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9.1 Summary Description

The radioactive waste systems are designed to collect, process and dispose of radioactive and potentially radioactive wastes in a controlled and safe manner without limiting plant power output or availability. The design objective for the radioactive waste systems is to provide equipment, instrumentation and operating procedures such that the discharge of radioactivity from the plant will not exceed the limits as set forth in 10CFR20 and meet the Design Objectives of Appendix I to 10CFR50.

The performance objectives of the radioactive waste control systems are:

a. To provide for the monitoring and the effective control of the release of radioactivity from the plant such that radiation dose rates to persons beyond the plant boundary are minimized below those limits set forth in 10CFR20.

b. To minimize the release of radioactivity to the environs and the corresponding radiation dose to persons beyond the plant boundary to satisfy the Design Objectives of Appendix I to 10CFR50.

c. To maximize the dispersal of the radioactivity released to the environs in terms of time and space; thereby reducing the average radioactivity concentration at the plant boundary.

d. To minimize the quantity of radioactive materials available for release and to reduce the potential mobility of the material released in the event of an operating error.

e. To maintain safe operating conditions by minimizing radiation hazards and exposure to plant operating personnel to as low as reasonably achievable (ALARA) levels.

The design basis for the radioactive waste systems is based on fuel leakage resulting in a 0.26 Ci/sec off-gas release rate following a 30-minute delay (design maximum fuel leakage condition). At this design maximum fuel leakage condition, the radwaste systems are selected to meet the design and performance objectives as stated above.

Normal radioactive input to the radwaste systems due to fuel leakage and activation products results in stack release rates much lower than those allowed by 10CFR20 and in off-site doses meeting the Design Objectives of Appendix I to 10CFR50. Under conditions of no fuel leaks, the radioactive input to the radwaste systems is due to activation products resulting from irradiation of the reactor water and impurities therein (principally metallic corrosion products). Corresponding release rates are even lower than the above rates.

Radioactive wastes resulting from station operation are classified as liquid, gaseous and solid. The following definitions apply to radioactive wastes:

a. Liquid Radioactive Wastes - Liquids directly from the reactor process and auxiliary systems or liquids which can become contaminated due to contact with these liquids from reactor process systems.

b. Gaseous Radioactive Wastes - Gases or airborne particulates vented directly from reactor and turbine equipment containing radioactive material.

c. Solid Radioactive Wastes - Solids from the reactor primary or auxiliary systems, solids in contact with reactor primary system liquids or gases, and solids used in reactor primary and turbine systems operations.

The systems which process these wastes for recycling, release or disposal are operated within limits established in the Technical Specifications.

9.2 <u>Liquid Radwaste System</u>

9.2.1 <u>Design Basis</u>

The liquid radioactive waste system is designed to collect, process and dispose of all radioactive liquid wastes generated in the operation of the plant. The system is designed to accommodate the radioactive input resulting from the design basis maximum fuel leakage condition.

9.2.2 <u>Description</u>

9.2.2.1 <u>General</u>

Liquid wastes from various drains and discharges from the reactor process and auxiliary systems are processed through the radwaste system. Final disposition of processed liquid may be one of the following:

a. return of the liquid to the condensate system for plant re-use, or

b. solidification of chemical liquid wastes and shipment of the resulting solid to an off-site location, or

c. release to the Mississippi River in accordance with limitations specified in the Offsite Dose Calculation Manual (ODCM).

Liquid wastes are collected in sumps and drain tanks in the various buildings and then transferred to the appropriate subsystem collection tanks in the Radwaste Building for subsequent treatment and disposal.

In order to keep the releases to a minimum, modifications were made to the liquid radwaste system to allow reclaiming of floor drains as well as equipment drains. The modified system limits the release of liquid effluents to the minimum practicable extent and to satisfy the Design Objectives of Appendix I to 10CFR50.

The radioactive and chemical contaminants are removed from the liquid waste streams by either filtration or filtration followed by mixed deep-bed demineralization. The filters remove insoluble particulate contaminants and the demineralizer is utilized to remove soluble materials. The filter and demineralizer sludge are backwashed into receiving tanks, dewatered and packaged as solid waste for disposal off-site at NRC approved sites.

Major flow paths and equipment are shown in Drawings NH-36043, NH-36044, NH-36045, NH-36046 and NH-36046-2 Section 15.

The radioactive waste effluent radiation monitor was installed for use during release of liquid radioactive waste to the discharge canal. Historically the use of this discharge path has not been required due to the design of the radioactive waste system. The liquid radwaste effluent radiation monitor has received reduced maintenance and calibration but remains available. Prior to use of this discharge path the ODCM requirements for the radiation monitoring equipment must be met. 01552374

9.2.2.2 Sources and Processing

9.2.2.2.1 High Purity and Low Purity Wastes

High Purity Wastes

Low conductivity (potentially high radioactivity) liquid wastes originating from piping and equipment drains are collected in:

a. the drywell equipment drain sump,

b. the reactor building equipment drain sump and tank, and

c. the turbine building equipment drain sump and tank.

Waste from the drywell equipment drain sump and the turbine building equipment drain sump and tank can be pumped directly to the condenser hotwell or to the waste collector tank in the radwaste building. The contents of the reactor building equipment drain sump and tank are transferred to the waste collector tank. The waste collector tank also collects wastes from sludge dewatering steps.

Low Purity Wastes

Low radioactivity and high conductivity liquid wastes, primarily from floor drains, are collected in:

a. the drywell floor drain sump,

b. the reactor building floor drain sump and tank,

c. the turbine building floor drain sump, and

d. the radwaste building floor drain sump.

Wastes are transferred from these collection points to the Floor Drain Collector Tank.

The treatment of both high purity and low purity wastes is combined into one processing chain. The waste collector tank and floor drain collector tank are cross-connected to allow mixing of contents.

The mixing is accomplished by recirculating their contents with a common pump simultaneously drawing suction from both tanks and processing the waste water through a filter-demineralizer unit and returning the liquid back to the tanks. In this way, an inventory of high purity water is maintained in the collection tanks to serve as a dilutent as fresh sump accumulation is discharged to the tanks. As the tanks are filled, the liquid is processed through a deep bed demineralizer and sent to one of two waste sample tanks. There, the contents are analyzed and, if acceptable, the effluent is reclaimed via the condensate storage tanks. If further processing is needed, the water is recycled back to the collector tanks and the process cycle repeated. The effluents may all be released to the river in accordance with the Technical Specifications.

9.2.2.2.2 Chemical Wastes

Chemical wastes are high conductivity wastes of variable activity levels originating from laboratory drains and various decontamination operations. These wastes are collected in the Chemical Waste Tank in the Radwaste Building. The wastes are sampled and neutralized as required and then processed in one of two ways depending upon the radioactivity level and the chemical nature of the wastes. If the radioactivity and chemical contamination concentrations are low, they are sent to the waste sludge tank, floor drain collector tank, or to the transportation liner via the B-centrifuge/B-hopper bypass line for additional processing and re-use. If the radioactivity concentrations in the chemical wastes are high and the chemical nature of the wastes does not lend itself to the conventional treatment of filtration and demineralization, then the wastes are solidified with cement or other acceptable technology for off-site disposal.

9.2.2.2.3 Detergent Wastes

Liquid Wastes originating from the laundering of radioactive clothing and from certain decontamination operations using detergents are high conductivity wastes and usually of very low activity level. These wastes are collected in the chemical waste tank and processed with the other chemical wastes.

9.2.2.2.4 Miscellaneous Liquid Wastes

Other liquid wastes resulting from refueling operations, plant start-up and equipment maintenance are classified according to conductivity, activity level, and chemical nature of the impurities. Based on the classification, these wastes are processed as either high purity or low purity, chemical or detergent wastes and may be released to the river in accordance with the Technical Specifications.

9.2.2.3 Instrumentation & Control of the Liquid Radwaste

Control of the liquid radioactive waste systems is exercised from a local control room situated in the Radwaste Building. The control room contains the instruments, control switches and alarms for the operation of the system. Included in the control room are valve position indicating lights and process and sump pump operating lights. A common radwaste trouble alarm is located in the plant main control room.

9.2.3 Performance Analysis

9.2.3.1 General

Protection against accident and/or off-standard discharge of wastes is provided by appropriate system interlocks, instrumentation for detection and alarm of off-standard conditions, batch sampling and analysis, and procedural controls. All radwaste tanks, filters and equipment are contained in concrete cells within the major concrete buildings of the plant in order to provide a substantial degree of immobility of the wastes within the plant. These arrangements are provided to assure that in the event of a failure of the liquid waste systems or errors in operation of the system the potential for inadvertent release of liquids is small. For example, the storage tanks, filter demineralizers and other equipment, are placed so that leakage is contained within the building. This assures control and containment of any leaks, spills or overflows from the equipment.

An evaluation was performed to determine the activity concentrations of principal radionuclides in the fluid streams under normal operation for Extended Power Uprate (EPU) conditions at 2004 MWt (References 14, 15, and 16). ANS/ANSI 18.1-1999 standard methodology was used to calculate the concentrations of nuclides in reactor water. The activity in the reactor water consists of activated corrosion products and fission products. The activated corrosion products are the result of metallic materials entering the water and being activated by neutron flux in the reactor region. The total activated corrosion product activity determined by the evaluation for EPU is approximately 41% of the total design basis activated corrosion product activity from USAR section 12.3.1.6. The fission product activity in the reactor water is the result of minute releases from the fuel rods. The cumulative calculated reactor water fission product activity level is less than 2% of the cumulative design basis reactor water fission product activity levels from USAR section 12.3.1.6. The NRC concluded in its review of the evaluation (Reference 17) that the resultant composition and quantity of radionuclides are appropriate for the evaluation of the radioactive waste management systems. The NRC further concluded that the radioactive source term meets the requirements of 10CFR20 and 10CFR50, Appendix I.

For the Maximum Extended Load Line Limit Analysis Plus (MELLLA+) operating domain there is no change in the FW flow, steam flow, or power. Production of coolant activation products is proportional to neutron flux and steam flow. Therefore, the MELLLA+ operating domain expansion does not affect the total activity concentration in the reactor coolant (Reference 19).

9.2.3.2 Tritium Significance

Tritium exists as a gas or combined in water. In the presence of water, the majority of the tritium will remain with the water and not appear as a gas.

The tritium release rates in the plant off-gas and liquid radwaste discharges result in concentration well below the 10CFR20 limits. The dose rate to the environs due to tritium is negligible and therefore not considered significant in the radioactive waste systems.

9.3 Gaseous Radwaste System

9.3.1 Design Basis

The gaseous radwaste systems design objective is to process and control the release of gaseous radioactive effluents to the site environs so that the off-site radiation dose rate does not exceed the limits specified in 10CFR20 and the Design Objectives of Appendix I to 10CFR50 are met.

The Monticello plant was originally designed with a simple off-gas treatment system. Radiolytic hydrogen and oxygen, including fission gases released from the fuel, and condenser in-leakage were removed at the air ejectors and allowed to decay by traveling down a long 42-inch delay pipe. The radioactive waste gas was then filtered and released from the plant 100-meter stack. A design holdup time of 30 minutes was provided by the system. The original design codes were ASA B31.1, 1955 Edition (Reference 6), and USAS B31.1.0 - 1967 (Reference 7).

Following plant startup, additions to the off-gas treatment system were made to bring the plant in compliance with NRC “as low as reasonably achievable” radioactive waste release guidelines. Recombiners were added to reduce the volume of radioactive waste gas. Compressors and storage tanks were added to provide a design holdup of 50 hours prior to release. Portions of the recombiner and compressed storage modification were originally designed and built to ASME Code Section III, Class 2 and Class 3, 1971 Edition (Reference 8). These systems were later reclassified in accordance with NRC Regulatory Guide 1.143, Revision 1 (Reference 5), which permits the use of ANSI B31.1 Code (Reference 9).

For recombiner and compressed storage system repairs and modifications (except non-nuclear auxiliary subsystems), the following Regulatory Guide 1.143, Revision 1, recommendations are adhered to:

(1) The governing code for design, materials, fabrication, and erection of all piping, piping components, and pipe support modifications or replacements for the Monticello gaseous radwaste treatment system is ANSI B31.1-1977 Code with addenda up to and including Winter of 1978 (Reference 9). This provides a consistent and contemporary basis for modifications and repairs.

(2) Pressure vessels, atmospheric tanks, 0-15 psig tanks, heat exchanges, and pumps utilized in modifications or replacements will be evaluated and shown to be in conformance with the recommendations contained in Table 1 of the Regulatory Guide, or suitable alternate criteria meeting the intent of the Regulatory Guide, prior to being installed in the recombiner or off-gas holdup systems.

(3) Materials utilized in recombiner or off-gas holdup system modifications or replacements will be evaluated and shown to be in conformance with the recommendations contained in Section 2.1.2 of the Regulatory Guide, or suitable alternate criteria meeting the intent of the Regulatory Guide, prior to being installed in the recombiner or off-gas holdup systems.

(4) Modifications and replacements related to the off-gas holdup tanks and related isolation valves, equipment, interconnecting piping, and components located between the upstream and downstream valves used to isolate these tanks from the rest of the system, will be analyzed for Operating Basis Earthquake (OBE) loads in accordance with Section 5 of the Regulatory Guide, or suitable alternate criteria meeting the intent of the Regulatory Guide.

(5) Recombiner and off-gas holdup system modifications and replacements will conform to the additional design, construction, and testing criteria related to leakage control and "as low as reasonably achievable" (ALARA) specified in Section 4 of the Regulatory Guide, or suitable alternate criteria meeting the intent of the Regulatory Guide.

(6) The quality assurance criteria described in Section 6 of the Regulatory Guide, or suitable alternate criteria meeting the intent of the Regulatory Guide, will be applied to recombiner and off-gas holdup system modifications and replacements. The extent of control required by 10CFR Part 50, Appendix B, is not required or appropriate. Specifically, the following quality assurance measures, or suitable alternative measures meeting the intent of the Regulatory Guide, are applicable to the recombiner and off-gas holdup systems:

a. Adherence to the quality control requirements of the applicable ASME or ANSI design and fabrication code.

b. Independent review of design and procurement documents.

c. Measures to ensure that suppliers of material, equipment, and construction services are capable of supplying these items to the quality specified in the procurement documents.

d. Instructions in procurement documents to control the handling, storage, shipping, and preservation of material and equipment to prevent damage, deterioration, or reduction of cleanliness.

e. Inspection of construction activities to verify conformance with the documented instructions, procedures, and drawings for accomplishing the activity. Items which have satisfactorily passed required inspections and tests are identified. Nonconforming items (failures, malfunctions, deficiencies, deviations, and defective material and equipment) are identified and action taken to correct such items. Sufficient records are maintained to furnish evidence that these measures are being implemented. Records should include results of reviews and inspections and should be identifiable and retrievable.

9.3.2 Description

9.3.2.1 General

Radioactive gases are delayed and filtered by the gaseous radwaste system and released through the plant off-gas stack. During normal operation, the gaseous radwaste system operates on a continuous basis with effective monitoring and control provided so as not to exceed the limits of 10CFR20 or the dose objectives of Appendix I to 10CFR50.

The gaseous radwaste system as shown in Drawings NH-36159, NH-54817-4, NH-54822-2, NH-54823-2, NH-54823-3 and NH-54823-4, Section 15, is a modification of the original system. The modification provides a compressed gas storage facility, capable of delaying main condenser off-gas for at least 50 hours compared with the original 30-minute hold-up. Greater than 100 hours of hold-up are typically achieved.

Details of the main condenser gas removal system are shown in Drawings NH-36034, NH-36035, and NH-36035-2, USAR Section 15.

9.3.2.2 Sources

Radioactive gases are collected from the following sources:

a. main condenser air ejector effluent,

b. steam packing exhaust system effluent,

c. plant start-up vacuum pump effluent,

d. HPCI gland seal effluent,

e. standby gas treatment system effluent, and

f. laboratory hood effluents.

The standby gas treatment system is discussed in Section 5.3.4.1 since it is directly associated with the plant building ventilation systems. The condenser air ejector effluent is collected and processed in the air ejector off-gas subsystem. The steam packing exhaust system effluent, the mechanical vacuum pump effluent and the HPCI gland seal effluent are all collected and processed in the steam packing exhauster off-gas subsystem.

9.3.2.3 Air Ejector Off-gas Subsystem

The air ejector offgases entering this system are the non-condensibles from the main condenser. They consist essentially of hydrogen and oxygen formed in the reactor by radiolytic decomposition of water, excess hydrogen from the Hydrogen Water Chemistry (HWC) System, air in-leakage to the turbine-condenser, water vapor, and fission gases (which are negligible in terms of volume).

Fission gas may arise from minor amounts of tramp uranium on the surface of the fuel element or from imperfections or perforations which might develop in the fuel cladding. The release rate of activation gases is dependent upon the thermal output of the reactor and the hold-up time provided in the gaseous radwaste system prior to release at the stack.

Steam diluted off-gas from the main condenser is processed through a recombiner subsystem where the hydrogen and oxygen react to form water reducing by a large factor the original volume of gases that must be processed and temporarily stored. Oxygen is added to the recombiner inlet stream by the HWC System to permit recombination of excess hydrogen. After removing the water for further treatment, the non-condensable gases pass through charcoal adsorbers and HEPA filters and then through the off-gas Holdup System where gases are compressed and stored in one of five hold up tanks. Prior to discharge through the main stack, the off-gases are passed once again through HEPA filters.

The off-gas exhaust stack filter system consists of two filter assemblies (one operating and the other in standby). Each filter assembly is composed of an integral HEPA filter element with a moisture separator all housed in one pressure vessel. The HEPA filter prevents all but a small fraction of the radioactive particulates from being released to the atmosphere.

These stack filter assemblies are housed in the off-gas stack and are shielded by at least 3 feet of concrete at the base of the stack.

Dilution fans are also provided in the base of the plant stack to maintain suitable exit velocities at the top of the stack.

9.3.2.4 Turbine and Air Ejector Steam Packing Exhauster Subsystem

The steam exhauster off-gases are discharged on a continuous basis into a 1.75 minute hold-up line and subsequently mixed with the air ejector off-gases and the dilution air at the base of the stack. The 1.75 minute hold-up time is established primarily to allow for radioactive decay of the activation gases, notably N-16 and 0-19.

The steam packing exhauster subsystem does not require filters. The short hold-up time and the small quantity of fission gases do not produce sufficient solid daughters to warrant filtration of these gases. Hydrogen and oxygen concentrations in this system are well below flammable concentrations and, therefore, the system is not designed to withstand explosive forces.

9.3.3 Performance Analysis

9.3.3.1 General

The modified off-gas system includes a compressed gas storage phase to provide an additional delay in effluent gas releases. The effect of this added delay is that a substantially greater fraction of the radioactive material will have decayed prior to release. Table 9.3-1 lists the estimated composition and quantities of the air ejector off-gases.

An evaluation was performed to determine the activity concentrations of principal radionuclides in the fluid streams under normal operation for Extended Power Uprate conditions at 2004 MWt (References 14, 15, and 16). ANS/ANSI 18.1-1999 standard methodology was used to calculate the concentrations of nuclides in reactor water and steam. The activity in the steam consists of noble gases released from the core plus carryover activity from the reactor water. This activity is the noble gas offgas that is included in the plant design. The expected total fission product offgas release rate determined by the evaluation is 0.018 Curies/sec which is bounded by the design total release rate of 0.26 Curies/sec. (See USAR section 12.3.1.6 for more information on design basis release rate). The NRC concluded in its review of the evaluation (Reference 17) that the resultant composition and quantity of radionuclides are appropriate for the evaluation of the radioactive waste management systems. The NRC further concluded that the radioactive source term meets the requirements of 10CFR20 and 10CFR50, Appendix I.

The plant operating parameters and design features which impact effluent gas releases are unchanged by plant operation in the maximum extended load line limit analysis plus (MELLLA+) domain. Therefore, the offsite release rate is unaffected by operation in the MELLLA+ domain (Reference 19).

The air ejector off-gas system is monitored and controlled to ensure that the radiation dose rate limits at the site boundary as prescribed in the Technical Specifications are not exceeded. Two off-gas pretreatment monitors are provided and when their trip point is reached, cause an automatic termination of air ejector off-gas flow. Isolation is initiated when both instruments reach their high trip point, one has an upscale trip and the other a downscale trip, or both have downscale trips. There is a 30-minute delay before off-gas flow is automatically terminated in which the reactor operator may take corrective action. Both instruments are required for trip. The trip settings of the instruments are set so that the maximum stack release rate limit allowed by the Offsite Dose Calculation Manual is not exceeded.

Radiation monitors are also provided in the plant main stack as a back-up detection of high activity release. Radiation levels, in excess of the allowable “instantaneous” release rate, alarm in the control room and isolate the hold-up line.

The plant stack allows atmospheric dispersion of the gas plume before it reaches ground level to reduce direct radiation dose rates. Natural dispersion of the gases into the atmosphere is achieved by a combination of plant stack height, exit velocity and plume buoyancy. Based on this natural dispersion, meteorological characteristics of the site, and the topography of the site environs, it has been determined that an annual average stack release rate of 0.26 Ci/sec can be accommodated without exceeding 500 mRem/year at the site boundary. Use of the gaseous radwaste treatment results in a stack release rate of less than 1% of this value to meet the Design Objectives of Appendix I to 10CFR50.

9.3.3.2 Atmospheric Release

The following points could possibly release radioactivity to the atmosphere and are provided with Wide Range Gas Monitors (WRGMs):

a. Plant main stack

b. Reactor building vent stack

Trip settings for the plant main stack WRGMs and alarm settings for the reactor building vent WRGM are calculated in accordance with NRC approved methods in the Offsite Dose Calculation Manual (ODCM) to ensure the maximum release rate is not exceeded.

The reactor building vent wide range gas monitors alarm in the control room. The control room operators can initiate prompt actuation of secondary containment and the standby gas treatment system to terminate releases which could result in exceeding radioactive effluent limits as identified in the ODCM.

Four radiation monitors are provided which initiate isolation of the reactor building and operation of the standby gas treatment system. The monitors are located in the reactor building vent plenum and on the refueling floor. Any one upscale trip will cause the desired action. Trip settings of ≤100 mR/hr for the monitors in the main exhaust plenum room and trip settings of ≤100 mR/hr for the monitors on the refueling floor are based upon initiating normal ventilation isolation and standby gas treatment system operation so that the activity released during a refueling accident is processed by the Standby Gas Treatment System. For the design basis refueling accident, analysis using Alternative Source Term methodology has demonstrated that accident doses remain within regulatory limits even without isolation of secondary containment, operation of SBGT, and isolation of the reactor building ventilation.

9.3.3.3 Off-gas Filters

In order to meet the NRC guidelines for the release of radioactive iodine from the off-gas system, charcoal filters which will remove about 95% of the iodine present in the off-gas at the entrance to the storage system were provided in addition to the HEPA stack filters. Although a single charcoal filter will normally remove in excess of 99% of the iodine present in a gas stream, the NRC generally does not allow design credit for more than 90% removal per filter due to the effects of variables such as charcoal material differences, moisture, low iodine concentrations, and deterioration of seals with age. Therefore, two charcoal filters in series have been specified.

Two charcoal beds in series are provided in the compressor suction filters. These beds are each 20” to 24” in diameter by at least 2” deep. There is a HEPA filter above the charcoal beds and a retention element below the beds. Each assembly is housed in an ASME Section III, Class 3 vessel of about 36” OD x 5 1/2’H. Two assemblies are provided for redundancy, with each designed for combined compressor capacity. The off-gas is preheated about 10°F before entering the filters to reduce the relative humidity below the 85% required for good methyl iodide removal. The ambient air around the filters is maintained above 70°F to prevent condensation on the interior surfaces of the vessel. Impregnated charcoals are specified as a further aid to methyl iodide removal.

The charcoal filter beds in the compressor suction are expected to last indefinitely. The stack HEPA elements are retained in service as long as testing verifies that they meet the filter efficiency requirements, and the filter pressure drop limit is not exceeded.

9.3.3.4 Hydrogen Explosion

The off-gas system, as originally installed, consisted of a 42-inch delay line, high efficiency filters, dilution fans and plant stack. This system was modified in 1974 to form two new systems. The Offgas Recombiner System functions to recombine the hydrogen and oxygen normally present in the off-gas, thus reducing the off-gas volume and rendering it non-combustible and safe for compressed storage. The Offgas Holdup System functions to provide additional delay time and removal of additional radioactive particulates and iodine from the gas (Reference 3).

The off-gas systems are designed to withstand the pressure encountered from a hydrogen detonation from an initial operating pressure of 20 psia assuming a stoichiometric hydrogen and oxygen mixture (Reference 4). NUREG-0800, Section 11.3, “Gaseous Waste Management System”, provides guidance for systems being designed to withstand a hydrogen explosion and recommends that piping be designed to 350 psia. As a minimum, all applicable piping in the off-gas system meets this recommendation except the compressed gas storage tanks. The compressed gas storage tanks are designed for a maximum pressure of 330 psig. The system upstream of the compressors normally operates at 12 psia.

Fully redundant hydrogen analyzers on the outlet from each Offgas Recombiner System initiate compressor system shutdown on sensing high hydrogen. The compressors isolate to prevent the addition of a potentially explosive mixture to the compressed storage tanks. Pressure then slowly builds up in the 42-inch delay line until corrective action is taken. These sensor and shutdown systems are designed with sufficient redundant equipment so that no one undetected fault in an active component will render the systems inoperable, and the systems are periodically tested to confirm continued operability.

A flammable mixture will not be allowed to reach the compressed gas storage tanks. The lower limit of flammability of 4% hydrogen is by volume; the detonation limit is 15% by volume. In the event of a complete failure of the recombiners, the volumetric hydrogen concentration at the suction of the compressors will not exceed 2.8% prior to compressor shutdown. This percentage conservatively assumes homogeneous mixing of off-gas in the 42 inch delay line downstream of the recombiners.

Should a number of unlikely events occur, it would be hypothetically possible for a hydrogen explosion to occur in the Offgas Recombiner System. Such an explosion within this system could result in an airborne shock wave propagating into the existing 42 inch decay pipe. The shock wave from the Offgas Recombiner System would be attenuated by its expansion into the 42 inch decay pipe and further attenuated in the interconnecting piping system and HEPA/charcoal filter in the compressor suction.

Calculations indicate that the overpressure associated with any shock wave propagating from the Offgas Recombiner System would be attenuated to about 1/10 of its initial value by the effects of expansion and subsequent propagation through the various piping systems to the air compressor inlet filter. To insure that no potential shock wave could reach the air compressors, the charcoal filter element has been designed to withstand the full 350 psig over-pressure without failure, and thus the filter element will reflect and/or absorb any potential shock wave to insulate the air compressor from the recombiner system.

An explosion in the Offgas Recombiner System could cause the rupture of the recombiner's H_2 analyzer equipment. The consequences of a radiological release due to such rupture has been analyzed and found to be within acceptable limits.

Table 9.3-1 Air Ejector Off-gas Composition

	cfm at 130°F, 1atm
Hydrogen	60[1,2]
Oxygen	30[1]
Air (assumed condenser leakage)	2-5
Water Vapor	22-25
Fission and activation gases	Negligible
Total	114-120

NOTES: 1. Hydrogen and oxygen composition values were derived by multiplying the expected radiolytic gas production rate value of 0.045 cfm/MWt by the Extended Power Uprate maximum power level of 2004 MWt and the respective fractions of hydrogen and oxygen gas contained in water. The design basis radiolytic gas production rate is 0.067 cfm/MWt (References 14 and 18).

2. Increases during HWC System operation by amount nominally equal to hydrogen injection rate.

9.4 Solid Radwaste System

9.4.1 Design Basis

The solid radwaste system is designed to process, package, store, monitor, and provide shielded storage facilities for solid wastes to allow for radioactive decay and/or temporary storage prior to shipment from the plant for off-site disposal. The solid radioactive wastes are shipped off-site in vehicles equipped with adequate shielding to comply with Department of Transportation (DOT) regulations.

The design bases of the solid radwaste system are to:

a. Minimize radiation exposure to plant operating personnel.

b. Prevent spillage and contamination spread of radioactive materials.

c. Provide safe and reliable means for handling the solid wastes.

9.4.2 Description

9.4.2.1 Sources

The following are radioactive solids wastes generated in the operation of the plant:

a. Process Wastes - filter sludge and spent resins from the liquid processing systems.

b. Reactor System Wastes - spent control rod blades, temporary control curtains, fuel channels, and in-core ion chambers.

c. Maintenance Wastes - contaminated clothing, tools, rags and small pieces of equipment which cannot be economically decontaminated.

d. Operating Wastes - laundry cartridge filters, paper, rags, off-gas filters, and ventilation filters.

e. Miscellaneous Wastes - solidified chemical and liquid wastes.

Major flow paths and equipment for the radwaste solids handling system and waste solidification system are shown in Drawings NH-36047 and NH-36047-2, Section 15, respectively.

9.4.2.2 Processing and Handling

9.4.2.2.1 Chemical, Liquid and Process Wastes

Chemical and Liquid Wastes

Chemical and liquid wastes which require solidification are added to containers and solidified using special procedures as necessary. The radiation level of these wastes is low enough to allow this operation to be done by contact handling in a well-ventilated area. After solidification, the wastes are sealed and placed in temporary storage for shipment along with the maintenance and operating wastes. Chemical and liquid wastes may also be used to maintain required moisture levels during the solidification of process wastes.

Process Wastes

The process wastes are the largest volume of solid wastes processed in the solid radwaste system. The process wastes consist of the filter sludge from the reactor clean-up, fuel pool, and condensate filter/demineralizer systems and the radwaste filters and the spent resins from the radwaste mixed-bed demineralizer. The filter sludge and spent resins are backwashed into their respective receiving tank, dewatered and processed in the waste solidification system. An initial dewatering step is accomplished on the condensate and clean-up sludge by sedimentation and decantation. This serves the two-fold purpose of providing decay storage of the sludge in the tanks and reduces the processing load on the centrifuges.

Two centrifuges and two drum-filling systems (“A” system and “B” system) are provided to allow for segregation of the high level and low level sludge in the drumming step. The “B” system includes the waste solidification system (see Section 9.4.2.2.2), and may also be used for processing sludge for solidification by an outside contractor. The “A” system is a standby system which may be used for de-watered sludge processing. The sludge are dewatered in the centrifuge and the moist solids fall directly into surge hoppers located beneath the centrifuges. The hoppers provide surge since only one 55 gal drum may be filled at a time. Empty drums are loaded onto a remotely operated conveyor system which carries one drum at a time to a selected filling station beneath a hopper. The drum is filled under visual observation and then transferred to a pre-selected storage conveyor as dictated by the radiation level of the drum contents. All operations in the handling of the process wastes, from the backwashing to the final load-out of the drums, are accomplished by remote means.

9.4.2.2.2 Waste Solidification System

The waste solidification system is part of the “B” hopper system and renders radioactive wastes into a homogeneous, solid product suitable for transportation and subsequent disposal. The system mixes moist radioactive wastes with dry portland cement and feeds the resulting mixture into standard 55-gallon drums. The system is remotely operated and fully automated to minimize radiation exposure to personnel and to minimize operator error. The system can also be manually operated. The control panel is located at elevation 935 in the radwaste conveyor operating galley.

The solidification system was manufactured by Atcor, Inc., and consists of three basic subsystems:

a. waste conditioning and metering,

b. dry cement storage and metering, and

c. product mixing and package filling.

The waste conditioning sub-system receives processed solid wastes from the centrifuge, adjusts the moisture content to assure a proper cement mixture, and meters the conditioned waste into the mixer/feeder unit for subsequent mixing with cement.

Dry cement is stored in a 75 cubic foot capacity bin from which it is fed to the mixer/feeder through a combination of two screw feeders. The combined waste product and dry cement are introduced to the mixer/feeder simultaneously. A screw flight and paddle arrangement within the mixer/feeder insures a thorough mixing action. After completion of the filling operation, the full drum is conveyed to the capping station and storage areas.

A solid radwaste drum decontamination station provides remote control washing of the drums after they are filled and capped so that radiation exposure to personnel during this operation can be eliminated. The solid radwaste drums are placed in temporary storage prior to shipment for disposal.

Portable equipment owned and operated by independent contractors has also been used at Monticello for the solidification of process and miscellaneous wastes. This equipment and its associated concrete solidification process is similar to that provided by Monticello in-plant equipment.

The interface connections from the plants process waste system to the independent contractors portable equipment consists of a waste hose from the radwaste system, a vent hose from the liner back to the radwaste building ventilation system, electrical, air and demineralized water for the process waste processing unit. A sketch of the system is shown in Figure 9.4-3. Alarms are provided to annunciate high sludge levels in the liner. Transfer hoses are leak-tested prior to use. The waste in the phase separators tanks is normally transferred to the processing liner in basically the same way as the ATCOR system. Waste is centrifuged to “B” hopper where liquid has been added, mixed and then pumped to the vendor’s liner. Solidification is then conducted in a liner contained inside a shield cask or a shipping cask.

The solidification system mixes separate feeds of radioactive wastes and dry Portland cement in the liner. Normal operation of the system includes transfer of dry cement from a cement storage bin or from a bulk cement truck into the waste container. The waste from the plant’s waste storage tank is pumped to the shipping container by the plant’s permanently installed pumps. After the correct metered amount has been transferred, the waste flow is stopped and the correct amount of additives is added to pre-condition the waste. Dry Portland cement is transferred to the liner with a pneumatic transfer system. Mixing of the waste with the additives and the cement is accomplished by a disposable agitator pre-mounted in the liner which is driven by a hydraulic motor. Instrumentation is available to monitor the level of waste and cement level in the liner and the temperature of the cement billet during solidification.

Prior to transferring the waste to the liner, a Process Control Program (PCP) is used to verify solidification and to determine the quantities of additives and cement required. In this procedure a sample of the waste is obtained and processed to ensure that a solidified product can be obtained.

The primary method for handling solid radwaste is to bypass the B-centrifuge and B-hopper via the bypass line. This method sends solid radwaste directly from the radwaste holding tanks to the waste dewatering system, the RDS-1000.

The RDS-1000, manufactured by Chem-Nuclear Systems, Incorporated (CNSI), is a waste processing system installed for use at the Monticello plant. This system is a waste dewatering system with connections to the plant systems via the bypass line as noted above. A water return line to the radwaste drains is used to route extracted liquid back to the plant for processing. A topical report for the waste dewatering system has been reviewed by the NRC. The NRC found this system and its operating procedures to be an acceptable process for the preparation of waste for burial to meet the requirements of 10CFR61 (Reference 13). Major flow paths and equipment are shown in Drawing NH-119062, USAR Section 15.

9.4.2.3 Storage and Shipment for Disposal

A radwaste storage building (32′ x 80.5′) is provided for the solid radwaste truck loading area. This sheet metal building is provided with shield walls, floor drains, heating and fire protection systems. An overhead crane (capacity of 10 tons) is located in the building. The building is designed to enclose the radwaste shipping truck and to facilitate loading of the truck.

In addition to normal pendant controls, the overhead crane is radio controlled to enable the crane operator to select the best location to handle the waste and minimize his radiation exposure.

A radwaste shipping building is erected along the west side of the radwaste storage building. The building is a metal, steel framed building 25′-6″ x 127′.

The waste is packaged, stored and shipped in accordance with applicable DOT and NRC regulations. All activities are performed in accordance with a Process Control Program and are under the control of the plant staff.

9.4.2.3.1 Hardware and Other Wastes Not Requiring Solidification

Reactor mechanical wastes are stored for decay in the spent fuel storage pool and subsequently packaged in suitable approved shipping containers for shipment to an approved off-site disposal site. Maintenance and operating wastes are collected in containers located in appropriate zones in the plant, as dictated by volume and degree of contaminated wastes. The activity level of these wastes is generally low enough to permit contact handling. After the containers are filled at their respective collection point, the waste is transferred to the Radwaste Building where it is either prepared for shipping off-site to a processing and/or disposal facility or compacted into drums by a hydraulic press bailing machine. If compacted, ventilation is provided to maintain control of contamination spread during the compacting and packaging operations. The compacted waste drums are then sealed and placed in temporary storage along with the non-compressible wastes until final shipment to the burial site.

Other equipment which is too large to be handled in this way requires special procedures. Since the need for handling of large equipment is quite infrequent, providing storage facilities in advance is not justified. Handling of such equipment depends upon the radiation level, transportation facilities, and available storage sites. Procedures for decontamination, shielding, shipment, monitoring and storage of such items are developed as necessary.

9.4.2.4 Process Control Program

All activities are performed in accordance with a Process Control Program (PCP) and are under the control of the plant staff. Waste classification and waste form requirements of 10CFR Part 61 are ensured by the PCP. The Process Control Program was approved by the NRC in a letter dated February 8, 1983 (Reference 10). A description of changes to the PCP are submitted to the NRC with the Annual Radioactive Effluent Release Report.

9.4.3 Performance Analysis

Radiation exposure to plant operating personnel is minimized by shielding around the sludge collection tanks, centrifuges, drum filling systems, drum storage conveyors and temporary storage areas. Methods are provided for gross decontamination of equipment which may require periodic maintenance.

The majority of the radioactivity released from the reactor process system is collected and retained on the filter sludge and spent resins. Following the dewatering step, the wastes are rendered relatively immobile and the potential for release is reduced. During solidification, radiation level detection is provided to preclude exceeding the curie limits of the transportation container. System interlocks are provided to prevent spills and improper operation. A loss of power de-energizes all equipment. Building ventilation is provided to reduce contamination spread with the building ventilation air being filtered through high-efficiency filters prior to discharge via the monitored Reactor Building vents. The release of activity from the solid radwaste system to the environs is extremely low.

9.5 References

1. Deleted.

2. Deleted.

3. NSP (D M Musolf) letter to the NRC, “License Amendment Request, Off-gas System Trip”, dated May 12, 1986.

4. NRC (J A Zwolinski) letter to NSP (D M Musolf), Technical Specification Change for Off-Gas System Trip (TAC No. 61659), dated December 1, 1986.

5. NRC Regulatory Guide 1.143, Revision 1, “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed In Light-Water-Cooled Nuclear Power Plants”, October 1979.

6. American Standards Association Code for Pressure Piping - Power Piping, ASA B31.1 - 1955 Edition.

7. USA Standard Code for Pressure Piping, USAS B31.1.0 - 1967, “Power Piping”.

8. ASME Boiler & Pressure Vessel Code, Section III, Class 2 and Class 3, 1971 Edition.

9. American National Standard Code for Pressure Piping, ANSI B31.1 “Power Piping”, 1977 Edition with all Addenda through Winter 1978 Addendum.

10. NRC (D B Vassallo) letter to NSP (D M Musolf), “Process Control Program”, dated February 8, 1983.

11. Deleted.

12. Deleted.

13. NRC (A C Thadani) letter to Chem-Nuclear Systems (W House), dated October 27, 1988, containing NRC Evaluation of Topical Report RDS-25506-01-P/NP “RDS-1000 Radioactive Waste Dewatering System”.

14. GE Hitachi Report NEDC-33322P, Revision 3, "Safety Analysis Report for Monticello Constant Pressure Power Uprate," October 2008.

15. NSPM letter L-MT-13-020 (M A Schimmel) to NRC, "Monticello Extended Power Uprate (EPU): Second Supplement for GAP Analysis Updates (TAC MD9990)," dated February 27, 2013.

16. GE Hitachi EPU Project Task Report NE-GE-0000-0061-9121-TR-R0, Revision 0, "Task T0807: Coolant Radiation Sources," December 2007 (Monticello calculation number 11-249).

17. NRC (T A Beltz) letter to NSPM (K D Fili), "Monticello Nuclear Generating Plant - Issuance of Amendment No. 176 to Renewed Facility Operating License Regarding Extended Power Uprate (TAC No. MD9990)," dated December 9, 2013.

18. GE Hitachi EPU Project Task Report GE-NE-0000-0062-9182-TR-R1, Revision 0, "Task T0801: Gaseous Waste Management," June 2007 (Monticello calculation number 11-244).

19. GE Hitachi Report NEDC-33435P, Revision 1, "Safety Analysis Report Monticello Maximum Extended Load Line Limit Analysis Plus," December 2009.

FIGURES

Figure 9.4-3 Vendor Portable Cement Solidification System