particulate, Iodine-131, and noble gas in the ventilation air exhausted from the auxiliary and intermediate buildings. The alarm setpoints are set to correspond to certain fractions of the release rate limits given in the Offsite Dose Calculation Manual (ODCM). For the particulate alarms, the release rate limit for Cesium-137 is used as representative of isotopes with half-lives greater than 8 days. The release limit values and alarm setpoints are listed in plant procedures.

11.5.2.2.8 Air Ejector and Gland Steam Exhaust Monitors

The air ejector and gland steam exhaust monitor (R-15) is used to detect primary-to-secondary leakage and to determine the release rate of noble gases from the main condenser. The release rate limit for noble gas from this source is calculated by methods described in the Off-site Dose Calculation Manual (ODCM). The air ejector and gland seal exhaust release concentration limit is calculated utilizing the release limit value and a conservatively large flow rate specified in the ODCM. The alarm setpoint is set at a fraction of the release concentration limit value. R-15 can be used to trend steam generator tube leakage or to determine an actual volumetric leak rate by correlating the monitor response to a leak rate determined by grab sample analysis.

The air ejector and gland steam exhaust monitor (R-48) is used to determine the release rate of noble gases from the main condenser. The release rate limit for noble gas from this source is calculated by methods described in the Offsite Dose Caluclation Manual (OCDM). The air ejector and gland steam exhaust release concentration limit is calculated utlilizing the release limit value and a conservatively large flow rate specified in the ODCM. The alarm setpoint is set at a fraction of the release concentration limit value. R-48 will alarm prior to Emergency Action Level (EAL) limits. R-48 mimics R-15 but provides a higher range required for accident conditions.

The air ejector noble gas monitor (R-47) is used to detect primary-to-secondary leaks. The alarm setpoints are set to correspond to primary-to-secondary leak rates in PPCS and ODCM limits in the rate-meter. R-47 directly monitors the condenser off-gas making it more sensitive than R-15 and R-48 which monitor the off-gas diluted by the gland air inleakage flow. In the event of significant fuel clad failures coincident with steam generator tube leakage, it is possible for R-47 to become over-ranged. In this case R-15 can be used for leak rate trending. Note that the increased reactor coolant activity with fuel failures more than offsets the reduction in sensitivity resulting from the combined air ejector and gland seal flow at R-15.

R-47 and R-48 provide displays in the TSC and data is transferred to the plant process computer system, which has high warning alarms. In addition, the plant process computer system calculates the primary-to-secondary leak rate continously from the R-47 monitor and

alarms at several levels as leakage increases to provide the operators with prompt warning of changing conditions.

11.5.2.2.9 Containment Service Water Monitor

The containment service water (containment fan coolers) monitor (R-16) monitors the service water from the containment and will see only background values, except for unusual accident conditions. A service water leak in the containment during an accident plus containment pressure greater than service water pressure is needed to contaminate the service water. The release rate limit is calculated by methods described in the Offsite Dose Calculation Manual (ODCM). The alarm setpoint is a fraction of the release limit value and prevents spurious alarms while affording good sensitivity. The release limit values and alarm setpoints are listed in plant procedures.

11.5.2.2.10 Component Cooling Water Monitor

The component cooling water monitor (R-17) is located in the Auxiliary Building in an area of varying background. The release rate limit is calculated by methods described in the Off-site Dose Calculation Manual (ODCM). The alarm setpoint is a fraction of the release limit value and is sufficient to detect leakage of primary coolant into the component cooling water. A high radiation level alarm on R-17 will cause the vent valve on the component cooling water surge tank to close. The release limit values and alarm setpoints are listed in plant procedures.

11.5.2.2.11 Liquid Waste Disposal Monitor

The liquid waste disposal monitor (R-18) can become internally contaminated causing increased background readings. The alarm setpoint is always added to the background reading of the monitor obtained when it contains clean water. For a maximum permissible concentration of 1×10^{-6} Ci/cm^{3}in the discharge canal, a minimum dilution flow of 170,000 gpm, and a maximum waste discharge rate of 75 gpm, the concentration limit in the waste tank is
2.2×10^{-3} μCi/cm^3. The release rate limit is calculated by methods described in the Offsite Dose Calculation Manual (ODCM). The alarm setpoint is a fraction of the release limit value. When monitor R-18 alarms, the liquid waste discharge valve (AOV-18) to the discharge canal closes. The release limit values and alarm setpoints are listed in plant procedures.

11.5.2.2.12 Steam-Generator Blowdown Monitor

The steam-generator blowdown monitor (R-19) monitors for primary to secondary leakage. The release rate limit is calculated by methods described in the Offsite Dose Calculation Manual (ODCM). The alarm setpoint is a fraction of the release limit value and will provide warning of significant steam-generator tube leakage and prevent blowdown activity of greater than 1×10^{-4} μCi/cm^3 from reaching the condenser. When R-19 reaches the high alarm setpoint, the blowdown system is automatically isolated by the closing of blowdown isolation valves AOV-5738 and AOV-5737, and the steam generator sampling lines are automatically isolated by the closing of valves AOV-5735 and AOV-5736 in the sampling lines. The release limit values and alarm setpoints are listed in plant procedures.

11.5.2.2.13 Spent Fuel Pool (SFP) Heat Exchanger Service Water Monitors

The spent fuel pool (SFP) heat exchanger service water monitors (R-20A and R-20B) monitor for leakage from the spent fuel pool (SFP) into the service water due to heat exchanger tube leaks. R-20A monitors spent fuel pool (SFP) heat exchanger A. R-20B monitors spent fuel pool (SFP) heat exchanger B. An activity of 2.4×10^{-4} $\mu Ci/cm^3$ in the service water due to pool water leakage would be required to reach a mixture of 1×10^{-6} $\mu Ci/cm^3$ in the discharge canal using a service water flow of 700 gpm and a dilution flow of 170,000 gpm. Since the normal activity in the pool water is 1×10^{-3} $\mu Ci/cm^3$, a leak of 165 gpm would be required to reach an activity of 2.4×10^{-4} $\mu Ci/cm^3$ in the service water discharge from the heat exchanger. A leak of this size would quickly be known due to pool low-level alarms. The release rate limits for monitors R-20A and R-20B are calculated by methods described in the Offsite Dose Calculation Manual (ODCM). The alarm setpoints are a fraction of the release limit values and will prevent spurious alarms and provide warning of maximum permissible concentration releases. The release limit values and alarm setpoints are listed in plant procedures.

11.5.2.2.14 Retention Tank Monitor

The retention tank monitor (R-21) monitors water from various floor and equipment drains in the turbine, service, and control, intermediate, and condensate demineralizer (AVT) buildings. For a mixture of maximum permissible concentration of 1×10^{-6} $\mu Ci/cm^3$ in the discharge canal, a minimum dilution flow of 170,000 gpm, and a maximum waste discharge rate of 500 gpm, the concentration limit in the retention tank is 3.4×10^{-4} $\mu Ci/cm^3$. The release rate limit is calculated by methods described in the Offsite Dose Calculation Manual (ODCM). The alarm setpoint is a fraction of the release limit value. Because the monitor can become internally contaminated, the alarm setpoint is always added to the background reading when the monitor contains clean water. When the monitor reaches its setpoint, it causes the retention tank pump to trip and the discharge valve automatically closes, and the recirculation valve automatically opens. The release limit values and alarm setpoints are listed in plant procedures.

11.5.2.2.15 High Conductivity Waste Tank Monitor

The high conductivity waste tank monitor (R-22) monitors a side stream from the high conductivity waste tank pump. For a mixture of maximum permissible concentration of 1×10^{-6} $\mu Ci/cm^3$ in the discharge canal, a minimum dilution flow of 170,000 gpm, and a maximum waste discharge rate of 300 gpm, the concentration limit in the high conductivity waste tank is 5.7×10^{-4} $\mu Ci/cm^3$. The release rate limit is calculated by methods described in the Offsite Dose Calculation Manual (ODCM). The alarm setpoint is a fraction of the release limit value. When the alarm setpoint is reached, it automatically closes the high conductivity waste tank discharge valve.

11.5.2.2.16 **Control Room Radiation Monitors**

The control room radiation monitors (R-45 and R-46) consist of redundant detectors mounted in the control room air intake duct. Digital ratemeters, PPCS and recorders provide indication and alarms in the control room. The redundant isolation logic is incorporated into the redundant toxic gas logic and will initiate a control room isolation if high radiation is detected in the air intake duct. Alarm setpoint allowable values for R-45 and R-46 are contained in the Technical Specifications.

11.5.2.3 Tritium Sampling

Tritium in the containment atmosphere is in the form of gaseous water, i.e., humidity. To collect a sample and determine the concentration in $\mu Ci/cm^3$ of containment air, a dehumidifier or air-water bubbler is used. The condensate from the process is purified by distillation to remove any dissolved or suspended radioactive solids. An aliquot of the resulting pure water is placed in a liquid scintillation counter. From the counts per minute, counter efficiency, containment humidity, and volume of the sample, the tritium concentration in the air can be calculated. The expected sensitivity of the counter is 10^{-4} $\mu Ci/cm^3$ for tritium in water. This sensitivity will allow readings within 1/10 of the maximum permissible concentration for tritium in air.

11.5.3 DESIGN EVALUATION

The Offsite Dose Calculation Manual (ODCM) requires alarm and/or trip setpoints for specified radiation monitors on each noble gas effluent line. Precautions, limitations, and setpoints applicable to the operation of Ginna Station gaseous effluent monitors are provided in plant procedures. Setpoints are conservatively established for each ventilation noble gas monitor so that dose rates in unrestricted areas corresponding to 10 CFR 20 limits will not be exceeded. Setpoints are determined so that dose rates from release of noble gases comply with the Offsite Dose Calculation Manual (ODCM) requirements, which stipulate that the instantaneous dose rate for noble gases shall be less than or equal to 500 mrem/yr to the total body and less than or equal to 3000 mrem/yr to the skin. The methodology for calculating alarm and trip action points for each radioactive gaseous effluent monitor is contained in the latest revision of the Offsite Dose Calculation Manual.

11.5.4 ENVIRONMENTAL RADIOACTIVITY MONITORING PROGRAM

The onsite and offsite environmental sampling locations for airborne particulates, radioiodine, and direct radiation are included in the Offsite Dose Calculation Manual. The requirements of the radiological environmental monitoring program are included in the Offsite Dose Calculation Manual (ODCM) as required by the Technical Specifications.