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11.0 WASTE DISPOSAL SYSTEMS AND RADIATION PROTECTION

Liquid, gaseous, and solid waste disposal facilities are designed so that discharge of effluents and off-site shipments are in accordance with applicable governmental regulations. Measures provided for the purpose of keeping releases of radioactive materials to unrestricted areas during normal reactor operations, including expected operational occurrences, as low as reasonably achievable are presented in Section 11.1, Section 11.2, and Section 11.3 to this document.

The waste disposal system collects and processes all potentially radioactive reactor plant wastes for disposal within limitations established by applicable governmental regulations. The waste disposal system includes the Waste Liquid (WL), Waste Gas (WG), Waste Solid (WS) Systems and the blowdown evaporator of the Blowdown Evaporator (BE) System. The waste disposal system outside containment is common to both units. Liquid and gaseous wastes are sampled and analyzed to determine the quantity of radioactivity, with an isotopic breakdown, if necessary. Depending on the results of the analysis, these wastes are processed further as required.

Liquid and gaseous wastes are released under controlled conditions. Radiation monitors are provided to maintain surveillance over the release operation, and a permanent record of activity releases is provided by radiochemical analysis of known quantities of waste. The system is capable of processing all wastes generated during continuous operation of the primary system assuming that fission products escape to the reactor coolant by diffusion through defects in the cladding of l% of the fuel rods.

The system is primarily controlled from a central panel in the auxiliary building. However, some equipment is provided with local control panels. Malfunction of the system is alarmed in the auxiliary building, and annunciated in the control room. All system equipment is located in or near the auxiliary building, except for the reactor coolant drain tanks which are located in the reactor containments, and the blowdown evaporator which is located in a separate building in the Unit 2 facade.

The waste disposal system obtains cooling water from the Unit 2 Component Cooling System. This cooling supply is automatically isolated by a Unit 2 Containment Isolation. Loss of the cooling supply will cause an automatic shutdown of the waste disposal system equipment that could be damaged by a loss of cooling.

11.0.1 REFERENCE

1. FPL Energy Point Beach Letter to NRC, NRC 2009-0030, “License Amendment Request 261 Extended Power Uprate,” dated April 7, 2009.

Table 11.0-1 WASTE DISPOSAL QUANTITIES

Annual liquid discharge	
Volume[1]	124.1E06 gal
Activity	See Table 11.1-3 for estimate
Annual gaseous discharge	
Activity	See Table 11.2-2 for estimate
Annual solids prepared for burial shipment	
Volume, not compacted[1]	12,337 ft^3
Activity[1]	92 curies

(1.) Based on the average annual values for both units during the 2002-2006 time period (Reference 1).

Table 11.0-2 WASTE DISPOSAL SYSTEM COMPONENT SUMMARY DATA
(Also See Table 11.1-1 and Table 11.2-1)

Tanks	Quantity	Type	Volume	Design Pressure	Design Temp	Material(1)
Reactor Coolant Drain (per unit)	1	Horiz	350 gal	25 psig	267	ss
Laundry & Hot Shower	1*	Vert	600 gal	Atm	180	ss
Chemical Drain	1*	Vert	600 gal	Atm	180	ss
Sump Tank	1*	Vert	600 gal	Atm	180	ss
Waste Holdup	1*	Horiz	21,444 gal	5 psig	200	ss
Waste Condensate	2*	Vert	1000 gal	Atm	180	ss
Reagent Tank	1*	Vert	6 gal	150 psig	250	ss

Pumps	Quantity	Type	Flow gpm	Head ft	Design Pressure	Design Temp	Material(1)
Reactor Coolant Drain (A) (per unit)	1	Horiz cent canned	50	175	150	267	ss
Reactor Coolant Drain (B) (per unit)	1	Horiz cent canned	150	175	150	267	ss
Chemical Drain	1*	Horiz cent(2)	20	100	150	150	ss
Laundry	1*	Horiz cent(2)	20	100	150	150	ss
Sump Tank	2*	Horiz cent(2)	20	100	150	150	ss
Waste Evaporator Feed	1*	Horiz cent(2)	20	100	150	150	ss
Waste Condensate	2*	Horiz cent(2)	20	100	150	150	ss

(1) Material contacting fluid
(2) Mechanical seal provided
* Shared by Units 1 and 2

11.1 LIQUID WASTE MANAGEMENT SYSTEM (WL)

The WL System collects, processes, and prepares for disposal potentially radioactive liquid wastes produced as a result of reactor operation.

11.1.1 DESIGN BASIS

The facility design shall include those means necessary to maintain control over the plant radioactive liquid effluents. Appropriate holdup capacity shall be provided for retention of liquid 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 on the basis of 10 CFR 20 requirements, for both normal operations and for any transient situation that might reasonably be anticipated to occur (GDC 70). A controlled release of liquid waste from the waste disposal system requires that at least two valves be manually opened, of which one of these valves is normally locked shut. In addition, a discharge control valve is provided which is designed to trip shut on an effluent high radioactivity signal from the discharge radiation monitor, thus preventing a release in excess of calculated amounts.

Radioactive fluids entering the waste disposal system are processed or collected in 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. Liquid wastes are processed as required and then released under controlled conditions. The system design and operation are 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.

11.1.2 SYSTEM DESIGN AND OPERATION

During normal plant operation, the waste disposal system processes liquids from the following sources:

1. Equipment drains, vents, and leaks;
2. Chemical laboratory drains;
3. Radioactive laundry and hot shower drains;
4. Decontamination area drains;
5. Chemical and Volume Control System (CVCS);
6. Sampling system drains and local sample sinks;
7. Normal letdown;
8. Steam generator blowdown (if required by radioactivity content);
9. Floor drains from the controlled areas of the plant; and
10. Liquids used to transfer solid radwaste.
11. Steam Generator Storage Facility sump (if required by radioactivity content)
12. Warehouse 7 sump (if required by radioactivity content)

The system also collects and transfers liquids from the following sources directly to the CVCS, to the -19' 3" auxiliary building sump, or back to the refueling water storage tank (depending on fluid content) for processing:

1. Pressurizer relief tank;
2. Reactor coolant pump secondary seals;
3. Excess letdown (during startup);
4. Accumulators;
5. Valve and reactor vessel flange leakoffs; and
6. Refueling canal drains.

These liquids flow to the reactor coolant drain tank and are discharged either directly to the CVCS holdup tanks or to the -19' 3" auxiliary building sump by either of the two reactor coolant drain tank pumps which are operated to control level in the tank. These pumps also may be aligned to return water from the refueling canal and lower cavity back to the refueling water storage tank. There is one reactor coolant drain tank inside each containment with the two reactor coolant drain tank pumps located outside each containment.

Where possible, other waste liquids drain to the waste holdup tank by gravity flow. Other waste liquids drain to the sump tank and are discharged to the waste holdup tank by pumps operated to control level in the tank.

Laundry and hot shower waste is pumped from the laundry and hot shower tank to the waste holdup tank via the laundry pump for processing with other waste liquids. Facilities are provided for discharging low-level waste from the waste holdup tank.

Liquids requiring cleanup before release are processed in batches by a filtration/demineralization system. The processed liquid is routed to one of the two waste distillate tanks. When one tank is filled, it is isolated and sampled for analysis while the second tank is in service. If analysis confirms that the activity level is suitable for discharge, the processed liquid is pumped through a flow meter and a radiation monitor to the service water discharge header. Exhausted filtration and demineralization media from this system is dewatered and packaged for shipping.

Liquids requiring cleanup before release can also be processed in batches by the blowdown evaporator. The concentrated bottoms are periodically pumped to the Primary Auxiliary Building (PAB) Truck Bay for shipment to vendor for processing. The condensate is routed to one of two waste distillate tanks. When one tank is filled, it is isolated and sampled for analysis while the second tank is in service. If analysis confirms the activity level is suitable for discharge, the condensate is pumped through a flow meter and a radiation monitor to the service water discharge header.

All routine liquid radioactive releases are made from waste disposal system distillate tanks or from CVCS monitor tanks. Prior to release, samples of the tank contents are taken and are analyzed for radioactivity by chemistry personnel. Results of analysis, waste liquid volume, dilution flow available, discharge rate, and total activities are recorded on a waste discharge permit. Administrative controls require comparison of analysis results with allowable limits by chemistry personnel and an authorizing signature of an operations group supervisor prior to initiation of waste liquid release. Although the radiochemical analysis forms the basis for recording activity releases, the radiation monitoring provides surveillance over the operation by closing the discharge valve if the liquid activity level exceeds a preset value.

Two blowdown vent condensers, one for each steam generator blowdown tank, condense the steam which would otherwise leave the tank vents. The condensers operate to maintain a pressure of one atmosphere in the tank and to return the condensate from flashed steam back to the tank. The vent from these condensers is piped to the plant vent header. The common 35 gpm Blowdown Evaporator (BE) may be used to concentrate blowdown or liquid wastes from either or both Unit 1 and Unit 2. Feed to the evaporator is collected in a surge tank and is pumped to the BE. BE internals and reflux to the evaporator tower reduce the carryover of boron solids or volatiles in the distillate. Conductivity control permits continuous monitoring of the cooled distillate effluent. Because liquid wastes can be processed by the filtration/demineralization system and significant steam generator tube leaks are not often present, the BE is normally not operating.

The performance of the filtration and demineralization media in the filtration/demineralization system is monitored through periodic sampling of the process effluent stream. Filtration and demineralization media are changed out when required, based on either the level of contaminants in the effluent stream or on a maximum activity level consistent with ALARA exposure while processing spent media. Exhausted filtration and demineralization media is dewatered and packaged for shipment off-site.

Steam generator chemistry treatment is all-volatile chemistry (AVT). Heat recovery exchangers for the steam generator blowdown allows for heat recovery, and connection to waste condensate demineralizers allow for processing of the blowdown from either unit by ion exchange if required prior to release.

When miscellaneous waste liquids are processed, batch control of both the processed liquid and bottoms is exercised. If necessary, processed liquid can be returned for reprocessing.

The following components are used in the Waste Liquid System. Additional component detail is provided in Table 11.0-2 and Table 11.1-1:

<u>Laundry and Hot Shower Tank</u> - One stainless steel tank collects liquid wastes originating from the laundry and hot shower. When the tank has been filled, its contents can be analyzed for gross beta-gamma activity. The tank contents are pumped to the waste holdup tank.

<u>Chemical Drain Tank</u> - The chemical drain tank is stainless steel and collects drainage from the chemistry laboratory. The tank contents are pumped to the waste holdup tank. A gross beta-gamma activity analysis can be obtained from the chemical drain tank to determine the radioactivity level.

(The above two tanks may be cross-connected for operational flexibility.)

<u>Reactor Coolant Drain Tanks</u> - The reactor coolant drain tanks are right circular cylinders with spherically dished heads. The tanks, which are all welded stainless steel, serve as a drain collecting point for the reactor coolant systems and other equipment located inside the reactor containments. The tank contents can be discharged to the CVCS holdup tanks, to the -19' 3" auxiliary building sump, or to the refueling water storage tanks.

Waste Holdup Tank - The waste holdup tank receives radioactive liquids from the chemical and volume control system, sump tank, chemical drain tank, -19' 3" auxiliary building sump, intermediate and operating floor drains, and laundry and hot shower tank. The tank is of welded stainless steel construction. The tank contents may be drained to the sump tank or pumped to the filtration/demineralization system.

Sump Tank and Pumps - The sump tank serves as a collecting point for waste discharged to the ground floor drain header. Two horizontal centrifugal sump pumps drain this tank to the waste holdup tank. All wetted parts of the pumps are stainless steel. The tank is all welded stainless steel.

Filtration/Demineralization System - The filtration/demineralization system is the primary means of processing radioactive liquid waste effluents. Through the use of deep bed filtration vessels and demineralization vessels, the filtration/demineralization system is designed to remove suspended particulate and ionic impurities from radioactive liquid waste. The system is common to both units. The major components include two stainless steel deep bed filtration vessels, four stainless steel demineralization vessels, a booster pump, stainless steel interconnecting piping and valves, and local instrumentation for process monitoring and control. The filtration/demineralization system has a maximum process capacity of 35 gpm.

Blowdown Evaporator - The blowdown evaporator system is designed to effectively remove radioactive particulate and gases from radioactive liquid waste and from steam generator blowdown water in the event of primary-to-secondary leakage. The Blowdown Evaporator is the alternate means of processing normal radioactive liquid waste effluents. The system is common to both units. Major components include one carbon steel blowdown surge tank; one stainless steel and Incoloy 825 blowdown evaporator; one carbon steel, steam heated reboiler; two carbon steel waste distillate tanks; two polishing demineralizers; and various interfacing heat exchangers, accumulators, and circulating pumps. The blowdown evaporator has a normal process capacity of 35 gpm.

Pumps - The wetted surfaces of all pumps are stainless steel or other materials of appropriate corrosion resistance. All pumps are either the canned motor type or mechanically sealed to minimize leakage.

Piping - Piping carrying liquid wastes is stainless steel. Piping connections are welded except where flanged connections are necessary to facilitate equipment maintenance.

Valves - All valves exposed to liquids 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. Isolation 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 free of gaseous activity are vented locally.

11.1.3 SYSTEM EVALUATION

Liquid wastes are generated primarily by plant maintenance and service operations, and consequently, the quantities and activity concentrations of influents to the system vary. A conservative estimate of the system's ability to limit dissolved and suspended activity released in the liquid phase is summarized in Table 11.1-3, Estimated Liquid Release by Isotope. The values in this table are for an annual release based on a thermal power level of 1811 MWt (1800 + 0.6% uncertainity.) (Reference 1) Refer to Appendices I.3 and I.9 for discussion of EPU impact on liquid and gaseous effluents. Activity concentrations in plant liquid effluents are monitored and controlled in accordance with the Offsite Does Calculation Manual (ODCM) and are reported to the NRC.

Steam generator blowdown is also released to the plant discharge system. Normally, blowdown does not require processing due to the high fuel integrity and steam generators which have very low leakage. However, if blowdown sampling showed elevated activities, the blowdown would be processed through the blowdown evaporator until the levels are low enough to release. In addition, many controls such as Technical Specification limits on leak rate and activity levels are in place that administratively and procedurally inhibit high blowdown activities.

Verification will be made to ensure that dilution flow sufficient to meet the requirements of 10 CFR 20 is available whenever radioactive liquid wastes are released to the plant discharge system. All liquid waste releases will be continuously monitored for gross activity during discharges to ensure that the activity limits specified in 10 CFR 20 for unrestricted areas are not exceeded. All radioactive liquid wastes will be sampled and analyzed prior to release to the plant discharge system.

Those secondary-side liquid wastes containing only tritium (for example, condenser hotwells) may be discharged without being continuously monitored if the volume of liquid to be released is a batch release and the amount of tritium has been isotopically quantified.

During release of liquid radioactive waste, the following conditions shall be met:

1. At least one condenser circulating water pump shall be in operation and the service water return header shall be lined up to the unit(s) whose circulating water pump is operating.
2. If the gross activity monitor in the discharge line is not operable or if the discharge is made via a pathway without an RMS monitor, the volume of liquid to be released is to be isotopically quantified pursuant to Radiological Effluent Control Manual (RECM) prior to release and periodically sampled during release.

11.1.4 REQUIRED PROCEDURES AND TESTS

The inservice testing requirements are described in the PBNP Inservice Testing Program and the IST Background Document.

11.1.5 ACCIDENTAL RELEASE-RECYCLE OR WASTE LIQUID

Accidents in the auxiliary building which would result in the release of radioactive liquids are those which may involve the rupture or leaking of system pipe lines or storage tanks. The largest

vessels are the three liquid hold up tanks, each sized to hold more than one reactor coolant liquid volume, which are used to process the normal recycle or waste fluids produced. The contents of one tank can be passed through the liquid processing train while another tank is being filled.

All liquid waste components except the reactor coolant drain tank and the blowdown evaporator are located in the auxiliary building and any leakage from the tank or piping will be collected in the building sump to be pumped back into the liquid waste system. Blowdown evaporator building drains will be directed to the liquid waste system via the sump tank. The building sump and basement volume are sufficient to hold the full volume of a liquid hold up tank without overflowing to areas outside the building. The full volume of either the volume control tank or the waste hold-up tank will be contained in the auxiliary building.

The holdup tanks are also equipped with safety pressure relief and designed to accept the established seismic forces at the site. Liquids in the chemical and volume control system flowing into and out of these tanks are controlled by manual valve operation and governed by prescribed administrative procedures.

The volume control tank design philosophy is similar in many respects to that applied for the holdup tanks. Level alarms, pressure relief valves and automatic tank isolation and valve control assure that a safe condition is maintained during system operation. Excess letdown flow is directed to the holdup tanks via the reactor coolant drain tank. The waste holdup tank is a horizontal tank which is continuously maintained at atmosphere pressure. Its vent is routed to the atmosphere through the auxiliary building exhaust ducts.

The potential hazard from these process or waste liquid releases is derived only from the volatilized components. The releases are described and their effects summarized in Section 11.2.5.

The evaluation of the credibility of the accidental release of radioactive fluids above the maximum permissible concentration from the waste disposal system discharge is based upon the following review of waste discharge operating procedure, monitoring function description, monitor failure mode and the consequences of a monitor failure.

The procedure for discharging liquid wastes is as follows:

1. A batch of waste is collected in a tank.

2. The tank is isolated.

3. The tank contents are recirculated to mix the liquid.

4. A sample is taken for radiochemical analysis.

5. Based on the analysis, the limiting discharge rate required to be in compliance with the PBNP Technical Specification upper discharge limit is calculated. If analysis indicates that release can be made within permissible limits, a discharge permit is completed indicating the quantity of activity to be released based on the liquid volume in the tank and its activity concentration. If release cannot be made within permissible limits, the waste is returned to the waste holdup tank for further processing.

6. To release the liquid, the last stop valve in the discharge line (which is normally locked shut) must be unlocked and opened; a second valve, which trips shut automatically on high radiation signal from the monitor, must be opened manually; and finally the recirculation valve must be closed.

7. Before the release, the operator verifies that the selected release path is isolated from all sources of potential discharge except the authorized source.

8. Soon after starting the discharge, the tank liquid levels are checked to ensure that discharge is occurring only from the approved tank and a calculation is performed to verify the discharge rate. The discharge rate is checked periodically during the discharge.

As the operating procedure indicates, the release of liquid waste is under administrative control. The monitor is provided to maintain surveillance over the release.

The monitor is provided with the following features:

1. A calibration source is provided to permit the operator to check the monitor before discharge by pressing a button in the control room to activate the circuitry.

2. If the monitor falls off scale at any time, an indicator visible to the operation in the control room lights.

3. If the power supply to the monitor fails a high radiation alarm is annunciated. The trip valve also closes.

4. The radiation trip valve fails closed, normally closed.

The administrative controls imposed on the operator combined with the safety features built into the equipment provide a high degree of assurance against accidental release of waste liquids.

Should a complete failure of any tank located in the auxiliary building occur, its content will remain in this building. Any subsequent discharge of radioactive liquid to the lake will be conducted under administrative controls and will not result in activity concentration into the lake in excess of the limits given in the Technical Specifications.

Dilution of off-site liquid releases to the lake are discussed in Section 2.5.

11.1.6 REFERENCES

1. Westinghouse Calculation Note, CN-CRA-99-15, Revision 1, September 30, 2009. (Confidential)

2. Westinghouse Report, WEP-98-077, “Wisconsin Electric Power Company Point Beach Unit 1 and 2 Chapter 9 and 11 – FSAR Updates,” December 8, 1998.

Table 11.1-1 COMPONENT DESIGN DATA FOR RADIOACTIVE LIQUID TREATMENT
(Also See Table 11.0-2)
Sheet 1 of 6

Booster Pumps

Number	2
Type	Centrifugal
Motor Horsepower	7.5
Seals	Mechanical
Capacity, gpm	36.5
Developed head at rated capacity, ft	135.6
Design Pressure, psig	150
Design Temperature, °F	300
Materials:	
Pump casing	Ductile Iron
Shaft	Carbon Steel
Impeller	Cast Iron

Blowdown Surge Tank

Number	1
Capacity, gal	500
Design Pressure, psig	50 and Full Vacuum
Design Temperature, °F	300
Material	Carbon Steel
Code	ASME VIII

Blowdown Evaporator

Number	1
Capacity, gpm	35
Design Pressure, psig	103 and Full Vacuum at 100°F
Design Temperature, °F	340
Material	Stainless Steel and Incoloy 825
Code	ASME VIII

Auxiliary Condensate Pump

Number	1
Type	Centrifugal
Motor Horsepower	100
Seals	Mechanical
Capacity, gpm	78
Developed Head, ft	822
Design Pressure, psig	600
Design Temperature, °F	307
Materials:	
Pump casing	Carbon Steel
Shaft	Carbon Steel
Impeller	Carpenter 20

Table 11.1-1 COMPONENT DESIGN DATA FOR RADIOACTIVE LIQUID TREATMENT
Sheet 2 of 6

Blowdown Evaporator Circulating Pump

Number	1
Type	Centrifugal
Motor Horsepower	50
Seals	Double Mechanical
Capacity, gpm	4,570
Developed head at rated capacity, ft	16
Design Pressure, psig	150
Design Temperature, °F	300
Materials:	
Pump casing	Carpenter 20
Shaft	Carbon Steel
Impeller	Carpenter 20

Blowdown Evaporator Reboiler

Number	1	
Design Duty, Btu/hr	20,697,000	
	Shell	Tube
Fluid	Steam	Evaporator Feed
Design Pressure, psig	150 and Full Vacuum	150 and Full Vacuum
Design Temperature, °F	375	375
Material	Carbon Steel	Incoloy 825
Design Code	ASME VIII	ASME VIII

Blowdown Evaporator Bottoms Pump

Number	1
Type	Centrifugal
Motor Horsepower	2
Seals	Double Mechanical
Capacity, gpm	10
Developed head at rated capacity, gpm	80.5
Design Pressure, psig	150
Design Temperature, °F	375
Materials:	
Pump casing	Carpenter 20
Shaft	Carbon Steel
Impeller	Carpenter 20

Table 11.1-1 COMPONENT DESIGN DATA FOR RADIOACTIVE LIQUID TREATMENT
Sheet 3 of 6

Blowdown Evaporator Bottoms Cooler Circulating Pump

Number	1
Type	Centrifugal
Motor Horsepower	2
Seals	Mechanical
Capacity, gpm	48.5
Developed head at rated Capacity, ft	49
Design Pressure, psig	200
Design Temperature, °F	250
Materials:	
Pump casing	Ductile Iron
Shaft	Carbon Steel
Impeller	Cast Iron

Blowdown Evaporator Distillate Pump

Number	1
Type	Centrifugal
Motor Horsepower	7.5
Seals	Mechanical
Capacity, gpm	40.7
Developed head at rated Capacity, ft	120
Design Pressure, psig	150
Design Temperature, °F	300
Materials:	
Pump casing	Stainless Steel
Shaft	Carbon Steel
Impeller	Stainless Steel

Blowdown Evaporator Distillate Cooler

Number	1
Design Duty, Btu/hr	2,100,000

	Shell	Tube
Fluid	Service Water	Distillate
Design Pressure, psig	150	150
Design Temperature, °F	200	300
Material	Carbon Steel	Stainless Steel
Design Code	ASME VIII	ASME VIII

Table 11.1-1 COMPONENT DESIGN DATA FOR RADIOACTIVE LIQUID TREATMENT
Sheet 4 of 6

Blowdown Evaporator Bottoms Cooler Preheater

Number	1
Fluid	Component Cooling Water
Design Pressure, psig	150
Design Temperature, °F	200
Material	Carbon Steel
Design Code	ASME VIII

Blowdown Evaporator Overhead Condenser

Number	1	
Design Duty, Btu/hr	18,200,000	
	Shell	Tube
Fluid	Distillate	Service Water
Design Pressure, psig	150	150
Design Temperature, °F	300	200
Material	Stainless Steel	Stainless Steel
Design Code	ASME VIII	ASME VIII

Blowdown Evaporator Distillate Accumulator

Number	1
Capacity, gal	500
Design Pressure, psig	50 and Full Vacuum
Design Temperature, °F	300
Material	Stainless Steel
Design Code	ASME VIII

Blowdown Vent Condensers

Number	2	
Design Duty, Btu/hr	970,000	
	Shell	Tube
Fluid	Saturated Steam	Service Water
Design Pressure, psig	150	150
Design Temperature, °F	300	200
Material	Carbon Steel	Carbon Steel
Design Code	ASME VIII	ASME VIII

Table 11.1-1 COMPONENT DESIGN DATA FOR RADIOACTIVE LIQUID TREATMENT
Sheet 5 of 6

Blowdown Evaporator Bottoms Cooler

Number	1	
Design Duty, Btu/hr	424,000	
	Shell	Tube
Fluid	Comp Cooling Water	12% Boric Acid
Design Pressure psig	150	150
Design Temperature, °F	200	300
Material	Carbon Steel	Incoloy 825
Design Code	ASME VIII	ASME VIII

Waste Distillate Pump

Number	1
Type	Centrifugal
Motor Horsepower	5
Seals	Mechanical
Capacity, gpm	75
Developed head at rated Capacity, ft	87.5
Design Pressure, psig	150
Design Temperature, °F	180
Materials:	
Pump casing	Stainless Steel
Shaft	Carbon Steel
Impeller	Stainless Steel

Blowdown Evaporator Sample Cooler

Number	1	
	Coolant Side	Process Side
Fluid	Comp Cooling Water	Evaporator Bottoms
Design Pressure, psig	225	100
Design Temperature, °F	200	300
Material	Cast Steel	Stainless Steel

Waste Distillate Tanks

Number	2
Capacity, gal	10,000
Design Pressure, psig	0.5
Design Temperature, °F	200
Material	Carbon Steel with Corrosion Resistant Lining
Design Code	AP1 650

Table 11.1-1 COMPONENT DESIGN DATA FOR RADIOACTIVE LIQUID TREATMENT
Sheet 6 of 6

<u>Condensate Receiver</u>

Number	1
Design Pressure, psig	150 & Full Vacuum
Design Temperature, °F	370
Material	Carbon Steel
Design Code	ASME VIII

<u>Deep Bed Filtration Vessel</u>

Number	2
Capacity, Gal	225
Design Pressure, psig	150
Design Temperature, °F	150
Material	Stainless Steel
Design Code	ASME Section VIII

<u>Demineralizer Vessel</u>

Number	4
Capacity, Gal	225
Design Pressure, psig	150
Design Temperature, °F	150
Material	Stainless Steel
Design Code	ASME Section VIII

<u>Filtration/Demineralizer Booster Pump</u>

Number	1
Type	Centrifugal
Motor Horsepower	5
Seal	Mechanical
Capacity, gpm	35
Developed Head, ft	275
Design Pressure, psig	150
Design Temperature, °F	150
Materials:	
Pump casing	Stainless steel
Shaft	Stainless steel
Impeller	Stainless steel

Table 11.1-2 ESTIMATED LIQUID DISCHARGE TO WASTE DISPOSAL

The information that was provided in this table is historical and can be found in FFDSAR Table 11.1-4.

Table 11.1-3 ESTIMATED LIQUID RELEASE BY ISOTOPE (TWO UNITS) (Reference 1)

ISOTOPE	Curies/year
Na-24	0.04766
Cr-51	0.0165
Mn-54	0.01034
Fe-55	0.00788
Fe-59	0.00172
Co-58	0.02774
Co-60	0.0035
Zn-65	0.00328
W-187	0.00362
Np-239	0.00518
Br-84	0.00062
Rb-88	0.04408
Sr-89	0.00082
Sr-90	0.00008
Y-90	0.00006
Sr-91	0.00068
Y-91m	0.00034
Y-91	0.00006
Y-93	0.00306
Zr-95	0.00234
Nb-95m	0.00002
Nb-95	0.00192
Mo-99	0.01632
Tc-99m	0.01524
Ru-103	0.0423
Rh-103m	0.04036
Ru-106	0.58396
Rh-106	0.56026
Ag-110m	0.00836
Ag-110	0.00104
Te-129m	0.00104
Te-129	0.0029
Te-131m	0.00252
Te-131	0.00064
I-131	0.02642
Te-132	0.00462
I-132	0.02836
I-133	0.04958
I-134	0.01988
Cs-134	0.83184
I-135	0.058
Cs-136	0.04858
Cs-137	1.11744
Ba-137m	1.02154
Ba-140	0.05726
La-140	0.08638
Ce-141	0.00082
Ce-143	0.00498
Pr-143	0.00086
Ce-144	0.02516
Pr-144	0.02414
Total release excluding tritium	4.862 Ci/yr
Tritium release	1300 Ci/yr

Table 11.1-4 ACTIVITY FROM STEAM GENERATOR BLOWDOWN WITHOUT AND WITH PROCESSING (Historical) [(4)]

Isotope	Blowdown Activity (1) µCi/cc	Activity in Circ Water Discharge Without Processing (2) µCi/cc	Activity in Circ Water Discharge With Processing (3) µCi/cc
Br-84	4.92E-05	4.77E-08	3.18E-11
I-131	9.25E-03	8.97E-06	5.98E-09
I-132	6.97E-03	6.76E-06	4.51E-09
I-133	1.43E-02	1.38E-05	9.20E-09
I-134	9.12E-04	8.85E-07	5.90E-10
I-135	6.63E-03	6.43E-06	4.29E-09
Rb-88	2.78E-03	2.70E-06	1.80E-09
Rb-89	1.30E-04	1.26E-07	8.40E-11
Sr-89	2.53E-05	2.45E-08	1.63E-11
Sr-90	1.31E-06	1.27E-09	8.47E-13
Sr-91	2.82E-05	2.74E-08	1.83E-11
Sr-92	4.12E-06	4.00E-09	2.67E-12
Y-90	2.23E-07	2.16E-10	1.44E-13
Y-91m	1.38E-05	1.34E-08	8.93E-12
Y-91	1.84E-06	1.78E-09	1.19E-12
Y-92	2.68E-06	2.60E-09	1.73E-12
Zr-95	2.13E-06	2.06E-09	1.37E-12
Nb-95	2.14E-06	2.07E-09	1.38E-12
Mo-99	2.67E-03	2.59E-06	1.73E-09
Tc-99m	2.47E-03	2.40E-06	1.60E-09
Te-132	1.07E-03	1.04E-06	6.93E-10
Te-134	3.86E-05	3.74E-08	2.49E-11
Cs-134	1.45E-02	1.14E-05	9.40E-09
Cs-136	1.52E-02	1.48E-05	9.87E-09
Cs-137	1.19E-02	1.15E-05	7.67E-09
Cs-138	1.29E-03	1.26E-06	8.40E-10
Ba-137m	1.11E-02	1.07E-05	7.13E-09
Ba-140	1.35E-05	1.31E-08	8.73E-12
La-140	4.81E-06	4.66E-09	3.11E-12
Ce-144	1.60E-06	1.55E-09	1.03E-12
Pr-144	1.60E-06	1.55E-09	1.03E-12
Cr-51	1.77E-05	1.71E-08	1.14E-11
Mn-54	1.31E-06	1.27E-09	8.47E-13
Mn-56	4.99E-05	4.84E-08	3.23E-11
Re-59	1.67E-06	1.62E-09	1.08E-12
Co-58	4.59E-05	4.45E-08	2.97E-11
Co-60	4.29E-06	4.16E-09	2.77E-12

1. Activity based on design parameters of 1650 MWt, 1% fuel defect, 1000 gpd leakage, and 100 gpm blowdown rate per unit. (Reference 2)

2. Apply a dilution factor for circulating water discharge of 9.70E-04, and no evaporator processing. (Reference 2)

3. Apply a decontamination factor of 1500 for the evaporator (Appendix I.2) and circ water dilution. (Reference 2)

4. This table was not updated for EPU. Table 11.1-3 does include expected blowdown system releases at EPU conditions but with different parameters from that used in this table.

Figure 11.1-1 UNITS 1 & 2 WASTE DISPOSAL SYSTEM PROCESS FLOW DIAGRAM (Sheet 1)

[redacted]

Figure 11.1-1 UNITS 1 & 2 WASTE DISPOSAL SYSTEM PROCESS FLOW DIAGRAM (Sheet 2)

[redacted]

Figure 11.1-2 UNITS 1 & 2 BLOWDOWN EVAPORATOR SYSTEM

[redacted]

Figure 11.1-3 UNITS 1 & 2 CONDENSATE WASTE POLISHING DEMINERALIZER

[redacted]

11.2 GASEOUS WASTE MANAGEMENT SYSTEMS (WG)

Various systems are provided for the processing of waste gases including: gas stripping and cryogenic separation* which remove radioactive gases and hydrogen from the primary coolant, condenser air ejector exhaust filtration and delay ductwork systems, which reduce radioactive gases in air ejector effluent in the event of primary-to-secondary leakage, and gas decay tanks which hold gases for an adequate period of time to allow decay. Cover gases are also considered part of the Waste Gas System and include the nitrogen blanketing system and parts of the hydrogen gas system.

11.2.1 DESIGN BASIS

The facility includes those means necessary to maintain control over the plant gaseous radioactive effluents. Appropriate holdup capacity shall be provided for retention of gaseous 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 on the basis of 10 CFR 20 requirements, for both normal operations and for any transient situation that might reasonably be anticipated to occur (GDC 70). Radioactive gases are processed to prevent their unmonitored release to the atmosphere. Gases are discharged intermittently at a controlled rate from the gas decay tanks through the monitored plant vent when required by plant inventory. A controlled release of gaseous waste from the waste disposal system requires that at least two valves be manually opened, one of which is normally locked shut. In addition, a discharge control valve is provided, which will trip shut on an effluent high radioactivity signal, thereby preventing an unanticipated release. Additional safety margin is provided by the use of ASME III, Class C materials and construction standards on significant components containing radioactive gases and USAS-B31.1 Section 1 piping and valves throughout the system.

11.2.2 SYSTEM DESIGN AND OPERATION

During plant operations, gaseous wastes will originate from:

1. Degassing reactor coolant discharged to the CVCS;
2. Displacement of cover gases as liquids accumulate in various tanks;
3. Miscellaneous equipment vents and relief valves; and
4. Sampling operations and gas analysis for hydrogen and oxygen in cover gases and gas decay tanks.

During normal operation, the waste disposal system also supplies nitrogen and hydrogen to primary plant components. The nitrogen gas system is divided into a low pressure and a high pressure side. The low pressure side (normally 40-70 psig) is supplied with nitrogen from the 3,000 gallon liquid nitrogen tank. Low pressure nitrogen can also be supplied from the high pressure side through pressure control valves. The low pressure nitrogen is used primarily in various tanks as a blanket above the liquid level to prevent air from entering the tanks and being absorbed into the liquid. The high pressure side is supplied from one or more clusters of 12 high pressure nitrogen bottles, or other pressurized nitrogen source, and is used to charge the safety injection system accumulators.

Most of the gas received by the waste disposal system during normal operation is cover gas displaced from the CVCS holdup tanks as they fill with liquid. Since this gas must be replaced when the tanks are emptied during processing, facilities are provided to return gas from the gas 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 minimal oxygen 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 in-leakage. Out-leakage from the system is minimized by using diaphragm valves, bellows seals, self-contained pressure regulators and soft-seated packless valves throughout the radioactive portions of the system.

Gases vented to the vent header flow to the waste gas compressor suction header. The waste gas compressors are operated only when conditions require their use. 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. When the tank in service becomes pressurized to approximately 95 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 so he may select a new backup tank. Pressure indicators are provided to aid the operator in selecting the backup tank. Gas held in the gas decay tanks can either be returned to the CVCS holdup tanks, or discharged to the atmosphere if it has decayed sufficiently for release. Generally, the last tank to receive gas will be the first tank emptied back to the holdup tanks which permits the maximum decay time before releasing gas to the environment. However, the header arrangement at the tank inlet gives the operator the option to fill, reuse, or discharge gas to the environment simultaneously without restriction by operation of the other tanks. During degassing of the reactor coolant prior to a cold shutdown, for example, it may be desirable to pump the gas purged from the volume control tank into a particular gas decay tank and isolate that tank for decay rather than reuse the gas in it. This is done by aligning the system to open the inlet valve to the desired tank and closing the outlet valve to the reuse header. Simultaneously, one of the other tanks can be opened to the reuse header if desired, while another is discharged to atmosphere.

Before a tank is discharged to the environment, it is sampled and analyzed to determine and record the radioactivity to be released, and then is discharged to the plant vent at a controlled rate through a radiation monitor. Results of analysis, waste gas volume, dilution flow available, discharge rate, and total activity are recorded on a waste discharge permit. Samples are taken manually by opening the appropriate sample isolation valve and permitting gas to flow to the gas analyzer and/or sampling hood where it can be collected in one of the sampling system gas sample vessels. After sampling, the isolation valve is closed until the tank contents are released. During release, a trip valve in the discharge line is closed automatically by a high radioactivity level indication in the plant vent.

During operation, gas samples can be drawn automatically from the gas decay tanks and automatically analyzed to determine their hydrogen and oxygen content. Manual sampling and analysis equipment is also available. The on-service gas decay tank is routinely sampled for oxygen in accordance with the Technical Specifications. The oxygen concentration is limited to avoid accumulation of explosive gas mixtures.

Separation and segregation of fission gases is accomplished by a high flow rate gas stripper for each unit and gas decay capabilities. Long lived fission gases may be removed by the cryogenic absorption system.* Following removal of fission gases, the remaining stripped gas is normally recycled to the reactor coolant systems. Should the plant gas inventory and requirements become unbalanced, this treated gas may be discharged under controlled conditions to the atmosphere following sampling and analysis. Decay of short-lived isotopes from a leaking steam generator to the condenser air ejector is accomplished by providing decay ductwork and an in-line filtration system for condenser air ejector exhaust gases prior to release. The effect of primary-to-secondary system leakage is minimized by continuous removal of coolant fission gases in the gas strippers and treatment of steam generator blowdown by use of vent condensers on the steam generator blowdown tanks.

Gaseous waste disposal system piping from the branch downstream of each reactor coolant filter through the strippers, charcoal decay tanks (CDT), and cryogenic separation system* to the stripped liquid return connections upstream of the 3-way valves for control of high level in the volume control tanks, the common supply and return line from the noble gas storage tank and the hydrogen recycle headers is considered Class I for seismic design purposes. The entire evaporator system and the condenser air ejection system are considered Class III for seismic design.

In order to reduce the reactor coolant fission gas inventory to its lowest equilibrium level and so that full letdown flows from each unit can be stripped during load follow operation, the gas strippers are sized to handle the maximum expected continuous letdown rates from each unit. The full letdown flow from each reactor plant, containing dissolved hydrogen and fission gases is directed from a point downstream of the reactor coolant filter to a gas stripper. Dissolved gases are separated from the liquid in the stripper, which is run continuously or intermittently depending on activity level in the primary coolant and up to a maximum flow of 90 gpm. Stripped liquid is pumped from the stripper back to the letdown line and is directed to either the volume control tank or to a CVCS holdup tank. Noncondensable hydrogen and fission gases from each stripper are pumped to a common decay system by a compressor. The gases may also be vented through a cryogenic separation system.*

The noncondensable gases are compressed, dehydrated, and passed through a system capable of removing by decay nearly all the Xenon-133. The decay system consists of a series of vertical tanks filled with charcoal (CDT) which causes a differential holdup of hydrogen and noble gases. This allows a decay of nearly all the Xenon-133 while allowing the hydrogen to pass quickly through the charcoal. The gases leaving the decay system are recycled to the volume control tank.

To reduce the condenser air ejector radioactivity releases which may be partially short-lived isotopes, decay ductwork is sized on the basis of 18 scfm of saturated air flow per unit, for two units, to decay all of the nuclides with half-lives less than about 15 minutes essentially to zero before release.

The following components are used in the Waste Gas System. Additional component detail is provided in Table 11.2-1:

Gas Decay Tanks - Four welded gas decay tanks are provided to contain compressed waste gases (hydrogen, nitrogen, and fission gases). After a period for radioactive decay, these gases may be released at a controlled rate to the atmosphere through the auxiliary building exhaust vent. All discharges to the atmosphere will be monitored.

Noble Gas Storage - One of the four carbon steel tanks located inside the gas decay tank cubicle can be used for noble gas storage from the cryogenic gas separation system.*

Waste Gas Compressors - Two compressors are provided for removal of gases from equipment that contains or can contain radioactive gases. These compressors are of the water-sealed centrifugal displacement type. The operation of the compressors can be controlled by the gas manifold pressure. Construction is primarily of carbon steel. A mechanical seal is provided to minimize leakage of seal water.

Letdown Gas Stripper - The letdown gas stripper system is designed to remove radioactive gases and hydrogen from the primary coolant normal letdown. The system consists of two trains which can be cross connected and have a common interface with the cryogenic gas separation system.* The major component of each gas stripper train is the gas stripper unit in which the process liquid is flashed to steam and then recondensed in an attached condenser. This effectively strips entrained gases from the primary coolant. The coolant is then returned to the letdown system and the stripped gases are directed to the cryogenic gas separation system.*

*Cryogenic Gas Separator - (The cryogenic separation system has never been used at PBNP but was installed to provide an additional means of removing radioactive krypton-85 gas. The required equipment is currently abandoned in place, however, the following functional description is provided for historical reference.)

The cryogenic gas separation system can be used to remove the radioactive krypton-85 from the process gases of the gas decay tanks and gas stripper system prior to exhausting to the atmosphere. This is accomplished by decay in decay and holdup tanks of short lived fission gases and by adsorption of long lived fission gases. The cryogenic gas separation system consists of two trains with common holdup and decay tanks. Major components of one train include one water separator and gas cooler; a series of three holdup and four gas decay tanks which allows the decay of the short lived xenon and krypton isotopes; a silver aluminum silicate adsorber which adsorbs the iodine isotopes which do not decay in the holdup and gas decay tanks; a catalytic recombiner which removes any oxygen and thus eliminates the potential explosion hazard in the charcoal of the cryosorber; a process gas dryer; a cryosorber unit which removes the long lived krypton-85 isotope by adsorption on activated charcoal; and a liquid nitrogen system which maintains the cryosorber at the required temperature to effectively remove the krypton-85 isotope.

Condenser Air Ejector Filtration - The filtration unit removes radioiodine and radioactive particulates which may be present in condenser air ejector offgas when significant primary to secondary leakage is present. The filtration unit is in line with the combined condenser air ejector vent line and consists of a moisture separator heater, high efficiency particulate air filter, and carbon adsorber bed. The moisture separator and heater are required to reduce relative humidity in order to keep this adsorbent dry, thereby preserving its function of radioiodine removal.

Nitrogen Manifold - High pressure nitrogen can be used as a backup supply to the low pressure system through a dual manifold and pressure control valves. The manifold is brass with brazed brass fittings. Nitrogen is supplied to the manifold from gas cylinders. The manifold and cylinders are located inside the west wall of the auxiliary building truck access area.

Hydrogen Manifold - Hydrogen is supplied to the volume control tanks and each generator from a central storage facility located outside the east wall of the turbine building. The facility includes a rack of six ASME vessels mounted horizontally and a pressure reducing station to maintain header gas pressure. Pressure controllers at each generator and the volume control tanks maintain required hydrogen pressure.

Gas Analyzer - A continuous gas analyzer is provided to monitor the concentrations of oxygen and hydrogen in the cover gas of tanks and vessels which might accumulate a hazardous mixture of the two gases. Upon indication of a high oxygen level, provisions are made to purge the equipment to the gaseous waste system with nitrogen gas.

11.2.3 SYSTEM EVALUATION

Gaseous wastes consist primarily of hydrogen stripped from coolant discharged to the CVCS holdup tanks during boron dilution, nitrogen and hydrogen gases purged from the CVCS volume control 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. Activity concentrations in plant effluents are monitored and controlled in accordance with the Offsite Dose Calculation Manual (ODCM) and reported to the NRC. Table 11.2-2 contains an estimate of annual gaseous activity release based on 1811 MWt power (Reference 2).
Table 11.2-3 details the failure analysis of the Waste Gas System components.

Condenser air ejector exhaust gases are filtered in order to prevent the release of radioactive isotopes to the atmosphere during periods when significant primary to secondary leakage exists. This filtration system consists of a moisture separator heater, HEPA, and charcoal filters. The removal of entrained water and reduction of relative humidity ensures that the charcoal bed will remain dry, thereby enabling effective removal of radioiodine.

During release of gaseous radioactive waste to the plant vent, the following conditions shall be met:

1. At least one PAB exhaust stack fan will be in operation.

2. The plant vent radioactivity monitor shall be operating.

The maximum allowable release rates of radioactive liquid and gaseous wastes are specified in the Technical Requirements Manual and the Radiological Effluent Control Manual (RECM).

11.2.4 REQUIRED PROCEDURES AND TESTS

The inservice testing requirements are described in the PBNP Inservice Testing Program and the IST Background Document.

11.2.5 ACCIDENTAL RELEASE-WASTE GAS

Gas Decay Tank Rupture - Causes and Assumptions

The gas decay tanks contain the gases vented from the reactor coolant system, the volume control tank, and the liquid holdup tanks. Sufficient volume is provided in each of four tanks to store the gases evolved during a reactor shutdown. The system is adequately sized to permit storage of these gases for 45 days prior to discharge. This period is selected as the maximum foreseeable holdup time because in this period the shorter-lived radioactive gaseous isotopes received by the waste system will have decayed to a level which is less significant than that of long-lived Kr-85.

The waste gas accident is defined as an unexpected and uncontrolled release to the atmosphere of the radioactive xenon and krypton fission gases that are stored in the waste gas storage system. Failure of a gas decay tank or associated piping could result in a release of this gaseous activity. This analysis shows that even with the worst expected conditions, the off-site doses following release of this gaseous activity would be very low.

The leakage of fission products through cladding defects can result in a buildup of radioactive gases in the reactor coolant. Based on experience with other operational, closed cycle, pressurized water reactors, the number of defective fuel elements and the gaseous coolant activity is expected to be low. The principal source of radioactive gases in the waste disposal system is the bleeding of effluents from the reactor coolant system.

Nonvolatile fission product concentrations are greatly reduced as the cooled coolant is passed through the purification demineralizers. The removal factor for iodine, for example, is at least 10. The decontamination factor for iodine between the liquid and vapor phases, for example, is expected to be on the order of 10,000. Based on the above analysis and operating experience at Yankee-Rowe and Saxton, activity stored in a gas decay tank consists of that from the noble gases released from the processed coolant and only negligible quantities of the less volatile isotopes.

As the components of the waste gas system are not subjected to any high pressures or stresses, are Class I seismic design, and are designed to the standards given in Table 11.2-1, a rupture or failure is highly unlikely. However, a rupture of a gas decay tank is analyzed to define the limit of the hazard that could result from any malfunction in the radioactive waste disposal system.

Gas Decay Tank Activity Release Characteristics

The activity in the gas decay tank (GDT) is taken to be the maximum amount that could accumulate from operation at the Technical Specification limit for reactor coolant system noble gas activity. The maximum activity concentration is obtained by assuming the noble gases, xenon and krypton, are accumulated with no release over a full core cycle of 18 months at 1810.8 MWt with a letdown flow of 120 gpm and no gas stripping. Table 11.2-4 lists the primary input parameters important to the source term development, (Reference 6).

Samples taken from gas storage tanks in pressurized water reactor plants in operation show no appreciable amount of iodine.

To define the maximum doses, the release is assumed to result from gross failure of any process system storage tank, here represented by a gas decay tank giving an instantaneous release of its volatile and gaseous contents to the atmosphere.

Volume Control Tank Rupture - Causes and Assumptions

The volume control tank contains fission gases and low concentrations of halogens which are normally a source of waste gas activity vented to a gas decay tank. The iodine concentrations and volatility are quite low at the temperature, pH and pressure of the fluid in the volume control tank. The same assumptions are detailed in the preceding subsection also apply to this tank.

As the volume control tank and associated piping are not subjected to any high pressures or stresses, failure is very unlikely. However, a rupture of the volume control tank is analyzed to define the limit of the exposure that could result from such an occurrence.

Volume Control Tank Activity Release Characteristics

The volume control tank (VCT) is assumed to fail, releasing the stored noble gas activity and a portion of the iodine in the tank instantaneously to the environment. In addition, it is assumed that the letdown flow to the VCT continues for 30 minutes before isolation would occur. All of the noble gas and 10 percent of the iodine activity in the letdown flow is released to the environment. The activity in the VCT is based on operation with cladding defects in 1% of the fuel elements at a core power level of 1810.8 MWt over a nominal 18-month fuel cycle. Table 11.2-5 lists the primary input parameters important to the VCT accident scenario.

The noble gas activity in the VCT is conservatively determined based on operation with a conservatively high letdown flow of 132 gpm (120 gpm + 10% uncertainty) and assuming no gas stripping of the letdown stream. It is further assumed the RCS activity is based on operation with no gas stripping such that the concentration is maximum. The iodine concentration in the RCS is assumed to be at the TS limit for equilibrium operation (i.e., equal to the limit for DE I-131). Credit is taken for the demineralizer in the letdown line reducing the coolant concentration by a factor of 10. Thus, the iodine concentration in the VCT liquid is 10% of the RCS activity, as is the concentration in the letdown flow that is released as a result of the accident.

It is conservatively assumed that all activity released to the environment is released instantaneously. This assumption is also applied to the activity releases associated with the thirty minutes of letdown flow (i.e., the activity in the thirty minutes of letdown flow is all released at time-zero).

Method of Analysis

In calculating offsite plume center-line exposure for both the GDT and VCT ruptures, it is assumed that the activity is discharged to the atmosphere at ground level and is dispersed as a Gaussian plume downwind taking into account building wake dilution. No credit is taken for the buoyant lift effect of the hydrogen present in the released gas. The Site boundary atmospheric dispersion factors (X/Q) are described in Table 14.3.5-2.

The whole body and thyroid doses are calculated using the dose conversion factors from Federal Guidance Reports 11 and 12 (Reference 3 and Reference 4).

Summary of Calculated Doses

The site boundary whole body doses are 0.08 rem and 0.1 rem due to the releases as described in the GDT accident scenario and the VCT accident scenario, respectively. The thyroid dose at the site boundary due to the release described in the VCT accident scenario is 0.04 rem. The whole body does meet the Branch Technical Position 11-5 (Reference 5) limit of 0.1 rem.

It is concluded that a rupture in the waste gas system or in the volume control tank would present no undue hazard to public health and safety.

Method of Analysis and Summary of Calculated Doses - Charcoal Decay Tank

An investigation was made of the off-site radiological doses resulting from a burst of both the charcoal-filled decay tank and the cryogenic absorber vessel, assuming the cryogenic separation system had been in use.

A rupture is assumed to occur in one of the three connected charcoal decay tanks or their associated piping resulting in the release of a portion of the activity stored on the charcoal in the tanks. The activity is assumed to be released instantaneously.

It is conservatively assumed that the RCS noble gas activity for both Unit 1 and 2 is based on operation at 1810.8 MWt with no gas stripping such that the RCS is at its maximum. It is then assumed that Units have gas stripping initiated combined with a conservatively high letdown flow rate of 132 gpm (120 gpm + 10% uncertainty). Table 11.2-6 lists the primary input important to the charcoal filled decay tank release. The stripped gases are directed to the shared charcoal decay tanks. In addition to the initial inventory of activity in the primary coolant, noble gas activity continues to enter the RCS from the fuel. This activity is also available to be stripped from the letdown flow and delivered to the charcoal decay tanks.

The whole body doses are calculated using the dose conversion factors from Federal Guidance Report 12 (Reference 4). The Site Boundary atmospheric dispersion factors (X/Q) are described in Table 14.3.5-2. The site boundary whole body dose is 0.07 rem.

Table 11.2-7 shows a summary of the calculated doses for GDT, VCT, and CDT ruptures at the site boundary (EAB), low population zone (LPZ), and control room (CR). The acceptance criteria for the EAB and LPZ doses are the 10 CFR Part 20 dose limits and NUREG-800 Standard Review Plan section 6.4 provides the appropriate accident-specific dose acceptance criteria for the control room (Reference 7).

The following information describes two accident scenarios involving the cryogenic absorber vessel. The cryogenic system was installed in the early 1970’s, however, was never used and is currently abandoned in place. The system description currently remains in the FSAR. Similarly, the accident analyses for the cryogenic absorber vessel is maintained for historical purposes and reflects power operations at 1518.5 MWt.

For the cryogenic vessel burst it is assumed that the cryogenic system has been in operation for a total of 180 days, at which time the noble gas inventory in the absorb vessel consists of 1,725 curies of Krypton-85 and 1,070 curies of Xenon-133, and the entire inventory is released

instantaneously. The site boundary whole body dose resulting from the above assumptions, and using the most conservative X/Q values shown in Figure 2.6-8, is less than 0.03 rem.

This discussion is provided for historical purposes. The cryogenic system was never used, and is not operational. Major portions of the system have been abandoned in place. Thus, the reference to a 40-year operating period would still bound a 60-year plant operating period (NRC SE dated 12/2005, NUREG-1839).

The noble gases absorbed in the cryogenic absorber vessel can be desorbed at the end of each 180 day cryogenic cycle and stored in one of the existing gas decay tanks. The resulting activity would, if accumulated over a 40-year period in this single gas decay tank, reach a maximum value of 50,000 curies Krypton-85. Xenon-133 would reach a maximum value of 2,100 curies. The whole body dose resulting from an instantaneous release of the gas decay tank contents would be 0.7 rem, which is less than that described previously for a single gas decay tank rupture.

11.2.6 REFERENCES

1. Letter PBW-WMP-416, Westinghouse to WE dated December 4, 1967.

2. Westinghouse Calculation Note, CN-CRA-99-15, WEP/WIS Annual Releases (GALE Code Analysis), Revision 1, September 30, 2009.

3. K.F. Eckerman et al, “Limiting Values of Radionuclide Intake and Air Concentration and dose Conversion Factors for Inhalation, Submersion, and Ingestion,” Federal Guidance Report No. 11, Environmental Protection Agency, September 1988.

4. K.F. Eckerman and J.C. Ryman, “External Exposure to Radionuclides in Air, Water, and Soil,” Federal Guidance Report No. 12, Environmental Protection Agency, September 1993.

5. Branch Technical Position 11-5, Revision 3, “Postulated Radioactive Releases due to a Waste Gas System Leak or Failure,” March 2007. (Contained in NUREG-0800.)

6. Westinghouse Calculation CN-REA-08-7, RCS, VCT, and GDT Sources for the Point Beach EPU, Revision 0, September 19, 2008.

7. NRC Safety Evaluation, “Point Beach Nuclear Plant (PBNP), Units 1 and 2 - Issuance of License Amendments Regarding Extended Power Uprate (TAC Nos. ME1044 and ME1045),” dated May 3, 2011.

8. Westinghouse Calculation, CN-CRA-08-45, Charcoal Delay Tank Doses for the Extended Power Uprate, Revision 0, December 17, 2008.

9. Westinghouse Calculation, CN-CRA-08-44, Volume Control Tank Rupture and Waste Gas Decay Tank Rupture Radiological Doses for the Extended Power Uprate, Revision 0, December 17, 2008.

Table 11.2-1 COMPONENT DESIGN DATA FOR RADIOACTIVE GAS TREATMENT
Sheet 1 of 8

Gas Decay Tanks

Number	4
Capacity, ft^3	525
Design Pressure, psig	150
Design Temperature, °F	150
Material	Carbon Steel
Design Code	ASME III-Class C

Waste Gas Compressor

Number	2
Type	Centrifugal, Liquid Ring
Motor Horsepower	25
Capacity, SCFM	40
Discharge Pressure at Capacity, psig	110
Design Pressure, psig	150
Design Temperature, °F	180
Materials	Carbon Steel

Air Ejector Iodine Filter

Number	1
Capacity, SCFM	40
Design Pressure, psig	150
Design Temperature, °F	125
Materials:	
Absorbent	Charcoal
Housing	Carbon Steel
Design Code	ANSI B31.1.0

Gas Stripper Recovery Heat Exchangers

Number	2	
Design Duty, Btu/hr	3,190,000	
	Shell	Tube
Fluid	Stripper Feed	Stripper Liquid Eff.
Design Pressure, psig	150	150
Design Temperature, °F	250	250
Material	Stainless Steel	Stainless Steel
Design Code	ASME III-Class 3	ASME III-Class 3

Table 11.2-1 COMPONENT DESIGN DATA FOR RADIOACTIVE GAS TREATMENT
Sheet 2 of 8

<u>Gas Stripper Preheaters</u>

Number	2	
Design Duty, Btu/hr	3,295,000	
	<u>Shell</u>	<u>Tube</u>
Fluid	Steam	Stripper Feed
Design Pressure, psig	Full vacuum & 150	150
Design Temperature, °F	375	375
Material	Carbon Steel	Stainless Steel
Design Code	ASME VIII	ASME III-Class 3

<u>Gas Stripper Vent Coolers</u>

Number	2	
Design Duty, Btu/hr	11,900	
	<u>Shell</u>	<u>Tube</u>
Fluid	Stripper Gas Eff.	Comp Cooling Water
Design Pressure, psig	Full vacuum & 150	150
Design Temperature, °F	300	300
Materials	Stainless Steel	Stainless Steel
Design Code	ASME III-Class 3	ASME VIII

<u>Gas Strippers</u>

Number	2
Capacity, gpm	90
Design Pressure, psig	103 & Full vacuum
Design Temperature, °F	340
Material	Stainless Steel
Design Code	ASME III-Class 3

<u>Gas Stripper Circulating Pumps</u>

Number	2
Type	Centrifugal
Motor Horsepower	20
Seals	Mechanical with Lip Seal
Capacity, gpm	87.5
Developed head at rated capacity, ft	236
Design Pressure, psig	150
Design Temperature, °F	300
Materials:	
Pump Casing	Stainless Steel
Shaft	Carbon Steel
Impeller	Stainless Steel

Table 11.2-1 COMPONENT DESIGN DATA FOR RADIOACTIVE GAS TREATMENT
Sheet 3 of 8

Regeneration Heater

Number	1
Design Duty, Btu/hr	900
Type	Tube enclosing heating element
Fluid	Nitrogen
Design Pressure, psig	250
Design Temperature, °F	750
Material	Stainless Steel

Gas Stripper Trim Coolers

Number	2	
Design Duty, Btu/hr	955,000	
	Shell	Tube
Fluid	Component Cooling Water	Stripper Liquid Eff.
Design Pressure, psig	150	150
Design Temperature, °F	200	200
Material	Carbon Steel	Stainless Steel
Design Code	ASME VIII	ASME III-Class 3

Gas Stripper Prefilters

Number	2
Retention Size, microns	2
Capacity, gpm	80
Design Pressure, psig	150
Design Temperature, °F	250
Materials: Housing	Stainless Steel
Design Code	ASME III-C

Gas Stripper Condensers

Number	2	
Design Duty, Btu/hr	1,700,000	
	Shell	Tube
Fluid	Component Cooling Water	Stripper Gas
Design Pressure, psig	150	Full vacuum & 150
Design Temperature, °F	200	200
Material	Carbon Steel	Stainless Steel
Design Code	ASME VIII	ASME III-Class 3

Table 11.2-1 COMPONENT DESIGN DATA FOR RADIOACTIVE GAS TREATMENT
Sheet 4 of 8

Chiller Pumps

Number	2
Type	Centrifugal
Motor Horsepower	1/8
Seals	Mechanical
Capacity, gpm	3
Design Pressure, psig	150
Design Temperature, °F	200
Material	Stainless Steel

Gas Subcoolers & Water Separators

Number	4	
Capacity, gpm	1.2	
	Shell	Tube
Fluid	Gas Effluent	Freon
Design Pressure, psig	200	200
Design Temperature, °F	150 & 35	150 & 35
Material	Stainless Steel	Stainless Steel
Design Code	ASME III-C	ASME III-C

Decay Tanks

Number	3
Capacity, ft^3	46
Design Pressure, psig	200
Design Temperature, °F	150
Material	Carbon Steel
Design Code	ASME III-Class C

Gas Afterfilters

Number	2
Retention size, microns	5
Capacity, scfm	1.2
Design Pressure, psig	200
Design Temperature, °F	150
Materials:	
Filter Element	Stainless Mesh
Housing	Stainless Steel
Design Code	ASME III-C

Table 11.2-1 COMPONENT DESIGN DATA FOR RADIOACTIVE GAS TREATMENT
Sheet 5 of 8

Gas Dryer

Number	2
Capacity, scfm	1.2
Absorbent Active Volume	4 cu ft
Design Pressure, psig	200
Design Temperature, °F	500
Materials:	
Absorbent	Silica Gel & Molecular Sieve
Housing	Stainless Steel
Design Code	ASME III-C

Cryogenic Precooler

Number	2
Design Duty, Btu/hr	250

	Shell (Stream #1)	Tube (Stream #2)	Tube (Stream #3)
Fluid	Gas Effluent In	Gas Effluent Out	Nitrogen
Design Pressure, psig	1,000	1,000	100
Design Temperature, °F	-320 & 500	-320 & 500	-320 & 500
Material	Stainless Steel	Stainless Steel	Stainless Steel
Design Code	ASME III-C	ASME III-C	ASME III-C

Cryogenic Absorber

Number	2
Capacity, scfm	1.2
Absorber Active Volume	0.25 cu ft
Design Pressure, psig	1,000
Design Temperature, °F	-320 & 400
Materials:	
Absorbent	Coconut Charcoal
Housing	Stainless Steel
Design Code	ASME III-C

De-Oxo Units

Number	2
Capacity, scfm	1.2
Catalyst Volume, cu ft	0.1
Design Pressure, psig	200
Design Temperature, °F	1100
Materials:	
Catalyst	Palladium Catalyst
Housing	Stainless Steel
Design Code	ASME III-C

Table 11.2-1 COMPONENT DESIGN DATA FOR RADIOACTIVE GAS TREATMENT
Sheet 6 of 8

Gas Prefilter

Number	2
Retention size, microns	5
Capacity, scfm	1.2
Design Pressure, psig	200
Design Temperature, °F	150
Materials:	
Housing	Stainless Steel

Cryogenic Gas Compressors

Number	2
Type	Diaphragm
Motor Horsepower	5
Capacity, scfm	1.2
Discharge Pressure at capacity, psig	150
Max. Design Pressure, psig	250
Materials:	
Diaphragm and parts contacting gas	Stainless Steel

De-Oxo Preheater

Number	2
Design Duty, Btu/hr	350
Type	Tubing coiled around heater element
Design Pressure, psig	200
Design Temperature, °F	1000
Material	Stainless Steel

Liquid Nitrogen Storage Tank

Number	1
Capacity, gal	3,000
Fluid	Liquid Nitrogen
Material	Stainless Steel
Design Code	ASME VIII

Table 11.2-1 COMPONENT DESIGN DATA FOR RADIOACTIVE GAS TREATMENT
Sheet 7 of 8

Liquid Nitrogen Surge Tank

Number	1	
Capacity, gal	50	
	Inner Dewar	Shell
Fluid	Liquid Nitrogen	Air
Design Pressure, psig	100	Full Vacuum and Atmospheric
Design Temperature, °F	-320 and 150	150
Material	Stainless Steel	Carbon Steel
Design Code	ASME VIII	ASME VIII

De-Oxo Aftercoolers

Number	2	
Design Duty, Btu/hr	750	
	Shell	Tube
Fluid	Comp. Cooling Water	Gas Effluent
Design Pressure, psig	200	200
Design Temperature, °F	1,100	1,100
Material	Stainless Steel	Stainless Steel
Design Code	ASME VIII	ANSI B31.1.0

Chiller Storage

Number	1	
Design Duty, Btu/hr	400	
	Shell	Tube
Fluid	Freon 11	Nitrogen
Design Pressure, psig	100	100
Design Temperature, °F	-200	-200
Material	Stainless Steel	Stainless Steel

Preabsorbers

Number	2
Capacity, scfm	1.2
Absorbent Volume, cu ft	0.2
Design Pressure, psig	200
Design Temperature, °F	500
Materials:	
Catalyst	Silver Treated Aluminum Silicate
Housing	Stainless Steel
Design Code	ASME III-C

Table 11.2-1 COMPONENT DESIGN DATA FOR RADIOACTIVE GAS TREATMENT
Sheet 8 of 8

Floor Equipment Drainage Sump Pumps

Number	2
Type	Vertical sump
Motor Horsepower	1
Capacity, gpm	30
Developed Head at Rated Capacity, ft	30
Materials:	
Pump Casing	Iron
Shaft	Carbon Steel
Impeller	Cast Iron

Table 11.2-2 ESTIMATED ANNUAL GASEOUS RELEASE BY ISOTOPE (TWO UNITS)

Isotope	Curies/yr
H-3	1.44E2
Ar-41	6.8E1
Kr-85m	4.4E1
Kr-85	7.4E2
Kr-87	8.0E0
Kr-88	2.0E1
Xe-131m	8.4E2
Xe-133m	4.0E0
Xe-133	8.0E2
Xe-135m	8.0E0
Xe-135	5.8E1
Xe-138	8.0E0
I-131	3.0E-1
I-133	9.4E-1
Cr-51	2.6E-4
Mn-54	1.3E-4
Co-57	1.6E-5
Co-58	1.3E-3
Co-60	3.2E-4
Fe-59	6.4E-5
Sr-89	4.6E-4
Sr-90	1.78E-4
Zr-95	2.0E-4
Nb-95	9.0E-5
Ru-103	3.8E-5
Ru-106	2.6E-6
Sb-125	1.92E-6
Cs-134	1.92E-4
Cs-136	7.4E-5
Cs-137	3.0E-4
Ba-140	8.0E-5
Ce-141	3.2E-5

Summary of Releases

Tritium release	1.44E2 Ci/yr
Total gaseous release	2.61E3 Ci/yr
Total iodine release	1.24E0 Ci/yr
Total particulate release	3.74E-3 Ci/yr

Table 11.2-3 GAS TREATMENT SYSTEM MALFUNCTION ANALYSIS

Components	Malfunction	Comments & Consequences
Entire gas treatment system modification	Fails to function	The cover gas, stripping and gas decay system is retained, so previously licensable performance is not affected by shutdown of modification equipment.
One gas stripper and associated exchangers, pumps and controls	Fails to function	Two stripper subsystems are provided, each at a capacity sufficient to process the normal letdown rate from both Unit 1 and Unit 2 reactors.
One gas compressor	Fails to function	Two units are provided; one in service, one in standby.
Decay tanks, surge tank and cryogenic absorber bed	Leak	These tanks are located in a tornado-proof Class I structure and are protected from overpressure by automatic controls and relief valves. Vent monitors and gas samples are used to detect leaks.
Cryogenic separation system	Fails to function	More than 90% of the fission gas removal is accomplished by components other than the cryogenic separation equipment. If the cryogenic portion were not operated, buildup of long-lived Krypton-85 in the reactor coolant would be very gradual. Alternatives to cryogenic processing and storage of fission product gases include controlled release methods and other processes described in this section.

Table 11.2-4 GAS DECAY TANK ACCIDENT ANALYSIS INPUT PARAMETERS

RCS Concentration Basis	
Power Level	1810.8 MWt
RCS Mass	1.147E8 gm
DE Xe-133	300 mCi/gm
Letdown Flow	120 gpm
Gas Stripping Rate	0 gpm
Releasable Activity from GDT	
Kr-85m	5.00E1 Ci
Kr-85	1.41E3 Ci
Kr-87	8.19E0 Ci
Kr-88	6.38E1 Ci
Xe-131m	2.07E2 Ci
Xe-133m	2.89E2 Ci
Xe-133	1.78E4 Ci
Xe-135	3.02E2 Ci
Xe-135m	1.33E1 Ci
Xe-138	9.66E-1 Ci

Table release values are from Reference 6 and modified by Reference 9. Reference 9 values were 42% of the Reference 6 values to account for the change RCS TS activity limit for DEX from 520 uCi/gm to 300 uCi/gm and corresponding change in the fuel defect level from 1% to 0.42%.

Table 11.2-5 VOLUME CONTROL TANK ACCIDENT ANALYSIS INPUT PARAMETERS

VCT Source Term Basis	
Power Level	1810.8 MWt
Fuel Cladding Defects	1%
Letdown Gas Stripping Rate	0 gpm
Noble Gas Basis (in the tank)	
Letdown Flow	132 gpm
Letdown Concentration Basis	
Letdown Flow	132 gpm
DE I-131	0.5 µCi/gm
Demineralizer DF for Iodine	10
DE Xe-133	520 µCi/gm
VCT Releasable Activities	
Kr-85m	96.7 Ci
Kr-85	1020 Ci
Kr-87	35.8 Ci
Kr-88	145 Ci
Xe-131m	195 Ci
Xe-133m	319 Ci
Xe-133	17800 Ci
Xe-135	503 Ci
Xe-135m	39.7 Ci
Xe-138	11.0 Ci
I-131	0.134 Ci
I-132	0.151 Ci
I-133	0.233 Ci
I-134	0.0355 Ci
I-135	0.134 Ci

Releasable Activity values are from Reference 9.

Table 11.2-6 CHARCOAL FILLED DELAY TANK ACCIDENT ANALYSIS INPUT PARAMETERS

CDT Source Term Basis	
Power Level	1810.8 MWt
DE Xe-133	300 µCi/gm
Letdown Gas Stripping Rate	132 gpm per Unit
CDT Releasable Activity	
Kr-85m	163 Ci
Kr-85	1806 Ci
Kr-87	46.3 Ci
Kr-88	231 Ci
Xe-131m	53.5 Ci
Xe-133m	84.0 Ci
Xe-133	4800 Ci
Xe-135	115 Ci
Xe-135m	0.86 Ci
Xe-138	0.64 Ci

Releasable Activity values are from Reference 8.

Table 11.2-7 CALCULATED DOSES FOR GDT, VCT, AND CDT RUPTURES

	Whole Body Dose (rem)	Thyroid Dose (rem)	Beta-Skin Dose (rem)
Gas Decay Tank (GDT) Rupture			
EAB	0.08	NA	NA
LPZ	0.02	NA	NA
CR	0.03	NA	1.2
Volume Control Tank (VCT) Rupture			
EAB	0.1	0.04	NA
LPZ	0.006	0.003	NA
CR	0.02	0.07	NA
Charcoal Decay Tank (CDT) Rupture			
EAB	0.07	NA	NA
LPZ	0.01	NA	NA
CR	0.02	NA	0.6
Acceptance Criteria			
EAB	0.1	1.5	NA
LPZ	0.1	1.5	NA
CR	5.0	30	30

Figure 11.2-1 UNITS 1 & 2 WASTE GAS DISPOSAL SYSTEM PROCESS FLOW DIAGRAM (Sheet 1)

[redacted]

Figure 11.2-1 UNITS 1 & 2 WASTE GAS DISPOSAL SYSTEM PROCESS FLOW DIAGRAM (Sheet 2)

[redacted]

Figure 11.2-1 UNITS 1 & 2 WASTE GAS DISPOSAL SYSTEM PROCESS FLOW DIAGRAM (Sheet 3)

[redacted]

Figure 11.2-2 UNITS 1 & 2 GAS STRIPPER SYSTEM

[redacted]

Figure 11.2-3 UNITS 1 & 2 CRYOGENIC GAS SEPARATION SYSTEM (Sheet 1)

[redacted]

Figure 11.2-3 UNITS 1 & 2 CRYOGENIC GAS SEPARATION SYSTEM (Sheet 2)

[redacted]

Figure 11.2-4 UNITS 1 & 2 CONDENSER AIR REMOVAL DECAY SYSTEM

[redacted]

11.3 SOLID WASTE MANAGEMENT SYSTEM (WS)

The Waste Solid System design and operation are directed toward minimizing releases of radioactive materials to unrestricted areas. The equipment is designed and operated to process solid radioactive wastes which result in a form which minimizes potential harm to personnel or the environment. Handling areas are appropriately monitored and safety features are incorporated to preclude releases in excess of the limits of 10 CFR 20.

11.3.1 DESIGN BASIS

The facility includes those means necessary to maintain control over the plant solid radioactive effluents. Appropriate holdup capacity shall be provided for retention of all solid effluents, particularly where unfavorable environmental conditions can be expected to affect the release of radioactive effluents to the environment. In all cases, the design for radioactivity control must be justified on the basis of 10 CFR 20 requirements, for both normal and transient operations. (GDC 70)

11.3.2 SYSTEM DESIGN AND OPERATION

Spent resins from the demineralizers, filter cartridges and the concentrates from the steam generator blowdown evaporator are packaged and stored on-site until shipment off-site for disposal. Miscellaneous materials such as paper, plastic, wood, and metal are collected and shipped offsite for vendor supplied volume reduction (i.e., incineration, supercompaction, metal melt, decon, etc.) followed by disposal.

Spent resins from CVCS and other system demineralizers are flushed to a shielded, lined stainless steel storage tank located in the auxiliary building basement. When the tank is full, the resin is dewatered and liquids from the dewatering operation are sent to the waste holdup tank. Following resin dewatering, the tank and its shield are transferred by the seismically qualified auxiliary building crane to the new fuel storage area where the resin is sluiced to a disposable cask liner. Spent filtration media and resin from the filtration/demineralization system is sluiced directly to a disposable cask liner in the truck access area. When a disposable liner is full, the liner is dewatered to meet disposal site or processor criteria. The disposable liner is then shipped offsite for processing or shipped offsite for disposal at a suitable burial site.

Dry active waste/radioactive materials may be stored in SeaLand containers, B-25/12 containers, or in other general design packages (49 CFR 173.410) in designated locations in the outside yard area of the RCA, in Warehouse 7 (Reference 2), and in the Steam Generator Storage Facility before shipment. Routine surveys and inspections are performed to verify container integrity.

11.3.3 SYSTEM EVALUATION

The quantity of solid radioactive waste shipped from PBNP is reported in the Annual Monitoring Report in accordance with the Radiological Effluent Control Manual (RECM). The typical solid radioactive waste volume shipped for offsite processing and disposal is given in Table 11.0-1.

11.3.4 REQUIRED PROCEDURES AND TESTS

The inservice testing requirements are described in the PBNP Inservice Testing Program and the IST Background Document.

11.3.5 REFERENCES

1. SE 99-007, Storage of Low-Level Dry Radioactive Waste, Green-Is-Clean Radioactive Waste, and Radioactive Material in PBNP RCA Yard Areas.

2. SCR 2012-0006, Storage of Radioactive Materials in Warehouse 7, January 12, 2012.

11.4 RADIATION PROTECTION PROGRAM

11.4.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURE IS AS LOW AS IS REASONABLY ACHIEVABLE (ALARA)

Policy Considerations

It is the policy of FPL Energy Point Beach to maintain occupational radiation exposure as low as is reasonably achievable (ALARA), consistent with plant construction, maintenance, and operational requirements, and within the applicable regulations. Regulatory Guide 8.8 is used as a basis for developing the ALARA and radiation protection programs.

FPL Energy Point Beach ALARA policy applies to total person-rem accumulated by personnel, as well as to individual exposures. FPL Energy Point Beach management provides the environment for this policy to function in a proper manner. Management's commitment to this policy is reflected in the design of the plant, the careful preparation of plant operating and maintenance procedures, the provision for review of these procedures and for review of equipment design to incorporate the results of operating experience, and most importantly, the establishment of an ongoing training program. Training is provided for all personnel so that each individual is capable of carrying out their responsibility for maintaining their own exposure ALARA consistent with discharging their duties and also that of others. The development of the proper attitudes and awareness of the potential problems in the area of radiation protection is accomplished by proper training of all plant personnel. The organizational structure related to assuring that occupational radiation exposure be maintained ALARA is described below.

Organization Structure

The operating organization structure of the Point Beach Nuclear Plant is described in Chapter 12. Reporting to the Radiation Protection Manager (RPM) are health physicists, radiological engineers, radiation protection specialists, supervisors and technicians.

The RPM is responsible for the overall radiation protection and ALARA programs. The RPM reports to the Plant Manager who reports to the Site Vice President. The RPM is a member of Senior Management. Radiation protection concerns are discussed at the Senior Management meetings. Also, the ALARA Review Board Chairperson holds periodic meetings to discuss ALARA concerns. Several station groups (e.g., Operations, Maintenance, station management, etc.) participate in these meetings.

Personnel Activities and Responsibilities

The RPM is responsible for the radiation protection program and for handling and monitoring radioactive materials, including source and byproduct materials.

Administrative Concerns

The radiation protection (health physics) program is based on regulations and experience which includes or considers the following:

a. Sufficiently detailed procedures are prepared and approved for radiation protection and are a part of the station health physics program.

b. Sufficiently detailed procedures are prepared approved for receiving and shipping of radioactive material and radioactive waste ensure compliance with 10 CFR and 49 CFR.

c. Radiological incidents are thoroughly investigated and documented in order to minimize the potential for recurrence. Reports are made to the NRC in accordance with 10 CFR 20.

d. Periodic radiation, contamination, and airborne activity surveys are performed and recorded to document radiological conditions. Records of the surveys are maintained in accordance with 10 CFR 20.

e. Records of occupational radiation exposure are maintained and reports are made to the NRC as required by 10 CFR 20, and to individuals as required by 10 CFR 19.13.

f. Posted areas are segregated and identified in accordance with 10 CFR 20. Positive control is exercised for each individual entry into high radiation and very high radiation areas.

g. Personnel are provided with personnel radiation monitoring equipment to measure their radiation exposure in accordance with 10 CFR 20.

h. Process radiation, area radiation, portable radiation, and airborne radioactivity monitoring instrumentation are periodically calibrated as required.

i. Access control points are established to separate potentially contaminated areas from uncontaminated areas of the station.

j. Protective clothing is used as required to help prevent personnel contamination and the spread of contamination from one area to another.

k. Tools and equipment used in radiologically controlled areas are surveyed for contamination before removal to an uncontrolled area. Contaminated tools and equipment removed from a contaminated area are packaged as necessary to prevent the spread of contamination to uncontrolled areas.

l. All entries to radiologically controlled areas at PBNP are controlled by a Radiation Work Permit (RWP). Personnel must be signed in on an RWP to perform work of any type in radiologically controlled areas. Jobs involving significant radiation exposure to personnel are pre-planned. Where available, mock-ups may be used for practice to reduce exposure time on the actual job. The use of special tools and temporary shielding to reduce personnel exposure is evaluated on a job-by-job basis.

m. A bioassay program is included as part of the radiation protection program. This program includes air sampling, whole body screens and counting, and/or in vitro analysis to determine the intake of radioactive material.

n. An environmental radiological monitoring program is in operation to measure any effect of the station on the surrounding environment.

o. All significant radioactive effluent pathways from the station are monitored and records are maintained.

p. “Hot spot” labeling is utilized on some localized radiation sources, as deemed appropriate, in efforts to reduce personnel time in regions of the exposure field and increase personnel distance from the source of exposure.

Implementation of Procedures and Techniques

The criteria or conditions under which various operating procedures and techniques for ensuring that occupational radiation exposures are ALARA for systems associated with radioactive liquids, gases, and solids, along with the means for planning and developing procedures for radiation exposure-related operations, are given in the following:

a. Section 11.4.1, Ensuring That Occupational Radiation Exposure are as ALARA;

b. Section 11.4.2, Radiation Protection

c. Section 11.5, Radiation Monitoring System

d. Section 11.6, Shielding Systems

Implementation of Exposure Tracking and Exposure Reduction Program

Self-reading and/or electronic dosimeters are used at Point Beach to record estimates of daily exposure received by each individual worker. This information enables the Radiation Protection group to spot significant individual exposures prior to processing other monitoring dosimetry. Work group person-rem summaries are generated by a computerized dose tracking program. The summaries serve to alert the plant radiation protection staff of the trends in person-rem expenditures. Point Beach tracks and reports occupational dose by work group, and the dose expenditure resulting from work performed on various plant systems and components.

The computerized dose tracking program applications are:

a. To provide timely radiological feedback information to the various work groups.

b. To identify and compile dose histories on specific sources of occupational dose that might be reduced through improved plant working and shielding procedures and training programs.

c. To provide data for comparison studies of specific sources of occupational exposure among similar nuclear stations with relevant factors such as reactor equipment and plant layout, etc., taken into account.

Point Beach has a Plant ALARA Review Board. This Review Board is composed of members of several station groups e.g., Operations, Maintenance, Engineering, the RPM and the ALARA Review Board Chairperson. The charter of the Plant ALARA Review Board is to advise the plant manager on ALARA matters. The Review Board reviews annual exposure goals and high dose jobs. The Review Board meets periodically. The Chairperson of the Review Board has decision-making responsibility.

Training Program

The radiation protection training program covers the following:

a. Plant/Contractor employee radiation worker training

b. Plant/Contractor employee respiratory protection training

c. Plant/Contractor employee radiation worker and respiratory protection retraining

d. Contractor radiation protection technician training

e. Contractor radiation protection technician retraining

f. Radiation protection technician training

g. Radiation protection technician retraining

All personnel must understand how radiation protection relates to their jobs and have reasonable opportunities to discuss radiation protection safety with a member of the Radiation Protection group whenever the need arises. Plant personnel are made aware of FPL Energy Point Beach's commitment to keep occupational radiation exposure as low as reasonably achievable.

11.4.2 RADIATION PROTECTION

Organization

The administrative organization of the radiation protection program and personnel responsibilities are referenced in Section 11.4.1.

The experience and qualifications of all station personnel are given in Technical Specifications, Section 5.3, Facility Staff Qualifications.

Facilities and Access Provisions, Equipment and Instruments

The plant site is divided into two categories, the Clean Area and the Radiation Control Area (RCA) as shown on Figure 11.4-1 through Figure 11.4-8 (RCA is shown cross-hatched).

The Radiation Control Area encompasses the Primary Auxiliary Building, both facades and Containment Buildings, portions of the South Service Building and outside yard area. Access to the Radiation Control Area is limited to those persons authorized for entry by plant supervisors and radiation protection personnel. Entry to and exit from the Radiation Control Area is normally through the designated access control point.

Radioactive materials may be stored in Sea Land containers in designated locations in the outside yard area of the RCA. Routine surveys and inspections are performed to verify container integrity.

Any area inside the Radiation Control Area in which radioactive materials and radiation are present shall be surveyed, and conspicuously posted where required by radiation protection procedures.

Any area established outside the radiation control area in which radioactive materials and radiation are present shall be surveyed and conspicuously posted in accordance with applicable Radiation Protection procedures.

The general arrangement of the service facilities is designed to provide adequate personnel decontamination and change areas. The clean locker rooms are used to store items of personal clothing not required or allowed in the Radiation Control Area. These locker rooms are employed as change areas from street clothes to modesty garments.

Several wall-mounted frisker-type monitors are available at strategic locations, particularly at or near the normal exit point of contaminated areas, to enable personnel to check themselves for contamination. Automated personnel contamination monitors are provided at the exit of the Radiation Control Area. All personnel are to use the personnel contamination monitors (or Geiger-Mueller count rate meters) to monitor themselves upon leaving the Radiation Control Area or other posted radiologically controlled areas as required by radiation protection procedures.

Decontamination showers are located in the Radiation Control Area (RCA). The decontamination of personnel is performed in accordance with the instructions listed in approved radiation protection procedures. The auxiliary building has facilities to handle the decontamination of large items or equipment. The decontamination area contains service facilities. A decontamination area is also provided within the RCA machine shop for the decontamination of tools and equipment.

Strict administrative control of radiation exposure includes those methods described in Section 11.4.3. Other administrative controls include locked high radiation areas, radiation work permits, timekeeping of personnel in high radiation areas when required by RWP, and measures including escorts for visitors within the plant radiologically controlled areas.

Locations where the dose to the whole body may exceed 1 rem in 1 hour are conspicuously posted, and have locked accesses to prevent unauthorized entry or are equipped with red flashing warning lights. Keys to these accesses are kept under special administrative control.

Facilities provided for the Chemistry and Radiation Protection groups, include the chemistry laboratories, counting rooms, the calibration and source storage room, and the Radiation Protection station. Laboratory radiation measuring instrumentation in the Radiation Protection count room is supplemented by chemistry laboratory and counting room instrumentation.

These facilities are equipped to conduct radiation protection and chemistry programs for the station; to detect, analyze, and measure ionizing radiation; and to evaluate any radiological problem that may reasonably be expected.

The chemistry lab is equipped with fume hoods, which exhaust through high efficiency particulate and charcoal filters to the auxiliary building vent stack. Other typical chemistry laboratory equipment includes analytical instruments and sample preparation equipment.

The Chemistry counting room is provided with walls sufficiently shielded to reduce background count rates to acceptable values. Counting room typical equipment includes gamma, and beta detection and quantification equipment.

The Radiation Protection count room is equipped to count routine air samples and contamination smear surveys for beta/gamma and alpha radiation. It also serves as a central location for Radiation Protection instrumentation and equipment, including: portable radiation survey instruments and air sampling equipment.

A variety of instruments are used to perform radiation measurements at Point Beach Nuclear Plant. These include instruments to detect and measure alpha, beta, gamma, and neutron radiation. Various isotopic sources are available for instrument calibration and functional tests. Calibration sources for chemistry laboratory radiation detection equipment conform to the various counting geometries used.

Assorted low volume and high volume gaseous, particulate, and iodine sampling equipment is available for routine use as well as for special purpose and emergency airborne radiation surveys. Table 11.4-1 lists the normal storage location of respiratory protection equipment, protective clothing, and portable and laboratory technical equipment and instrumentation.

Quantities of respirators vary throughout the life of the plant and are sufficient to meet personnel needs. The following types of respirators are used: air purifying full-face mask respirators, air-line full-face mask respirators, air-line hood respirators, and positive pressure self-contained breathing apparatus (SCBA). In addition, to the equipment listed in Table 11.4-1, an emergency breathing air system exists for control room personnel, and backup is provided by additional SCBAs with bottled air.

Typical detectors and monitors and the quantity, range, and frequency and methods of calibration for radiation protection instrumentation and technical equipment are specified in Table 11.4-2.

Radiation protection and radio chemistry facilities are described in Table 11.4-3.

Procedures

All personnel who are to work in the Radiation Control Area (RCA) receive radiation protection general access training prior to their assignment to work in the RCA. Radiation protection general access training includes all pertinent radiation practices and procedures to a degree that allows an employee to perform his/her assignment without incurring unnecessary radiation exposure.

In addition to general access training and periodic safety meetings, radiation safety instructions, policies, and procedures are made available to plant workers. Radiation control standards and procedures for working with radioactive materials are designed for protection of all personnel involved in the operating and maintenance of the facility. The Radiation Protection group provides additional detailed operational health physics procedures for use.

11.4.3 PERSONNEL MONITORING

Personnel External Exposure Program

The personnel external exposure program consists of multiple methods of reviewing external radiation levels and controls within the plant. These provide plant personnel status information required to maintain an ALARA program.

Area radiation monitors are located throughout the plant and provide general area indication of gamma radiation levels. These levels are continuously monitored and are alarmed in the Control Room. Some monitors also have local indication and alarm at certain in-plant locations. Process radiation monitors with control room indication and alarms also provide for immediate recognition of significant increases in in-plant dose rate levels.

Routine beta-gamma dose rate surveys are made of general access areas of the plant. This provides detailed dose rate information for normal in-plant exposure evaluation. The surveys are reviewed to note unusual trends and for determination of additional controls that may be required due to new or increased radiation dose rates.

Special beta-gamma dose rate surveys are made on an as-needed basis for jobs that take place in normally inaccessible (i.e., high radiation) areas. These areas may not normally be surveyed on a routine basis to keep doses as low as possible. Continuous or intermittent surveys are provided on an as-needed basis as determined by radiation protection for radiation work permits.

Neutron dose rate surveys and personnel neutron dose monitoring is performed when entrance is made into neutron areas as required by radiation protection procedures.

Radioactive materials and special nuclear materials are handled and stored under the direction of personnel as specified in Section 11.4.1.

Dosimeter records furnish data for administrative control of radiation exposure. The official and permanent record of accumulated external radiation exposure is obtained principally from the interpretation of the thermoluminescent dosimeter (TLD) badge. The TLDs are normally processed at routine frequencies. TLD badge results are reviewed and are entered in a computerized radiation exposure records system. These official and permanent records furnish the exposure data for the administrative control of radiation exposure. Required reports are made by radiation protection personnel through use of this records system.

TLDs of personnel who have been or may have been overexposed are processed immediately. The direct reading pocket ionization chamber, commonly called a self-reading dosimeter or SRD, or electronic dosimeter (ED) also may be used to provide an indication of external radiation exposure. Additional monitoring devices are issued as required by radiation protection personnel to provide further monitoring under special conditions.

The use and issuance of personnel monitoring equipment such as SRDs, EDs, and TLDs as well as the evaluation and recording of personnel monitoring data are controlled by written procedures. All persons subject to occupational radiation exposure and having authorized access to radiologically controlled areas are required to wear TLD badges and SRDs or EDs whenever they enter a radiologically controlled area. The RP Manager may make exceptions to this on a case

basis. Some situations that may be appropriate include rescue and medical emergency situations, and others that are deemed appropriate and documented in accordance with radiation protection procedures. Persons not subject to occupational radiation exposure and who do not enter radiologically controlled areas may be exempted from the use of personnel monitoring devices. Area TLDs located within the PBNP protected area are used to ensure compliance with the exemption monitoring requirements of 10 CFR 20 for those personnel exempted from monitoring.

The NRC has approved the use of a multiple dosimetry method for determining external radiation exposure using the weighting factors listed in Table 1 of ANSI/HPS N13.41-1997, “Criteria for Performing Multiple Dosimetry” as an optional means of demonstrating compliance with the TEDE based requirements in 10 CFR Part 20 (Reference 4).

Personnel Internal Exposure Program

The personnel internal exposure program consists of multiple methods of reviewing airborne radioactivity concentrations and controls within the plant. These provide plant personnel status information required to maintain an ALARA program.

The plant vent stack monitors (one for each of the two containment vent stacks) have detectors for air particulate, gas (low, mid, and high range), and iodine. These detectors are monitored by control room operators.

Continuous air monitors also monitor auxiliary building ventilation exhausts, containment purge systems, and the drumming area/spent fuel pool ventilation exhausts. These are used to measure, indicate, and record levels of airborne radioactivity in air exhausted from plant areas.

Portable grab samples are normally taken in accessible areas of the plant on a periodic basis. Special samples are taken as required by radiation protection personnel prior to issuing Radiation Work Permits and before other jobs as necessary. These air sample results are reviewed by radiation protection personnel and are used to determine respiratory protective equipment requirements in accordance with the plant radiation control standards and procedures.

Personnel Bioassay Program

The personnel bioassay program at Point Beach Nuclear Plant is administered by radiation protection management personnel. Bioassay (in vivo measurement and in vitro measurement of radioactive material) are conducted as necessary to aid in determining the extent of an individual’s internal exposure to concentrations of radioactive material. The need for and frequency of bioassay are determined by the duration that a person works with radioactive materials or in an airborne radioactive materials area. Specific frequencies are determined and controlled by procedures. Bioassay results are recorded when required by radiation protection procedures.

1. Whole Body Screen

   Portal Monitors are used to qualitatively detect internal contamination greater than one percent of an ALI (passive monitoring). Entrance and exit whole body screens are performed and documented in accordance with radiation protection procedures.

2. Bioassay Techniques

Bioassay techniques may include any or all of the following: whole body counting, urinalysis or fecal sampling and analysis.

The internal radiation exposure assessment program is implemented in compliance with 10 CFR 20. Work restrictions shall be imposed as needed to ensure that occupational radiation doses are minimized. External and internal doses are limited pursuant to 10 CFR 20. Evaluation of bioassay results is primarily based upon the identification and quantification of radioactive material intake. At the discretion of radiation protection management, on a case basis, dose equivalents are estimated from bioassay data. The actual calculation methods utilized are based on EPA Federal Guidance Reports and International Commission on Radiological Protection (ICRP) reports.

11.4.4 CONTAMINATION CONTROL PROGRAM

The contamination control program consists of multiple methods of controlling the spread of contamination to personnel and equipment within the plant. Routine smear surveys are periodically made of normally accessible areas of the plant and are recorded. These results are reviewed by a radiation protection supervisor. Special smear surveys are performed as required by Radiation Work Permits and for unconditional release of equipment, tools, and materials being removed from radiologically controlled areas. Items which are contaminated are required to be decontaminated to within release limits or packaged and tagged in accordance with the plant radiation protection procedures.

Workers in contaminated areas are required to be monitored for contamination as soon as possible after leaving a contaminated area and prior to exiting the RCA. Additionally, portal-type monitors are utilized to monitor individuals leaving the RCA via the main access area and again when leaving the site (in the security gatehouses). Actual instrumentation used for the contamination surveys is determined by plant radiation protection personnel.

Personnel Protective Equipment

The nature of the work to be done is the governing factor in the selection of protective clothing to be worn by individuals. The protective apparel available are shoe covers, head covers, gloves and coveralls or lab coats. Additional items of specialized apparel such as plastic suits, face shields, and respirators are available for operations involving high level contamination. Radiation Protection personnel evaluate the radiological conditions and specify the required items of protective clothing to be worn.

Personnel Respiratory Protection

The Point Beach Nuclear Plant is designed to minimize concentrations of airborne radioactivity due to inadvertent leaks, spills or other causes through filtered ventilation systems and isolation of equipment in compartments. Further, a radiation protection program is provided to minimize airborne concentrations by detecting and controlling potential sources of airborne radioactivity. The normal concentrations present in areas occupied by personnel are much less than derived air concentration (DAC) levels, and the use of respiratory protective equipment is, therefore, normally not necessary.

Respiratory protective devices are required, however, in any unusual situation arising from plant operations in which airborne radioactivity exceeding the action levels described below, exists or is expected. In such cases, the airborne concentrations are monitored by radiation protection personnel and the necessary protective devices specified according to concentration and type of airborne contaminants present.

Several types of respiratory protective equipment are utilized for radiological control in the respiratory protection program. The type used for a particular circumstance will be determined by the concentration in the air and the protection factor needed to prevent personnel from breathing or being exposed to airborne radioactivity in excess of that specified by 10 CFR 20.

The specifics of the respiratory protection program are directed by Radiation Protection procedures that are maintained current to the Code of Federal Regulation and OSHA requirements using NRC NUREGs, IE Circulars, and Information Notices for guidance, as well as Industry Events and NIOSH notices.

The use of Delta Protection Mururoa V4 F1 and V4 MTH2 supplied air suits has been approved for use at Point Beach with an assigned protection factor (APF) of 2,000. The use of Delta Protection Mururoa V4 F1 R supplied air suits has been approved for use at Point Beach with an assigned protection factor (APF) of 5,000. Approval of the Mururoa suits was based on testing which demonstrated the suits met the applicable European standard for the requirements and test methods for ventilated protective clothing used against particulate radioactive contamination. The testing demonstrated the suits have an overall measured fit factor of 50,000. The Mururoa suits will not be used in an environment immediately deleterious to life and health and will be discarded after one use. Any problems with the suits will be documented in the site's corrective action program and communicated to the manufacturer and to the US nuclear industry. Additional commitments were made regarding suit use, system testing, procedures and training. The requirement of 10 CFR 20.1703(f), to provide standby rescue persons whenever one-piece atmosphere supplying suits are used, does not apply when the Mururoa suits are used in accordance with the manufacturer's instructions because of the suit's self-rescue features. (Reference 2 and Reference 3)

In addition, Self Contained Breathing Apparatuses (SCBAs) that are used for fire-fighting and emergency situations are maintained and cleaned by the Operations group in accordance with Operations procedures.

11.4.5 CORRESPONDENCE AND COMMITMENTS

1. NRC Information Notice 90-33: Sources of Unexpected Occupational Radiation Exposures At Spent Fuel Storage Pools.

2. NRC Generic Letter 94-04: Voluntary Reporting of Additional Occupational Radiation Exposure Data.

3. NRC Information Notice 97-036: Unplanned Intake by Worker of Transuranic Airborne Radioactive Materials and External Exposure Due to Inadequate Control of Work.

4. NRC Information Notice 97-066: Failure to Provide Special Lenses for Operators Using Respirator or Self-Contained Breathing Apparatus During Emergency Operations.

11.4.6 REFERENCES

1. SE 99-007 Storage of Low-Level Dry Radioactive Waste, Green-Is-Clean Radioactive Waste, and Radioactive Material in PBNP RCA Yard Areas.

2. NRC Safety Evaluation, Duane Arnold Energy Center, Monticello Nuclear Generating Plant, Palisades Nuclear Plant, Point Beach Nuclear Plant, Units 1 and 2, Prairie Island Nuclear Generating Plant, Units 1 and 2 - Use of Delta Protection Respiratory Protection Equipment (TAC NOS. MC8744, MC8745, MC8746, MC8747, MC8748, MC8749, and MC8750), dated December 28, 2005.

3. NRC Safety Evaluation, St. Lucie Nuclear Plant, Units 1 and 2; Turkey Point Nuclear Plant, Units 3 and 4; Seabrook Station; Duane Arnold Energy Center; and Point Beach Nuclear Plant, Units 1 and 2, Request for the Use of Delta Protection Mururoa V4F1 R Supplied Air Suits (TAC Nos. ME1156 through ME1163) dated, August 31, 2009.

4. NRC Safety Evaluation, Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Approval to Use Effective Dose Equivalent Weighting Factors for External Radiation Exposure, dated July 13, 2009.

Table 11.4-1 STORAGE LOCATION OF EQUIPMENT

Equipment	Normal Storage Location
Full-face Masks and Hoods (Air Purifying) Full-face Masks and Hoods (Airline) Hoods (Airline)	RP Station Area
Protective Clothing	Auxiliary Building/RP Station
ß-γ Air Ionization Chambers G-M Survey Instruments Neutron Detectors	Auxiliary Building/ RP Instrument Facility
Chemical Analysis equipment	Hot Laboratory, Cold Laboratory

Table 11.4-2 RADIATION PROTECTION EQUIPMENT

Type Detector/Monitor	Estimated Number	Range[1]	Calibration Frequency[2]	Calibration Method
Multichannel Analyzer	1	Various	Annual	Standard Reference Materials
Air Ion Chamber Exposure Rate meter	30	Various	Annual	Standard Reference Materials
G-M Survey Count Rate Instrument	50	Various	Annual	Standard Reference Materials
Alpha Detector	1	0-2E6 cpm	Annual	Standard Reference Materials
High Range, Exposure rate	25	Various	Annual	Standard Reference Materials
Neutron Detector	2	0-5 rem/hr minimum	Annual	Standard Reference Materials
Air Sampler	10	Various	Annual	Standard Reference Materials
Portable Area Radiation Monitors	15	Various	Annual	Standard Reference Materials
Portable Continuous Air Monitor	5	Various	Annual	Standard Reference Materials
Dosimeters				
Direct Ion Chamber	As Required	Various	Semiannual	Standard Reference Materials
Electronic	300	0.001-1000 rem; 0.003 - 100 rem/hr	Annual	Standard Reference Materials

(1) A variety of models are in use with a variety of ranges.

(2) Denotes minimum requirement. More frequent calibrations may be required by Radiation Protection Instrument Calibration procedures.

Table 11.4-3 RADIATION PROTECTION AND RADIOCHEMICAL FACILITIES

Name	Location	Primary Function
Calibration Facility	South Service Building	Calibration and Storage of Portable Radiation Survey and Air Sampling Equipment
Hot Laboratory	South Service Building	Chemical Analysis and Radiochemical Separations
Cold Laboratory	North Service Building	Chemical Analysis
Counting Rooms	South Service Building	Radioactivity and Radiological Determination of Samples
Laundry and Respirator Cleaning Facility	South Service Building	Cleaning, Inspection, and Storage of Respiratory Protection Equipment
Radiation Protection Offices	South Service Building	Location of Radiation Protection Information

Figure 11.4-1 UNIT 1 CONTAINMENT OPERTING FLOOR AND MISCELLANIOUS UPPER FLOORS SOUTH

[redacted]

Figure 11.4-2 UNIT 1 RADIATION CONTROL AREA - OPERATING FLOOR

[redacted]

Figure 11.4-3 UNIT 1 RADIATION CONTROL AREA - INTERMEDIATE FLOOR

[redacted]

Figure 11.4-4 UNIT 1 RADIATION CONTROL AREA - GROUND FLOOR

[redacted]

Figure 11.4-5 UNIT 2 CONTAIMENT OPERATING FLOOR AND MISCELLANOUS UPPER FLOORS NORTH

[redacted]

Figure 11.4-6 UNIT 2 OPERATING FLOOR LEVELS NORTH

[redacted]

Figure 11.4-7 UNIT 2 INTERMEDIATE FLOOR LEVELS NORTH

[redacted]

Figure 11.4-8 UNIT 2 GROUND FLOOR NORTH

[redacted]

11.5 RADIATION MONITORING SYSTEM

11.5.1 DESIGN BASES

Monitoring Radioactivity Releases

Criterion: Means shall be provided for monitoring the containment atmosphere and the facility effluent discharge paths for radioactivity released from normal operations, from anticipated transients, and from accident conditions. An environmental monitoring program shall be maintained to confirm that radioactivity releases to the environs of the plant have not been excessive. (GDC 17)

The containment atmosphere, the auxiliary building vent, the drumming area vent, the condenser air ejector exhaust, the gas stripper building exhaust, the containment fan-coolers service water discharge, blowdown from the steam generators, the steam relief lines to atmosphere, the component cooling water, the waste disposal system liquid effluent, the spent fuel pool heat exchanger service water discharge, and the service water discharge are monitored for radioactivity concentration during normal operations, anticipated transients, and accident conditions. High radiation in any of these is indicated and alarmed in the control room.

All gaseous effluent from possible sources of accidental radioactive release external to the reactor containment (e.g., the spent fuel pool and waste handling equipment) are exhausted from vents which are monitored. All accidental spills of liquids are contained within the reactor auxiliary building and collected in a sump. Any contaminated liquid effluent released to the condenser circulating water is monitored. For any leakage from the reactor containment, under accident conditions, the plant radiation monitoring system supplemented by portable survey equipment provides adequate monitoring of radioactivity release during an accident. An outline of the procedures and equipment to be used in the event of an accident is presented in Section 11.5 and Section 11.6. The environmental monitoring program is described in Section 2.7.

Radiation Monitoring for leakage detection is described in Section 6.5.

Monitoring Fuel and Waste Storage Areas

Criterion: Monitoring and alarm instrumentation shall be provided for fuel and waste storage and associated handling areas for conditions that might result in loss of capability to remove decay heat and to detect excessive radiation levels (GDC 18).

Monitoring and alarm instrumentation is provided for fuel and waste storage and handling areas to detect inadequate cooling and excessive radiation levels. Radiation monitors are provided to maintain surveillance over the release of radioactive gases and liquids, and the permanent record of activity releases is provided by radiochemical analysis of known quantities of waste.

The spent fuel pool cooling system is flow monitored to ensure proper operation, as described in Section 9.9. Radiation monitors are provided in the storage and associated handling areas when fuel is present to detect excessive radiation levels and to initiate appropriate safety actions as required by 10 CFR 50.68(b)(6) (Reference 3 and Reference 4).

A controlled ventilation system removes gaseous radioactivity from the atmosphere of the fuel storage and waste treating areas of the auxiliary building and discharges it to the atmosphere via the drumming area vent. Radiation monitors are in continuous service in these areas to actuate high radiation alarms in the control room as described in Section 11.5.2.

Protection Against Radioactivity Release from Spent Fuel and Waste Storage

Criterion: Provisions shall be made in the design of fuel and waste storage facilities such that no undue risk to the health and safety of the public could result from an accidental release of radioactivity. (GDC 69)

Waste handling and storage facilities located within the containment building or primary auxiliary building are contained and equipment is designed so that accidental releases directly to the atmosphere are monitored and will not exceed the limits of 10 CFR 20, Subpart D as discussed in Section 11.1.5, and Section 11.2.5, with the exception of Warehouse 7, Steam Generator Storage Facility and the Independent Spent Fuel Storage Facility. All waste storage facilities located outside the containment building or primary auxiliary building are constrained so that accidental releases directly to the atmosphere will not exceed a small fraction of the dose limits of 10 CFR 100 (as invoked by NRC Generic Letter 81-38), and will not exceed the dose limits of 40 CFR 190 and 10 CFR 20 (Reference 5).

11.5.2 SYSTEM DESIGN AND OPERATION

The radiation monitoring system monitors radiation levels and fluid activities at various locations throughout the plant. It is designed to accomplish three functions under normal and accident conditions:

1. Provide direct indication of and, if necessary, warning of radiation levels in the plant;

2. Measure gas releases from the plant vent stacks to provide indication of potential airborne activity; and

3. Initiate isolation and control functions on certain effluent streams.

In conjunction with regular and special radiation surveys and with radio-chemical analyses performed by the plant staff, the radiation monitoring system provides information to the operator to determine plant conditions and/or emergency status. It also provides adequate information and warning for the safe operation of the plant and assurance that personnel exposure does not exceed 10 CFR 20 limits.

Radiation detectors, microprocessors, and operator input/output terminals are integrated in the radiation monitoring system in order to achieve the desired functions. Figure 11.5-1 and Figure 11.5-2 provide block diagrams of the radiation monitoring system and illustrate the functional relationships of the components.

The radiation detectors sense radiation through one of the physical processes of either ionization or scintillation. The radiation detectors can be further characterized by their monitoring function:

1. Area Monitor
2. Process Monitor
3. System-Level Particulate, Iodine, and Noble Gas Monitor (SPING)

Area monitors calibrated in mR/hour (or mrem/hr) provide direct indication of area radiation dose rates in various parts of the plant. Table 11.5-1A and Table 11.5-1B provide a description, i.e., monitor name, location, indication, and control function; detector type and range; and associated alarm units, of the area monitors.

The process monitors in the radiation monitoring system provide an indication of increasing radiation levels in various fluid streams. Table 11.5-2A and Table 11.5-2B provide a description of the process monitors in format similar to that provided for the area monitors.

The SPING monitors measure particulate, iodine and noble gas discharges from the plant. This provides an indication of potential airborne activity in areas surrounding the plant. Table 11.5-3 provides a description of the SPING monitors.

The radiation monitoring system is a microprocessor-based radiation detection system. Eight Data Acquisition Modules (DAMs) and four SPING monitors provide the necessary microprocessing capability for the plant's radiation detectors. Each SPING has a DAM built into it, and each DAM is capable of serving nine detector (digital) inputs and six analog inputs. Each DAM also has a microcomputer which performs the tasks of data acquisition, history file management, operational status check, alarm determination and interface with the input/output terminals.

The operator has three interfaces with the Radiation Monitoring System: a) plant process computer system (PPCS), b) system server (SS), and c) annunciator panels. The PPCS is designed to be the primary operator interface with RMS. The PPCS polls the SS for information and status. The SS, also an operator interface, has a primary function of polling the DAMs and SPINGs. Annunciator panels are provided that alert the operator to system high alarms.

The only components of this system which are located in the containment are the detectors for certain area monitoring channels. Except for the containment high range monitors which are part of a separate qualified system, they would not be expected to operate following a major loss-of-coolant accident and are not designed for this purpose.

The entire radiation monitoring system is powered from vital busses. The instrument bus provides power to each DAM; the DAM provides power to each of its associated channels. In addition; each DAM is equipped with a battery which provides for eight hours of continuous operation in the event of a power failure.

As can be seen in Figure 11.5-2, the radiation monitoring system consists of eight data acquisition modules (DAMs); four system-level particulate, iodine and noble gas monitors (SPINGs); two system servers (SSs); and interfaces to the PPCS.

The RMS detectors sense radiation either through ionization or scintillation. The detector produces a pulse output that is related to the radiation detected. This signal is input to an interface box, which acts as a signal conditioner for the DAM channel. The interface box regulates voltage to the detector and amplifies the detector signal for input to the DAM/SPING microprocessor.

The microcomputer in the DAM/SPING counts these pulse inputs and converts them into a count rate. The microcomputer performs mathematical calculations to convert the count rate to proper units and to apply a background compensation factor, if required.

Each DAM/SPING is designed to operate its detectors in a stand alone manner. It is capable of doing the following major functions:

1. Accumulate and store historical information from its detectors in the form of 24-one minute, ten minute, one hour and one day average detector readings.

2. Provide instantaneous detector readings on demand.

3. Provide alarm indication and/or control function actuation if the instantaneous detector reading exceeds the programmed alarm setpoint. Detectors that are connected to a DAM or SPING will also provide control function actuation when in a fail low or fail high status.

4. Provide a trend alarm if the rate of change of averaged readings exceeds a programmed trend alarm setpoint.

5. Provide an alert alarm if the instantaneous detector reading exceeds a programmed alert alarm setpoint.

6. Provide an external failure, a low or high fail alarm if a detector system parameter indicates the detector is inoperable.

7. For each detector, maintain a programmed file which serves as the source of detector calibration constants, engineering units, various alarm setpoints and channel file number.

8. Communicate RMS data and alarm information to SS's for audio and visual display and printout.

9. Operate detector check sources.

10. Provide remote on-off control of one pallet-mounted sample pump.

Four DAMs are located in each of the Unit 1 and Unit 2 rod drive rooms. Three SPINGs are also located in the Unit 1 and Unit 2 rod drive rooms; one SPING is located near the drumming area vent stack. Each DAM and SPING has a local readout panel. The local readout device is capable of accessing the current status of any channel associated with that particular DAM.

The DAMs are also connected to two SSs. The SSs provide RMS data to the plant process computer system (PPCS). The Point Beach control room and Technical Support Center are each equipped with a SS.

Each SS has its own keyboard, printer, and system status annunciator. The SS provides a redundant communication and display capability with each DAM/SPING. The SS also has a microcomputer which provides the following functions:

1. Remote programming capability of each DAM channel file.
2. Automatic logging of one minute, ten minute, one hour, or one day averages, if desired.
3. Logging and, on demand, printout of history files.
4. Printout of alarm or failure data when transmitted by DAMs.
5. Printout of current values on demand.
6. Audible and visual alarm indications and reset functions.
7. Annunciating communications error messages.
8. Provides data transmission to the PPCS.

The PPCS operates independently of the SS and has access to all of the SS information. It is programmed to process and display RMS data in an efficient manner which allows the operator rapid and easy access to all system data. The PPCS may be used to display all channels which are in alarm and gives the operator the capability of graphically trending any channel.

The PPCS displays information in the form of status grids. Status grids are block diagrams of the plant showing the detectors in their appropriate locations in the plant. Color coding is employed to show monitor status.

Area Radiation Monitoring System

This system consists of multiple channels which monitor radiation levels in various areas of the plant. These areas are as follows:

Area Monitor
Control Room
Containment 66′ El (one per unit)
Radiochemistry Laboratory
Charging Pump Area (one per unit)*
Spent Fuel Pool Area*
Sampling Room (one per unit)*
Seal Table Containment 46′ El (one per unit)
Drumming Station
Letdown Line (one per unit)
SI Pump Area*
C-59 Area
Central Auxiliary Building Area
CVCS Holdup Tank Area
Valve Gallery

* A redundant high range radiation monitor is also installed in these areas.

Each low range channel contains a fixed position gamma sensitive G-M tube detector assembly. In addition to the G-M tube, the detector assembly also contains its own high voltage supply, pulse amplifier, low voltage regulator, line driver and check source assembly. The high voltage supply develops the potential applied to the G-M detector. When radiation reacts in the detector a negative pulse is generated and coupled to the amplifier. This pulse is amplified and processed by the line driver. The signal is then carried on a twisted, shielded cable pair to its electronics channel on the DAM where it is further processed.

The high range detectors are pressurized ion chamber types, designed to be used in high-level gamma fields of 1 mR/hr to 10,000 R/hr. The detector assembly, like the low range detectors, contains its own high voltage power supply, amplifier, low voltage regulator, line driver, and check source plus a charge-to-pulse converter. The detecting element, an ion chamber, operates in the proportional region. When radiation reacts in the chamber, a current flow is developed that is proportional to the intensity of the radiation field. The charge-to-pulse rate converter develops a pulse rate proportional to the current. The pulses are amplified and then processed by the line driver. A twisted, shielded pair cable then carries the signal to the appropriate electronics channel in the DAM.

Radioactive check sources, 0.5 μCi $(Sr\text{-}Y)^{90}$, are provided with each detector to enable periodic checking of the detectors and electronics for proper response.

The range of all the area radiation detectors is provided in Table 11.5-1B.

Remote alarm for the low range area monitors are provided by an Area Monitor Alarm Unit (AMAU). When high alarm condition exists in the detector channel, a red beacon will flash and a horn will sound on the AMAU. The audible alarm can be silenced on the AMAU by pressing the alarm acknowledge switch. The high alarm condition, through the DAM-SS network, also causes an annunciator in control to alarm. Area Monitor Beacon Units (AMBU), installed in areas where the AMAU beacon and horn are not visible and audible throughout the area being monitored, also respond to the high alarm condition and further serve to alert plant operators to high radiation conditions in the plant.

Process Radiation Monitoring System

This system consists of channels which monitor radiation levels in various plant operating systems. Table 11.5-2A lists the detectors and the systems which are monitored.

The liquid process monitors used for effluent monitoring are “offline samplers.” The sampler is typically a lead shielded detector housing and a sample container for monitoring gamma emitters in liquids. The lead shield configuration is such that the container can be easily changed should it become contaminated.

Each liquid monitor is normally equipped with two detectors. One resides in the lead shielded sampler well and measures the activity of the liquid. The other detector is a general area monitor and measures ambient radiation levels. The detector that is inserted in the liquid sample chamber is a scintillation counter. The detector assembly consists of a photomultiplier tube, high voltage power supply, preamplifier, and discriminator and pulse shaper. In a scintillation detector, when radiation reacts with an inorganic crystal such as NaI, it causes emission of light from the crystal. The photomultiplier tube “sees” this light, amplifies current through electron multiplication, and develops a pulse output. The detector output is then amplified by a preamplifier, processed by a discriminator and pulse shaper and then carried to its electronics channel on the DAM where it is counted and processed. The background detector for each liquid monitor is a G-M tube type detector. The operation of this type of detector is like that explained previously for the area monitor G-M tube detectors.

The gaseous process monitors may be either scintillation or G-M tube type detectors. Some of the detector channels do not contain a background compensation channel.

The primary function of the process monitors is to monitor effluent streams and provide a control function should radiation levels exceed applicable setpoints. Table 11.5-2A lists the control function of the various process monitors. The process monitors are not nuclide specific. Nuclide concentrations are determined by routine analysis of reactor coolant samples for fission and corrosion product activities.

Radiation Monitoring System - SPING Monitors

The SPING monitors (see Table 11.5-3) are used to monitor the exhaust gas of the:

1. Unit 1 containment purge exhaust stack *
2. Unit 2 containment purge exhaust stack *
3. Auxiliary building exhaust stack
4. Radwaste packaging (drumming) area exhaust stack **

* Although blind flanges are installed inside containment on the purge supply and exhaust penetrations during normal operation the containment purge exhaust stacks are monitored because some exhaust gas discharges from RE211/212 through them in order to maintain containment atmosphere at a reduced pressure.

** The radwaste packaging area SPING is the only SPING not equipped with a high range noble gas chamber.

The SPING is used to sample and monitor particulates, iodine and noble gas in the air. The sample connect points for each of these monitors is downstream of the exhaust stack filters. The sample intake goes through a filter paper on which particulates are deposited, then through a charcoal cartridge to trap iodines and then into the gas chamber for low and medium range noble gas measurement. The sample then passes through a high-range noble gas chamber, through the pump and to the sample outlet.

The SPING features stainless steel plumbing through the sampler stages, a photohelic flow indicator with low and high flow setpoints, remote flush valves, a manual grab sample port with hose barbs and an air pump and a connection plug for a portable terminal.

Instrumentation

1. The particulate filter is monitored by a beta scintillation detector. Counts from the beta detector are a measure of the amount of beta-emitting isotopes on the filter.

2. The charcoal cartridge is monitored by a 2″ × 2″ NaI gamma scintillation detector. This detector is gain stabilized to minimize the effects of drift caused by fluctuations in temperature and/or aging. The measurement is accomplished using a single channel analyzer with its window calibrated to the 364 keV energy of I-131.

3. The low-range noble gas monitor is a beta scintillation detector.

4. An energy compensated G-M detector monitors the gas volume for the medium-range noble gas measurement. Its output is proportional to the gamma emission of the sample.

5. An energy compensated G-M detector monitors the gas volume of a section of 1″ stainless steel tubing in the high-range noble gas sampler of the SPING. Its output is proportional to the gamma emission of the sample.

6. Each SPING is equipped with a local area monitor. This detector is an energy compensated G-M tube which is calibrated in radiation dose rate and provides a measure of the gamma field at the instrument.

7. Radioactive check sources are provided to enable periodic checking of the detectors and electronics for proper response. The following list summarizes the channels with check sources.

	CHANNEL	CHECK SOURCE
1.	Beta Particulate	30 μCi Cs-137
3.	Iodine	0.5 μCi Ba-133
5.	Low-Range Noble Gas	30 μCi Cs-137
6.	Area Monitor	0.5 μCi Sr-90, Y-90
9.	High-Range Noble Gas	0.5 μCi Sr-90, Y-90

The upper counting range of the particulate, iodine, and low-range noble gas channels is 5.1×10^5 cpm. The beta particulate channel is approximately 3% (4π) efficient for Tc-99 beta particles. The I-131 gamma scintillation channel is approximately 3% (4π) efficient for the 364 keV gamma from I-131 decay. The low-range noble gas channel's useable range is approximately from 1×10^{-7} to 2×10^{-2} μCi/cc for Xe-133. The medium-range noble gas channels range is approximately from 3×10^{-3} to 1×10^{3} μCi/cc for Xe-133. The high-range noble gas channel has an approximate range of 1×10^{1} to 5×10^{5} μCi/cc for Xe-133. An area monitor measures ambient radiation levels and has an approximate range of 0.01 mR/hr. to 1000 mR/hr.

The radiation monitoring system also includes monitors for each steam line of each unit. The monitors are comprised of a lead-shielded detector which monitors the main steam line upstream of the safety valves for gamma radiation. The detector is an energy compensated G-M tube; its output, therefore, is proportional to the gamma emission from the steam line. The detector output is input to a single electronics channel on a data acquisition module (DAM). The purpose of this detector is to monitor steam line activity in the event steam reliefs are challenged and steam is dumped to the atmosphere.

Radiation Monitoring System Detector Alarms

The radiation monitoring system has the possibility of having three setpoints for each channel: alert alarm, trend alarm, and high alarm. The applicability of each alarm is determined for every monitor. Similarly, the determination of the appropriate setpoint is dependent on the monitor in question.

For the low range area monitors, in general, the high alarm setpoints are chosen to signal unusual radiation conditions. In the event that unusual conditions would persist for a long period of time the alarm setpoint may be raised with proper administrative approval and after appropriate HP precautions have been taken. The setpoint is returned to its normal value after conditions return to normal.

Each liquid process monitor is normally equipped with two detectors: one to measure activity of the liquid and the other monitors ambient radiation levels.

The SPINGs have multiple monitors with a variety of considerations affecting the alarm setpoints. The particulate monitor is a fixed filter type monitor. Therefore, the setpoints for the particulate

monitor must accommodate the accumulation of particulates. Similarly, the iodine monitor has a fixed filter and, therefore, the setpoints must accommodate the accumulation of iodine. The noble gas monitor is equipped with a sampler assembly. In general, purpose of the monitor, location and shielding, range, and sensitivity and ambient background are considered in determining the alarm setpoints. Both liquid and gaseous monitors at release points have setpoints conservatively based on not exceeding Technical Specification limits.

Isokinetic Stack Sampling System

An Isokinetic Stack Sampling System (ISSS) has been installed providing the capability to sample both the Auxiliary Building Vent Stack and the Drumming Area Vent Stack for iodine and particulates during normal operations and accident conditions. One system is installed in each of the above locations.

An air pump draws a suction on a probe inserted in the stack. The air is drawn through a filter where particulates in the stack atmosphere are deposited. The air flowrate is determined by a solenoid operated flow control valve. Both stack velocity and probe velocity signals are sent to a flow controller which controls the position of the flow control valve. The controller matches probe velocity with stack velocity, thus providing a truly representative isokinetic sampling of stack particulate. The filter must be removed manually for laboratory analysis.

Accident Monitoring - Containment High-Range Radiation Monitor

In addition to the low-range radiation monitors in containment, three high-range radiation monitors per containment are provided for accident monitoring. Though the high-range radiation monitors are assigned RMS sequential detector numbers, they do not interface with the DAM-SS network in the radiation monitoring system. Three detectors per containment with an eight decade range are located on floor or beam-mounted seismic supports located on the 66′ El. in each containment. Power for each detector is via a separate safety-grade instrument bus. Separation and seismic support provide IE qualification for the detector channels. The output of the detector is supplied to the plant computer and the respective unit ASIP.

Emergency Plan Facility Monitoring

Radiation monitoring is provided for the Technical Support Center and the Emergency Operations Facility in the Site Boundary Control Center. The monitoring meets the requirements of NUREG-0696 and is described in the Emergency Plan.

11.5.3 SYSTEM EVALUATION

All liquid waste releases shall be continuously monitored for gross activity during discharge to ensure that the activity limits specified in 10 CFR 20 for unrestricted areas are not exceeded.

Those secondary-side liquid wastes containing only tritium (for example, condenser hotwells) may be discharged without being continuously monitored if the volume of liquid to be released is a batch release and the amount of tritium has been isotopically quantified.

11.5.4 REQUIRED PROCEDURES AND TESTS

Monthly checks of all process and area radiation monitors are performed using remotely operated or portable radioactive check sources. Results of the monthly checks are used to determine the need for recalibration or maintenance of the monitors. Calibration of the process and area radiation monitors is done at refueling intervals as a minimum, and may be more frequent as required by maintenance or replacement of instrumentation.

11.5.5 REFERENCES

1. Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Revision 2.

2. WE Letter to NRC, "Implementation of Regulatory Guide 1.97," dated September 1, 1983.

3. 10 CFR 50.68, Criticality Accident Requirements.

4. NRC 2008-0044, License Amendment Request 247: Spent Fuel Pool Storage Criticality Control, dated July 24, 2008.

5. 10 CFR 50.59, SER 96-037, "Radioactive Materials Storage Area In the South Bay of the Steam Generator Storage Facility," dated May 9, 1996.

Table 11.5-1A RADIATION MONITORING SYSTEM AREA MONITORS

Sheet 1 of 3

DETECTOR NUMBER	NAME	LOCATION	INDICATION	CONTROL FUNCTION
RE-101	Control Room Area Monitor	West wall of control room.	Indicates dose rates in control room.	Shifts control room ventilation to Mode 5.
1/2-RE-102	Containment Low-Range Area Monitor	El. 66′ near access hatch on east side.	Provides dose rates within containment near access hatch.	
RE-103	Chemistry Lab Area Monitor	East wall of chemistry lab near counting room door.	Provides indication of dose rates in chemistry lab and associated hallways.	
1/2-RE-104	Charging Pump Room Low-Range Area Monitor	West side of shield wall east of cubicles on El. 8′ of aux. bldg.	Indicates dose rates in hallways east of charging pump cubicles.	
RE-105	Spent Fuel Pool Low-Range Area Monitor	Mounted on railing just northeast of spent fuel pool on El. 66′ of aux. bldg.	Provides indication of dose rates in the vicinity of the spent fuel pool.	
1/2-RE-106	Primary Side Sample Room Low-Range Area Monitor	West wall towards north corner of sample room on El. 26′ of aux. bldg.	When sampling system is in operation, it indicates dose rate inside sample room.	
1/2-RE-107	Seal Table Area Monitor	Mounted on wall just above table on El. 46′ of containment	Provides an indication of general area dose rate near seal table.	
RE-108	Drumming Station Area Monitor	Mounted inside the drumming station area waste processing cubicle.	Provides dose rate indication within the drumming station.	
1/2-RE-109	Post-Accident Sample Line Monitor	South wall near east corner of primary side sample room on El. 26′ of aux. bldg.	Provides an indication of failed fuel by monitoring the primary coolant sample activity.	

Table 11.5-1A RADIATION MONITORING SYSTEM AREA MONITORS
Sheet 2 of 3

DETECTOR NUMBER	NAME	LOCATION	INDICATION	CONTROL FUNCTION
RE-110	Safety injection Pump Room Low-Range Monitor	North wall just west of passageway in SI pump room.	Provides an indication of the dose rate in general area of SI pumps.	
RE-111	C-59 Panel Area Monitor	Mounted on top of C59 instrument panel on El. 26′ of aux. bldg.	Provides general area dose rate near C59 panel.	
RE-112	Central Aux. Bldg. Area Monitor	North wall just east of pipeway No. 3 on El. 8′ of aux. bldg.	Indicates general area dose rate on El. 8' on aux. bldg.	
RE-113	Aux. Bldg. El. 19' Area Monitor	General area of El. 19′ aux. bldg.	Provides an indication of the dose rate in aux. bldg. sump and general area of El. 19′.	
RE-114	CVCS Holdup Tank Area Monitor	Mounted on wall at entrance of cubicle	Indicates general area dose in cubicle.	
RE-116	Letdown System Valve Gallery Area Monitor	Mounted by north entrance to valve gallery on El. 26′ of aux. bldg.	Indicates general area dose rate in letdown demin valve gallery.	
1/2-RE-126 1/2-RE-127 1/2-RE-128	Unit 1/2 Containment High-Range Radiation Monitors	Containment El. 66′ spaced approximately 120º apart along the outer wall.	Indicates and alarms in computer room and on ASIP panels 1(2) C20.	
1/2-RE-134	Charging Pump Room High-Range Area Monitor	Next to RE-104 on west side of shield wall.	Provides an indication of general area dose rates in the event low-range monitor saturates.	
RE-135	Spent Fuel Pit High-Range Area Monitor	Next to RE-105 on railing just northeast of spent fuel pit.	Provides an indication of general area dose rates in the event low-range monitor saturates.	

Table 11.5-1A RADIATION MONITORING SYSTEM AREA MONITORS

Sheet 3 of 3

DETECTOR NUMBER	NAME	LOCATION	INDICATION	CONTROL FUNCTION
1/2-RE-136	Primary Side Sample Room High-Range Area Monitor	Mounted next to RE-106 on west wall.	Provides an indication of general area dose rates in the event low-range monitor saturates.	
RE-140	Safety Injection Pump Room High-Range Area Monitor	Next to RE-110 on north wall just west of passageway.	Provides an indication of general area dose rates in the event low-range monitor saturates.	
RE-239	TSC Area Monitor	North wall of TSC	Indicated general area TSC dose rates	
RE-240	TSC El. 18.5′ Assembly Area Monitor	North wall of 18.5′ assembly area	Indicates general area El. 18.5′ dose rates	
RE-243	EOF Area Monitor	East wall of EOF	Indicates general area EOF dose rates	

Table 11.5-1B RADIATION MONITORING SYSTEM AREA MONITORS

Sheet 1 of 2

DETECTOR NUMBER	NAME	DAM UNIT-CHANNEL	DETECTOR TYPE	DETECTOR RANGE	AMAU	AMBU (# OF UNITS)
RE-101	Control Room	07-09	G-M Tube (DA1-6CC)	10^{-1}-10^{4}mR/hr		
1-RE-102	Unit 1 Containment Low Range	03-01	G-M Tube (DA1-6CC)	10^{-1}-10^{4}mR/hr	Yes	
2-RE-102	Unit 2 Containment Low Range	04-01	G-M Tube (DA1-6CC)	10^{-1}-10^{4}mR/hr	Yes	
RE-103	Radiochemistry Lab	05-06	G-M Tube (DA1-6CC)	10^{-1}-10^{4}mR/hr	Yes	
1-RE-104	Unit 1 Charging Pump Room Low Range	01-01	G-M Tube (DA1-6CC)	10^{-1}-10^{4}mR/hr	Yes	Yes (3)
2-RE-104	Unit 2 Charging Pump Room Low Range	02-01	G-M Tube (DA1-6CC)	10^{-1}-10^{4}mR/hr	Yes	Yes (3)
RE-105	Spent Fuel Pool Low Range	06-05	G-M Tube (DA1-6CC)	10^{-1}-10^{4}mR/hr	Yes	
1-RE-106	Unit 1 Sampling Room Low Range	03-04	G-M Tube (DA1-6CC)	10^{-1}-10^{4}mR/hr	Yes	
2-RE-106	Unit 2 Sampling Room Low Range	04-04	G-M Tube (DA1-6CC)	10^{-1}-10^{4}mR/hr	Yes	
1-RE-107	Unit 1 Seal Table	01-09	G-M Tube (DA1-6CC)	10^{-1}-10^{4}mR/hr	Yes	
2-RE-107	Unit 2 Seal Table	02-09	G-M Tube (DA1-6CC)	10^{-1}-10^{4}mR/hr	Yes	
RE-108	Drumming Station	07-07	G-M Tube (DA1-6CC)	10^{-1}-10^{4}mR/hr	Yes	Yes (1)
1-RE-109	Unit 1 Sample Line	05-01	G-M Tube (DA1-6CC)	10^{-1}-10^{4}mR/hr		
2-RE-109	Unit 2 Sample Line	06-01	G-M Tube (DA1-6CC)	10^{-1}-10^{4}mR/hr		
RE-110	S.I. Pump Room Low Range	08-09	G-M Tube (DA1-6CC)	10^{-1}-10^{4}mR/hr	Yes	
RE-111	C-59 Panel Area	08-07	Ion Chamber (DA1-4CC)	10^{-2}-10^{2} R/hr	Yes	
RE-112	Central PAB	08-04	Ion Chamber (DA1-4CC)	10^{-2}-10^{2} R/hr	Yes	

Table 11.5-1B RADIATION MONITORING SYSTEM AREA MONITORS
Sheet 2 of 2

DETECTOR NUMBER	NAME	DAM UNIT-CHANNEL	DETECTOR TYPE	DETECTOR RANGE	AMAU	AMBU (# OF UNITS)
RE-113	Auxiliary Building Sump	08-01	G-M Tube (DA1-6CC)	10^{-1}-10^{4}mR/hr	Yes	
RE-114	CVCS Holdup Tank	07-08	Ion Chamber (DA1-4CC)	10^{-2}-10^{2} R/hr	Yes	
RE-116	Valve Gallery	05-05	Ion Chamber (DA1-4CC)	10^{-2}-10^{2} R/hr	Yes	Yes (1)
1-RE-126 1-RE-127 1-RE-128	Unit 1 Containment High Range	None	Ion Chamber	10^{0}-10^{8} R/hr		
2-RE-126 2-RE-127 2-RE-128	Unit 2 Containment High Range	None	Ion Chamber	10^{0}-10^{8} R/hr		
1-RE-134	Unit 1 Charging Pump Room High Range	03-08	Ion Chamber (DA1-5CC)	10^{0}-10^{4} R/hr		
2-RE-134	Unit 2 Charging Pump Room High Range	04-08	Ion Chamber (DA1-5CC)	10^{0}-10^{4} R/hr		
RE-135	Spent Fuel Pool High Range	08-08	Ion Chamber (DA1-5CC)	10^{0}-10^{4} R/hr		
1-RE-136	Unit 1 Sampling Room High Range	01-08	Ion Chamber (DA1-5CC)	10^{0}-10^{4} R/hr		
2-RE-136	Unit 2 Sampling Room High Range	02-08	Ion Chamber (DA1-5CC)	10^{0}-10^{4} R/hr		
RE-140	S. I. Pump Room High Range	05-09	Ion Chamber (DA1-5CC)	10^{0}-10^{4} R/hr		
RE-239	TSC Area	None	GM Tube (DA1-6CS)	10^{-1}-10^{4} mR/hr		
RE-240	TSC El. 18.5′ Assembly Area	None	GM Tube (DA1-6CS)	10^{-1}-10^{4} mR/hr		
RE-243	EOF Area	None	GM Tube (DA1-6CS)	10^{-1}-10^{4} mR/hr		

AMAU - Area Monitor Alarm Unit: Provides primary alerting beacon and horn and acknowledge and testing switches for the unit.

AMBU - Area Monitor Beacon Unit: Provides secondary, remote alerting beacon for high radiation levels where the primary beacon is not visible throughout the area being monitored.

Table 11.5-2A RADIATION MONITORING SYSTEM PROCESS MONITORS
Sheet 1 of 5

DETECTOR NUMBER	NAME	LOCATION	INDICATION	CONTROL FUNCTION
1/2-RE-211	Containment Air Particulate Monitor	In cubicle on east side of containment façade at El. 52′	Indicates particulate activity inside containment, or in purge exhaust stack when on purge supply & exhaust.	
1/2-RE-211B	Background Monitor for RE-211	Next to RE-211	Indicates ambient radiation levels around RE-211/RE-212 pallet	
1/2-RE-212	Containment Noble Gas Monitor	Located in series with RE-211 on detector skid in the same cubicle on El. 52′ of containment façade	Indicates noble gas activity inside containment, or in purge exhaust stack when on purge supply & exhaust.	Initiates containment ventilation isolation upon high activity, which in turn will close purge valves, and secure continuous vent.
RE-214	Aux. Bldg. Vent Stack Noble Gas Monitor	On aux. bldg. exhaust stack at about El. 80′ in Unit 1 façade just south of elevator	Indicates gaseous activity from the primary auxiliary building, service building, chemistry laboratory, SGBD tank vent condensers, or air ejectors.	Shuts vent gas release valve (WG-014) and initiates exhaust vent filtration through filter bank F-23
1/2-RE-215	Condenser Air Ejector Noble Gas Monitor	West wall of turbine hall on El. 46′ west of MSR's	Indicative of steam generator primary-to-secondary leak. May be indicative of a potential airborne radiation exposure in turbine hall.	
1/2-RE-216	Containment Fan Coolers Liquid Process Monitor	Unit 1 - West and slightly south of C59 panel Unit 2 - West and slightly north of C59 panel	Provides indication of potential contamination of service water outlet from containment fan coolers.	
1/2-RE-216B	Background Monitor for RE-216	Next to RE-216	Monitors radiation levels near RE-216.	

Table 11.5-2A RADIATION MONITORING SYSTEM PROCESS MONITORS
Sheet 2 of 5

DETECTOR NUMBER	NAME	LOCATION	INDICATION	CONTROL FUNCTION
1/2-RE-217	Component Cooling Water Liquid Process Monitor	Unit 1 - In overhead pipe just north of stairs going from El. 8′ of aux. bldg. to panel C59 Unit 2 - In overhead pipe just west of Unit 2 component cooling water pumps	Provides indication of component cooling water contamination	Shuts component cooling water surge tank vent valve, CC-017.
RE-218	Waste Disposal System Discharge Liquid Process Monitor	East wall of waste condensate tank cubicle across from component cooling water pump, El. 8′ of aux. bldg.	Monitors waste condensate tank or monitor tank activity being discharged.	Secures waste condensate tank or monitor tank discharge by shutting WL-018.
RE-218B	Background Monitor	Next to RE-218	Monitors radiation levels near RE-218.	
1/2-RE-219	Steam Generator Blowdown Liquid Process Monitor	Outside of each primary side sample room on El. 26′ of aux. bldg.	Provides indication of steam generator blowdown activity and steam generator tube leak rates.	Shuts steam generator blowdown and blowdown tank outlet valves: MS-5958/5959, 2040, and steam generator blowdown sample valves 2083/2084.
1/2-RE-219B	Background Monitor for RE-219	Next to RE-219	Monitors radiation levels near RE-219.	
RE-220	Spent Fuel Pool Heat Exchanger Service Water Liquid Process Monitor	North wall just west of door to Unit 2 containment façade on El. 46′ of aux. bldg.	Provides indication of service water contamination from a spent fuel pool heat exchanger tube leak.	
RE-220B	Background Monitor for RE-220	Next to RE-220	Monitors radiation levels near RE-220.	

Table 11.5-2A RADIATION MONITORING SYSTEM PROCESS MONITORS
Sheet 3 of 5

DETECTOR NUMBER	NAME	LOCATION	INDICATION	CONTROL FUNCTION
RE-221	Drumming Area Vent Stack Noble Gas Monitor	In exhaust ducting above drumming area SPING in northwest corner of Unit 1 façade	Indicates noble gas activity released from spent fuel pool and drumming area, which may be indicative of a potential aux. bldg. airborne release.	
1/2-RE-222	Steam Generator Blowdown Tank Outlet Liquid Process Monitor	East side of steam generator blowdown tank on El. 26′ of aux. bldg.	Indicates activity level in blowdown tank.	Shuts steam generator blowdown and blowdown tank outlet valves: MS-5958/5959, and 2040.
RE-223	Waste Distillate Discharge Liquid Process Monitor	East side of "D" component cooling water heat exchanger on El. 46′ of aux. bldg.	Monitors activity of waste distillate being discharged.	Shuts discharge valve, BE-LW15.
RE-223B	Background Monitor for RE-223	Next to RE-223	Monitors radiation levels near RE-223.	
RE-224	Gas Stripper Bldg. Exhaust Noble Gas Monitor	In exhaust duct in northeast corner of Unit 2 containment façade, ~ El. 87'.	Indicates activity of gaseous release from letdown gas stripper bldg.	
RE-225	Combined Air Ejector Low-Range Noble Gas Monitor	Above door on El. 46′ of Unit 1 turbine hall west of MSR's.	Indicative of primary-to-secondary leak in steam generators. May also indicate potential radiation exposure sources within turbine bldg.	
RE-226	Combined Air Ejector High-Range Noble Gas Monitor	Refer to RE-225	Refer to RE-225.	

Table 11.5-2A RADIATION MONITORING SYSTEM PROCESS MONITORS
Sheet 4 of 5

DETECTOR NUMBER	NAME	LOCATION	INDICATION	CONTROL FUNCTION
1/2-RE-229	Service Water Discharge Liquid Process Monitor	Unit 1: On El. 8′ of aux. bldg. in vent area Unit 2: In aux. feed pump room on east side of tunnel.	Monitors activity of service water discharge.	
1/2-RE-229B	Background Monitor for RE-229	Next to RE-229	Monitors radiation levels near RE-229.	
RE-230	Wastewater Effluent Process Monitor	El. 8′ of turbine hall outside the entrance to water treatment.	Monitors activity level in wastewater effluent.	
RE-230B	Background Monitor for RE-230	Next to RE-230	Monitors radiation levels near RE-230.	
1/2-RE-231	Steam Line “A” Monitor	El. 88′ of containment façade in area of atmospheric steam reliefs.	Monitors activity of Steam Line A.	
1/2-RE-232	Steam Line “B” Monitor	El. 88′ of containment façade in area of atmospheric steam reliefs.	Monitors activity of Steam Line B.	
RE-234	Control Room Iodine Monitor	Top of control room bldg. El. 46′ of turbine hall	Monitors Iodine activity in control room.	
RE-235	Control Room Noble Gas Monitor	Adjacent to RE-234	Monitors noble gas activity in control room.	Shifts control room ventilation to Mode 5.

Table 11.5-2A RADIATION MONITORING SYSTEM PROCESS MONITORS
Sheet 5 of 5

DETECTOR NUMBER	NAME	LOCATION	INDICATION	CONTROL FUNCTION
RE-237	Technical Support Center (TSC) Iodine Monitor	In HVAC ductwork in northwest corner of TSC bldg. at El. 18.5′	Monitors the iodine activity of the supply air to the TSC, displayed locally only.	
RE-238	Technical Support Center (TSC) Noble Gas Monitor	Adjacent to RE-237	Monitors the noble gas activity of the supply air to the TSC.	
RE-241	SBCC Iodine Monitor	SBCC Mechanical/HVAC Room	Monitors iodine activity of air supply to SBCC	
RE-242	SBCC Noble Gas Monitor	Adjacent to RE-241	Monitors noble gas activity of air supply to SBCC.	

Table 11.5-2B RADIATION MONITORING SYSTEM PROCESS MONITORS

Sheet 1 of 3

Detector Number	Name	DAM Unit-Channel	Detector Type	Medium
1(2)RE-211	Containment Air Particulate Monitor	01-02 (02-02)	Scintillation (RDA-3)	Air
1(2)RE-211B	Background Monitor for 1(2)RE-211	01-04 (02-04)	GM Tube (DA1-1)	Air
1(2)RE-212	Containment Noble Gas Monitor	01-03 (02-03)	Scintillation (RDA-3)	Air
RE-214	Aux. Bldg. Exhaust Ventilation Gas Monitor	07-04	Scintillation (RDA-3)	Air
1(2)RE-215	Condenser Air Ejector Gas Monitor	03-05 (04-05)	Scintillation (RDA-3)	Air
1(2)RE-216	Containment Fan Coolers Liquid Monitor	01-05 (02-05)	Scintillation (RDA-5)	Water
1(2)RE-216B	Background Monitor for 1(2)RE-216	01-06 (02-06)	GM Tube (DA1-1)	
1(2)RE-217	Component Cooling Water Liquid Monitor	03-06 (04-06)	Scintillation (RDA-5)	Water
RE-218	Waste Disposal System Liquid Monitor	07-02	Scintillation (RDA-5)	Water
RE-218B	Background Monitor for RE-218	07-03	GM Tube (DA1-1)	
1(2)RE-219	S/G Blowdown Liquid Monitor	05-03 (06-03)	Scintillation (RDA-5)	Water
1(2)RE-219B	Background Monitor for 1(2)RE-219	05-04 (06-04)	GM Tube (DA1-1)	

Table 11.5-2B RADIATION MONITORING SYSTEM PROCESS MONITORS
Sheet 2 of 3

Detector Number	Name	DAM Unit-Channel	Detector Type	Medium
RE-220	Spent Fuel Pool Liquid Monitor	07-05	Scintillation (RDA-5)	Water
RE-220B	Background Monitor for RE-220	07-06	GM Tube (DA1-1)	
RE-221	Drumming Area Vent Gas Monitor	05-07	Scintillation (RDA-3)	Air
1(2)RE-222	Blowdown Tank Outlet Monitor	01-07 (02-07)	GM Tube (DA1-6)	
RE-223	Waste Distillage Overboard Liquid Monitor	08-05	Scintillation (RDA-5)	Water
RE-223B	Background Monitor for RE-223	08-06	GM Tube (DA1-1)	
RE-224	Gas Stripper Building Exhaust Monitor	06-06	Scintillation (RDA-3)	Air
RE-225	Combined Air Ejector Low Range Monitor	07-01	Scintillation (RDA-3)	Air
RE-226	Combined Air Ejector High Range Monitor	05-08	Ion Chamber (DA1-4)	Air
1(2)RE-229	Service Water Discharge Monitor	03-02 (04-02)	Scintillation (RDA-5)	Water
1(2)RE-229B	Background Monitor for 1(2)RE-229	03-03 (03-04)	GM Tube (DA1-1)	
RE-230	Wastewater Effluent Monitor	08-02	Scintillation (RDA-5)	Water
RE-230B	Background Monitor for RE-230	08-03	GM Tube (DA1-1)	

Table 11.5-2B RADIATION MONITORING SYSTEM PROCESS MONITORS
Sheet 3 of 3

Detector Number	Name	DAM Unit-Channel	Detector Type	Medium
1(2)RE-231	Steam Line Monitors - Line A	03-09 (04-09)	GM Tube	
1(2)RE-232	Line B	05-02 (06-02)	GM Tube	
RE-234	Control Room Iodine Monitor	06-07	Scintillation (RDA-2)	Air
RE-235	Control Room Noble Gas Monitor	06-09	Scintillation (RDA-3)	Air
RE-237	TSC Iodine Monitor	None	Scintillation (RDA-2)	Air
RE-238	TSC Noble Gas Monitor	None	Scintillation (RDA-3)	Air
RE-241	SBCC Iodine Monitor	None	Scintillation (RDA-2)	Air
RE-242	SBCC Noble Gas Monitor	None	Scintillation (RDA-3)	Air

RDA-2 2″ diameter X 2″ thick NaI(TI) crystal with an AM-241 seed embedded for automatic gain stabilization. The TI is an impurity added for low energy gamma stabilization.

RDA-3 2″ diameter X.01″ thick plastic crystal.

RDA-5 Same as an RDA-2 minus the Am-241 seed.

Table 11.5-3 RADIATION MONITORING SYSTEM SPECIAL PARTICULATE IODINE AND NOBLE GAS MONITORS SPINGS
Sheet 1 of 2

Number	Name	SPING Number	Location	Detector Type
			Unit 1(2) Containment Purge Exhaust Monitor	
1(2)RE-301	Beta Particulate	21-01 (22-01)	Unit 1(2) Rod Drive Room	RDA-3
1(2)RE-303	Iodine	21-03 (22-03)	Unit 1(2) Rod Drive Room	RDA-2
1(2)RE-305	Low Range Gas	21-05 (22-05)	Unit 1(2) Rod Drive Room	RDA-3
1(2)RE-306	Area Monitor	21-06 (22-06)	Unit 1(2) Rod Drive Room	DA1-1
1(2)RE-307	Mid Range Gas	21-07 (22-07)	Unit 1(2) Rod Drive Room	GM Tube
1(2)RE-309	High Range Gas	21-09 (22-09)	Unit 1(2) Rod Drive Room	GM Tube

Table 11.5-3 RADIATION MONITORING SYSTEM SPECIAL PARTICULATE IODINE AND NOBLE GAS MONITORS SPINGS
Sheet 2 of 2

Number	Name	SPING Number	Location	Detector Type
			Auxiliary Building Exhaust Monitor	
RE-311	Beta Particulate	23-01	Unit 1 Rod Drive Room	RDA-3
RE-313	Iodine	23-03	Unit 1 Rod Drive Room	RDA-2
RE-315	Low Range Gas	23-05	Unit 1 Rod Drive Room	RDA-3
RE-316	Area Monitor	23-06	Unit 1 Rod Drive Room	DAI-1
RE-317	Mid Range Gas	23-07	Unit 1 Rod Drive Room	GM Tube
RE-319	High Range Gas	23-09	Unit 1 Rod Drive Room	GM Tube
			Drumming Area Exhaust Monitor	
RE-321	Beta Particulate	24-01	Top of Drumming Area	RDA-3
RE-323	Iodine	24-03	Top of Drumming Area	RDA-2
RE-325	Low Range Gas	24-05	Top of Drumming Area	RDA-3
RE-326	Area Monitor	24-06	Top of Drumming Area	DA1-1
RE-327	Mid Range Gas	24-07	Top of Drumming Area	GM Tube

Figure 11.5-1 TYPICAL RMS CHANNEL FUNCTIONAL BLOCK DIAGRAM

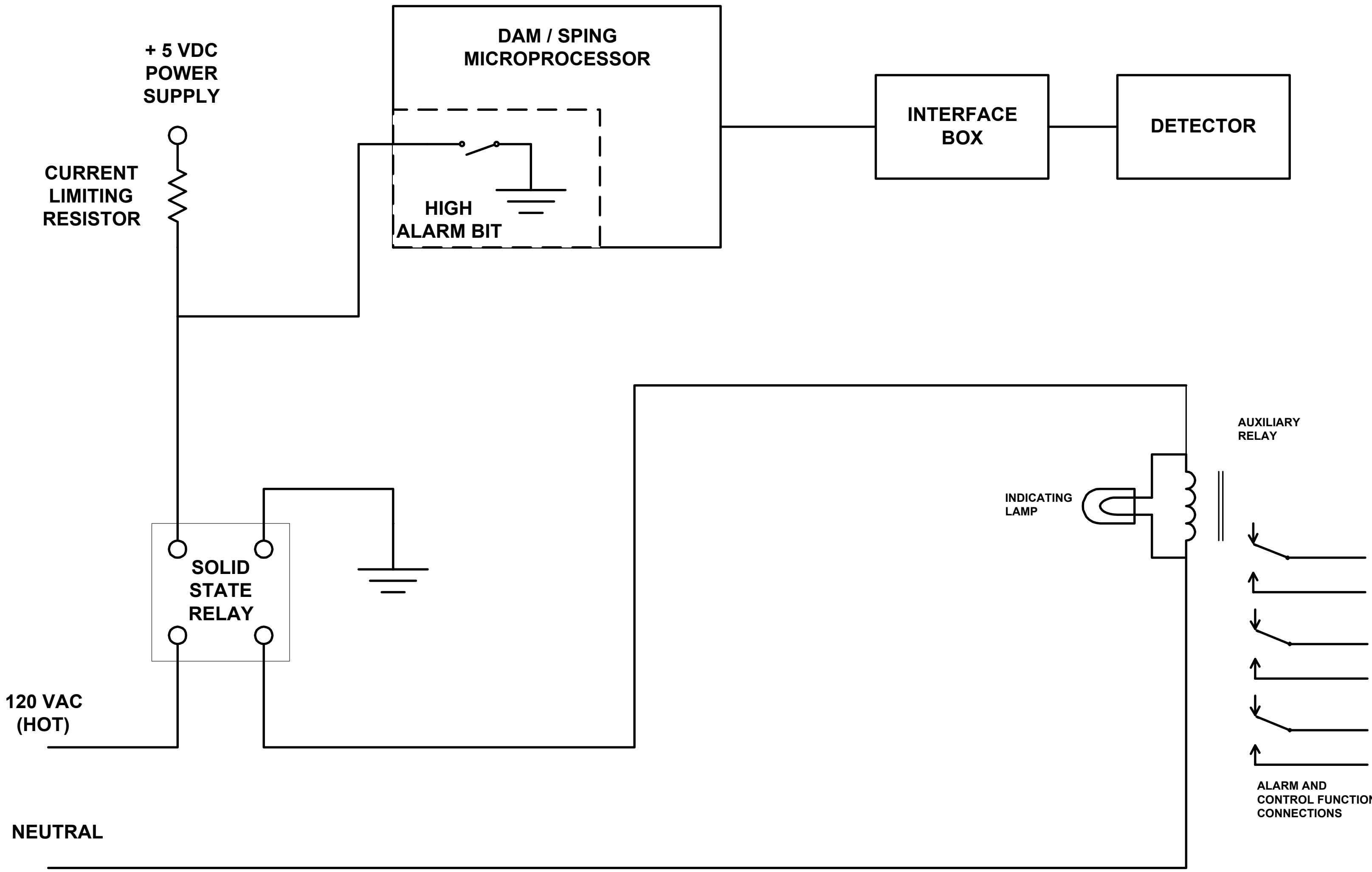
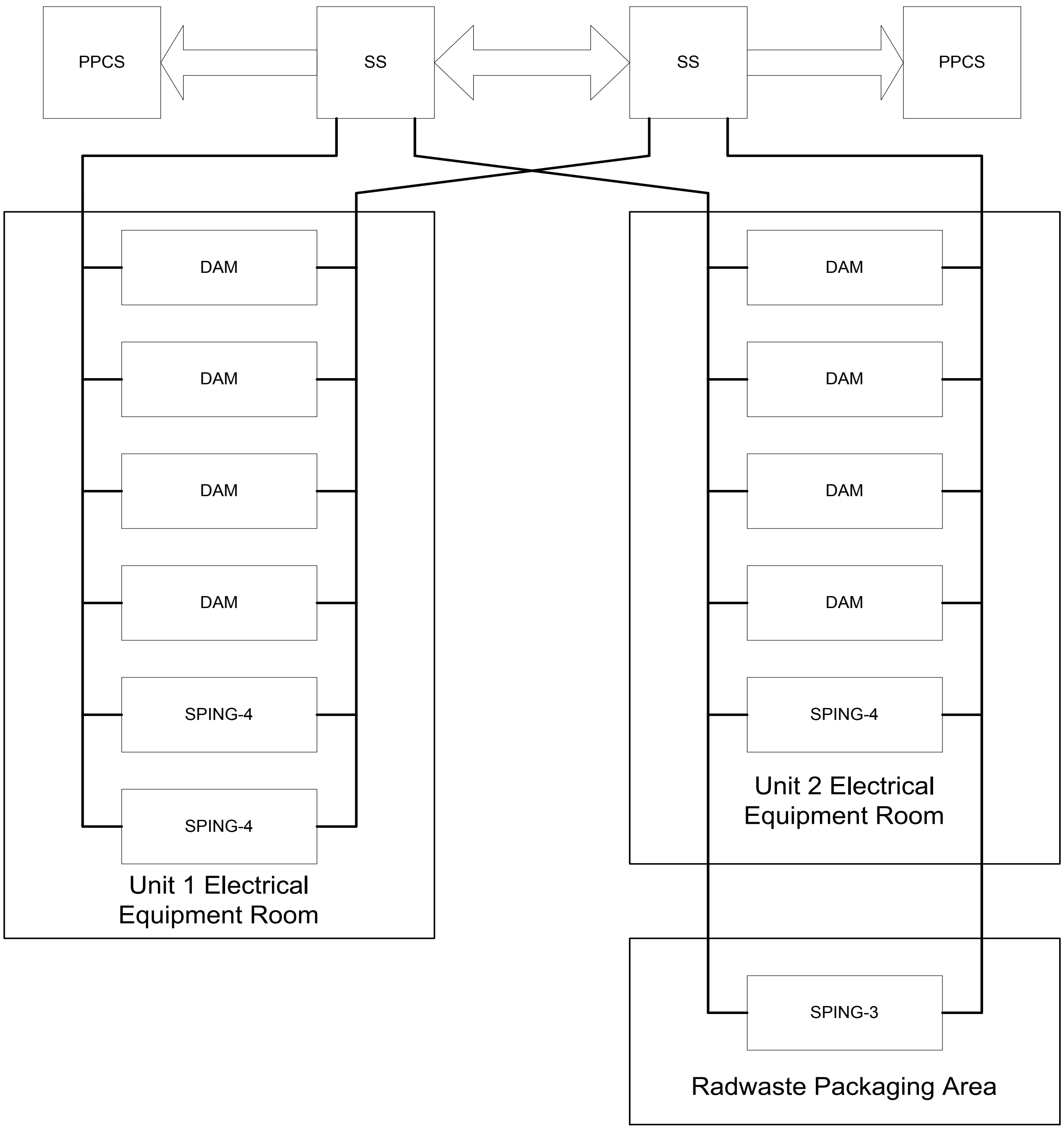


Figure 11.5-2 RADIATION MONITORING SYSTEM FUNCTIONAL BLOCK DIAGRAM



11.6 SHIELDING SYSTEMS

11.6.1 DESIGN BASES

Fuel and Waste Storage Radiation Shielding

Criterion: Adequate shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities. (GDC 68)

Auxiliary shielding for the waste disposal system and its storage components is designed to limit the dose rate to levels not exceeding 1 mr/hr in normally occupied areas and to levels typically <5 mr/hr in periodically occupied areas. Areas having levels in excess of 100 mr/hr, for example, the packaged waste storage area, are posted and barricaded.

Gamma radiation is continuously monitored in the auxiliary building. A high level signal is alarmed locally and annunciated in the control room.

11.6.2 SYSTEM DESIGN AND OPERATION

Radiation shielding is designed for operation at maximum calculated thermal power and to limit the normal operation radiation levels at the site boundary to below those levels allowed for continuous non occupational exposure. The plant is capable of continued safe operation with 1% fuel element defects.

In addition, the shielding provided ensures that in the unlikely event of a maximum design accident, the contained activity does not result in any harmful off-site radiation exposures.

Original design of the plant shielding was performed for a licensed core power level of 1518.5 MWt and a 12-month fuel cycle length. The plant shielding was re-evaluated for the extended power uprate (EPU) assuming a core thermal power of 1810.8 MWt and an 18-month fuel cycle using scaling techniques and the information presented in the following sections and original design reports. Taking into consideration the conservative analytical techniques used to establish the original shielding design and the plant Technical Specifications, which restrict the reactor coolant activity to levels significantly less than 1% fuel defects, it is concluded that the increase in the core power level and in the fuel cycle length will have no significant impact on plant shielding adequacy and safe plant operation. (Reference 12, Reference 13, Reference 16)

Operating personnel at the plant are protected by adequate shielding, monitoring, and operating procedures. Each area in the plant is classed according to the dose rate allowable in the area, based on the expected frequency and duration of occupancy. All plant areas capable of personnel occupancy are classified as one of the five zones of radiation level listed in Table 11.6-1.

Typical Zone 0 areas are the turbine building and turbine plant service areas. Typical Zone I areas are the offices and control room. Zone II areas include the local control spaces in the auxiliary building, and the operating deck of the containment during reactor shutdown. Areas designated Zone III include the sample room, valve galleries, fuel handling areas, and intermittently occupied work areas. Typical Zone IV areas are the shielded equipment compartments in the Auxiliary Building, waste drum storage area, and the primary loop compartments after shutdown.

All radiation and high radiation areas are appropriately marked and isolated in accordance with 10 CFR 20 and other applicable regulations.

The shielding is divided into five categories according to function. These functions include the primary shielding, the secondary shielding, the accident shielding, the fuel transfer shielding, and the auxiliary shielding.

11.6.2.1 Shielding Functions

Primary Shielding

The primary shielding is designed to:

1. Reduce the neutron fluxes incident on the reactor vessel to reduce neutron embrittlement of the reactor vessel beltline region.

2. Attenuate the neutron flux sufficiently to prevent excessive activation of plant components.

3. Limit the gamma flux in the reactor vessel and the primary concrete shielding to avoid excessive temperature gradients or dehydration of the primary shield.

4. Reduce the residual radiation from the core, reactor internals and reactor vessel to levels which will permit access to the region between the primary and secondary shields after plant shutdown.

5. Reduce the contribution of radiation leaking to obtain optimum division of the shielding between the primary and secondary shields.

Secondary Shielding

The main function of the secondary shielding is to attenuate the radiation originating in the reactor and the reactor coolant. The major source in the reactor coolant is the Nitrogen-16 activity, which is produced by neutron activation of oxygen during passage of the coolant through the core. The secondary shielding will limit the full power dose rate outside the containment building from radioactivity inside the containment to less than 1 mr/hr.

Accident Shield

The main purpose of the accident shield is to ensure safe radiation levels outside the containment building following a maximum design accident.

Fuel Handling Shield

The fuel handling shield permits the safe removal and transfer of spent fuel assemblies and control rod clusters from the reactor vessel to the spent fuel pool. It is designed to attenuate radiation from spent fuel, control clusters, and reactor vessel internals to less than 2.5 mr/hr at the refueling cavity water surface and less than 1.0 mr/hr in the auxiliary building.

Auxiliary Shielding

The function of the auxiliary shielding is to protect personnel working near various system components in the chemical and volume control system, the residual heat removal system, the waste disposal system and the sampling system. The shielding provided for the auxiliary building is designed to limit the dose rate to less than 1 mr/hr in normally occupied areas, and at or below 2.5 mr/hr in periodically occupied areas.

11.6.2.2 Shielding Design

Primary Shielding

The primary shielding consists of the reactor internals, the reactor vessel wall, and a concrete structure surrounding the reactor vessel.

The primary shielding immediately surrounding the reactor vessel consists of a reinforced concrete structure extending from the base of the containment to an elevation of 66.0 ft. The lower portion of the shield is a minimum thickness of 6.5 ft. of concrete and is an integral part of the main structural concrete support for the reactor vessel. It extends upward to the operating floor, forming a portion of the refueling cavity. This cavity is approximately rectangular in shape, and has concrete sidewalls which are 5 ft. 5 in. thick adjacent to areas in which fuel is transported.

The primary concrete shielding is air cooled to prevent overheating and dehydration from the heat generated by radiation absorption in the concrete. Eight “windows” have been provided in the primary shield for insertion of the out-of-core nuclear instrumentation. Cooling for the primary shield concrete, nuclear instrumentation, and vessel supports is provided by circulating 26,000 cfm of containment air between the reactor vessel wall and the surrounding concrete structure.

The original primary shield neutron fluxes and design parameters are listed in Table 11.6-2. The parameters listed in Table 11.6-2 are the original design parameters used to assess the adequacy of the Primary Shielding while operating at 1518.5 MWt. The calculations of neutron and gamma ray leakage from the reactor were based on a design basis core configuration that included fresh fuel on the periphery of the core, thus maximizing the neutron and gamma radiation levels external to the reactor vessel. In actual operations, low-low leakage fuel management is used which places burned fuel on the periphery of the core. This fuel management strategy acts to reduce radiation leakage by at least a factor of two. With continued use of low leakage fuel management, the original primary shielding remains adequate following EPU (Reference 12).

Secondary Shield

The secondary shield surrounds the reactor coolant loops and the primary shield. It consists of interior walls within the containment building, the operating floor, and the reactor containment building itself. The containment building also serves as the accident shield.

The lower portion of the secondary shield above grade consists of the 3 ft. 6 in. thick cylindrical portion of the reactor containment and a minimum of 3 ft. thick concrete interior walls surrounding the reactor coolant loops. The secondary shield will attenuate the radiation levels in the primary loop compartment from a value of 25 rem/hr to a level of less than 1 mr/hr outside the reactor containment building. Penetrations in the secondary shielding are protected by supplemental shields.

The original secondary shield design parameters are listed in Table 11.6-3. The parameters listed in Table 11.6-3 are the original design parameters used to assess the adequacy of the Secondary Shielding while operating at 1518.5 MWt. As discussed earlier, the secondary shielding was designed to attenuate the radiation originating from the N-16 activity. N-16 is produced as the oxygen in the water moderator is exposed to the neutron flux present in the reactor core. The amount of activation is defined by the flux (or power) density of the core and the amount of time the moderator is resident in the core. After the moderator exits the core (and neutron field), decay of the N-16 will occur. The amount of decay at any given point in the coolant loop is defined by the time subsequent to exiting the core.

The key parameter affected by power uprate is the change (increase) in the core flux level. This is quantified by the "Core Power Density" design parameter. The change in this parameter will be directly proportional to the change in core power; therefore, the amount of N-16 would also be expected to increase in the same proportion as the core power density and in turn the dose rates in areas inside the secondary shielding surrounding the reactor coolant loops would increase in the same proportion. At EPU conditions, the N-16 source is estimated to increase by approximately 19% compared to original design. The N-16 activity level is not impacted by fuel cycle length. The impact of the estimated 19% increase in source terms is bounded by the conservative analytical techniques typically used to establish plant shielding design (such as ignoring the shadow shielding effect of the neighboring sources, rounding up the calculated shield thickness to a higher whole number, etc.). The current reactor coolant loop shielding and containment structure is determined to be adequate for safe operation following EPU (Reference 12).

Accident Shield

The accident shield consists of the 3 ft. 6 in. prestressed concrete cylinder capped by a shallow, prestressed concrete dome 2 ft. 6 in. thick. Supplemental shielding has been provided for the containment penetrations.

The equipment access hatch is shielded by a 3 ft. thick concrete shadow shield, and a 1 ft. concrete roof to reduce scattered radiation. The personnel lock is provided with an internal lead shield, 3 inches thick, to reduce streaming through the hatch doors following an accident. Smaller penetrations associated with piping and electrical cables are directed into pipe tunnels which are shielded with a minimum of 18 in. of concrete. The control room is protected with concrete sidewalls 18 in. thick, and a concrete roof 14 in. thick.

The original accident shield design parameters are listed in Table 11.6-4.

The post EPU contribution of direct radiation from the radioactivity inside containment on control room dose and the habitability of other plant areas following a LOCA are discussed in Section 11.6.3.

Fuel Handling Shield

The refueling cavity is formed by the upper portions of the primary shield concrete, and other sidewalls of varying thicknesses. A portion of the cavity is used for storing the upper and lower internals packages; these are shielded with concrete walls 5 ft. thick. The remaining walls vary from 3 ft. to 5 ft. 5 in. thick, and provide the shielding required for handling spent fuel.

The refueling cavity, flooded with borated water during refueling operations, provides a temporary water shield above the components being withdrawn from the reactor vessel. The water height during refueling is greater than 21 ft. above the reactor vessel flange. This height ensures that a more than 8 ft. of water will be above the top of a withdrawn fuel assembly. Under these conditions, the dose rate is less than 2.5 mr/hr at the water surface, due to the fuel assembly.

The spent fuel assemblies and control rod clusters are remotely removed from the reactor containment through the horizontal spent fuel transfer tube and placed in the spent fuel pool. The spent fuel transfer tube shielding is designed to protect personnel from radiation during the time a spent fuel assembly is passing through the main concrete support of the reactor containment and the transfer tube.

Radial shielding during fuel transfer is provided by the water and concrete walls of the fuel transfer canal. Sufficient shielding is provided to ensure a maximum dose value of 1.2 mr/hr. in the auxiliary building areas adjacent to the spent fuel pool, next to the exterior of the vertical pool walls.

Spent fuel is stored in the spent fuel pool which is located adjacent to the containment building. Radial shielding for the spent fuel is provided by 5 ft. thick concrete walls plus a minimum of 4 in. of water. The pool is flooded with borated water to a level such that the water height above the stored fuel assemblies is approximately 25 ft. The shielding design parameters for the spent fuel pool include a core unload of 121 assemblies with a cooling time of three days and a pre-EPU average burnup of 40,000 MWD/MTU for all assemblies.

Level and radiation alarms provide assurance that exposure of fuel assemblies cannot occur during transfer operations. A water level sensor in the spent fuel pool provides a low level alarm in the plant control room at a water elevation of 62 ft. 8 in. At this low level alarm point, there would still be more than 7 ft. of water over any withdrawn fuel assembly. A radiation monitor located on the bridge of the fuel handling and transfer manipulator crane alarms locally when radiation levels increase to a pre-determined level above normal background. If an irradiated fuel assembly were to approach the surface of the refueling cavity, the monitor would sense the increase in radiation level and actuate the alarm.

With the analyzed core power increase to 1810.8 MWt, the gamma source from the irradiated fuel is estimated to increase by approximately 19%. The 18-month fuel cycle will also increase the inventory of long-lived isotopes in the irradiated fuel. However, this is not a concern as the estimated maximum dose rates near the refueling canal and the spent fuel pool are dominated by the shorter half-life isotopes in the freshly discharged spent fuel assemblies. The impact of the estimated 19% increase in source terms used in the EPU analysis versus the original shielding analysis is bounded by the conservative analytical techniques which were used to establish plant shielding design. Consequently, the current spent fuel shielding is determined adequate for safe operation following EPU. (Reference 12)

Auxiliary Shielding

The auxiliary shield consists of concrete walls around certain components and piping which process reactor coolant. In some cases, the concrete block walls are removable to allow personnel access to equipment during maintenance periods. Each equipment compartment is individually shielded so that compartments may be entered without having to shut down and, possibly, to decontaminate the adjacent system.

The shield material provided throughout the auxiliary building is concrete. The principal original auxiliary shielding provided is tabulated in Table 11.6-5.

A power uprate will impact the radiation source terms in the core and the "expected" radiation source terms in the coolant. "Expected" source terms are generally less than those allowed by the plant Technical Specifications and are usually significantly less than the "design basis" source terms.

The EPU assessment concluded that the estimated increase in the dose rate for shielded configurations based on the design basis EPU reactor coolant activity versus the pre-uprate coolant activity is compensated by the plant Technical Specifications that will limit the EPU reactor coolant source terms and associated dose rates to less than the original design basis values. Therefore the shielding design based on the original design basis primary coolant activity remains acceptable for the EPU condition. (Reference 12)

11.6.3 SYSTEM EVALUATION

CONTROL ROOM HABITABILITY

In accordance with the requirements set forth in NUREG-0737, the habitability of the PBNP control room has been evaluated. With respect to radiological conditions, the habitability of the control room is most challenged by the large break loss of coolant accident (LOCA) described by FSAR Chapter 14.3. This evaluation and subsequent evaluations have taken credit for the shielding features described in this chapter to evaluate the direct radiation dose and have taken credit for the ventilation-filtration features described in FSAR Chapter 9.8 to evaluate the dose to the control room operator caused by the radioactivity introduced to the control room atmosphere. Refer to FSAR Chapter 9.8 and FSAR Chapter 14.3.5 for more detailed analysis of the post-accident dose caused by radioactivity introduced to the control room. The following discussion generally describes the contributing factors which ensure the habitability of the control room following a design basis accident.

Habitability analyses consider the following contributions to control room operator radiation dose:

1. Inhalation of radioactivity emitted from the post-LOCA containment atmosphere. Leakage from containment is postulated to escape to the environment and then be drawn into the control room through the control room ventilation system. The performance of the control room ventilation system and the assumptions of the analysis are described in FSAR Chapter 9.8 and Chapter 14.3.5.

2. Inhalation of radioactivity emitted from emergency core cooling system (ECCS) piping leaks during the recirculation phase of the accident. Leakage from the ECCS into the primary auxiliary building (PAB) atmosphere and leakage from the refueling water storage tank (RWST) vent is postulated. Some of the activity is released to the environment and then drawn into the control room through the control room ventilation system. The performance of the control room ventilation system and the assumptions of the analysis are described in FSAR Chapter 9.8 and Chapter 14.3.5.

3. Direct radiation from the cloud of radioactivity inside the containment. This is not a significant contribution, but has been calculated in Reference 5.

4. Direct radiation from the cloud of radioactivity that may escape the containment. This contribution has been calculated in Reference 5.

5. Direct radiation from the cloud of radioactivity that may escape from ECCS piping leaks in the PAB. This contribution has been calculated in Reference 5.

6. Direct radiation from the control room emergency filter source. This contribution has been calculated in Reference 5.

In addition, the contribution of scattered radiation from air ("sky-shine") and scattering from large surfaces in the vicinity of containment had been considered in original analyses. However, estimates indicated that the scattered radiation levels would contribute less than 10% of the direct dose. Therefore, scattered radiation has not been considered to be a contributor to control room dose analyses.

Direct Radiation Dose Due to the Radioactive Cloud Inside Containment

Radiation emitted directly from containment is a contributor to post-accident control room gamma doses. The direct dose rate in the control room due to the activity dispersed within the containment is calculated by a computer program which is based on a point kernel attenuation model. The source region is divided into a number of incremental source volumes and the associated attenuation, gamma ray buildup, and distance through regions between each source point and the control room are computed.

The source term for this evaluation is based on operation at 1810.8 MWt for 18 months, and release to the reactor containment of fission products with the fractions and the timing/duration of releases as described in RG 1.183. The fission products are assumed to be homogeneously distributed within the free volume of the reactor containment.

The calculated thirty-day integrated gamma dose to general areas of the control room attributable to direct containment radiation is insignificant. The containment wall, control room wall, and other major intervening walls and floors were considered as shielding. (From Reference 5).

Direct Radiation Dose Due to Control Room Emergency Ventilation Filters

The halogens and particulates in the plume resulting from containment leakage and the halogens in the ECCS/RWST leakage plume are transported to the control room intake and deposited in the emergency ventilation filters. The control room filter unit is located in the northeast corner of the

equipment room, shielded from the control room by 4-inch concrete pads below the filter unit and the nearby heat exchanger units, and by the 14-inch control room concrete ceiling. The point-kernel computer code QAD-CGGP (Reference 6) was used to calculate the direct radiation dose due to the filter source. The calculated 30-day dose is 0.04 rem at the northeast corner of the control room. The contribution of the filter source to the rest of the control room is negligible. (Reference 5)

Direct Radiation Dose Due to the Radioactive Cloud from ECCS Leakage in PAB

Analyses have conservatively included the direct radiation dose from a postulated cloud which may form from the ECCS piping leakage during the recirculation phase of the accident. The equipment leakage rate is assumed to be 300 cc/minute for this analysis. The source term for this evaluation is based on operation at 1810.8 MWt. The fraction of core inventory of iodine in the recirculation water is assumed to be 40 percent. It is assumed that the cloud formed from this leakage is dispersed toward the control room. The contribution of this leakage is added to the postulated radioactive cloud formed from the containment leakage. The analysis of that cloud and the total radiation dose from the radioactive clouds are described below.

Direct Radiation Dose Due to the Radioactive Cloud from Back-Leakage into RWST

Following a postulated loss-of-coolant accident, a small amount of recirculating sump water may back-leak into the Refueling Water Storage Tank (RWST). Some of the radioiodines from the RWST liquid phase may become airborne and be released to the atmosphere. Analysis have conservatively included the direct radiation dose from a postulated cloud which may form from this release. The back-leakage rate is assumed to be 500 cc/minute. The source term is based on operation at 1810.8 MWt. The fraction of core inventory of iodine in the recirculating water is assumed to be 40 percent. The contribution of this leakage is added to the postulated radioactive cloud formed from the containment leakage. The analysis of that cloud and the total radiation dose from the radioactive clouds are described below.

Direct Radiation Dose Due to the Radioactive Cloud Which Escapes Containment

The direct radiation dose due to the postulated radioactive cloud outside of the control room (also called the "passing plume") was calculated using the computer program QAD-CGGP (Reference 6), a point-kernel code. The contribution from the postulated containment leakage, ECCS equipment leakage, and RWST back-leakage were summed. The assumed radioactive source terms are based on those used in the large break LOCA dose calculations described in Chapter 14.3.5. The source term for this evaluation is based on operation at 1810.8 MWt. (Reference 5)

Radiation from the passing plume may stream through the control room doors and window. The "door" relates to the 9 x 10 foot bullet-proof fire wall structures used for ingress and egress located at the northeast and southeast corners of the control room. The "window" relates to the 9 × 12 foot bullet-proof fire wall structure on the east wall of the control room. The post-accident dose rates resulting from radiation emissions from the passing plume decrease considerably with increasing distances inside the control room doors and window. Dose rates near the control room window are also more restrictive because radiation emanating through the window impinges on central areas of the control room where occupancy times are expected to be higher. Areas located immediately

inside the control room doors are expected to be occupied for limited and infrequent periods. To facilitate the calculation of estimated integrated doses, it is necessary to assume conservative occupancy factors. Dose rates inside the control room were calculated at locations 10 feet from the north and south control room doors, and 5 feet from the control room window. The dose resulting from an operator occupancy time of 100% at a location 5 feet from the control room window is used in the control room total dose determination. (Reference 5)

Analyses assume placement of 3 inches of steel shielding for the window and 2 inches of steel shielding (7 inches of concrete equivalent) for the south door to reduce the post-accident dose from the passing plume (Reference 5). Portable lead shielding was originally used to shield the window and south door but was replaced by permanent shielding per Engineering Change EC 11691 (Reference 11). The equivalent-lead thickcness of the permanent shielding exceeds that of the portable lead shielding.

Conclusions

Analyses showed that the direct radiation dose accumulated over the 30-day duration of the accident will be less than 0.32 rem (Reference 5). To address the remaining contributors to the control room radiation, FSAR Chapter 14.3.5 describes the dose contributed from radioactivity which is drawn into the control room during the accident. As described therein, the 30-day Total Effective Dose Equivalent (TEDE) dose from the radionuclides within the control room for the large break LOCA event is 4.4 rem.

Therefore, the direct radiation dose to operators from radiation outside the control room in combination with the radiation dose from radioactivity inside the control room is maintained below the 5 rem TEDE dose limit for the duration of the event.

Habitability of Other Operating Areas: Prior to NUREG-0737 (Historical)

Although the whole body dose rate to personnel entering and exiting the facility buildings would be expected to be higher than that in the control room, two factors assure that the applicable criterion will not be exceeded. First the times required for entry and exit are short. Secondly, the selection of entry and exit times can make use of favorable atmospheric dispersion conditions and wind directions, and information available from activity monitors.

To determine the possible dose that an operator could receive under accident conditions while operating a manual backup item (e.g., valve), it is estimated rather conservatively that it will require 15 minutes to operate the valve. In addition, it is assumed that an additional 15 minutes is required to get to and from the manual equipment. The total integrated whole body dose that an operator would receive performing the above operation would be about 8 rem. This dose is calculated for the first half hour immediately following the accident and assumes that the equipment being operated or services is adjacent to the containment surface. Doses in the vicinity of equipment located within the auxiliary building would be much less due to the shielding afforded by the concrete walls of the auxiliary building.

All components necessary for the operation of the external recirculation loop following a loss-of-coolant accident are capable of remote manual operation from the control room and can be powered by the emergency diesel-generators so that it should not be necessary to enter the auxiliary building in the vicinity of the recirculation loops.

The radiation sources used with the auxiliary shielding design criteria result from a loss of coolant accident caused by a double-ended rupture of a reactor coolant loop where the engineered safety features function to prevent melting of fuel cladding and to limit the cladding metal-water reaction to a negligible amount. This would result in only the fission products which are in the fuel rod gaps being released to the containment. The nongaseous activity would be absorbed in the sump water which flows in the residual heat removal loop and associated equipment. The radiation sources circulating in the residual heat removal loop, shown in Table 11.6-6, form the basis for radiation doses in the auxiliary building.

The radioactivity in the containment building could be an additional source of radiation to the auxiliary building following a loss-of-coolant accident. However, the radiological exposure rate in the auxiliary building from this source would be less than 1% of that from heat removal system piping.

An evaluation was made of direct radiation levels surrounding a 14 in. RHR pipe. The evaluation was based on the radiation sources and evaluation parameters tabulated in Table 11.6-6.

The results of the evaluation are presented in Figure 11.6-1, showing the dose rates for an unshielded and shielded pipe as function of distance. The sensitivity of radiation levels external to the pipe to different degrees of activity released is expressed in Figure 11.6-2. The dose ratio obtained from Figure 11.6-2 may be multiplied by dose rates from Figure 11.6-1 to account for activity levels in the piping which are different from activities resulting from release of fuel rod clad gap activity.

If maintenance of equipment near the recirculation loop is absolutely essential to the continued operation of the engineered safety features during the recirculation phase, local shielding would permit some operations in vicinity of the loop with attendant dose rates of less than 25 rem per hour within one hour following the accident.

If maintenance directly on the loop proper is required, such operations would be limited in duration as radiation levels adjacent to equipment containing the sump water and fission products might be as high as 200 to 300 rem per hour shortly after the initiation of recirculation. Any such emergency maintenance operations described above could be carried out using portable breathing equipment to limit the inhalation hazard from possibly leaking components.

The RHR piping direct doses were performed assuming a power level of 1518.5 MWt. Increasing the power by 2% (~1548.9 MWt) would not result in direct dose rates in excess of those presented in Figure 11.6-1 because of the conservatism in the original assumption made for the RHR piping diameter. The RHR piping associated with recirculation of sump water has a maximum diameter of ten inches as opposed to fourteen inches as assumed in the original design evaluation. The source term assuming power operations at 1518.5 MWt and a 14-inch diameter pipe is large enough to bound a source term assuming 1548.9 MWt and 10 inch pipe.

Habitability of Other Operating Areas: NUREG-0737 Requirements

Additional shielding has been installed in areas of the plant identified by a design review per the requirements NUREG-0737, Item II.B.2 (Reference 9, Reference 10). The purpose of the design review of plant shielding was to identify the location of vital areas and equipment in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded during post-accident operations. The criteria for dose rates and for accessibility to vital areas are based

on 10 CFR 50, Appendix A, GDC 19, which limits the dose to an operator to 5 rem TEDE during the course of an accident in accordance with 10 CFR 50.67. Following changes associated with EPU implementation, there are no vital areas requiring short-term access during the post LOCA recirculation phase other than the control room and technical support center. (Reference 13, Reference 14, Reference 15).

11.6.4 REQUIRED PROCEDURES AND TESTS

Complete radiation surveys were made throughout the plant containment and auxiliary building during initial phases of plant startup. Survey data were taken and compared to design levels at power levels ranging from approximately .01% to 100% rated full power. Survey data at each power level were reviewed for conformance to design before increasing to a higher power level.

11.6.5 REFERENCES

1. NRC Information Notice 83-064: Lead Shielding Attached to Safety-Related Systems Without 10 CFR 50.59 Evaluations, dated September 29, 1983.

2. NRC Information Notice 90-033: Sources of Unexpected Occupational Radiation Exposures At Spent Fuel Storage Pools, dated May 9, 1990.

3. NRC Information Notice 93-039: Radiation Beams from Power Reactor Biological Shields dated May 25, 1993.

4. WE Letter to NRC, “Additional Response To NUREG-0737,” dated September 4, 1984.

5. Calculation 129187-M-0105, “Control Room Direct Shine Dose Due to Loss of Coolant Accident Following Extended Power Uprate and Using Alternate Source Term Methodology,” Revision 1, dated April 27, 2011.

6. QAD-CGGP, “A Combinatorial Geometry Version of QAD-P5A, A Point Kernel Code System for Neutron and Gamma-Ray Shielding Calculations Using the GP Buildup Factor.”

7. WE Letter to NRC, NPL-97-0315, “Supplement to Technical Specifications Change Request 192,” dated June 3, 1997.

8. Westinghouse Report, WEP-98-077, “Wisconsin Electric Power Company, Point Beach Units 1 and 2 Chapter 9 and 11 - FSAR Updates,” December 8, 1998.

9. NUREG-0737, “Clarification of TMI Action Plan Requirements,” dated October 31, 1980.

10. NRC Safety Evaluation of NUREG-0737 Item II.B.2.2, “Plant Shielding Modifications for Vital Area Access,” Point Beach Nuclear Plant, Unit Nos. 1 and 2, dated November 3, 1983.

11. Engineering Change EC 11691 (258119), Revision 1, “Addition of Control Room Shielding,” Approved March 30, 2010.

12. FPL Energy Point Beach Letter to NRC, NRC 2009-0030, “License Amendment Request 261 Extended Power Uprate,” dated April 7, 2009.

13. NRC Safety Evaluation, "Point Beach Nuclear Plant (PBNP), Units 1 and 2 - Issuance of License Amendments Regarding Extended Power Uprate (TAC Nos ME1044 and ME1045)," dated May 3, 2011.

14. NRC Safety Evaluation, "Point Beach Nuclear Plant (PBNP), Units 1 and 2 - Issuance of License Amendments Regarding Use of Alternate Source Term (TAC Nos. ME0219 and ME0220)," dated April 14, 2011.

15. Nextera Energy Point Beach Letter to NRC, NRC 2010-0042, "License Amendment Request 261 Extended Power Uprate Response to Request for Additional Information," dated May 14, 2010.

16. SCR 2011-0275, "Revise FSAR 14.3.5, Radiological Consequences of a LOCA for EPU per AR 1688483," dated October 18, 2011.

Table 11.6-1 SHIELDING DESIGN ZONE CLASSIFICATIONS

Zone	Condition of Occupancy	Maximum Dose Rate (1% failed fuel) mrem/hr
0	Unlimited occupancy	0.1
I	Normal continuous occupancy	1.0
II	Periodic occupancy	2.5
III	Controlled occupancy	15
IV	Controlled access	>15

Table 11.6-2 ORIGINAL PRIMARY SHIELD NEUTRON FLUXES AND DESIGN PARAMETERS (Historical)

Calculated Neutron Fluxes

Energy Group	Incident Fluxes ($n/cm^2/sec$)	Leakage Fluxes ($n/cm^2/sec$)
E < 1 Mev	2.2×10^9	7.5×10^2
5.3 Kev ≤ E ≤ 1 Mev	2.3×10^{10}	1.6×10^3
.625 ev ≤ E ≤ 5.3 Kev	1.4×10^{10}	2.7×10^3
E < .625 ev	1.9×10^{10}	9.8×10^5

Design Parameters

Core thermal power	1518.5 MW
Active core height	144 in.
Effective core diameter	96.50 in.
Baffle wall thickness	1.125 in.
Barrel wall thickness	1.75 in.
Thermal shield wall thickness	3.50 in.
Reactor vessel I.D.	132.0 in.
Reactor vessel wall thickness	6.50 in.
Reactor coolant cold leg temperature	559.5°F
Reactor coolant hot leg temperature	614.5°F
Maximum thermal neutron flux exiting primary concrete	10^6 $n/cm^2/sec$.
Reactor shutdown dose exiting primary concrete	<15 mR/hr

Table 11.6-3 ORIGINAL SECONDARY SHIELD DESIGN PARAMETERS (Historical)

Core power density	85 w/cc
Reactor coolant liquid volume	6450 ft^3
Reactor coolant transit times:	
Core	0.9 sec.
Core exit to steam generator inlet	2.0 sec.
Steam generator inlet channel	0.6 sec.
Steam generator tubes to vessel inlet	2.6 sec.
Vessel inlet to core	2.2 sec.
Total out of core	10.6 sec.
Full power dose rate outside secondary shielding	<1 mR/hr.

Table 11.6-4 ORIGINAL ACCIDENT SHIELD DESIGN PARAMETERS (Historical)

Core thermal power	1518.5MW
Minimum full power operating time	1000 days
Equivalent fraction of core melting	1.0
Fission product fractional releases:	
Noble gases	1.0
Halogens	0.5
Remaining fission product inventory	0.01
Clean up rate following accident	0
Maximum integrated dose (infinite exposure) in the control room	<2 rem

Table 11.6-5 ORIGINAL PRINCIPAL AUXILIARY SHIELDING (Historical)

Component	Concrete Shield Thickness, Ft. - In.
Demineralizers	4 - 0
Charging pumps	2 - 2
Liquid holdup tanks	2 - 6
Volume control tank	3 - 6
Reactor coolant filter	2 - 9
Gas stripper	2 - 6
Gas decay tanks	3 - 6
Gas compressor	3 - 0
Waste evaporator	2 - 0
Liquid waste holdup tank	2 - 0

Design parameters for the auxiliary shielding include:

Core thermal power	1518.5 MWt
Fraction of fuel rods containing small clad defects	0.01
Reactor coolant liquid volume	6450 ft^3
Letdown flow (normal purification)	40 gpm
Effective cesium purification flow (intermittent)	4.0 gpm
Cut-in concentration deborating demineralizer	160 ppm
Dose rate outside auxiliary building	1 mR/hr
Dose rate in the building outside shield walls	2.5 mR/hr

Table 11.6-6 ORIGINAL RESIDUAL HEAT REMOVAL SYSTEM RADIATION SOURCES AND EVALUATION PARAMETERS (Historical)

Radiation Sources - MEv/cc-sec

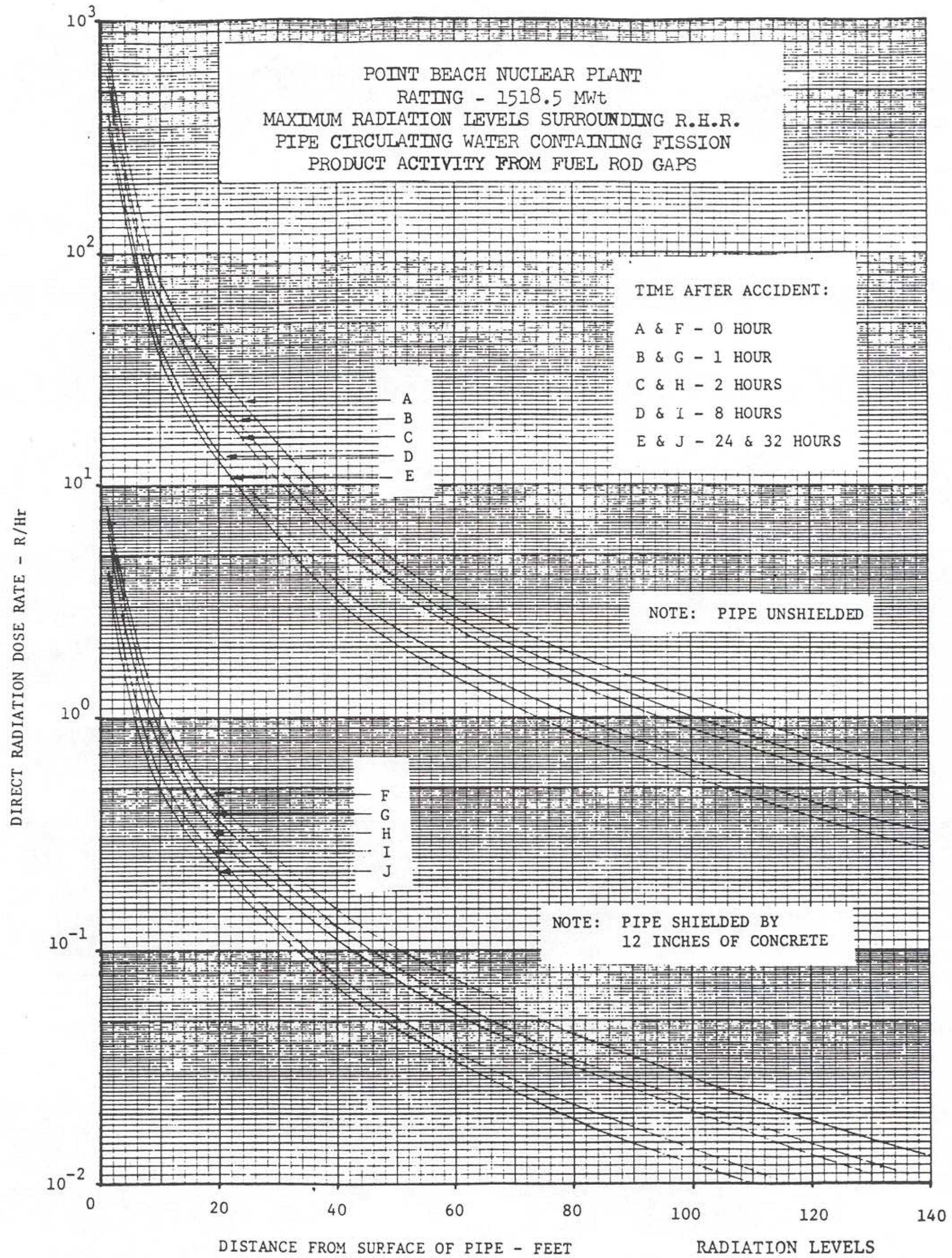
Energy	Time After Release					
Mev	0	1 hr	2 hrs	8 hrs	24 hrs	32 hrs
0.4	6.04+7	8.88+6	8.28+6	7.08+6	6.60+6	6.48+6
0.8	8.28+7	7.32+7	6.60+7	4.92+7	4.47+7	4.35+7
1.3	5.46+6	3.95+6	2.94+6	4.57+5	3.09+4	1.54+4
1.7	3.13+6	2.21+6	1.62+6	2.65+5	1.17+4	1.13+4
2.2	2.94+6	2.34+6	1.92+6	3.59+5	1.12+5	5.40+4
2.5	1.38+6	9.06+5	6.55+5	9.90+4	4.56+4	3.14+4

Note: 1.04+7=1.04 $\times 10^7$

Evaluation Parameters

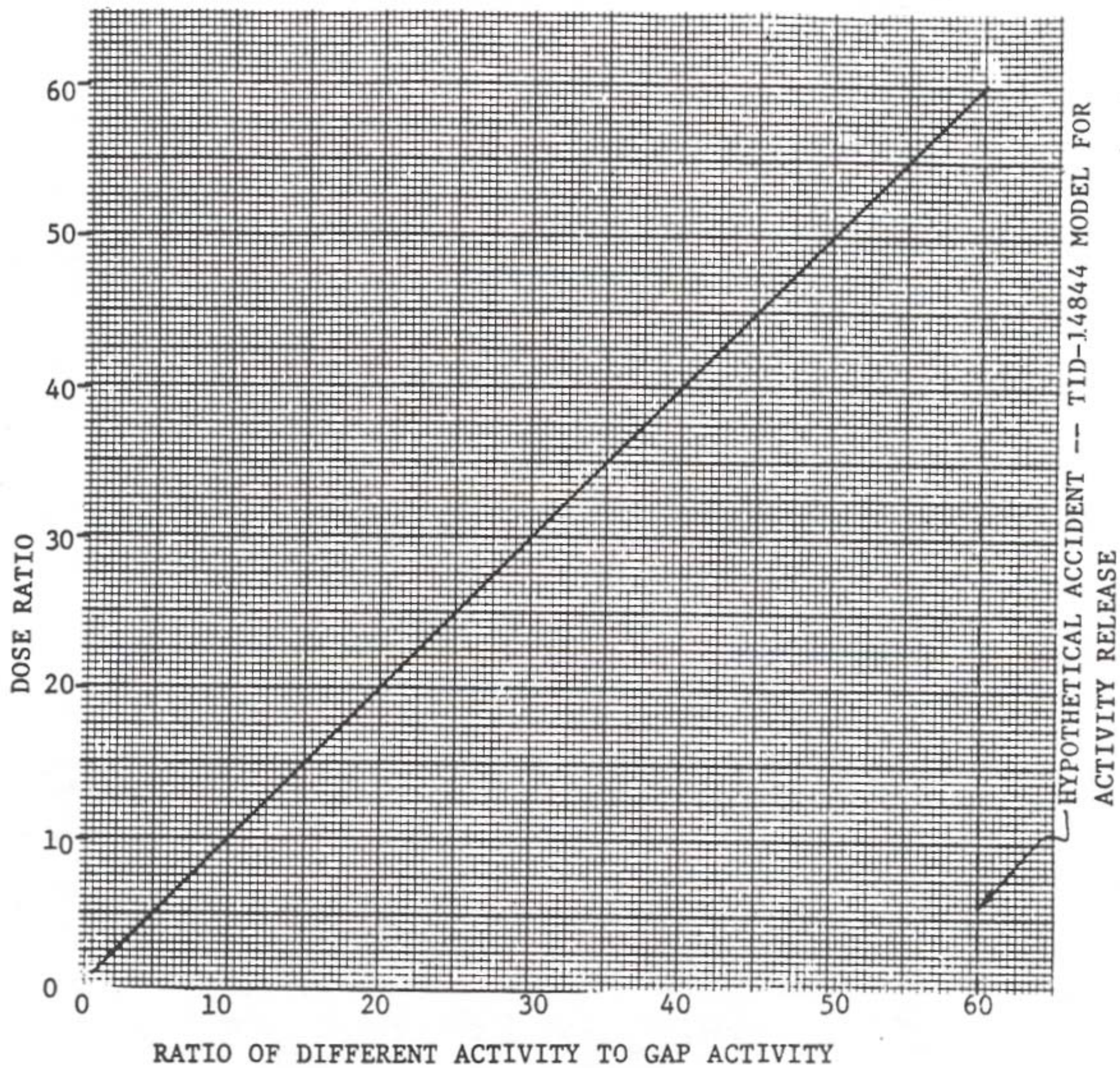
Core thermal power, MWt	1518.5
Percent of gap activity absorbed by the sump water	
Noble gases	0
All others	100
Fission product clean-up rate	0
Reactor coolant volume, ft^3	6450
Refueling water volume, ft^3	38,100

Figure 11.6-1 MAXIMUM RADIATION LEVELS SURROUNDING 14 IN. DIAMETER R.H.R. PIPE CIRCULATING WATER CONTAINING FISSION PRODUCT ACTIVITY FROM FUEL ROD GAPS (Historical)



RADIATION LEVELS
Figure 11.6-1

Figure 11.6-2 SENSITIVITY OF DOSE TO ACTIVITY IN THE RESIDUAL HEAT REMOVAL WATER (Historical)



11.7 EQUIPMENT AND SYSTEM DECONTAMINATION

11.7.1 CONTAMINATION SOURCES

Activity outside the core could result from fission products from defective fuel elements, fission products from tramp uranium left on the cladding in small quantities during fabrication, products of n - γ or n - p reactions on the water or impurities in the water, and activated corrosion products. Fission products in the reactor coolant associated with normal plant operation and tramp uranium are generally removed with the coolant or in subsequent flushing of the system to be decontaminated. The products of water activation are not long lived and may be removed by natural decay during reactor cooldown and subsequent flushing procedures. Activated corrosion products are the primary source of the remaining activity.

The corrosion products contain radioisotopes from the reactor coolant which have been absorbed on, or have diffused into, the oxide film. The oxide film, essentially magnetite (Fe_3O_4) with oxides of Cr and Ni, can be removed by chemical means presently used in industry.

Water from the primary coolant system and the spent fuel pool is the primary potential source of contamination outside of the corrosion film of the primary coolant system. The contamination could be spread by various means when access is required. Contact while working on primary system components could result in contamination of the equipment, tools and clothing of the personnel involved in the maintenance. Also, leakage from the system during operation or spillage during maintenance could contaminate the immediate areas and could contribute to the contamination of the equipment, tools, and clothing.

11.7.2 METHODS OF DECONTAMINATION

Surface contaminants which are found on equipment in the primary system and the spent fuel pool that are in contact with the water are removed by conventional techniques of flushing and scrubbing as required. Tools are decontaminated by flushing and scrubbing since the contaminants are generally on the surface only of nonporous materials. Personnel and their clothing are decontaminated according to the standard health physics requirements.

Those areas of the plant which are susceptible to spillage of radioactive fluids are painted with a sealant to facilitate decontamination that may be required. Generally, washing and flushing of the surfaces are sufficient to remove any radioactivity present.

The corrosion films generally are tightly adhering surface contaminants, and must be removed by chemical processes. The removal of these films is generally done with the aid of commercial vendors who provide both services and formulations. Since decontamination experience with reactors is continually being gained, specific procedures may change for each decontamination case. For corrosion films, the APAC (alkaline permanganate-diammonium citrate) treatment, or an organic acid variation of the APAC treatment, is considered to be the most effective for removal.

Portable components may be cleaned with a combination of chemical and ultrasonic methods if required.

11.7.3 DECONTAMINATION FACILITIES

Decontamination facilities on site consist of an equipment cleaning room in the machine shop and a cask pit located adjacent to the spent fuel storage pool. These facilities are shared by Units 1 and 2.

In the cask decontamination pit, the outside surfaces of the shipping casks are decontaminated, if required, by using water detergent solutions and manual scrubbing to the extent required. When the outside of the casks are decontaminated, the casks are removed from the pit area by the auxiliary building crane, loaded on an approved shipping trailer, and shipped offsite.

In the equipment cleaning room, located in the machine shop area, small equipment and tools can be decontaminated by using water, detergent solutions, and manual scrubbing to the extent required and by ultrasonic techniques.

The decontamination pit is also used for decontamination of the spent fuel dry storage casks used for temporary storage of spent fuel at the Point Beach Independent Spent Fuel Storage Installation. The outside surfaces of the storage cask are decontaminated using a high pressure water spray and if required a detergent solution. When the outside of the storage cask is decontaminated, the storage cask is removed from the decontamination pit, loaded into an overpack, and moved to the ISFSI.

For personnel, a decontamination shower and washroom are located adjacent to the potentially contaminated (controlled) area locker room. An emergency decontamination shower for highly contaminated personnel is located in the health physics station.

11.8 RADIOACTIVE MATERIALS SAFETY

11.8.1 MATERIALS SAFETY

Procedures are implemented at Point Beach Nuclear Plant to assure safe storage, handling, and use of sealed and unsealed source, special nuclear, and by-product materials.

Special Nuclear Material

To minimize the possibility of diversion of special nuclear material for unauthorized use, to detect any potential diversion as quickly and accurately as possible, and to provide information for fuel management purposes, responsibilities and procedures for handling special nuclear material are provided in the Point Beach Administrative procedures. Primary responsible groups include Reactor Engineering, Radiation Protection, Operations, and Maintenance Instrumentation and Control.

In addition, both normal plant procedures and policies under the control of Chemistry and Radiation Protection address responsibilities for the radiological control of special nuclear materials for receipt, handling, storage, surveillance, and shipment.

The licensing basis for these materials and their storage is based upon the requirements of 10 CFR 50.68. This license basis was approved by the NRC in March of 2010. (Reference 1)

By-product Materials

Responsibilities and procedures for the handling, transfer, storage, and use of radioactive by-product materials are provided in the Chemistry and Radiation Protection groups procedures. Compliance with all applicable regulations and conditions is assured by the Chemistry and Radiation Protection groups.

All non-exempted sealed sources are leak tested as required by the PBNP Technical Requirements Manual (TRM), Section 3.7.4.

Radioactive source inventories and methods for leak testing radioactive sources, shipment and receipt of radioactive materials, and control and accountability of radioactive materials are discussed in policy documents and procedures under the control of Chemistry and Radiation Protection.

11.8.2 REQUIRED MATERIALS

By-product, source, and special nuclear materials are retained by Point Beach Nuclear Plant in amounts required for reactor operation in the form of reactor fuel, startup sources, neutron flux detectors, and sealed sources for calibration of reactor instrumentation and radiation monitoring equipment. In addition, certain by-product, source, and special nuclear materials are required for calibration of other plant instrumentation not directly associated with reactor operation. These sources and by-product material are discussed in and controlled under applicable Chemistry and Radiation Protection procedures.

11.8.3 REFERENCE

1. NRC Safety Evaluation, "Point Beach Nuclear Plant, Units 1 and 2 - Issuance of Amendments Re: Spent Fuel Pool Storage Criticality Control," dated March 5, 2010.