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9.0 AUXILIARY AND EMERGENCY SYSTEMS

The auxiliary and emergency systems are supporting systems required to insure the safe operation or servicing of the reactor coolant system (described in Section 4.0). Various components in some of these systems are shared by Unit l and Unit 2. Appendix A6 discusses this sharing and lists the shared components.

In some cases, the dependable operation of several systems is required to protect the reactor coolant system by controlling system conditions within specified operating limits. Certain systems are required to operate under emergency conditions.

This section considers systems in which component malfunctions, inadvertent interruptions of system operation, or a partial system failure must be designed for, to prevent a hazardous or unsafe condition.

The systems considered under this category are:

Chemical and Volume Control System: This system provides for boric acid injection, chemical additions for corrosion control, reactor coolant cleanup and degasification, reactor coolant makeup, reprocessing of water letdown from the reactor coolant system, and reactor coolant pump seal water injection.

Residual Heat Removal System: This system removes the residual heat from the core and reduces the temperature of the reactor coolant system during the second phase of plant cooldown.

Spent Fuel Cooling System: This system removes the heat generated by spent fuel elements stored in the spent fuel pool.

Component Cooling System: This system removes heat from the reactor coolant system, via the residual removal system during plant shutdown, cools the letdown flow to the chemical and volume control system during power operation and provides cooling to dissipate waste heat from various primary plant components.

Sampling System: This system provides the equipment necessary to obtain liquid and gaseous samples from the reactor plant systems.

Facility Service Systems: These systems include fire protection and service water systems.

Fuel Handling System: This system provides for handling fuel assemblies, control rod assemblies, and material irradiation specimens.

Missile Protection

Criterion: Adequate protection for those engineered safety features, the failures of which could cause an undue risk of the health and safety of the public, shall be provided against dynamic effects and missiles that might result from plant equipment failures. (GDC 40)

This plant-specific General Design Criteria is very similar to 10 CFR 50 Appendix A GDC 4. Under the provisions of that criterion, the dynamic effects associated with postulated pipe ruptures of the RCS may be excluded from the design basis when appropriate analyses approved by the NRC demonstrate that the probably of such ruptures is extremely low. Analyses have been completed for PBNP for the RHR return line, and the RHR suction line (Reference 1 and Reference 2). The NRC has approved the analyses (Reference 3, Reference 4 and Reference 5). As such, the original design features of the facility to accommodate the dynamic effects of a pipe rupture are no longer applicable for these lines.

9.0.1 GENERAL DESIGN CRITERIA

Criteria which are specific to one of the auxiliary or emergency systems are listed and discussed in the appropriate system design basis subsection. Criteria which apply primarily to other systems (and are discussed in other sections) are also listed and cross-referenced below because details of closely related systems and equipment are given in this section.

Reactivity Control Systems Malfunction

Criterion: The reactor protection systems shall be capable of protecting against any single malfunction of the reactivity control system, such as unplanned continuous withdrawal (not ejection or dropout) of a control rod, by limiting reactivity transients to avoid exceeding acceptable fuel damage limits. (GDC 31)

As described in Section 7.0 and justified in Section 14.0, the reactor protection systems are designed to limit reactivity transients to DNBR no less than the design basis limit due to any single malfunction in the deboration controls.

Engineered Safety Features Performance Capability

Criterion: Engineered Safety Features, such as the emergency core cooling system and the containment heat removal system, shall provide sufficient performance capability to accommodate the failure of any single active component without resulting in undue risk to the health and safety of the public. (GDC 41)

Each of the auxiliary cooling systems which serves an emergency function provides sufficient capability in the emergency mode to accommodate any single failure of an active component and still function in a manner to avoid undue risk to the health and safety of the plant personnel and the public.

Containment Heat Removal Systems

Criterion: Where an active heat removal system is needed under accident conditions to prevent exceeding containment design pressure, this system shall perform its required function, assuming failure of any single active component. (GDC 52)

Each of the auxiliary cooling systems, which serves an emergency function to prevent exceeding containment design pressure, provides sufficient capability in the emergency mode to accommodate any single failure of an active component and still perform its required function.

REFERENCES

1. WCAP-15107-P-A, Revision 1 "Technical Justification for Eliminating Accumulator Lines Rupture as the Structural Design Basis for Point Beach Units 1 and 2 Nuclear Plants" (Proprietary) dated June 1, 2001.

2. WCAP-15105-P-A, Revision 1 "Technical Justification for Eliminating Residual Heat Removal (RHR) Lines Rupture as the Structural Design Basis for Point Beach Units 1 and 2 Nuclear Plants" (Proprietary) dated June 1, 2001.

3. NRC SER 2000-0011 "Safety Evaluation of the Request to Apply Leak-Before-Break Status to the Accumulator Line Piping at PBNP, Units 1 and 2," dated November 7, 2000.

4. NRC SER 2000-0014, "Safety Evaluation of the Request to Apply Leak-Before-Break Status to Portions of the Residual Heat Removal Piping at Point Beach Nuclear Plant, Units 1 and 2," December 18, 2000.

5. NRC SER 2000-0011/14/15-S1, PBNP, Units 1 and 2, Supplement to Safety Evaluation on Leak-Before-Break Regarding Correction of Leak Detection Capability," February 7, 2005.

9.1 COMPONENT COOLING WATER (CC)

The component cooling water system consists of four pumps, four heat exchangers, two surge tanks and the piping, valves, instrumentation, and controls necessary to provide the heat removal capability to support the operation of the units and equipment. The component cooling water loop in each unit consists of two pumps (P-11A&B), two heat exchangers (HX-12A&B in Unit 1 and HX-12C&D in Unit 2), a surge tank (T-12), a supply header, and a return header. Heat exchangers HX-12B&C may be used in either loop as cooling conditions require. The capability to use the pumps assigned to one loop to supply both loops is also provided.

9.1.1 DESIGN BASIS

The CC system performs the following safety-related functions (Reference 8, Reference 9, Reference 10, and Reference 11):

- Remove residual and sensible heat from the reactor coolant system, via the residual heat removal (RHR) heat exchangers during the recirculation phase of Safety Injection to support long-term core cooling.
- Remove heat from the RHR heat exchangers to terminate the steam releases associated with the license basis dose analyses for the postulated rupture of a steam pipe (MSLB; Chapter 14.2.5), steam generator tube rupture (SGTR; Chapter 14.2.4), and reactor cooling pump locked rotor (Chapter 14.1.8) accidents.
- Remove heat from the RHR, SI, and containment spray pump seal coolers to maintain the integrity of the pump seals.
- Preclude leakage of the containment atmosphere into the CC system piping to limit the release of radioactive materials.

The CC system is credited in the event of a fire and has been evaluated in the at-power and non-power analyses (Reference 3).

- In the event of plant fires, including those requiring evacuation of the control room, the CC system shall be capable of supporting plant cool down to cold shutdown conditions within 72 hours post-fire. Repairs of fire damage may be performed to support this function (Reference 3).

9.1.2 SYSTEM DESIGN AND OPERATION

Normally the component cooling loops of each of the two units operate independently such that two component cooling pumps and one component cooling heat exchanger, HX-12A in Unit 1 and HX-12D in Unit 2, are available for use, and two heat exchangers, HX-12B&C, serve as shared standby units. The description contained herein applies to the component cooling loop of one unit operating independently of the component cooling loop of the second unit. The sharing of components is discussed further in Appendix A.6 (Reference 1).

One pump and one heat exchanger are normally operated to provide cooling water for various components located in the auxiliary and containment buildings. The second pump is in standby and will auto start on low discharge pressure and a second heat exchanger is normally aligned to the unit with CC flow cut in and service water isolated at the discharge. The automatic start function of the CC pumps on low pressure is provided for operational convenience and is not relied upon by the safety analyses to mitigate accidents or events (Reference 16). During the recirculation phase following a loss-of-coolant accident, either or both component cooling water pumps deliver flow to the shell side of the RHR heat exchangers. With the exception of the CC supply to the RHR heat exchangers, cooling water is normally supplied to components served by CC even though a component may be out of service.

Component cooling is provided for the following components:

1. Residual heat exchangers (RHR)
2. Reactor coolant pumps (RCS)
3. Nonregenerative heat exchanger (CVCS)
4. Excess letdown heat exchanger (CVCS)
5. Seal water heat exchanger (CVCS)
6. Sample heat exchangers (SG blowdown and SC)
7. Waste evaporator (WDS) (abandoned in place)
8. Waste gas seal water heat exchangers (WDS)
9. Residual heat removal pumps (RHR)
10. Safety injection pumps (SI)
11. Containment spray pumps (SI)
12. Blowdown Evaporator
13. Cryogenic gas compressors
14. Cryogenic after coolers (abandoned in place)
15. Letdown gas stripper condensers

Although component cooling water piping is still in place for Items 7 Waste evaporator (WDS) and 14 Cryogenic after coolers, these components are considered abandoned and therefore do not require component cooling water.

Makeup water is normally taken from the plant makeup water system by manual valve operation as required and delivered to the component cooling surge tank via the surge line. An emergency backup source of water is provided from the reactor makeup water tank by remote operation of a motor operated valve.

The operation of the loop is monitored with the following instrumentation:

1. A temperature detector in the outlet line for the component cooling heat exchangers
2. A pressure detector on the line between the component cooling pumps and the component cooling heat exchangers
3. A flow indicator in the outlet line from the component cooling heat exchangers
4. A radiation monitor on the return header to the component cooling pumps.

The component cooling loop serves as an intermediate system between the reactor coolant and service water systems during cooldown, transferring heat from the reactor coolant to the service water system. This double barrier arrangement reduces the potential for leakage of radioactive reactor coolant to the service water system.

During normal full power operation, one component cooling pump and one component cooling heat exchanger accommodate the heat removal loads and the standby pump and the shared heat exchangers provide 100% backup. Two pumps and two heat exchangers are used to remove the residual and sensible heat during plant shutdown. If one of the pumps or two of the heat exchangers are not operable, safe shutdown of the plant is not affected; however, the time for cooldown is extended.

The component cooling surge tank accommodates expansion, contraction, and inleakage of water, providing a reservoir for continuous component cooling water supply until either a leaking cooling line can be isolated, or system make-up can be initiated. System overpressure protection is provided by a relief valve. A radiation monitor in the component cooling system return header annunciates in the control room and closes the surge tank vent valve in the unlikely event that the radiation level reaches a preset level above the normal background.

The component cooling system branches to and from the above listed Radwaste Equipment (Items 12, Blowdown Evaporator through 15, Letdown gas stripper condensers) are seismic Class III piping. Class change isolation valves are provided for both the supply and return headers; equipped with control room control, control room indication, and a containment isolation actuation signal. This provides the class break isolation required by Appendix A.5 (Reference 2).

Component Cooling Loop Components

Several of the components in the component cooling loop are fabricated from carbon steel. The component cooling water contains a corrosion inhibitor to protect the carbon steel. Welded joints and connections are used except where flanged closures are employed to facilitate maintenance. The system is Seismic Class I design with the exception of CC branch lines to radwaste equipment. Design parameters for the component cooling loop components are presented in Table 9.1-1

In addition, the components are not subjected to any high pressure or stresses, hence a rupture or failure of the system is very unlikely. Active components which are relied upon to perform the emergency core cooling function are redundant. The design provides for detection of radioactivity and also provides for isolation means.

The component cooling water system is normally aligned such that Unit 1 and Unit 2 have hydraulically independent systems. However, the CC systems for each unit were designed with the capability to be hydraulically connected under abnormal conditions. Therefore, CC loop components are discussed below at a plant-level.

Component Cooling Heat Exchangers

Four component cooling heat exchangers are of the shell and straight tube (fixed tubesheet) type. Service water circulates through the tubes while component cooling water circulates through the shell side. Normally one heat exchanger is used with each unit with the two idle heat exchangers serving as standby units. The standby unit alignment is CC water cut in and service water isolated at the discharge. The tubes are SA-268 (26-3-3) metal and the shells are carbon steel.

Component Cooling Pumps

The four component cooling pumps, which circulate component cooling water through the component cooling water system, are horizontal, centrifugal units. The pump casings are made from cast iron (ASTM A48) or carbon steel (ASTM A216) based on the corrosion-erosion resistance and the ability to obtain sound castings. The material thickness is dictated by high quality casting practice and ability to withstand mechanical damage and as such are substantially overdesigned from a stress level standpoint. Normally two pumps are designated to each unit, but a crosstie may be opened under abnormal conditions to allow unit-designated pump(s) to supply both units.

Component Cooling Surge Tank

The component cooling surge tank, one per each unit, accommodates changes in component cooling water volume and is constructed of carbon steel. Potassium Chromate is added to the component cooling loops to prevent corrosion.

Component Cooling Valves

The valves used in the component cooling loop are constructed of carbon steel with bronze or stainless steel trim. Since the component cooling water is not normally radioactive, special valve features (such as leakoff connections to the waste disposal system) to prevent leakage to the atmosphere are not provided.

Several component cooling water air operated valves are provided with Close/Auto/Open switches which provide the following control functions when in Auto: (1) the excess letdown HX cooling water outlet valve (CC-769) closes on a containment isolation signal, (2) the “A” and “B” reactor coolant pump thermal barrier cooling water outlet valves (CC-761A&B) close on a high flow signal, and (3) the radwaste component cooling water supply and return valves (LW-63&64) close on a Unit 2 containment isolation signal. The auto closure of LW-63&64 enhances CC system integrity. This feature is not credited for the mitigation of any analyzed accident. Manual action, including completely securing the CC system for repairs, is the analyzed method for CC system restoration, even under DBA conditions (Reference 16). The non-regenerative HX cooling water flow control valve (CC-130) has an Auto/Manual switch which, when in Auto, controls letdown temperature. This is a fail open valve with a manual gag red-locked to limit flow.

Self-actuated spring loaded relief valves are provided for piping and components that could be pressurized to their design pressure.

Component Cooling Piping

All component cooling loop piping is carbon steel with welded joints and connections except at certain components where flanged connections are used to facilitate maintenance. All component cooling lines inside containment have been analyzed for protection from missiles, pipe whip, and jet impingement. Using Leak-Before-Break methodology, no credible missiles exist and therefore the component cooling piping is considered missile protected. The component cooling water system is considered a closed system inside containment with respect to containment isolation capability (Reference 5, Reference 6, Reference 7, Reference 8, Reference 12, and Reference 13).

9.1.3 SYSTEM EVALUATION

For continued cooling of the reactor coolant pumps and the excess letdown heat exchanger inside the containment, most of the piping, valves, and instrumentation are located outside the primary system concrete shield at an elevation well above the anticipated post-accident water level in the bottom of the containment. Cooling lines and equipment in the annular area near the reactor coolant pumps are protected against credible missiles and from being flooded during post-accident operation. Also, this location provides radiation shielding which allows maintenance and inspections to be performed during power operation. The only component cooling water lines that are not shielded by the primary system concrete shield wall are the cooling lines near the reactor coolant pumps. These lines have been analyzed for protection from missiles, pipe whip, and jet impingement. Using Leak-Before-Break methodology, no credible missiles exist and therefore, the component cooling piping is considered missile protected (Reference 5, Reference 6, Reference 7, Reference 8, Reference 12 and Reference 13). Additionally, system leakage would be detected and these lines can be isolated, if necessary, by two valves in series.

Outside the containment, the component cooling pumps and heat exchangers, and associated valves, piping and instrumentation can be maintained and inspected during power operation. System design provides for the replacement of one pump or one heat exchanger while the other units are in service.

Welded construction is used where possible throughout the component cooling loop piping, valves and equipment to minimize the possibility of leakage.

The component cooling water could become contaminated with radioactive water due to a leak in any heat exchanger tube in the chemical and volume control system, the sampling system, the residual heat removal system, or a leak in the cooling coil for the reactor coolant pump thermal barrier. Tube or coil leaks in components being cooled would be detected during normal plant operation by the leak detection system described in Section 6.5. Such leaks are also detected by a radiation monitor located on the main return header.

Should a large tube-side to shell-side leak develop in a residual heat exchanger, the water level in the component cooling surge tank would rise or fall, depending on RCS conditions, and the operator would be alerted by a level alarm. The atmospheric vent on the tank is automatically closed (if the vent was open) in the event of high radiation level at the component cooling water pump suction header. If the leaking residual heat exchanger is not isolated from the component cooling loop before the inflow completely fills the surge tank, the relief valve on the surge tank lifts. The discharge of this relief valve is routed to the auxiliary building waste holdup tank.

The relief valves on the component cooling water header downstream from each of the reactor coolant pumps are designed with a capacity equal to the maximum rate at which reactor coolant can enter the component cooling loop from a severance type break of the reactor coolant pump thermal barrier cooling coil. The valve set pressure is less than or equal to the design pressure of the component cooling piping.

The relief valves on the cooling water lines downstream from the sample, excess letdown, seal water, nonregenerative and residual heat exchangers are sized to relieve the volumetric expansion occurring if the heat exchanger shell side is isolated when cool, and high temperature coolant flows through the tube side. The set pressure is less than or equal to the design pressure of the shell side of the heat exchangers.

The relief valve on the component cooling surge tank is sized to relieve the maximum flow rate of water which enters the surge tank following a rupture of a reactor coolant pump thermal barrier cooling coil.

Leakage from the component cooling loop can be detected by a falling level in the component cooling surge tank. The rate of water level decrease and the area of the water surface in the tank permit determination of the leakage rate. Normal makeup is from the demineralized water header through a manual valve at the tank. Emergency makeup is from reactor makeup water through an isolation valve which is remotely operated from the control room. The component which is leaking can be located by sequential isolation or inspection of equipment in the loop. If the leak is in the on-line component cooling water heat exchanger, a standby exchanger would be placed in service and the leaking heat exchanger isolated. Two standby heat exchangers can be used on either unit in the event the normal heat exchanger develops a leak during a high heat load period when two heat exchangers are desired.

Each of the cooling water supply lines to the reactor coolant pumps contains a check valve inside and a remotely operated valve outside the containment. The cooling water supply line to the excess letdown heat exchanger contains a check valve inside and a manual valve outside the containment. The common supply line to both the reactor coolant pumps and the excess letdown heat exchanger also contains a remotely operated valve outside containment. Additionally, these lines have been analyzed for protection from missiles, pipe whip, and jet impingement. Using Leak-Before-Break methodology, no credible missiles exist and therefore, the component cooling piping is considered missile protected (Reference 5, Reference 6, Reference 7 and Reference 8).

Except for the normally closed makeup lines and equipment vent and drain lines, there are no direct connections between the component cooling water and other systems. The equipment vent and drain lines outside the containment have manual valves which are normally closed unless the equipment is being vented or drained for maintenance or repair operations. The vent lines are also capped when not in use as an additional leakage protection feature.

Following a loss-of-coolant accident, one component cooling pump and one component cooling heat exchanger can accommodate the heat removal loads. If either a component cooling pump or component cooling heat exchanger fails, the standby pump and one of two standby heat exchangers provide 100% backup. Valves in the component cooling supply and return lines for the safety injection, containment spray, and residual heat removal pump seal coolers are normally open. However, each of the component cooling inlet lines to the residual heat exchangers has a normally closed remotely operated valve. If one of those valves fails to open at initiation of long-term recirculation, the other RHR heat exchanger will be provided with sufficient cooling to remove the heat load.

Additional information concerning component cooling design requirements during a loss-of-coolant accident can be found in FSAR Section 6.2, Safety Injection System.

A break of a component cooling line occurring outside the containment could either be isolated and repaired, or the system could be shutdown for repairs, depending on the location at which the break occurred. Although it is anticipated engineering controls would be required, access is available to required components. Once the leak is isolated or the break has been repaired, makeup water is supplied from the plant makeup water system or by the reactor makeup water tank. During the recirculation phase of an accident, repairs to the component cooling system (loss) would not significantly impair reactor core cooling if, Containment Fan Coolers operate to remove containment heat, and core decay heat is transferred to the containment atmosphere by coolant boiling.

The normal power supplies for the component cooling water pumps P-11A and P-11B are safety-related 480 volt buses B-03 and B-04 respectively. In the event of a loss of off-site power without a coincident safety injection signal, at least one CC pump will be automatically started immediately when power is restored to the safeguards buses. If the loss of off-site power is coincident with a safety injection signal, automatic starting of the CC pumps will be blocked on the unit with the safety injection signal. The CC pumps are anticipated to be operating for the recirculation phase of an accident, with the alignment accomplished by operator action. The pumps also have a designated alternate source of power via B-08 or B-09 and an electrical disconnect switch. The alignment requires alternate power supply cables to be run from the disconnect switches to the pump motors.

A failure analysis of pumps, heat exchangers, and valves is presented in Table 9.1-2.

9.1.4 REQUIRED PROCEDURES AND TESTS

The active components of the component cooling system are in either continuous or intermittent use during normal plant operation. Periodic visual inspections and preventive maintenance can be conducted as necessary without interruption of cooling system operation. The inservice testing requirements are described in the PBNP Inservice Testing Program and the IST Background Document. The Closed-Cycle Cooling Water System Surveillance Program (FSAR Section 15.2.10) will be implemented during the period of extended operation (NRC SE dated 12/2005, NUREG-1839)

9.1.5 REFERENCES

1. PBNP FSAR Appendix A.6, Shared Systems Analysis

2. PBNP FSAR Appendix A.5, Seismic Design Analysis

3. NFPA 805 Fire Protection Program Design Document (FPPDD).

4. Not Used

5. “Safety Evaluation of the Request to Apply Leak-Before-Break Status to the Accumulator Line Piping at Point Beach Nuclear Plant, Units 1 and 2,” November 7, 2000.

6. “Safety Evaluation of the Request to Apply Leak-Before-Break Status to Portions of the Residual Heat Removal Piping at Point Beach Nuclear Plant, Units 1 and 2,” December 18, 2000.

7. “Safety Evaluation of the Request to Apply Leak-Before-Break Status to the Pressurizer Surge Line Piping at Point Beach Nuclear Plant, Units 1 and 2,” December 15, 2000.

8. PBNP SE 2001-007, “Component Cooling Water System Closed Loop Inside Containment,” February 24, 2001.

9. VPNPD-92-378, B. Link to NRC, "Classification of Auxiliary Systems Necessary to Assure Safe Plant Shutdown at Point Beach, Units 1 and 2," December 22, 1992.

10. VPNPD-93-115, B. Link to NRC, "Classification of Auxiliary Systems Necessary to Assure Safe Plant Shutdown at Point Beach, Units 1 and 2," June 17, 1993.

11. NPL 97-0401, D.F. Johnson to NRC, "Component Cooling Water System Issues Update, Point Beach Nuclear Plant, Units 1 and 2," July 7, 1997.

12. "PBNP, Units 1 and 2, Issuance of Amendments Re: Leak-Before-Break Evaluation for Primary Loop Piping, (TAC NOS. MC1279 and MC1280)" June 6, 2005.

13. "PBNP, Units 1 and 2, Supplement to Safety Evaluation on Leak-Before-Break Regarding Correction of Leak Detection Capability," February 7, 2005.

14. Not Used.

15. Not Used.

16. "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 201 to Facility Operating License No. DPR-24 and Amendment 206 to Facility Operating License No. DPR-27 NMC, LLC PBNP, Units 1 and 2" dated August 8, 2001.

17. Calculation CN-SEE-III-08-10, "Point Beach Units 1 and 2 RHRS Cooldown Analysis for EPU to 1806 MWT NSSS Power," approved February 1, 2013.

Table 9.1-1 COMPONENT COOLING SYSTEM COMPONENT DATA

Component Cooling Pumps	
Quantity	4*
Type	Horizontal centrifugal
Nominal flow rate (each), gpm	3650
Total developed head, ft H_2O	220
Motor horsepower, hp	250
Casing material	Cast iron or Carbon steel
Design pressure, psi	250
Design temperature, °F	250
Component Cooling Heat Exchangers	
Quantity	4**
Type	Fixed tube sheet, horizontal
Design heat transfer, BTU/hr.	50.1×10^6 ******
Code Requirements	ASME VIII ****
Shell side (component cooling water)	
Design inlet Temp., °F	158 ******
Design flow rate, lb/hr	1.346×10^6 ******
Design temperature, °F	200
Design pressure, psig	250
Material	Carbon steel
Tube side (service water)	
Design inlet temperature, °F	85 ******
Design flow rate, lb/hr	1.446×10^6 ******
Design pressure, psig	150
Design temperature, °F	200
Material	SA-268 (26-3-3,Trent Sea-Cure)
Component Cooling Surge Tank	
Quantity (per unit)	1
Volume, gal.	2000
Normal water volume, gal.	1000
Design pressure (internal), psig	100
Design pressure (external), psig	Vacuum breaker provided
Design temperature, °F	200
Material	Carbon steel
Code Requirement	ASME VIII ****
Component Cooling Loop Piping and Valves	
Design pressure, psig	150
Design temperature, °F	200
Code Requirement	USAS B31.1 *****

* Two pumps are normally used with each unit. The capability exists for sharing of the 4 pumps between the two units.

** One heat exchanger is normally used with each unit. Two heat exchangers are used as shared standby units.

**** ASME VIII - American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section VIII.

***** USAS B31.1 - Code for Pressure Piping, and special nuclear cases where applicable.

****** The heat transfer value is the design basis value that is used to ensure that the CC HXs can perform their design function. They are dependent on the denoted fluid mass flowrates and inlet temperatures. They are determined in accordance with Reference 17.

Table 9.1-2 FAILURE ANALYSIS OF PUMPS, HEAT EXCHANGERS, AND VALVES

	Components	Malfunction	Comments and Consequences
1.	Component cooling water pumps	Rupture of a pump casing	The casing is designed for 250 psi and 250°F which exceeds maximum operating conditions. Pump is inspectable and protected against missiles. Rupture due to missiles is not considered credible. Each unit is isolable. The second unit can carry the total decay heat load.
2.	Component cooling water pumps	Pump fails to start	One operating pump supplies sufficient water for emergency cooling.
3.	Component cooling water pumps	Manual valve on a pump suction line closed	This is prevented by prestartup and operational checks. Further, during normal operation, each pump is checked on a periodic basis which would show if a valve is closed.
4.	Component cooling water pumps	Valve on discharge line sticks closed	The valve is checked open during periodic operation of the pumps during normal operation.
5.	Component cooling heat exchanger	Tube or shell rupture	Rupture is considered improbable because of low operating pressures. Each unit is isolable. Either of two standby units can carry total emergency heat load.
6.	Demineralized water makeup line check valve	Sticks open	The check valve is backed up by the manually operated valve. The manual valve is normally closed.
7.	Component cooling heat exchanger vent or drain valve	Left open	This is prevented by prestartup and operational checks. On the operating unit such a situation is readily assessed by makeup requirements to system. On the standby unit such a situation is ascertained during periodic testing or startup.
8.	Component cooling water valve to residual heat exchanger	Fails to open	There is one valve on each inlet line to each heat exchanger. One heat exchanger remains in service and provides adequate heat removal during long term recirculation. During normal operation the cooldown time is extended.

Figure 9.1-1 UNIT 1 AUXILIARY COOLANT SYSTEM

[redacted]

Figure 9.1-2 UNIT 1 AUXILIARY COOLANT SYSTEM

[redacted]

9.2 RESIDUAL HEAT REMOVAL (RHR)

The residual heat removal system is designed to remove decay heat from the core and reduce the temperature of the reactor coolant system during the second phase of plant cooldown. During the first phase of cooldown, the temperature of the reactor coolant system is reduced by transferring heat from the reactor coolant system to the steam and power conversion system. Separate and independent residual heat removal systems are supplied for the two units. The description contained herein is equally applicable to either unit.

The equipment utilized for residual heat removal is also used for emergency core cooling during loss-of-coolant accident conditions. All active system components which are relied upon to perform the emergency core cooling function during an accident are redundant. Components not required for this function may or may not be redundant.

9.2.1 DESIGN BASIS

The residual heat removal system is designed to provide the following safety-related functions: (1) deliver borated cooling water to the reactor coolant system during the injection phase of safety injection, (2) recirculate and cool the water that is collected in the containment sump and return it to the reactor coolant system or containment spray pump suction during the recirculation phase of safety injection, (3) provide the means to preclude containment leakage through the RHR system piping penetrations following accidents, and (4) for piping and components that are part of the reactor coolant pressure boundary, maintain pressure boundary integrity during all MODES of plant operation.

The RHR system is credited in the event of a fire and has been evaluated in the at-power and non-power analyses (Reference 1). It is also designed to provide the following augmented quality functions; (1) provide low temperature overpressure protection of the reactor coolant system when the reactor coolant system is solid and the RHR system is in operation, and (2) provide indication of plant conditions during accident situations.

9.2.2 SYSTEM DESIGN AND OPERATION

The residual heat removal system is a dual purpose system. During power operation, the system is aligned to perform its low head safety injection function. As such, the system is split providing two independent trains, each containing a pump and heat exchanger. Suction and discharge valves for this function and long term sump recirculation are part of the safety injection system as described in Chapter 6. During a plant shutdown to cold shutdown conditions, the RHR pumps and heat exchangers perform the residual heat removal functions for the reactor. To accomplish this alignment, several manual valves must be opened to cross-connect the outlet of the heat exchangers and the discharge of the pumps and to provide a suction path for each of the pumps. After the reactor coolant system temperature and pressure have been reduced to less than or equal to 350°F and less than 400 psig respectively, residual heat removal is initiated by aligning the pumps to take suction from the “A” hot leg reactor coolant loop and discharge through the heat exchangers into the “B” cold leg reactor coolant loop. If only one pump and one heat exchanger are available, reduction of reactor coolant temperature is accomplished at a lower rate.

A connection between the residual heat removal system and the reactor coolant system letdown line permits purification of the reactor coolant when the reactor coolant system temperature and pressure is reduced. The system design includes provisions to enable hydrostatic testing to applicable test pressures during shutdown. System components, whose design pressure and temperature are less than the reactor coolant system design limits, are provided with overpressure protective devices and redundant isolation means. All piping and components of the residual heat removal system are designed to the applicable codes and standards listed in Table 9.2-1. Austenitic stainless steel piping is used in the residual heat removal loop, which contains reactor coolant.

The residual heat removal system consists of heat exchangers, pumps, piping and the necessary valves and instrumentation. During plant shutdown reactor coolant flows from the reactor coolant system to the residual heat removal pumps, through the tube side of the residual heat removal heat exchangers and back to the reactor coolant system. The inlet line to the residual heat removal system starts at the hot leg of one reactor coolant loop and the return line connects to the cold leg of the other loop. The heat loads are transferred by the residual heat removal heat exchangers to the component cooling water. The residual heat removal heat exchangers are also used to cool the recirculated water during the recirculation phase of safety injection system operation. These duties are defined in Chapter 6.

During plant shutdown, the cooldown rate of the reactor is controlled by regulating the reactor coolant flow through the tube side of the residual heat removal heat exchangers. A bypass line and an automatic flow control valve around the residual heat removal heat exchangers are used to maintain a constant flow through the residual heat removal system.

RHR system operational methods preclude any significant reduction in the overall design reactor shutdown margin when the loop is brought into operation for residual heat removal, or for emergency core cooling by recirculation.

Redundant, remotely operated valves (RH-700 and RH-701) in the residual heat removal system inlet line are provided to isolate the system from the reactor coolant system. A remotely operated valve (RH-720) and a check valve (SI-867B) isolate the return line to the reactor coolant system cold leg from the residual heat removal system. When reactor coolant system pressure exceeds the design pressure of the residual heat removal system, an interlock between the reactor coolant system wide range pressure channel (PT-420) and valves RH-700 and RH-720 prevents them from opening. Overpressure in the system is further prevented by two relief valves in the inlet piping to the residual heat removal system (valves RH-861B and RH-861C) and by a relief line from the outlet piping of the residual heat removal system to the CVCS letdown line.

Relief valve RH-861B is set at 600 psig and discharges to the pressurizer relief tank. Relief valve RH-861C is a high capacity relief valve. This relief valve is adjusted to open at 500 psig and provides a relief flow rate of 1106 gpm at 10% accumulation. This capacity was set to handle the maximum assumed injection flow which could occur by operation of a single safety injection pump and two charging pumps (Reference 4). The RH-861C valve discharges to the containment sump B.

In addition to protecting the residual heat removal system from overpressure, these relief valves are also available for water relief whenever the residual heat removal system is connected to the reactor coolant system i.e., during low temperature and low pressure conditions. These relief valves can thus be considered a diverse relief system and backup to the overpressure mitigating system for low temperature overpressure protection of the reactor coolant system during cold shutdown and solid pressurizer conditions (Reference 2).

To minimize the effects of pressure locking and to provide overpressure protection for the section of piping between the two residual heat removal loop suction isolation valves, a 3/8-inch hole has been drilled in the RCS-side disc of each of the RHR loop isolation motor-operated valves (RH-700 and RH-701). A 3/8-inch hole has also been drilled in the RCS-side disc of the RHR return isolation valve (RH-720) to minimize the effects of pressure locking.

Residual Heat Removal System Components

Each unit is provided a residual heat removal (RHR) system that is independent of the other unit's RHR system. The components are discussed below on a per-unit basis.

Residual Heat Removal Heat Exchangers

The two residual heat removal heat exchangers are of the shell and U-tube type with the tubes welded to the tube sheet. Reactor coolant circulates through the tubes, while component cooling water circulates through the shell side. The tubes and other surfaces in contact with reactor coolant are austenitic stainless steel and the shell is carbon steel.

Residual Heat Removal Pumps

The two residual heat removal pumps are horizontal, centrifugal units with mechanical seals to limit reactor coolant leakage to the atmosphere. All pump parts in contact with reactor coolant are austenitic stainless steel or equivalent corrosion resistant material.

Residual Heat Removal Valves

The valves used in the residual heat removal system are constructed of austenitic stainless steel or equivalent corrosion resistant material. Manual stop valves are provided to isolate equipment for maintenance. Throttle valves are provided for remote and manual control of the residual heat exchanger tube side flow. Check valves prevent reverse flow through the residual heat removal pumps.

Valves that perform a modulating function or have a diameter greater than 2 1/2 in. are equipped with sufficient packing to minimize leakage to the atmosphere. Manually operated valves have backseats to facilitate repacking and to limit the stem leakage in the event the packing fails or leaks excessively. Backseats are not normally relied upon as the primary leakage barrier.

Residual Heat Removal Piping

All residual heat removal system piping is austenitic stainless steel. The piping is welded with flanged connections at some components for ease of maintenance.

9.2.3 SYSTEM EVALUATION

Two pumps and two heat exchangers are provided to remove residual and sensible heat during plant cooldown. If one of the pumps and/or one of the heat exchangers is not operative, safe operation of the plant is not affected; however, the time for cooldown is extended. The function of this equipment following a loss-of-coolant accident is discussed in Chapter 6. The entire system is seismic Class I design. The components are designed to the codes given in Table 9.2-1.

Welded construction is used where possible throughout the residual heat removal system piping, valves, and equipment to minimize the possibility of leakage. During reactor operation all equipment of the residual heat removal system is idle, and the associated isolation valves are closed. During the loss-of-coolant accident condition, water from the containment sump is recirculated through the outside containment piping system. Both of the lines from the containment sump to the individual residual heat removal pumps have two remotely operated isolation valves in series. To quantify the possible total radiation dose to the public due to leakage from this system, the potential leaks have been evaluated and are discussed in Chapter 6 and Chapter 14.

Each RHR pump is located in an individual shielded compartment which is equipped with a floor drain and separated equipment drains. The floor drain from each compartment flows through an individual pipe to the sump. Two 75 gpm sump pumps transfer the leakage to the waste disposal system. The supply and discharge piping and valves for the RHR pumps are located in a pipeway adjacent to the pump compartments. A seven ft. high wall divides the pipeway into two sections, each of which drains into a pump compartment through a 4-inch by 4-inch opening at floor level. The RHR pump seal failure rate is 50 gpm.

The RHR cubicle drain valves are maintained in the closed position. If a RHR pump seal failure occurred with the drain valves in the closed position, a RHR pump room high level alarm would eventually be indicated in the control room. The cubicle could then be drained to the sump by opening the remotely operated drain valve. If flooding in EL.-19′ occurred due to a source other than a failed RHR pump seal, the fluid would collect in the center cubicle (cubicle between the Unit 1 and Unit 2 RHR pumps) and flow to the sump via the floor drains. The flow path to the RHR pump cubicle would remain isolated.

The residual heat removal system is connected to the hot leg of one reactor coolant loop on the suction side and to the cold leg of the other reactor coolant loop on the discharge side. On the suction side, the connection is through two electric motor operated gate valves in series with the first valve interlocked with reactor coolant system pressure. On the discharge side, the connection is through a check valve in series with an electric motor operated gate valve which is also interlocked with reactor coolant system pressure. All of these valves are closed during normal operation and the power to the MOVs is removed.

The RHR pumps are powered from 480 Volt safety-related buses with emergency diesel generator backup.

A reactor power of 1800 MWt was used for evaluation of the RHR system for the extended power uprate for both the normal cooldown requirements and post-fire considerations (Reference 2, Reference 3, Reference 5, Reference 6).

9.2.4 REQUIRED PROCEDURES AND TESTS

The residual heat removal pump's flow instrument channels can be calibrated during shutdown. Periodic visual inspections and preventive maintenance can be conducted as necessary without interruption of cooling system operation. The inservice testing requirements are described in the PBNP Inservice Testing Program and the IST Background Document.

Generic Letter 2008-01, "Managing Gas Accumulation in Emergency Core cooling Decay Heat Removal, and Containment Spray Systems," was issued to evaluate the systems to ensure gas accumulation is maintained less than the amount that challenges operability. The Gas Accumulation Management Program (GAMP) ensures that gas accumulation within the RHR system is identified, evaluated, trended and effectively controlled to prevent unacceptable degradation of performance. (Reference 7, Reference 8, Reference 9, and Reference 10)

9.2.5 REFERENCES

1. NFPA 805 Fire Protection Program Design Document (FPPDD).
2. NRC Safety Evaluation by the Office of Nuclear Regulation related to Amendment No. 45 to Facility Operating License No. DPR-24 and Amendment No. 50 to Facility Operating License No. DPR-27, dated May 20, 1980.
3. 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.
4. Calculation M11165-112-RH.1, Relief Valve 1/2 RH-00861C Setpoint and Capacity Determination, Approved May 22, 2009.
5. Westinghouse Calculation CN-SEE-III-08-10, "Point Beach Units 1&2 RHR Cooldown Analysis for EPU to 1806 MWt NSSS Power," Rev 3, Approved April 27, 2011.
6. Calculation 97-0118, "Capability to Achieve Cold Shutdown in Both Units with One CCW Pump and Two CCW Heat Exchangers."
7. Letter NRC 2008-0075, "Nine-Month Response to NRC Generic Letter 2008-01 Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems," dated October 14, 2008.
8. NRC letter, "Point Beach Nuclear Plant, Units 1 and 2 Closeout of Generic Letter 2008-01 Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal and Containment Spray Systems" (TAC Nos. MD7864 and MD7865), dated January 7, 2010.
9. Letter NRC 2009-0015, "Point Beach Nuclear Plant, Unit 1, Nine-Month Supplemental (Post-Outage) Response to NRC Generic Letter 2008-01," dated February 11, 2009.
10. NRC Safety Evaluation by the Office of Nuclear Reactor Regulation related to Amendment Nos. 251 and 255, "Managing Gas Accumulation," dated January 27, 2015.

Table 9.2-1
RESIDUAL HEAT REMOVAL LOOP COMPONENT DATA

Reactor coolant temperature at startup of residual heat removal, °F	350
Time to cool reactor coolant system from 350°F to 140°F, hr (single RHR train while in MODE 4, 72°F SW temp)****	113
Decay heat generation standard****	ANS 5.1-1979
Reactor cavity fill time, hr	1.5
Reactor cavity drain time, hr	4
H_3BO_3 concentration in refueling water storage tanks, ppm boron	2800-3200
Residual heat removal pumps	
Quantity (per unit)	2
Type	Horizontal centrifugal
Design flow rate (each), gpm	1560
Total developed head, ft of water	280
Motor horsepower, hp	200
Material	Stainless Steel
Design pressure, psig	600
Design temperature, °F	400
Residual heat removal pump room sump pumps (WL System)	
Quantity	2
Type	Vertical, duplex
Capacity, gpm	75
Head, ft of water	55
Material (wetted surface)	Stainless steel
Residual Heat Removal Heat Exchangers	
Quantity (per unit)	2
Type	Shell and U-tube, vertical
Design heat transfer, Btu/hr	24.15×10^6
Shell side (component cooling water)	
Design inlet temperature, °F	100
Design outlet temperature, °F	117.3
Design flow rate, lb/hr	1.375×10^6
Design pressure, psig	150
Design temperature, °F	350
Material	Carbon steel
Tube side (reactor coolant)	
Design inlet temperature,°F	160
Design outlet temperature, °F	128.4
Design flow rate, lb/hr	7.63×10^5
Design pressure, psig	600
Design temperature, °F	400
Material	Stainless steel
Code Requirements	
Piping and valves	USAS B31.1*
RHR heat exchangers	ASME III**, Class C, tube side ASME VIII***, shell side

* USAS B31.1 - Code for Pressure Piping, and special nuclear cases where applicable
** ASME III - American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III
*** ASME VIII - American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section VIII
**** Reference 5

Figure 9.2-1 UNIT 1 AUXILIARY COOLANT SYSTEM

[redacted]

9.3 CHEMICAL AND VOLUME CONTROL SYSTEM (CV)

The chemical and volume control system described in this section includes descriptions of the boron recycle (BS), reactor makeup water (RMW), and boric acid heat tracing (HTRACE) systems.

The chemical and volume control system (CVCS) (a) adjusts the concentration of chemical neutron absorber for chemical reactivity control, (b) maintains the proper water inventory in the reactor coolant system (RCS), (c) provides the required seal water flow for the reactor coolant pump shaft seals, (d) maintains the desired concentration of corrosion controlling chemicals in the reactor coolant, (e) keeps the reactor coolant activity to within the design levels and (f) provides for RCS degasification. The system is also used to fill, drain, and hydrostatically test the reactor coolant system.

To accomplish the above functions, this system has provisions for supplying:

1. Hydrogen to the volume control tank.

2. Nitrogen to the volume control tank (for purging during shutdown operations).

3. Chemicals, as required, via the chemical mixing tank to the charging pumps' suction.

9.3.1 DESIGN BASES

The CVCS System performs the following safety-related functions:

a. CVCS System piping and components interfacing with pressure boundaries for the (1) reactor coolant system, (2) component cooling water system, (3) safety injection system (refueling water storage tank), and (4) residual heat removal system shall maintain the pressure boundary integrity to support the safety function of these systems.

b. CVCS System containment isolation valves and portions of the CVCS System that function as a closed system outside containment shall maintain containment integrity following accidents that require containment isolation.

The CVCS System is credited in the event of a fire and has been evaluated in the at-power and non-power analyses (Reference 6).

<u>Redundancy of Reactivity Control</u>

Criterion: Two independent reactivity control systems, preferably of different principles, shall be provided. (GDC 27)

In addition to the reactivity control achieved by the rod cluster control (RCC) described in Section 3.0, reactivity control is provided by the CVCS which regulates the concentration of boric acid solution neutron absorber in the reactor coolant system. The system is designed to prevent uncontrolled or inadvertent reactivity changes which might cause system parameters to exceed design limits.

Reactivity Hot Shutdown Capability

Criterion: The reactivity control system provided shall be capable of making and holding the core subcritical from any hot standby or hot operating condition. (GDC 28)

The reactivity control systems provided are capable of making and holding the core subcritical from any hot standby or hot operating condition, including conditions resulting from power changes. The maximum excess reactivity expected for the core occurs for the cold, clean condition at the beginning of life of the core.

The rod cluster control assemblies (RCCAs) are divided into two categories comprising control and shutdown groups. The control group, used in combination with chemical shim, provides control of the reactivity changes of the core throughout the life of the core at power conditions. This group of RCCAs is used to compensate for short term reactivity changes at power such as those produced due to variations in reactor power requirements or in coolant temperature. The chemical shim control (CVCS) is normally used to compensate for the more slowly occurring changes in reactivity throughout core life such as those due to fuel depletion, fission product buildup and decay, and load follow. The safety injection system, used in conjunction with the reactor vessel head vent system, provides a safety-related backup for CVCS.

Reactivity Hold-Down Capability

Criterion: The reactivity control systems provided shall be capable of making the core subcritical under credible accident conditions with appropriate margins for contingencies and limiting any subsequent return to power such that there will be no undue risk to the health and safety of the public. (GDC 30)

Normal reactivity shutdown capability is provided by control rods, with boric acid injection used to compensate for the xenon transients, and for plant cooldown. When the plant is at power, the quantity of boric acid retained in the boric acid tanks and/or the refueling water storage tank (RWST) and ready for injection will always exceed that quantity required for the normal cold shutdown. This quantity will always exceed the quantity of boric acid required to bring the reactor to hot shutdown and to compensate for subsequent xenon decay.

The boric acid solution is transferred from the boric acid tanks by boric acid transfer pumps (or via gravity feed from the RWST) to the suction of the charging pumps which inject the solution into the reactor coolant. Any charging pump and any boric acid transfer pump can be manually transferred to diesel generator power on loss of off-site AC power. Boric acid can be injected by one charging pump and one boric acid transfer pump at a rate which shuts the reactor down, with no control rod insertion, in less than 120 minutes. In 150 additional minutes, enough boric acid can be injected to compensate for xenon decay although xenon decay below the equilibrium operating level will not begin until approximately 12-15 hours after shutdown. If two boric acid pumps and two charging pumps are available, the injection time periods are halved. Additional boric acid is employed if it is desired to bring the reactor to cold shutdown conditions.

On the basis of the above, the injection of boric acid is shown to afford backup reactivity shutdown capability, independent of control rod clusters which normally serve this function in the short term situation. Shutdown for long term and reduced temperature conditions can be accomplished with boric acid injection using redundant components.

9.3.2 SYSTEM DESIGN AND OPERATION

Various components of the chemical and volume control system are shared by the two units. These components are shown in Table 9.3-3 and a discussion concerning the sharing is given in Appendix I.6. The following discussion is for the CVCS for one unit but applies equally to either unit.

The CVCS, shown in Figure 9.3-1 through Figure 9.3-5, provides a means for injection of the neutron control chemical in the form of boric acid solution, chemical additions for corrosion control, and reactor coolant cleanup and degasification. This system also adds makeup water to the reactor coolant system, reprocesses water letdown from the reactor coolant system, and provides seal water injection to the reactor coolant pump seals. Materials in contact with the reactor coolant are austenitic stainless steel or equivalent corrosion resistant materials. System components whose design pressure and temperature are less than the reactor coolant system design limits are provided with overpressure protective devices. System discharges from overpressure protective devices (safety valves) and system leakages are directed to closed systems. Effluents removed from such closed systems are monitored and discharged under controlled conditions.

During normal plant operation, reactor coolant flows through the letdown line from the ‘B’ loop cold leg on the suction side of the reactor coolant pump and, after processing is returned either to the cold leg of the ‘A’ loop on the discharge side of the reactor coolant pump via a charging line, or via reactor coolant pump seal injection. An alternate charging connection is provided on the cold leg of the ‘B’ loop (on the discharge side of the pump). An excess letdown line is also provided for removing coolant from the reactor coolant system when normal letdown is not available.

Each of the connections to the reactor coolant system has an isolation valve located close to the loop piping. In addition, a check valve is located downstream of each charging line isolation valve. Reactor coolant letdown entering the CVCS flows through the shell side of the regenerative heat exchanger where its temperature is reduced. The coolant then flows through letdown flow control orifices which reduce the coolant pressure. The cooled, low pressure water leaves the reactor containment and enters the auxiliary building where it undergoes a second temperature reduction in the tube side of the nonregenerative heat exchanger. Following that cooling, a second pressure reduction is accomplished in the low pressure letdown valve. Mixed bed, cation, and deborating demineralizers follow. Normally one mixed bed demineralizer is aligned for ionic impurity control. Coolant then flows through a reactor coolant filter, and letdown gas stripper before entering the volume control tank.

A letdown gas stripper is located downstream of the letdown demineralizers and prior to the volume control tank. The letdown gas stripper is capable of removing entrained gasses from the letdown stream. The degassed water is then returned to the volume control tank. The stripped gasses (primarily hydrogen, but including any fission gasses) from both units are drawn from the gas stripper towers by a compressor. The discharge of the compressors is routed to heavily shielded delay/decay tanks, where a backpressure is maintained to achieve the desired radioactive decay of the gas. The gas exits the delay/decay tanks to be recycled into the volume control tanks to again establish an excess of hydrogen in the reactor coolant.

The letdown gas decay portion of the system additionally has a section (unused) capable of noble gas retention by cryogenic adsorption. Utilizing the cryogenic capability of the system would have resulted in accumulation of long-lived and highly radioactive noble gasses, which would require continuous maintenance of cryogenic conditions, and a leak tight noble gas container. There are therefore, no plans to utilize the cryogenic capability of the system.

The letdown gas stripper and gas recycle system are more completely described in Section 11.2.

Hydrogen is automatically supplied, as determined by pressure control, to the vapor space in the volume control tank. The VCT vapor space is predominantly hydrogen and water vapor. The hydrogen overpressure within the tank causes its absorption into the reactor coolant to aid in maintaining a low oxygen concentration.

From the volume control tank the coolant flows to the charging pumps which raise the pressure above that in the reactor coolant system. The coolant then enters the containment, passes through the tube side of the regenerative heat exchanger, and is returned to the reactor coolant system.

A cation bed demineralizer, located downstream of the mixed bed demineralizers, is used intermittently to control cation activity in the coolant, primarily to remove excess lithium formed from the B^{10} (n, a) Li^7 reaction, or cesium.

A pair of deborating demineralizers, also located downstream of the mixed beds, are aligned near the end of core life (low boron concentration) to achieve reduction in RCS boron concentration which would be otherwise unattainable without significant reductions in desirable chemicals, given the high volume dilution required for a similar change in boron concentration.

Boric acid is dissolved in heated water in the batching tank to the desired concentration. The lower portion of the batching tank is jacketed to permit heating of the batching tank solution with low pressure steam. A transfer pump may be used to transfer the batch to the boric acid tanks or a gravity drain process may be used. Electric immersion heaters may be used to maintain the temperature of the boric acid tanks solution high enough to prevent precipitation. The boric acid solution is metered from the discharge of an operating transfer pump and either blended with reactor makeup water as makeup for system level control, or added without dilution if reactor coolant boron concentration is being increased.

Excess liquid effluents from the reactor coolant system are collected in the holdup tanks. As liquid enters the holdup tanks, the cover gas is displaced to the gas decay tanks in the waste disposal system through the waste gas vent header. The concentration of boric acid in the holdup tanks varies throughout core life from the refueling concentration to essentially zero at the end of the core cycle. A recirculation pump is provided to transfer liquid from one holdup tank to another and to recirculate the contents of individual holdup tanks.

A holdup tank can be aligned in a recirculation line-up through demineralizers and back to the holdup tank to allow for cleanup and sampling. Processing can be performed either as a batch operation or as a continuous bleed operation. In either case, liquid is pumped through ion exchangers which primarily remove lithium hydroxide and fission-products such as long-lived cesium. Following the ion exchangers, flow can be routed to the monitor tanks, for reuse or release assessment.

Subsequent handling of the holdup tank water is dependent on the results of sample analysis. Discharge from the monitor tanks may be pumped to the reactor makeup water storage tank, recycled through demineralizers, returned to the holdup tanks for reprocessing, or discharged to the environment (via the condenser circulating water system and the service water return header,) when within the allowable activity concentration as discussed in Section 11.1. If the sample analysis of the monitor tank contents indicates that it may be discharged to the environment, at least two valves must be opened to provide a discharge path. As the effluent leaves, it is continuously monitored by the waste disposal system liquid effluent monitor. If an unexpected increase in radioactivity is sensed, one of the valves in the discharge line to the service water discharge header closes automatically and an alarm sounds in the control room.

When the residual heat removal loop is operating and the reactor coolant system is depressurized, a flow path is provided to remove corrosion impurities and fission products. A portion of flow leaving the residual heat removal pumps can be directed through the nonregenerative heat exchanger, mixed bed demineralizers, and reactor coolant filter. The fluid can then bypass the volume control tank, pass through the charging pumps, and then either through normal or auxiliary charging lines, into the RCS. A flow path can also be provided to the CVCS to remove corrosion impurities and fission products via the refueling water circulating pump.

Expected Operating Conditions

Table 9.3-2, Table 9.3-3, Table 9.3-4 and Table 9.3-5 list the system performance requirements, data for individual system components and reactor coolant equilibrium activity concentration.

Reactor Coolant Activity Concentration

The parameters used in the calculation of the reactor coolant fission product inventory, including pertinent information concerning the coolant cleanup flow rate and demineralizer effectiveness, are presented in Table 9.3-4. The results of the calculations are presented in Table 9.3-5. In these calculations, 1% defects in the fuel rods are assumed to be present at initial core loading and are uniformly distributed throughout the core. (Reference 3) The fission product escape rate coefficients are therefore based upon an average fuel temperature.

Tritium is produced in the reactor from ternary fission in the fuel, irradiation of boron in the burnable poison rods and irradiation of boron, lithium and deuterium in the coolant. The parameters used in the calculation of tritium production rate and results are presented in Table 9.3-6. This table reflects tritium produced from twice and thrice burned IFBA fuel assembles at a power level of 1650 MWt. RCS tritium level was not specifically evaluated for EPU conditions, but can be expected to increase approximately proportional to the power level. Tritium effluents at EPU conditions are discussed in Section 11.2 and Appendix I.3.

Reactor Makeup Control

The reactor makeup control consists of a group of instruments arranged to provide a manually preselected makeup water composition to the charging pump suction header or the volume control tank. The makeup control functions are to maintain desired operating fluid inventory in the volume control tank and to adjust reactor coolant boron concentration for reactivity control. Makeup for normal plant leakage is regulated by the reactor makeup control which is set by the operator to blend water from the reactor makeup water tank with concentrated boric acid from the BAST to match the reactor coolant boron concentration.

The makeup system also provides concentrated boric acid or reactor makeup water to either increase or decrease the boric acid concentration in the reactor coolant system. Since a constant volume is maintained in the reactor coolant system by the pressurizer level control system, it is the volume control tank level which will rise or fall as makeup is added or leakage occurs. If volume control tank level increases to beyond the control band, letdown will be diverted to the holdup tanks until level returns to within the control band. If volume control tank level decreases to the lower limit of the control band, an automatic makeup will occur to ensure charging pump suction is maintained. Should the letdown line be out of service during operation, sufficient volume exists in the pressurizer to accept into the reactor coolant system the amount of borated water necessary for hot shutdown. Makeup to the reactor coolant system is provided by the chemical and volume control system from the following sources:

1. The reactor makeup water tank, which provides water for dilution when the reactor coolant boron concentration is to be reduced.

2. The boric acid storage tanks, which supply concentrated boric acid solution when reactor coolant boron concentration is to be increased.

3. The refueling water storage tank, which supplies borated water for emergency makeup.

The reactor makeup control is operated from the control room by manually preselecting makeup composition to the charging pump suction header or the volume control tank. Makeup is provided to maintain the desired operating fluid inventory in the reactor coolant system and to adjust the reactor coolant boron concentration for reactivity control. The operator can stop the makeup operation at any time in any operating mode. One reactor makeup water pump and one boric acid transfer pump are normally lined up for automatic operation as required by the makeup controller.

A portion of the high pressure charging flow is injected into the reactor coolant pumps between the pump impeller and the No. 1 shaft seal. The seal supply flow is greater than seal leakage flow so that the seals are not exposed to high temperature reactor coolant. The flow which does not leave the system via the RCP controlled leakage seal, enters the RCS through a labyrinth seal surrounding the pump shaft. The shaft seal leakage flow cools the lower radial bearing, passes through the seals, is cooled in the seal water heat exchanger, filtered, and returned to the volume control tank.

In the event of a loss of seal injection and CCW flow to the thermal barrier heat exchanger, reactor coolant begins to travel along the RCP shaft and displace the cooler seal injection water. The shut down seal (SDS) is designed to function only when exposed to an elevated fluid temperature downstream of the RCP number 1 seal. The SDS deploys via retraction of a thermal actuator, which causes the SDS seal ring to constrict around the pump shaft. SDS deployment controls shaft seal leakage and limits the loss of reactor coolant via the RCP seal package.

Seal water inleakage to the reactor coolant system requires a continuous letdown of reactor coolant to maintain the desired inventory. In addition, bleed and feed of reactor coolant are required for removal of impurities, chemical control, and adjustment of boric acid in the reactor coolant. The excess letdown line is sized to accommodate seal injection flow if normal letdown is out of service.

Automatic Makeup

The "automatic makeup" mode of operation of the reactor makeup control provides boric acid solution preset by the operator to match the boron concentration in the reactor coolant system when VCT level reaches preset values. The automatic makeup compensates for minor leakage of reactor coolant without causing significant changes in the coolant boron concentration.

Under steady state plant operating conditions, the mode selector switch is set in the "automatic makeup" position. When the low level signal from the volume control tank level controller reaches a preset low setpoint, it causes the automatic makeup control action to open the makeup stop valve to the charging pump suction, open the concentrated boric acid control valve, and the reactor makeup water control valve. A boric acid transfer pump and reactor makeup water pump will start automatically. The flow controllers then blend the makeup stream according to the preset concentration. Makeup addition to the charging pump suction header causes the water level in the volume control tank to rise. At a preset high level setpoint, the makeup is stopped; the reactor makeup water control valve closes, the concentrated boric acid control valve closes, the makeup stop valve to charging pump suction closes, and the reactor makeup water and boric acid transfer pumps stop automatically if they were started automatically.

Dilution

The "dilute" mode of operation permits the addition of a preselected quantity of reactor makeup water at a preselected flow rate to the reactor coolant system. The operator sets the mode selector switch to "dilute," the reactor makeup water flow controller set point to the desired flow rate, and the reactor makeup water batch integrator to the desired quantity. Upon manual start of the system the makeup stop valve opens, the reactor makeup water control valve opens, and a reactor makeup water pump starts. Makeup water is added at the volume control tank inlet. Excessive rise of the volume control tank water level is prevented by automatic actuation (by the tank level controller) of a three-way diversion valve, which routes the reactor coolant letdown flow to the holdup tanks. When the preset quantity of reactor makeup water has been added, the batch integrator causes the reactor makeup water pump to stop, the makeup stop valve to close, and the reactor makeup water control valve to close.

Alternate Dilute

The "alternate dilute" mode is similar to the dilute mode except that the dilution water splits after passing through the blender. A portion flows directly to the charging pump suction and a portion flows into the volume control tank inlet. The operator sets the mode selector switch to "alternate dilute," the primary makeup water flow controller set point to the desired flow rate, the reactor makeup water batch integrator to the desired quantity and actuates the makeup start. The start signal causes the makeup control action to start a selected reactor makeup water pump and opens the makeup stop valve to the volume control tank and the makeup stop valve to the charging pump suction header and the reactor makeup control valve. Reactor makeup water is simultaneously added to the volume control tank and to the charging pump suction header. This mode is used for load follow and permits the dilution water to follow the initial xenon transient and simultaneously dilute the volume control tank. Excessive water level in the volume control tank is prevented by automatic actuation of the volume control tank level controller which routes the reactor coolant letdown flow to the holdup tanks. When the preset quantity of reactor makeup water has been added, the batch integrator causes the reactor makeup water pump to stop and the reactor makeup water control valve and the reactor makeup stop valves to close. The operation may be stopped manually by actuating the makeup stop valve.

Boration

The “borate” mode of operation permits the addition of a preselected quantity of concentrated boric acid solution at a preselected flow rate to the reactor coolant system. The operator sets the mode selector switch to “borate,” the concentrated boric acid flow controller set point to the desired flow rate, and the concentrated boric acid batch integrator to the desired quantity. Upon manual start of the system, the stop valve to the charging pumps opens, the selected boric acid transfer pump starts, if not already running, and the concentrated boric acid is added to the charging pump suction header. The total quantity added in most cases is so small that it has only a minor effect on the volume control tank level. When the preset quantity of concentrated boric acid solution has been added, the batch integrator causes the boric acid transfer pump to stop, the boric acid control valve to close and the makeup stop valve to the charging pump suction to close.

Concentrated boric acid can be injected into the primary coolant system via several different flow paths. Boric acid storage tank level meters and in-line flow meters allow the operator to verify injection of concentrated boric acid into the primary system.

The capability to add boron to the reactor coolant is such that it imposes no limitation on the rate of cooldown of the reactor upon shutdown. The maximum rates of boration and the equivalent coolant cooldown rates are given in Table 9.3-2. One set of values is given for the addition of boric acid from a boric acid storage tank at 3.5 weight percent boric acid with one transfer and one charging pump operating. The other set assumes the use of refueling water but with two of the three charging pumps operating. The rates are based on hot zero power temperature and on the end of the core life when the moderator temperature coefficient is most negative.

By manual action of the operator, the boric acid transfer pump can discharge directly to the charging pump suction and bypass the blender and volume control tank.

Blend

The “blend” mode of operation provides for manually initiated makeup of boric acid solution preset by the operator. The operator sets the mode selector switch to “blend,” the reactor makeup water flow controller setpoint to the desired flow rate, and the boric acid flow controller setpoint to the desired flow rate. Upon a manual start of the system, the makeup stop valve opens (to charging pump suction), the reactor makeup water control valve opens, and the boric acid control valve opens. A boric acid transfer pump and reactor makeup water pump will start automatically. The flow controllers blend the makeup stream according to the preset concentration. Makeup addition causes the water level in the volume control tank to rise. The operator manually stops the “blend” function when desired. The boric acid control valve closes, the reactor makeup control valve closes, and the reactor makeup water and boric acid transfer pumps stop.

The “blend” mode of operation may also be used to provide blended makeup to other plant systems.

Alarm Functions

The reactor makeup control is provided with alarm functions to call the operator's attention to the following conditions:

1. Deviation of reactor makeup water flow rate from the control set point.
2. Deviation of concentrated boric acid flow rate from the control set point.
3. Low level in the volume control tank. This alarm alerts the operator to a failure in the auto makeup controls or improper selector switch position.
4. Dilution in progress while at cold shutdown with the control rods inserted.

Charging Pump Control

Three positive displacement variable speed (capacity) drive charging pumps are available to supply charging flow to the reactor coolant system. The speed of each pump can be controlled manually or automatically. One charging pump normally operates with the pump control switch in START and its speed controller in MAN, a second charging pump operates with its control switch in START and its speed controller in AUTO, and a third charging pump is normally in STOP. The pump in MAN pumps at a constant rate, normally 18 gpm. The pump in AUTO pumps at a variable rate in response to pressurizer level differences.

The pressurizer level set point is varied by changes in average coolant temperature. If the pressurizer level increases above the setpoint, the speed of the pump decreases; likewise, if the level decreases below the setpoint, the speed increases. If the charging pump under automatic control reaches the high or low speed limit, an alarm is actuated. Each charging pump has a suction pressure stabilizer and discharge pulsation dampener to minimize vibration and pump inlet chamber pressure fluctuations.

To ensure that the charging pump flow is always sufficient to meet the seal water flow requirements, the pump has a low speed stop which does not permit pump flow lower than the seal injection minimum.

The charging pump controls include an automatic trip on low suction pressure trip to protect the pumps due to a loss of suction source from the volume control tank or refueling water storage tank. The controls consist of the tripping function on sustained low pressure and the capability to manually override the trip after adequate suction pressure has been verified.

Components

A summary of principal component data is given in Table 9.3-3.

Regenerative Heat Exchanger

The regenerative heat exchanger is designed to recover the heat from the letdown stream by reheating the charging stream during normal operation. This exchanger also limits the temperature rise, which occurs at the letdown orifices during periods when letdown flow exceeds charging flow by a greater margin than at normal letdown conditions. The letdown stream flows through the shell of the regenerative heat exchanger and the charging stream flows through the tubes. The unit is made of austenitic stainless steel, and is of all-welded construction.

Letdown Orifices

One of three letdown orifices can control flow of the letdown stream during normal operation, reducing letdown pressure to a value which maintains subcooling and is compatible with the nonregenerative heat exchanger design. Either of two letdown orifices, each 40 gpm, is used to pass normal letdown flow. The third orifice, 80 gpm, is designed to be used for maximum purification flow at normal reactor coolant system operating pressure. The orifices are placed in and taken out of service by remote manual operation of their respective isolation valves. One or both of the standby orifices may be used in parallel with the normally operating orifice in order to provide normal letdown flow when the reactor coolant system pressure driving force is below normal. This arrangement provides a full standby capacity for control of letdown flow. Each orifice consists of bored stock made of austenitic stainless steel.

Nonregenerative (Letdown) Heat Exchangers

The nonregenerative heat exchangers cool the letdown stream to the operating temperature of the mixed bed demineralizers. Reactor coolant flows through the tube side of the exchanger while component cooling water flows through the shells. The letdown stream outlet temperature is automatically controlled by a temperature control valve in the component cooling water outlet stream. This temperature control valve has a handwheel installed. This handwheel serves as a mechanical gag to limit the maximum flow to 685 gpm, the design flow, should a loss of instrument air occur. The unit is a multiple-tube, multiple-pass two-shell heat exchanger. All surfaces in contact with the reactor coolant are austenitic stainless steel, and the shells are carbon steel.

Mixed Bed Demineralizers

Two flushable mixed bed demineralizers maintain reactor coolant purity. A lithium-7 (or H^+ form) cation resin and a hydroxyl form anion resin are initially charged into one of the demineralizers. Both forms of resin remove fission and corrosion products, and in addition, the reactor coolant causes the anion resin to be converted to the borate form. The resin bed is designed to reduce the concentration of ionic isotopes in the purification stream, except for cesium, yttrium, and molybdenum, by a minimum factor of 10.

Each demineralizer is sized to accommodate the normal letdown flow. One demineralizer serves as a standby unit for use should the operating demineralizer become exhausted during operation and as the preferred method to remove lithium ions from the reactor coolant.

The demineralizer vessels are made of austenitic stainless steel, and are provided with suitable connections to facilitate resin replacement when required. The vessels are equipped with a resin retention screen. Each demineralizer has sufficient capacity, after operation for one core cycle with one per cent defective fuel rods, to reduce the activity of the primary coolant to refueling concentration.

Cation Bed Demineralizer

A flushable cation resin bed in the hydrogen form is located downstream of the mixed bed demineralizers and is used as a method to control the concentration of lithium-7 which builds up in the coolant from the B^{10} (n, a) Li^{7} reaction. The demineralizer also has sufficient capacity to maintain the cesium-137 concentration in the coolant below 1.0 μCi/cc with one percent defective fuel. The demineralizer is used intermittently to control cesium. The demineralizer is made of austenitic stainless steel and is provided with suitable connections to facilitate resin replacement when required. The vessel is equipped with a resin retention screen.

Deborating Demineralizers

When required, two demineralizers are available to be aligned to remove boric acid from the reactor coolant system fluid. The demineralizers are provided for use near the end of a core cycle, but can be used at any time. Hydroxyl-form ion-exchange resin can be used to reduce reactor coolant system boron concentration when loaded into these demineralizers. Facilities are provided for regeneration; however, the resin is normally flushed to the spent resin transfer cask for processing in the waste disposal system. Each demineralizer can remove the quantity of boric acid that must be removed from the reactor coolant system to maintain full power operation near the end of core life without the use of the holdup tanks.

Resin Fill Tank

The resin fill tank is no longer used. The resin fill tank was used to charge fresh resin to the demineralizers. The line from the conical bottom of the tank is fitted with a dump valve and may be connected to any one of the demineralizer fill lines. The demineralized water and resin slurry can be sluiced into the demineralizer by opening the dump valve. The tank is made of austenitic stainless steel.

Reactor Coolant Filter

The filter collects resin fines and particulates larger than 5 microns from the letdown stream. The vessel is made of austenitic stainless steel, and is provided with connections for draining and venting. Design flow capacity of the filter is equal to the maximum purification flow rate. Disposable filter elements are used. The bases for determining when the reactor coolant filter is replaced are pressure differential across the filter, and/or radiation levels.

Volume Control Tank

The volume control tank collects the excess water released from zero power to full power that is not accommodated by the pressurizer. It also receives the excess coolant release caused by the deadband in the reactor control temperature instrumentation. Overpressure of hydrogen gas is maintained in the volume control tank to control the hydrogen concentration in the reactor coolant to the specification listed in the EPRI PWR Primary Water Chemistry Guidelines (Reference 5).

A spray nozzle is located inside the tank on the inlet line from the reactor coolant filter. This spray nozzle provides intimate contact to equilibrate the gas and liquid phases. A remotely operated vent valve discharging to the waste disposal system permits removal of gaseous fission products which are stripped from the reactor coolant and collected in this tank. The volume control tank also acts as the suction supply for the charging pumps and a reservoir for the leakage from the reactor coolant pump controlled leakage seal. The tank is constructed of austenitic stainless steel.

Charging Pumps

Three charging pumps inject coolant into the reactor coolant system. The pumps are the variable speed positive displacement type, and all parts in contact with the reactor coolant are fabricated of austenitic stainless steel or other materials of adequate corrosion resistance. Special low-chloride packing is used in the pump glands. These pumps have mechanical packing followed by a leakoff to collect reactor coolant before it can leak to the outside atmosphere. Pump leakage is piped to a local drain for disposal to the waste disposal system. The pump design precludes the possibility of lubricating oil contaminating the charging flow, and the integral suction and discharge valves (check valves) allow water to pass through the pumps when idle, if the RCS is depressurized.

Each pump is designed to provide the full charging flow and the reactor coolant pump seal water supply during normal seal leakage. Each pump is designed to provide rated flow against a pressure equal to the sum of the reactor coolant system maximum pressure (existing when the pressurizer power operated relief valve is operating) and the piping, valve and equipment pressure losses of the charging system at the design charging flows. Any of the three charging pumps can be used to hydrotest the reactor coolant system. Additionally, a small motor can be directly coupled to one pump if hydrostatic test requirements demand.

The charging pump controls include an automatic trip on low suction pressure trip to protect the pumps due to a loss of suction source from the volume control tank or refueling water storage tank. The controls consist of the tripping function on sustained low pressure and the capability to manually override the trip after adequate suction pressure has been verified.

Chemical Mixing Tank

The primary use of the chemical mixing tank is in the preparation of pH control chemical solutions, hydrazine for oxygen scavenging, and hydrogen peroxide for cold shutdown corrosion product source term reduction or hydrogen removal. The capacity of the chemical mixing tank is determined by the quantity of 35% hydrazine solution necessary to increase the concentration in the reactor coolant by 10 ppm. This capacity is more than sufficient to prepare the solution of pH control chemical for the reactor coolant system. The chemical mixing tank is made of austenitic stainless steel.

Excess Letdown Heat Exchanger

The excess letdown heat exchanger is capable of cooling a reactor coolant letdown flow stream equal to the nominal injection rate through the reactor coolant pump labyrinth seals. The unit is designed to reduce the excess letdown stream temperature from the cold leg temperature to 195°F. The excess letdown stream flows through the tube side and component cooling water is circulated through the shell side. All surfaces in contact with reactor coolant are austenitic stainless steel and the shell is carbon steel. All tube joints are welded.

RCP Seal Water Heat Exchanger

The seal water heat exchanger removes heat from the reactor coolant pump seal water and from the excess letdown heat exchanger flow stream, prior to returning them to the volume control tank. Reactor coolant flows through the tubes and component cooling water is circulated through the shell side. The unit is designed to cool the excess letdown flow and the seal water flow to the temperature normally maintained in the volume control tank if all the reactor coolant pump seals are leaking at the maximum design leakage rate. The tubes are welded to the tube sheet to prevent leakage in either direction, which would result in undesirable contamination of the reactor coolant or component cooling water. All surfaces in contact with reactor coolant are austenitic stainless steel and the shell is carbon steel.

RCP Seal Water Return Filter

The filter collects particulates from the reactor coolant pump seal water return and from the excess letdown heat exchanger flow. The filter is designed to pass the sum of the excess letdown flow and the maximum design leakage from the reactor coolant pump controlled leakage seals. The vessel is constructed of austenitic stainless steel and is provided with connections for draining and venting. Disposable synthetic filter cartridges are used.

Seal Water Injection Filters

Two filters are provided in parallel, each sized for the injection flow. They collect particulates from the water supplied to the reactor coolant pump seal.

Boric Acid Filter

The boric acid filter collects particulates from the boric acid solution being pumped to the charging pump suction line. The filter is designed to pass the design flow of two boric acid transfer pumps operating simultaneously. The vessel is constructed of austenitic stainless steel and the filter elements are disposable synthetic cartridges. Provisions are available for venting and draining the filter.

Boric Acid Storage Tanks

Either the refueling water storage tank or the boric acid storage tanks may be used to contain the Technical Requirements Manual (TRM) required volume of borated water needed for cold shutdown. This volume is sufficient to provide the required shutdown margin at cold shutdown; xenon-free conditions with the most reactive RCCA not inserted from any expected operating condition.

The concentration of boric acid solution in the boric acid storage tanks is maintained within one of the operational bands described in the TRM. The nominal concentration of these bands varies from 3.25% to 12% by weight. Periodic sampling is performed in accordance with the TRM to ensure the desired concentration range is maintained. As a consequence, measured amounts of boric acid solution can be delivered to the reactor coolant to control the chemical poison concentration. The combination overflow and breather vent connection has a water loop seal to minimize vapor discharge during storage of the solution. The tanks are constructed of austenitic stainless steel.

Each of the three boric acid storage tanks has a capacity of 5000 gallons. The tank capacities can be shared by Unit 1 and Unit 2. The tanks have sufficient capacity for the TRM cold shutdown volume for one unit plus load follow capability volume for both units. One tank is normally aligned to each unit of the two-unit station and the third tank acts as a standby.

Each tank is provided with two level indicators and differential pressure cells. If the boric acid storage tanks are being relied upon for the TRM cold shutdown volume, the operator may enable a low level computer alarm that will alert him to an approach to the TRM minimum volume requirement. Level indication is provided in the control room and locally.

Boric Acid Storage Tank Heaters

Two 100% capacity electric immersion heaters located near the bottom of each boric acid storage tank are designed to maintain the temperature of the boric acid solution at 165°F with an ambient air temperature of 40°F; thus ensuring a temperature in excess of the solubility limit. Preferentially, plant parameters are controlled such that the required boric acid concentration in the tank is soluble at room temperatures. Operating at lower boric acid concentrations will reduce the need for tank heating. The temperature is monitored and is alarmed (high and low temperature alarms) in the control room. The heaters are sheathed in austenitic stainless steel.

Batching Tank

The batching tank, shared by Units 1 and 2 will hold about 2 1/2 day's makeup supply of 3% boric acid solution for the boric acid tank. The basis for makeup is reactor coolant leakage of ½ gpm near the beginning of core life. The tank may also be used for solution storage. A local sampling point is provided for verifying the solution concentration prior to transferring it to the boric acid storage tank. The tank manway is provided with a removable screen to prevent entry of foreign material. In addition, the tank is provided with an agitator to improve mixing during batching operation. The tank is constructed of austenitic stainless steel, and is not used to handle radioactive substances. The tank is provided with a steam jacket for heating the boric acid solution up to 165°F.

Boric Acid Transfer Pumps

Two canned centrifugal pumps are used to circulate or transfer boric acid. The pumps circulate boric acid solution through the boric acid storage tanks and inject boric acid into the charging pump suction header.

Although one pump may be used for boric acid batching and transfer and the other for boric acid injection, either pump may function as standby for the other. The design capacity of each pump is equal to the normal letdown flow rate. The design head is sufficient, considering line and valve losses, to deliver rated flow to the charging pump suction header when volume control tank pressure is at the maximum operating value (relief valve setting). All parts in contact with the solutions are austenitic stainless steel or other adequately corrosion-resistant material.

The transfer pumps are operated either automatically or manually from the main control room or from a local control point. The reactor makeup control operates one of the pumps automatically when boric acid solution is required for makeup or boration.

Boric Acid Recirculation Pump

One pump per unit can be used to continuously circulate boric acid. The pumps circulate boric acid from the boric acid storage tanks, throughout most of the piping, then back to the boric acid storage tanks to assure temperature equalization and positive evidence of boric acid fluidity throughout the piping.

The design capacity of the pump is based on providing uniform concentration and temperature. The design head is sufficient to overcome recirculation line losses. A minimum recirculation line from the discharge of the pump to the pump suction is provided to prevent pump damage should a boric acid transfer pump start and be aligned to the charging pump suction header while the recirculation pump is running. All parts in contact with the solutions are austenitic stainless steel or other adequately corrosion-resistant material. The recirculation pumps are manually operated from a local control point.

Boric Acid Blender

The boric acid blender promotes thorough mixing of boric acid solution and reactor makeup water from the reactor makeup supply circuit. The blender consists of a conventional pipe tee fitted with a perforated tube insert. The blender decreases the pipe length required to homogenize the mixture. All material is austenitic stainless steel.

Recycle Process - Holdup Tanks

Three holdup tanks can be shared by Units 1 and 2, to contain radioactive liquids from the letdown line and other sources. Most of the liquid is released from the reactor coolant system during startup, shutdown, load changes, and from boron dilution to compensate for burnup. The contents of one tank are normally being processed by at least a portion of the ion exchanger train while another tank is available as a standby. The tanks are constructed of austenitic stainless steel. A pressure switch on each tank will trip the Holdup Tank Recirculation Pump to provide vacuum protection for the tank.

Holdup Tank Recirculation Pump

The holdup tank (HUT) recirculation pump is shared between Units 1 and 2 and is used to mix the contents of a holdup tank or transfer the contents of one holdup tank to another or transfer the spent fuel pool transfer canal water to the HUTs or spent fuel pool. This pump can also be used to transfer water from a holdup tank to the spent fuel pool to increase pool inventory as required. The wetted surface of this pump is constructed of austenitic stainless steel.

Recycle Process Gas Stripper Feed Pumps

The two recycle process gas stripper feed pumps can be shared by Units 1 and 2, to supply feed from a holdup tank to the ion exchanger process train. The nonoperating pump is a standby and is available for operation in the event the operating pump malfunctions. These canned centrifugal pumps are constructed of austenitic stainless steel.

<u>Evaporator Feed Ion Exchangers</u>

Four flushable ion exchangers are shared by Units 1 and 2 to remove ionic impurities from the holdup tank effluent. The ion exchangers may be operated in parallel or in series with the alignment chosen given the ion load in the water, and the type and location of the resin needed to reduce that ion load. Each vessel is constructed of austenitic stainless steel and contains a resin retention screen.

The ion exchanger effluent may be routed directly to the monitor tanks, where sampling and a determination concerning environmental release or reprocessing can be made. Once approved for release, a monitor tank may be discharged utilizing the waste disposal system release point.

<u>Ion Exchanger Filters</u>

The filters collect resin fines and particulates from the evaporator feed ion exchangers. The vessel is made of austenitic stainless steel and is provided with connections for draining and venting. Disposable synthetic filter cartridges are used.

<u>Boric Acid Gas Stripper Equipment</u>

The boron recycle gas strippers were abandoned-in-place and bypassed in about 1972, after determining the quantity of entrained gasses were very low, and that the strippers were difficult to operate. Entrained gasses were low due to the combination of low cover gas pressures in the waste gas system and the 'bottom suction' utilized by the gas stripper feed pumps.

<u>Evaporator Condensate Demineralizers</u>

The evaporator condensate demineralizers are no longer used.

<u>Condensate Filter</u>

The condensate filters are no longer used.

<u>Monitor Tanks</u>

Four monitor tanks can be shared by Unit 1 and Unit 2. The monitor tanks accept processed water from the holdup tanks and provide a location where sampling and a determination concerning environmental release can be made. Once approved for release, a monitor tank may be discharged utilizing the waste disposal system release point. When tanks are filled, the contents are analyzed and either reprocessed, discharged via the waste disposal system, or pumped to the reactor makeup water tank. The monitor tanks can also be filled with water from the makeup water treatment plant. These tanks contain a diaphragm membrane and are constructed of stainless steel.

<u>Monitor Tank Pumps</u>

Two monitor tank pumps, shared by Units 1 and 2, discharge water from the monitor tanks. The pumps are constructed of austenitic stainless steel.

Reactor Makeup Water Tank

One reactor makeup water tank can be shared between the two units and is used to store makeup water, which is primarily supplied from the water treatment plant, but can also be supplied from the monitor tanks. The tank contains a diaphragm membrane and is constructed of coated carbon steel.

Reactor Makeup Water Pumps

Two reactor makeup water pumps, shared between Unit 1 and Unit 2, take suction from the reactor makeup water tank. These pumps are used to feed dilution water to the boric acid blender and are also used to supply makeup water for intermittent flushing of equipment and piping.

Each pump is sized to match the combined maximum letdown flow from each unit. One pump serves as a standby for the other. These centrifugal pumps are constructed of austenitic stainless steel.

Concentrates Filter

The concentrates filters are no longer used.

Concentrates Holding Tank

The concentrates holding tank is no longer used.

Concentrates Holding Tank Transfer Pumps

The concentrates holding tank transfer pumps are no longer used.

Electrical Heat Tracing

Electrical heat tracing is installed under the insulation on piping, valves, line-mounted instrumentation, and components that may contain highly concentrated boric acid solution (up to 12 weight percent). The heat tracing provides the capability to prevent boric acid precipitation due to cooling of highly concentrated solution, by compensating for heat loss. Exceptions are:

1. Lines which may transport concentrated boric acid but are subsequently flushed with reactor coolant or other liquid of low boric acid concentration during normal operation.

2. The boric acid storage tanks, which are provided with immersion heaters.

3. The batching tank, which is provided with a steam jacket.

4. Various pumps, which normally contain concentrated boric acid solution, are installed in electrically heated enclosures.

5. Portions of system containing boric acid at concentrations less than approximately 4 weight percent where ambient temperatures are adequate to prevent boric acid precipitation.

Valves

Valves that perform a modulating function and utilize packing are equipped with sufficient packing to minimize leakage to the atmosphere. Basic material of construction is stainless steel for all valves except the batching tank steam jacket valves, which are carbon steel.

Isolation valves are provided at all connections to the reactor coolant system. Lines with flow into the reactor containment also have check valves inside the containment to prevent reverse flow from the containment.

Relief valves are provided for lines and components that might be pressurized above design pressure by improper operation or component malfunction. Pressure relief for the tube side of the regenerative heat exchanger is provided by the auxiliary spray line isolation valve, which is designed to open to limit the upstream pressure.

Piping

All chemical and volume control system piping handling radioactive liquid is austenitic stainless steel. All piping joints and connections are welded, except where flanged connections are required to facilitate equipment removal for maintenance and hydrostatic testing. Piping which normally contains concentrated boric acid solution is temperature monitored to verify solubility of boric acid. Low temperatures are alarmed in the control room.

Codes and Classifications

All pressure retaining components (or compartments of components) which are exposed to reactor coolant, comply with the following codes:

1. System pressure vessels - ASME Boiler and Pressure Vessel Code, Section III, Class C, including Para. N-2113.

2. System valves, fittings and piping - USAS B31.1, including nuclear code cases.

System component code requirements are tabulated in Table 9.3-1.

The tube and shell sides on the regenerative heat exchanger and the tube side of the excess letdown heat exchanger are designed to ASME III, Class C. This designation is based on the following considerations: (a) each exchanger is connected to the reactor coolant system by lines equal to or less than 3", and (b) each is located inside the reactor containment. Analyses show that the accident associated with a 3" line break does not result in clad damage or failure. Additionally, previously contaminated reactor coolant, escaping from the reactor coolant system during such accident is confined to the reactor containment building and no public hazard results.

9.3.3 SYSTEM EVALUATION

Availability and Reliability

A high degree of functional reliability is assured in this system by providing standby components where performance is vital to safety and by assuring failsafe response to the most probable mode of failure. Solubility of boric acid at concentrations of less than about 4 weight percent is maintained at ambient temperature without additional heating. The system has three high pressure charging pumps, each capable of supplying the normal reactor coolant pump seal and makeup flow.

The electrical equipment of the chemical and volume control system is arranged so that multiple items receive their power from various 480 volt buses (See Figure 8-2). Two of the three charging pumps are powered by a 480 volt bus while the third charging pump is powered from a separate 480 volt bus.

The two boric acid transfer pumps are powered from separate 480 volt buses. One charging pump and one boric acid transfer pump are capable of meeting cold shutdown requirements shortly after full-power operation. In cases of loss of A.C. power, a charging pump and a boric acid transfer pump can be manually started if necessary after their buses have been reenergized by the emergency diesel generators. The transfer pumps are powered from MCCs that are stripped from their normal power supply by a safety injection signal. They can be recovered, but are not automatically re-energized from emergency diesels.

Control of Tritium

The chemical and volume control system is used to control the concentration of tritium in the reactor coolant system. Essentially all of the tritium is in chemical combination with oxygen as a form of water. Therefore, any leakage of coolant to the containment atmosphere carries tritium in the same proportion as it exists in the coolant. Thus, the level of tritium in the containment atmosphere, when it is sealed from outside air ventilation, is a function of tritium level in the reactor coolant, the cooling water temperature at the cooling coils, (which determines the dew point temperature of the air), and the presence of leakage other than reactor coolant as a source of moisture in the containment air.

There are two major considerations with regard to the presence of tritium:

1. containment atmosphere. It is desirable to limit the accumulation to allow containment access for periodic equipment inspection.)

2. Possible release of tritium to the environment.

Neither of these considerations is limiting at Point Beach Nuclear Plant. The concentration of tritium in the reactor coolant is maintained at a level which precludes personnel hazard during access to the containment. Acceptable tritium levels are achieved by eliminating a portion of the processed letdown stream from the recycle process.

The refueling water surface ventilation system can be utilized during refueling operations to minimize containment air tritium concentrations. Periodic determinations of tritium concentrations may be made by liquid scintillation counting of condensed water vapor from the containment and by calculations based on humidity measurements. Tritium release to the atmosphere via the containment purge system will be made in accordance with limits given in the Technical Specifications. Normally, tritium releases are much lower than allowed by the referenced limits. During periods other than refueling, personnel exposure to tritium while in the containment will be limited in accordance with applicable sections of 10 CFR 20.103.

Leakage Prevention

Quality control of the material and the installation of the chemical and volume control system valves and piping, which are designated for radioactive service, is provided in order to essentially eliminate leakage to the atmosphere. Except for component maintainability concerns, components designated for radioactive service are normally provided with welded connections to prevent leakage to the atmosphere. Flanged connections are provided in each charging pump suction and discharge, including the pressure fluctuation dampeners, on each boric acid pump suction and discharge, on relief valve inlets and outlets, on three-way valves, on the flow meters and elsewhere where necessary for maintenance. The positive displacement charging pumps stuffing boxes are provided with leakoffs to floor drains.

All valves which are larger than 2″ and are designated for radioactive service at an operating fluid temperature above 212°F are provided with sufficient packing to minimize leakage to the atmosphere. Leakage to the atmosphere is essentially zero for these valves.

All control valves are either provided with stuffing box and leakoff connections or are totally enclosed. Leakage to the atmosphere is essentially zero for these valves. Diaphragm valves are provided where the operating pressure and the operating temperature permit the use of these valves. Leakage to the atmosphere is essentially zero for these valves.

Incident Control

The letdown line penetrates the reactor containment building. The letdown line contains one motor-operated valve inside the reactor containment and three parallel air-operated orifice block valves for isolation from the RCS. Additionally, an automatic containment isolation signal closes an air-operated valve inside the reactor containment and another outside containment.

The reactor coolant pumps seal water return line contains one motor-operated isolation valve outside the reactor containment and an air-operated valve inside containment which are automatically closed by the containment isolation signal.

The two seal water injection lines to the reactor coolant pumps, the normal charging line, and the auxiliary charging line are inflow lines penetrating the reactor containment. Each line contains redundant containment isolation features to accommodate a potential break in these lines outside the reactor containment. Refer to Section 5.2.

Malfunction Analysis

To evaluate system safety, failures or malfunctions were assumed concurrent with a loss-of-coolant accident and the consequences analyzed and presented in Table 9.3-7. As a result of this evaluation, it is concluded that proper consideration has been given to plant safety in the design of the system.

If a rupture were to take place between the reactor coolant loop and the containment isolation valve or check valve, this incident would result in an uncontrolled loss of reactor coolant. The analysis of loss of coolant accidents is discussed in Section 14.

Should a rupture occur in the chemical and volume control system outside the containment, or at any point beyond the first check valve or containment isolation valve, actuation of the valve would limit the release of coolant and assure continued functioning of the normal means of heat dissipation from the core. For the general case of rupture outside the containment, the largest source of radioactive fluid subject to release is the contents of the volume control tank. The consequences of such a release are considered in Section 11. Should a LOCA occur, piping inside containment which is isolated in accordance with procedural system alignments, is protected against thermal overpressurization by relief devices.

When the reactor is subcritical, i.e., during cold or hot shutdown, refueling and approach to criticality, the relative reactivity status (neutron source multiplication) is continuously monitored and indicated by BF_3 counters and count rate indicators. Any appreciable increase in the neutron source multiplication, including that caused by the maximum physical boron dilution rate is slow enough to give ample time to start a corrective action (boron dilution stop and/or emergency boron injection) to prevent the core from becoming critical. The maximum dilution rate is based on the abnormal condition of two charging pumps operating at full speed delivering unborated primary water to the reactor coolant system at a particular time during refueling when the boron concentration is at the maximum value and the water volume in the system is at a minimum (see Section 14.1.4).

At least three separate and independent flow paths are available for reactor coolant boration from the CVCS system; i.e., the charging line, the auxiliary charging line, or the reactor coolant pump labyrinth seals. The malfunction or failure of one component will not result in the inability to borate the reactor coolant system. An alternate flow path is always available for emergency boration of the reactor coolant. As backup to the boration system the operator can align the refueling water storage tank outlet to the suction of the charging pumps, or can depressurize the RCS to utilize the safety injection system.

Boration during normal operation to compensate for power changes is indicated to the operator from two sources: (a) the control rod movement and (b) the flow indicator in the boric acid transfer pump discharge line. When the emergency boration path is used, two indications to the operator are available. The charging line flow indicator could indicate boric acid flow if the charging pump suction is aligned to the boric acid transfer pump suction alone, and the change in boric acid tank level is another indication of boric acid injection.

On loss of seal injection water to the reactor coolant pump seals, seal water flow may be reestablished by manually starting a standby charging pump. Even if the seal water injection flow is not reestablished, the plant can be operated indefinitely since the thermal barrier cooler has sufficient capacity to cool the flow passing through the RCP seals, as long as that leakage flow remains within the range of seal leak-off indications.

<u>Galvanic Corrosion</u>

The only types of materials which are in contact with each other in borated water are stainless steels, Inconel, Stellite valve materials, Zircaloy (STD & OFA fuel) and ZIRLO®/Optimized ZIRLO™ (422V+ fuel) fuel element cladding. Stainless steels, Inconel, Stellite and Zircaloy have been shown (Reference 2) to exhibit only an insignificant degree of galvanic corrosion when coupled to each other. This is also true for the ZIRLO® or Optimized ZIRLO™ fuel element claddings, which when coupled with the materials noted above, exhibit an insignificant amount of galvanic corrosion. 422V+ fuel uses ZIRLO® or Optimized ZIRLO™ instead of Zircaloy for cladding and ZIRLO for other fuel assembly components. The use of ZIRLO® was approved for use commencing with Unit 1, Cycle 27 and Unit 2, Cycle 25. The use of Optimized ZIRLO™ fuel cladding was approved for use commencing with Unit 1, Cycle 37 and Unit 2, Cycle 35 (See FSAR Section 3.3).

For example, the galvanic corrosion of Inconel versus 304 stainless steel resulting from high temperature tests (575°F) in lithiated, boric acid solution was found to be less than -20.9 mg/dm^2 for the test period of 9 days. Further galvanic corrosion would be trivial since the cell currents at the conclusion of the tests were approaching polarization. Zircaloy versus 304 stainless steel was shown to polarize in 180°F lithiated, boric acid solution in less than 8 days with a total galvanic attack of -3.0 mg/dm^2. Stellite versus 304 stainless steel was polarized in 7 days at 575°F in lithiated boric acid solution. The total galvanic corrosion for this couple was -0.97 mg/dm^2.

As can be seen from the tests, the effects of galvanic corrosion are insignificant to systems containing borated water.

9.3.4 REQUIRED PROCEDURES AND TESTS

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

9.3.5 REFERENCES

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

2. WCAP 1844 "The Galvanic Behavior of Materials in Reactor Coolants," D. G. Samarone, August, 1961.

3. Calculation CN-REA-08-7, "RCS, VCT, and GDT Sources for the Point Beach EPU," Westinghouse Electric Co. LLC, Revision 0, dated September 19, 2008.

4. 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.

5. EPRI 1014986, "Pressurized Water Reactor Primary Water Chemistry Guidelines."

6. NFPA 805 Fire Protection Program Design Document (FPPDD).

Table 9.3-1 CHEMICAL AND VOLUME CONTROL SYSTEM CODE REQUIREMENTS

Regenerative heat exchanger	ASME III*, Class C
Nonregenerative heat exchanger	ASME III, Class C, tube side, ASME VIII, shell side
Mixed bed demineralizers	ASME III, Class C
Reactor coolant filter	ASME III, Class C
Volume control tank	ASME III, Class C
Seal water heat exchanger	ASME III, Class C, tube side, ASME VIII, shell side
Excess letdown heat exchanger	ASME III, Class C, tube side, ASME VIII, shell side
Chemical mixing tank	ASME VIII
Deborating demineralizers	ASME III, Class C
Cation bed demineralizer	ASME III, Class C
Seal water injection filters	ASME III, Class C
Holdup tanks	ASME III, Class C
Boric acid filter	ASME III, Class C
Gas stripper package	ASME III, Class C
Evaporator condensate demineralizers	ASME III, Class C
Concentrates filter	ASME III, Class C
Evaporator feed ion exchanger	ASME III, Class C
Ion exchanger filter	ASME III, Class C
Condensate filter	ASME III, Class C
Piping and valves	USAS B31.1**

* ASME III - American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

** USAS B31.1 - Code for Pressure Piping and special nuclear cases where applicable.

Table 9.3-2 CHEMICAL AND VOLUME CONTROL SYSTEM PERFORMANCE REQUIREMENTS*

Original plant design life, years	40
Seal water supply flow rate, normal, gpm	16
Seal water return flow rate, normal, gpm	6
Normal letdown flow rate, gpm	40
Maximum letdown (purification) flow rate, gpm	120
Normal charging pump flow (one pump), gpm	46
Normal flow to reactor coolant pumps, gpm	16
Normal charging line flow, gpm	30
Maximum rate of boration with one transfer and one charging pump, ppm/min (EOL)	7.1
Equivalent cooldown rate to above rate of boration, °F/min (EOL)	2.1
Maximum rate of boron dilution (two charging pumps), ppm/min (BOL)**	5.9
Two-pump rate of boration, using refueling water, ppm/min (EOL)	9.8
Equivalent cooldown rate to above rate of boration, °F/min (EOL)	2.9
Design temperature of reactor coolant entering system at full power, °F	559.5
Design temperature of coolant return to reactor coolant system at full power, °F	509.5
Normal coolant discharge temperature to holdup tanks, °F	127.0

* Volumetric flow rates in gpm are based on 127°F and 15 psig.

** At HFP, Equilibrium Xenon, 1673 ppm boron.

Table 9.3-3 PRINCIPAL COMPONENT DATA SUMMARY

Heat Exchangers	Quantity[1]	Design Heat Transfer Btu/hr	Flow lb/hr	Letdown ΔT °F	Letdown Pressure psig Shell/tube	Design Temperature °F Shell/tube
Regenerative	1	$5.81x10^6$	19,760	268.5	2485/2735	650/650
Nonregenerative	2	$10.1x10^6$	19,760	164	150/600	250/400
Seal water	1	$1.137x10^6$	79,040	14.5	150/150	250/250
Excess letdown	1	$1.92x10^6$	4,940	364	150/2485	250/650

Pumps	Quantity[1]	Type	Capacity Each gpm	Head ft	Design Pressure psig	Design Temperature °F
Charging	3	Pos.displ.	60.5	2385 psi	3000	250
Boric acid transfer	2	Canned	40	152	150	250
Holdup tank recirc.	1*	Centrifugal	500	100	75	200
Reactor makeup water	2*	Centrifugal	270	300	150	250
Monitor tank	2*	Centrifugal	60	235	150	250
Concentrates holding tank transfer**	2*	Canned	20	150	100	250
Gas stripper feed**	2*	Canned	25	183	150	200
Gas stripper bottoms**	2	Canned	12.5	93	75	300

Tanks	Quantity[1]	Type	Volume, Each Gal. or as noted	Design Pressure psig	Design Temperature °F
Volume control	1	Vertical	220 ft^3	75Int/15Ext	250
Charging pump suction stabilizer	3	Vertical	5.0	150	250
Charging pump discharge pulsation dampener.	3	Vertical	2.5	3000	250
Boric acid	3*	Vertical	5000	Atmos.	250
Chemical mixing	1	Vertical	3.0	200	200
Batching	1*	Jacket Btm.	800	Atmos.	250
Holdup	3*	Vertical	7800 ft^3	15	200
Reactor makeup water	1*	Diaphragm	100,000	Atmos.	125
Concentrates holding	1**	Vertical	900	Atmos.	250
Monitor	4*	Diaphragm	10,000	Atmos.	125

Demineralizers	Quantity[1]	Type	Resin Volume ft^3	Flow gpm	Design Pressure psig	Design Temperature °F
Mixed bed	2	Flushable	20	90	200	250
Cation bed	1	Flushable	12	40	200	250
Evaporator feed	4*	Flushable	12	12.5	200	250
Evaporator condensate	3**	Fixed	12	12.5	200	250
Deborating	2	Fixed	30	109	200	250

1 Quantity per unit unless otherwise specified.
* Shared or capable of being shared by Unit 1 and Unit 2; items not marked are duplicated for each unit.
** Equipment no longer used.

Table 9.3-4 PARAMETERS USED IN THE CALCULATION OF REACTOR COOLANT FISSION PRODUCT ACTIVITIES (Reference 3)

1.	Core thermal power, MWt	1810.8
2.	Fraction of fuel containing clad defects	0.01
3.	Reactor coolant liquid volume/mass, ft^3/g	$5689/1.147 \times 10^8$
4.	Reactor coolant core average temperature, °F	581
5.	Purification flow rate (normal), gpm	40
6.	Effective cation demineralizer flow, gpm	4
7.	Volume control tank volumes	
	a. Vapor, cu ft	122
	b. Liquid, cu ft	98
8.	Fission product escape rate coefficients:	
	a. Noble gas isotopes, sec^{-1}	6.5×10^{-8}
	b. Br, I and Cs isotopes, sec^{-1}	1.3×10^{-8}
	c. Te isotopes, sec^{-1}	1.0×10^{-9}
	d. Mo isotopes, sec^{-1}	2.0×10^{-9}
	e. Sr and Ba isotopes, sec^{-1}	1.0×10^{-11}
	f. Y, La, Ce and Pr isotopes, sec^{-1}	1.6×10^{-12}
9.	Mixed bed demineralizer decontamination factors:	
	a. Noble gases	1.0
	b. Br ,I, Sr, Ba isotopes	10.0
	c. Other isotopes	1.0
10.	Cation bed demineralizer decontamination factor for Rb-86, Cs-134, Cs-137	10.0
11.	Volume control tank noble gas stripping fraction (closed system):	

Isotope	Stripping Fraction
Kr-83m	7.5×10^{-1}
Kr-85	5.8×10^{-5}
Kr-85m	5.5×10^{-1}
Kr-87	8.1×10^{-1}
Kr-88	6.6×10^{-1}
Kr-89	9.9×10^{-1}
Xe-131m	1.3×10^{-2}
Xe-133	2.9×10^{-2}
Xe-133m	6.6×10^{-2}
Xe-135	2.9×10^{-1}
Xe-135m	9.4×10^{-1}
Xe-137	9.8×10^{-1}
Xe-138	9.4×10^{-1}

Table 9.3-5 REACTOR COOLANT SYSTEM EQUILIBRIUM ACTIVITIES (Reference 3)

Activation Products

	(µCi/gm)
Cr-51	5.40E-03
Mn-54	1.60E-03
Fe-55	2.10E-03
Fe-59	5.10E-04
Co-58	1.40E-02
Co-60	1.30E-03

Gaseous Fission Products

	(µCi/gm)
Kr-83m	5.30E-01
Kr-85	1.05E+01
Kr-85m	2.17E+00
Kr-87	1.44E+00
Kr-88	4.01E+00
Kr-89	1.15E-01
Xe-131m	3.23E+00
Xe-133	2.91E+02
Xe133m	5.23E+00
Xe-135	9.25E+00
Xe-135m	5.96E-01
Xe-137	2.20E-01
Xe-138	7.94E-01

Nonvolatile Fission Products

	(µCi/gm)
Br-83	1.12E-01
Br-84	5.81E-02
Br-85	6.87E-03
Rb-86	2.72E-02
Rb-88	4.97E+00
Rb-89	2.31E-01
Sr-89	4.57E-03
Sr-90	2.15E-04
Sr-92	1.44E-03
Y-90	5.96E-05
Sr-91	6.66E-03
Y-91	5.88E-04
Y-91m	3.56E-03

Nonvolatile Fission Products(cont'd)

	(µCi/gm)
Y-92	1.25E-03
Y-93	4.23E-04
Zr-95	6.68E-04
Nb-95	6.65E-04
Mo-99	8.50E-01
Tc-99m	7.83E-01
Ru-103	5.64E-04
Rh-103m	5.64E-04
Ru-106	1.79E-04
Rh-106	1.79E-04
Ag-110m	1.07E-03
Te-125m	3.85E-04
I-127[a]	8.57E-11
Te-127	1.17E-02
Te-127m	3.35E-03
Te-129	1.28E-02
Te-129m	1.13E-02
I-129	5.02E-08
I-130	2.16E-02
I-131	2.82E+00
Te-131	1.45E-02
Te-131m	3.38E-02
I-132	3.17E+00
Te-132	3.15E-01
I-133	4.90E+00
Te-134	3.76E-02
I-134	7.46E-01
Cs-134	2.46E+00
I-135	2.81E+00
Cs-136	2.57E+00
Cs-137	2.09E+00
Ba-137m	1.98E+00
Cs-138	1.21E+00
Ba-140	4.26E-03
La-140	1.40E-03
Ce-141	6.39E-04
Ce-143	5.69E-04
Pr-143	6.32E-04
Ce-144	4.88E-04
Pr-144	4.88E-04

[a] Gram of I-127 per gram of coolant.

Table 9.3-6 TRITIUM PRODUCTION IN THE REACTOR COOLANT ONE UNIT (Reference 1)
Note: RCS tritium level was not specifically evaluated for EPU conditions, but can be expected to increase approximately proportional to the power level. Tritium effluents at EPU conditions are discussed in Section 11.2 and Appendix I.3.

Basic Assumptions (Plant Parameters):

1.	Core thermal power, MWt	1650
2.	RCS water volume (at T_{HOT}), ft^3	5880
3.	RCS core water mass, kg	6990
4.	Plant full power operating time for equilibrium (days)	500
5.	Boron Concentrations (equilibrium cycle), ppm	1435
6.	Fuel Rod / burnable poison release fraction	0.1
7.	RCS lithium concentration, ppm	2.2
8.	Li purity (atom percent Li-7)	99.9

RESULTS

Tritium Source	Total Produced Ci/cycle	Design Release to Coolant Ci/cycle
Ternary fission	8250	825
Fuel containing boron	882	88
Coolant soluble boron	286	286
Coolant soluble lithium	76	76
Coolant deuterium	2	2
TOTALS	9495	1277

Table 9.3-7 MALFUNCTION ANALYSIS OF CHEMICAL AND VOLUME CONTROL SYSTEM

	Component	Failure	Comments and Consequences
1.	Letdown line	Rupture in the line inside the reactor containment	The remote motor-operated valve located near the main coolant loop is closed on low pressurizer level to prevent supplementary loss of coolant through the letdown line rupture. The orifice block valves are closed when the motor operated valve closes. The containment isolation valves in the letdown line inside and outside the reactor containment are automatically closed by the containment isolation signal initiated by the safety injection signal. The closure of either containment isolation valve prevents leakage of the reactor containment atmosphere to the outside atmosphere.
2.	Normal and auxiliary charging line	See above	The check valves located near the main coolant loops prevent supplementary loss of coolant through the line rupture. The remote-operated valve located upstream of the check valve in the defective line also may be closed to isolate the reactor coolant system from the rupture. The check valves located at the boundary of the reactor containment prevent leakage of the reactor containment atmosphere outside the reactor containment.
3.	Seal water return line	See above	The motor-operated isolation valve located outside the containment and the air-operated valve located inside containment are manually closed or are automatically closed by the containment isolation signal initiated by the safety injection. The closure of either valve prevents leakage of the reactor containment atmosphere outside the reactor containment.

Figure 9.3-1UNIT 1 CHEMICAL AND VOLUME CONTROL

[redacted]

Figure 9.3-2 UNIT 1 CHEMICAL AND VOLUME CONTROL

[redacted]

Figure 9.3-3 UNIT 1 CHEMICAL AND VOLUME CONTROL

[redacted]

Figure 9.3-4 UNIT 1 CHEMICAL AND VOLUME CONTROL

[redacted]

Figure 9.3-5 UNIT 1 CHEMICAL AND VOLUME CONTROL

[redacted]

9.4 FUEL HANDLING SYSTEM (FH)

The fuel handling system provides a safe and effective means of transporting and handling fuel from the time it reaches the plant in an unirradiated condition until it leaves the plant after postirradiation cooling. The system is designed to minimize the possibility of mishandling or maloperations that cause fuel damage and potential fission product release.

The fuel handling system consists basically of:

1. The reactor cavity, which is flooded only during plant shutdown for refueling
2. The spent fuel pool, shared by the two units, which is kept full of water during and after the first refueling and is always accessible to operating personnel
3. The fuel transfer system, consisting of an underwater conveyor that transports fuel assemblies between the reactor cavity and the spent fuel pool.

9.4.1 DESIGN BASIS

Prevention of Fuel Storage Criticality

Criterion: Criticality in the new and spent fuel storage pits shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls. (GDC 66)

The new fuel storage area has accommodations as defined in Table 9.4-1 and is designed so it is impossible to insert assemblies in locations other than storage locations in the new fuel racks. Administrative controls are used to ensure that fuel stored in the new fuel storage racks complies with the requirements of the criticality analyses described in FSAR Section 9.4.2, including the use of a 3 out of 4 checkerboard arrangement when required. The fuel in the New Fuel Storage Vault is stored vertically and in an array with sufficient center-to-center distance between assemblies to assure $k_{eff} \leq 0.95$ as described in Technical Specification 4.3.1.

The spent fuel storage pool has accommodations as defined in Table 9.4-1. A criticality analysis was performed in Reference 1, submitted in Reference 11 with addendum in Reference 19 and approved in Reference 17. This amendment changed the licensing basis for spent fuel pool criticality requirements to 10 CFR 50.68(b) (Reference 8). The criticality analysis demonstrates that the spent fuel pool meets the requirements of 10 CFR 50.68(b)(4) to maintain keff less than 1.0 if filled with unborated water and keff less than or equal to 0.95 (including all biases and uncertainties) by maintaining at least 402 ppm boron in the spent fuel pool (Technical Specification 4.3.1.1 (c)). The analysis in Reference 1 assumes no Boraflex is present in the spent fuel storage racks to maintain subcriticality. Subcritical requirements are maintained by storing fuel in the “Acceptable” range of Technical Specification 3.7.12, Figure 3.7.12-1, considering initial enrichment, burnup and decay time of each fuel assembly. Fuel in the “Unacceptable” range is stored in accordance with Technical Specification 4.3.1.1.

Reference 1, Figure 3-5 also identified allowable IFBA patterns of 52 or less IFBA pins that can be credited for determining acceptable storage in the Point Beach spent fuel pool. IFBA patterns of 52 or less IFBA pins other than those shown in Reference 1 will require a 10 CFR 50.59

evaluation to validate that the conclusions from the analysis documented in Reference 1 remain unchanged. Such an evaluation was performed in Reference 21 and Reference 22. This evaluation identified additional IFBA patterns, with less than 52 pins that can be credited for acceptable storage in the Point Beach spent fuel pool. Any IFBA loadings of greater than 52 pins per assembly up to 120 pins are allowed with no IFBA pattern restrictions (Reference 23). Note that any IFBA length 120 inches or greater and any loading of 1.OX IFBA or greater (e.g., 1.5X, 2.OX, etc) are acceptable, as identified in Reference 1.

Reference 1 also considered the amount of soluble boron necessary to mitigate a misloaded 5.0 w/o fresh fuel assembly into a location intended for a burned fuel assembly. The amount of boron necessary to mitigate this accident and maintain the spent fuel pool keff less than or equal to 0.95 (including all biases and uncertainties) is 664 ppm. 664 ppm is well within the 2100 ppm minimum boron concentration required in the spent fuel pool by Technical Specification 3.7.11. Administrative controls ensure that fuel stored in the spent fuel pool meets the requirements of Technical Specifications and the criticality analysis described above and meets the requirements of 10 CFR 50.68(b)(1).

Because Reference 1 credits soluble boron, an additional analysis for boron dilution was included in Reference 11. The analysis concluded that a substantial volume of water is required to dilute the spent fuel pool from 2100 ppm to 664 ppm. This volume of water would be detected by the high level alarm, plant flooding or by operator rounds through the spent fuel pool area.

The Fuel Upgrade/Power Uprating Reload Transition Safety Report (Reference 18) concluded that the storage of 422V+ fuel also meets the required criteria for spent fuel and new fuel storage.

In addition, the spent fuel pool has an area set aside for accepting spent fuel shipping casks or dry storage casks. Cask loading is also done under water. Borated water is used to fill the spent fuel storage pool at a concentration to match or exceed that used in the reactor cavity and refueling canal during refueling operations. The fuel in the spent fuel pool is stored vertically in an array with sufficient center-to-center distance to assure keff <1.0 even if unborated water were to fill the space between the assemblies. The maximum nominal U-235 enrichment of the fresh fuel assemblies is limited to five (5.0) weight by percent per the requirements of 10 CFR 50.68(b)(7) and Technical Specification 4.3.

Detailed instructions are available for use by refueling personnel. These instructions, the minimum operating conditions, and the design of the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety.

Fuel and Waste Storage Decay Heat

Criterion: Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities and to waste storage tanks that could result in radioactivity release which would result in undue risk to the health and safety of the public. (GDC 67)

The refueling water provides reliable and adequate cooling medium for spent fuel transfer. Heat removal from the spent fuel pool is provided by the spent fuel cooling system specifically installed for this purpose. Natural radiation and convection is adequate for cooling the holdup tanks.

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)

Adequate shielding for radiation protection is provided during reactor refueling by conducting all spent fuel transfer and storage operations under water. This permits visual control of the operation at all times while maintaining low radiation levels, typically <5 mr/hr, for periodic occupancy of the area by operating personnel. Spent fuel pool water level is indicated by a level transmitter which causes an audible alarm in the control room on high or low level, and water removed from the pool must be pumped out since there are no gravity drains. Shielding is provided for waste handling and storage facilities to permit operation within requirements of 10 CFR 20.

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

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)

All fuel storage facilities are contained and equipment designed so that accidental releases of radioactivity directly to the atmosphere are monitored and do not exceed the dose criteria of 10 CFR 50.67 (Reference 20).

The reactor cavity, refueling canal and spent fuel storage pool are reinforced concrete structures with seam-welded stainless steel plate liners. These structures are designed to withstand the anticipated earthquake loadings as Class I structures so that the liner prevents leakage even if the reinforced concrete develops cracks. All operating areas in the fuel storage facilities are adequately ventilated. The exhausts of the ventilation system in the waste storage and drumming areas are monitored for radioactivity and are discharged via stacks through the top of the auxiliary building and facade.

All vessels in the waste disposal system which are used for waste storage are Class I seismic design.

9.4.2 SYSTEM DESIGN AND OPERATION

Various sections of the fuel handling system are shared by Units l and 2. These include a common spent fuel storage pool and a common new fuel storage area. This is discussed further in Appendix A.6.

The reactor is refueled with equipment designed to handle the spent fuel under water from the time it leaves the reactor vessel until it is placed in a cask for shipment from the site. Boric acid is added to the water to ensure subcritical conditions during refueling.

The fuel handling system, shown in Figure 9.4-1, may be generally divided into two areas: the refueling cavity (which is flooded only during plant shutdown for refueling) and the spent fuel pool (which is full of water during and after the first refueling and is always accessible to operating personnel). These two areas are connected by the fuel transfer system consisting of an underwater conveyor that carries the fuel through an opening in the units containment.

The refueling cavity is flooded with borated water from the refueling water storage tank. In the refueling cavity, fuel is removed from the reactor vessel, transferred through the water and placed in the fuel transfer system by a manipulator crane. In the spent fuel pool the fuel is removed from the transfer system and placed in storage racks with a long manual tool suspended from an overhead hoist. After a sufficient decay period, the fuel may be removed from storage and loaded into a shipping cask for removal from the site or loaded into a dry storage cask for temporary storage at the Point Beach Independent Spent Fuel Storage Installation (ISFSI) under 10 CFR 72. Both the manipulator crane and the long handled tool can handle only one fuel assembly at a time.

New fuel assemblies are received and stored in racks in the new fuel storage area or in the spent fuel pool. New fuel is delivered to the reactor by transferring it into the spent fuel pool and taking it through the transfer system. The new fuel storage area is sized for storage of the fuel assemblies and control rods normally associated with the replacement of one-third of a core plus space for another one-third core. Fuel handling data are given in Table 9.4-1.

<u>Major Structures Required for Fuel Handling</u>

<u>Refueling Cavity</u>

The refueling cavity is a reinforced concrete structure that forms a pool above and adjacent to the reactor when it is filled with borated water for refueling. The cavity is filled to a depth that limits the radiation at the surface of the water to less than 5 millirems per hour during fuel assembly transfer.

The reactor vessel flange is sealed to the bottom of the upper refueling cavity by a clamped, gasketed seal ring which prevents leakage of refueling water from the cavity. This seal is fastened and closed after reactor cooldown but prior to flooding the cavity for refueling operations. Potential leakage past the seal would go to the keyway under the reactor vessel and cause an alarm on the sump level instrument.

The lower refueling cavity is large enough to provide storage space for the reactor upper and lower internals, several control cluster drive shafts removed from the upper internals, and miscellaneous refueling tools. The floor and sides of the refueling cavity are lined with stainless steel. A skimmer pump system improves the surface water conditions during refueling.

<u>Transfer Canal</u>

The transfer canal is a passageway extending from the lower refueling cavity to the inside surface of the reactor containment where it aligns with the transfer tube and from the outside surface of containment along the East side of the spent fuel pool. The transfer canal walls and floor are lined with stainless steel.

In containment, the floor of the canal is approximately five feet below the floor of the lower refueling cavity to provide the greater depth required for the fuel transfer system upending device. The containment side of the transfer canal is drained after a refueling.

Outside containment, the walls of the transfer canal extend upward to the same elevation as the top of the spent fuel pool. Two gates in the wall between the transfer canal and the spent fuel pool allow for transfer of fuel assemblies from one area to the other while maintaining the fuel assembly below water for shielding purposes. The gates maintain spent fuel pool inventory and allow the transfer canal to be drained for maintenance of fuel handling equipment. The elevation of the bottom of the gates is above the top of the spent fuel racks. The gates employ inflatable seals supplied by Instrument Air and a redundant static seal that is seated to the door jamb by hydrostatic force.

The transfer tube connects the two portions of the transfer canal and is isolated by a Transfer Tube Closure assembly inside containment and a gate valve outside containment. Each unit has a transfer tube going from containment to the transfer canal.

Refueling Water Storage Tank

The normal function of the refueling water storage tank is to supply borated water to the refueling cavity for refueling operations. In addition, the tank provides borated water for delivery to the core following either a loss-of-coolant or a steam line rupture accident. This is described in Section 6.2.

The capacity of the tank is based upon the requirement for filling the refueling cavity. The water in the tank is borated to a concentration which assures reactor shutdown by at least 5% $\delta k/k$ when all RCC assemblies are inserted and when the reactor is cooled down for refueling. The tank design parameters are given in Section 6.2.

Spent Fuel Storage Pool

The spent fuel storage pool is designed for the underwater storage of spent fuel assemblies and control rods and other inserts after their removal from the reactor. New fuel assemblies may also be stored in the pool. Spent fuel pool accommodations are listed in Table 9.4-1. Spent fuel assemblies are handled by a long-handled tool suspended from an overhead hoist and manipulated by an operator standing on the movable bridge over the pool. Storage racks are provided to hold spent fuel assemblies and are erected on the pool floor. Fuel assemblies are held in a rectangular array, and placed in vertical cells. The racks are designed so that it is impossible to store fuel assemblies within the racks in other than a storage module, thereby ensuring the necessary spacing between assemblies. Control rod clusters are stored in place inside the spent fuel assemblies. One inspection location in the spent fuel pool allows rotation of a fuel assembly for visual inspection, but not for storage. The spent fuel storage pool is constructed of reinforced concrete and is Class I seismic design. The entire interior basin face and transfer canal is lined with stainless steel plate.

The spent fuel pool is divided into two parts by an internal dividing wall whose lowest point is approximately 3 ft. above the top of the stored spent fuel. The north portion of the pool contains an area reserved for the loading of the spent fuel shipping cask or dry storage cask. Administrative controls are such that no heavy loads, such as a spent fuel shipping cask or spent resin shipping cask, are transported over or placed in either part of the pool when fuel is stored in that part, unless suitable precautions are taken.

The spent fuel storage racks for the Point Beach Nuclear Plant are designed in accordance with Regulatory Guide 1.29, Revision 2, as seismic Category I components. The structural analysis of the racks has considered all the loads and load combinations specified in the NRC Standard Review Plan. The steel structure of the rack not only provides a smooth, all welded stainless steel box structure to preclude damage during normal and abnormal load conditions, but also provides an additional margin of safety in the form of internal structural damping created by the large areas of bearing surface between boxes in the array.

Auxiliary Building Crane

The auxiliary building crane has been modified to conform with single-failure-proof criteria. This modification evolved as a result of concern over the movement of heavy loads over or near the spent fuel pool when spent fuel is stored there (Reference 2 and Reference 5). The crane is designed to not allow a load drop as a result of any single constituent component failure.

The PAB superstructure has been analyzed for the capability of the structure to support and hold the crane with its full rated lift load of 125 tons plus a roof snow load and a concurrent seismic (OBE or SSE) event or a lift of 125 tons plus a roof snow load and design wind loads. (Reference 15)

New Fuel Storage

New fuel assemblies and control rods can be stored in a separate area that facilitates the unloading of new fuel assemblies or control rods from trucks. This storage vault is designed to hold new fuel assemblies in specially constructed racks and is utilized primarily for the storage of the replacement fuel assemblies. The new fuel assemblies are stored in dry racks arranged to space the fuel assemblies such that the maximum k_{eff} should the new fuel storage area be inadvertently filled with the most reactive water density is less than 0.95.

The new fuel storage area was evaluated by Westinghouse calculation CAB-98-292 submitted by Westinghouse letter 98WE-G-0052 (Reference 10). The analysis has shown that based on a center to center distance of 19 inches, k_{eff} remains below 0.95 for the fully flooded condition, and 0.98 for the optimum moderation condition. This meets the requirements of 10 CFR 50.68(b)(2) and 10 CFR 50.68(b)(3) and Technical Specification 4.3.1.2. All new fuel assemblies with an enrichment of 5 w/o U-235 or less and containing a minimum of 32 1.25X IFBA rods may utilize all available storage locations in the new fuel storage area.

The new fuel storage area was also evaluated by Westinghouse calculation CAB-99-318 submitted by Westinghouse letter 99WE-G-0043 (Reference 16). The analysis has shown that based on a center to center distance of 19.5 inches, k_{eff} remains below 0.95 for the fully flooded condition, and 0.98 for the optimum moderation condition. This meets the requirements of 10 CFR 50.68(b)(2) and 10 CFR 50.68(b)(3) and Technical Specification 4.3.1.2. All new fuel assemblies with an enrichment of 5.00 w/o U-235 or less may occupy cells in a 3 out of 4 checkerboard arrangement. The 3 out of 4 storage arrangement with empty cell means that three fuel assemblies can occupy three storage cells with the other cell being empty in any 2 x 2 array of storage cells. This analysis is valid for Westinghouse STD, OFA, and various advanced fuel products.

Major Equipment Required for Fuel Handling

Reactor Vessel Stud Tensioner

The stud tensioner is a hydraulically operated (oil is the working fluid) device provided to permit preloading and unloading of the reactor vessel closure studs at cold shutdown conditions. Stud tensioners were chosen in order to minimize the time required for the tensioning or unloading operations. Three tensioners are applied simultaneously to three studs 120° apart. One hydraulic pumping unit operates the tensioners which are hydraulically connected so equal pressure is applied simultaneously to the three studs. The studs are tensioned to their operational load using a controlled procedure to prevent high stresses in the reactor vessel and head flange region and unequal loadings in the studs.

A pressure control valve and relief valve are provided on the hydraulic pump assembly to prevent over tensioning the studs due to excessive hydraulic pressure.

Tables of tensioning sequence and oil pressure are included in the operating instructions. Stud elongation measuring equipment is provided to measure the elongation of the studs after tensioning to determine the acceptability of the final tensioning.

Reactor Vessel Head Lifting Device

The reactor vessel head lifting device consists of a welded and bolted structural steel frame with lifting tripod to enable the crane operator to lift the head and store it during refueling operations. The lifting device, including the lifting tripod, remains attached to the reactor vessel head during power operation. (Reference 13 and Reference 14)

Reactor Internals Lifting Device

The reactor internals lifting device is a fixture provided to remove the upper reactor internals package and to move it to a storage location in the refueling cavity. The device is lowered onto the guide tube support plate of the internals and is manually bolted to the support plate by three bolts. The bolts are controlled by long torque tubes extending up to an operating platform on the lifting device. Bushings on the fixture engage guide studs mounted on the vessel flange to provide close guidance during removal and replacement of the internals package. This lifting device can also be used to remove the lower internals once the vessel has been cleared of all fuel assemblies.

Manipulator Crane

The manipulator crane is a rectilinear bridge and trolley crane with a vertical mast extending down into the refueling water. The bridge spans the reactor cavity and runs on rails set into the floor along the edge of the reactor cavity. The bridge and trolley motions are used to position the vertical mast over a fuel assembly in the core. A long tube with a pneumatic gripper on the end is lowered out of the mast to grip the fuel assembly. The gripper tube is long enough so the upper end is still contained in the mast when the gripper end contacts the fuel. A hoist mounted on the trolley raises the gripper tube and fuel assembly up into the mast tube. The fuel is transported while inside the mast tube to its new position. The manipulator can lift only one fuel assembly at a time. An unlatching stand is installed in the cavity to enable unlatching the gripper underwater and avoid having to drain the cavity should the gripper be accidentally put into the latch position while not engaged in a fuel assembly.

All controls for the manipulator crane are mounted on a console on the trolley. The bridge is positioned on a coordinate system consisting of index plates installed along the refueling cavity. A video camera located on the bridge truck indicates the position of the bridge via a TV monitor located on the control console. The trolley is positioned with the aid of a scale on the bridge structure. The scale is read directly by the operator at the console. The bridge, trolley, and hoist motors are controlled with a variable frequency drive. This allows for variable speed control of the motors as well as separate slow speed (jog) control for each motor. Electrical interlocks and limit switches on the bridge and trolley drives protect the equipment. In an emergency, the bridge, trolley, and hoist can be operated manually using a handwheel on the motor shaft.

The suspended weight on the gripper tool is monitored by an electric load cell indicator mounted on the control console. An excessive load stops the hoist drive from moving in the up direction. The gripper is interlocked through a weight sensing device and also a mechanical spring lock so that it cannot be opened when supporting a fuel assembly.

Safety features are incorporated in the system as follows:

1. Travel limit switches on the bridge and trolley drives.

2. Bridge, trolley, and hoist drives which are mutually interlocked to prevent simultaneous operation of any two drives.

3. A position safety switch (GRIPPER TUBE UP) prevents bridge and trolley motion when the gripper is in the ENGAGED position on a fuel assembly except when it is actuated. Also, a geared limit position switch allows for bridge and trolley motion when the inner mast is just inside the outer mast and the gripper is in the DISENGAGED position without the weight of an assembly added to the mast. This allows for faster refueling movements as an empty inner mast is not required to travel to the top of the outer mast before the bridge and trolley are allowed to move.

4. An interlock which prevents the opening of a solenoid valve in the air line to the gripper except when a programmed suspended weight is indicated by a digital readout on the control console. As backup protection for this interlock, the mechanical weight-actuated lock in the gripper prevents operation of the gripper under load even if air pressure is applied to the operating cylinder.

5. The OVERLOAD interlock switch, which opens the hoist drive circuit in the up direction when the loading is excessive.

6. An interlock on the hoist drive circuit in the up direction, which permits the hoist to be operated only when either the ENGAGED or DISENGAGED indicating switch on the gripper is actuated.

7. An interlock of the bridge and trolley drives, which prevents the bridge drive from traveling beyond the edge of the core unless the trolley is aligned with the refueling canal centerline. The trolley drive is locked out when the bridge is beyond the edge of the core.

Suitable restraints are provided between the bridge and trolley structures and their respective rails to prevent derailing. The manipulator crane is designed to prevent disengagement of a fuel assembly from the gripper in the event of a maximum potential earthquake. The auxiliary hoist is used for the rod latching tool, plug device tool, and other tools used in the refueling cavity.

Spent Fuel Pool Bridge

The spent fuel pool bridge is a wheel-mounted walkway, spanning the spent fuel pool which carries an electric monorail hoist on an overhead structure. A fuel assembly is moved within the spent fuel pool by means of a long handled tool suspended from the hoist. The hoist travel and tool length are designed to limit the maximum lift of a fuel assembly to a safe shielding depth.

The engineering specification for the design of the Spent Fuel Bridge included a 0.20g seismic loading in the horizontal and vertical directions. The limiting stress criteria employed with this seismic load results in a 3 to 1 factor of safety with respect to yield of the bridge steel. The maximum floor horizontal acceleration at the point of bridge support for the Design Basis Earthquake is 0.22g.

Fuel Transfer System

The fuel transfer system, shown in Figure 9.4-1 is an alternating current (AC) motor driven conveyor car that runs on tracks extending from the lower refueling cavity through the transfer tube and into the transfer canal next to the spent fuel pool. The conveyor car receives a fuel assembly in the vertical position from the manipulator crane. The fuel assembly is lowered to a horizontal position for passage through the tube, and then is raised to a vertical position for transfer to the spent fuel pool.

During plant operation, the conveyor car is stored in the fuel transfer canal outside of containment. The gate valve is closed and the Transfer Tube Closure hatch is installed on the transfer tube to seal the reactor containment.

Rod Cluster Control Changing Fixture

A fixture is mounted on the refueling cavity wall for removing rod cluster control (RCC) elements from spent fuel assemblies and inserting them into new fuel assemblies for reuse. The fixture consists of two main components; a guide tube mounted to the wall for containing and guiding the RCC element, and a wheel-mounted carriage for holding the fuel assemblies and positioning fuel assemblies under the guide tube. The guide tube contains a pneumatic gripper on a winch that grips the RCC element and lifts it out of the fuel assembly. By repositioning the carriage, another fuel assembly is brought under the guide tube and the gripper lowers the RCC element and releases it. There is a third position in the basket used for a temporary storage of an insert. The manipulator crane loads and removes the fuel assemblies into and out of the carriage. The gripper is also used for source and power suppression assembly changes.

9.4.3 SYSTEM EVALUATION

Underwater transfer of spent fuel provides safety in handling operations. Water is an effective and transparent radiation shield and a reliable cooling medium for removal of decay heat.

Basic provisions to ensure the safety of refueling operations are:

1. Gamma radiation levels in the containment, control room, and fuel storage areas are continuously monitored (see Section 11.5). These monitors provide an audible alarm at the initiating detector indicating an unsafe condition. Continuous monitoring of reactor neutron flux provides immediate indication and alarm in the control room, and in the containment, of an abnormal core flux level.

2. A minimum boron concentration, specified in the COLR, is required for MODE 6 refueling operations.

3. Whenever fuel is added to the reactor core, the source range neutron count rate is monitored to verify the subcriticality of the core.

Incident Protection

Direct communication between the control room and the operating floor of the containment is available whenever changes in core geometry are taking place. This provision allows the control room operator to inform the core loading supervisor of any impending unsafe condition detected from the main control board indicators during fuel movement.

The walls and the base of the pool will withstand all design tornado missiles. Calculations demonstrate that tornado generated winds will not remove any critical amount of water from the spent fuel pool. Any water removed in this way will leave adequate coverage to maintain cooling of the stored fuel elements. (Reference 12)

No special design features had been made for the spent fuel pool as far as turbine missiles were concerned because it had been believed that the worst low-trajectory missile could not have sufficient translational kinetic energy to reach the spent fuel pool. However, model tests initiated by Westinghouse contradicted this theory in the case of a turbine overspeed. Therefore, a completely independent turbine speed detection and valve trip initiation system for the turbine generators of Units 1 and 2 was provided to minimize the likelihood of a turbine generator unit overspeeding above the design speed. FSAR Section 14.1.12 gives more insight into this event.

Malfunction Analysis

An analysis is presented in Section 14.2.1 concerning damage to all of the fuel rods in an assembly, assumed as a conservative limit for evaluating the environmental consequences of a fuel handling accident.

9.4.4 REQUIRED PROCEDURES AND TESTS

Calibrations and operational tests of the fuel handling equipment are performed as required by the Technical Requirements Manual.

The minimum boron concentration in the spent fuel pool is monitored in accordance with Techncial Specification 3.7.11. (Reference 8)

9.4.5 REFERENCES

1. Westinghouse Report WCAP-16541-P, Revision 2, Point Beach Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis, dated June 2008.
2. NUREG-0612, Control of Heavy Loads at Nuclear Power Plants
3. WE letter VPNPD-96-029, Response to NRC Bulletin 96-02, Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment, dated May 9, 1996.
4. NRC SER, Re NUREG-0612, Control of Heavy Loads at Nuclear Power Plants, dated March 27, 1984.
5. NRC SER, Re Amendment Nos. 96 and 100, dated September 3, 1985.
6. NRC SER, Re Issuance of Amendments Nos. 35 and 41, dated April 4, 1979.
7. NRC SER, Re Issuance of Amendments Nos. 77 and 81, dated October 5, 1983.
8. 10 CFR 50.68, Criticality Accident Requirements.
9. NRC Letter to WE, Issuance of the Exemption from the Requirements of 10CFR70.24, dated October 6, 1997.
10. Westinghouse Letter 98WE-G-0052, Fresh Fuel Rack Criticality Analysis, dated October 21, 1998.
11. NRC 2008-0044, License Amendment Request 247, Spent Fuel Pool Storage Criticality Control, July 24, 2008.
12. Bechtel Topical Report B-TOP-3, Design Criteria for Nuclear Power Plants Against Tornadoes, (Proprietary) dated March 12, 1970.
13. Westinghouse Calculation CN-RVHP-04-10, “Point Beach Units 1 and 2 HAUP - Missile Impact Analysis,” (Westinghouse Proprietary), Rev 5, dated May 18, 2006.
14. Westinghouse Calculation CN-RVHP-04-12, “Point Beach HAUP Structural Analysis,” (Westinghouse Proprietary), Rev 2, dated March 24, 2006.
15. Automated Engineering Services Corp. Calculation PBNP-305336-S01, Rev. 1, “Structural Analysis of Central PAB with Crane Load of 125 Tons,” dated April 3, 2006.
16. Westinghouse Letter 99WE G 0043, Point Beach Fresh Fuel Storage 3-out-of-4 Configuration Final Criticality Analysis Report, dated August 23, 1999.
17. 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.
18. Westinghouse Fuel Upgrade/Power Uprating Reload Transition Safety Report for Point Beach Units 1 and 2, Revision 2, dated November 1, 1999.
19. Westinghouse Report WCAP-16541-NP Revision 2, Addendum 1, Point Beach Units 1 and 2 Spent fuel Pool Criticality Analysis - Addendum, dated November 2009.

20. 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.

21. Point Beach Nuclear Plant Evaluation 2011-007, “10 CFR 50.59 Evaluation - Justification of IFBA fuel rod patterns for the SFP Criticality Analysis.”

22. Engineering Evaluation EC 273511, Revision 0, “Justification of IFBA Pattern for the SFP Criticality Analysis.”

23. FPL Energy Point Beach Letter to NRC, NRC 2009-0057, “Response to Request for Additional Information, License Amendment Request 247, Spent Fuel Pool Storage Criticality Control,” dated May 22, 2009.

Table 9.4-1 FUEL HANDLING DATA

New Fuel Storage Area

Core storage capacity	≈2/3
Equivalent fuel assemblies	84
Center-to-center spacing of assemblies, in.	19 (min)
Maximum k_{eff} with the most reactive water density	<0.95
Maximum keff with optimum moderation	<0.98

Spent Fuel Storage Pool

South pool fuel assembly storage capacity	803
North pool fuel assembly storage capacity	699
Number of space accommodations for spent fuel cask loading	1
Maximum k_{eff} with borated water (402 ppm)	≤0.95
Maximum k_{eff} with unborated water	<1.0

Miscellaneous Details

Width of transfer canal, ft.	3
Wall thickness for spent fuel storage pool, ft.	4 to 6
Weight of fuel assembly with RCC (dry), lb.	1750
Capacity of refueling water storage tank, each, gal.	289,504
Quantity of water required for refueling, gal.	275,000

Figure 9.4-1 FUEL TRANSFER SYSTEM

[redacted]

9.5 PRIMARY AUXILIARY BUILDING VENTILATION SYSTEM

9.5.1 DESIGN BASIS

The Primary Auxiliary Building Ventilation (VNPAB) system is not required to perform any Safety Related functions. VNPAB system operation is credited for Primary Auxiliary Building (PAB) heat removal. No credit is taken in any accident analysis or habitability study for the filtration capability of the system. The PAB Ventilation system is credited in the event of a fire and evaluated in the at-power and non-power analyses (Reference 5).

9.5.2 SYSTEM DESIGN AND OPERATION

The auxiliary building ventilation air is supplied by a central supply fan which includes an air filter, heating coils, and service water supplied cooling coils. Sufficient outside air is supplied to maintain a once-through system with provisions available to recirculate air from the PAB central area. The system is balanced to maintain the auxiliary building at slightly negative pressure with respect to outside pressure and adjacent building pressures. This is accomplished by providing an exhaust flow capacity larger than the supply capacity. All the exhaust air is filtered through roughing and high efficiency filters for removal of particulates. Areas which have possible contamination from iodine vapor have the capability to be exhausted through activated carbon beds in addition to high efficiency filters if required. All air exhausted from these areas is then discharged through the auxiliary building vent stack, which is monitored for radiation. A radiation detector output above its set point will initiate exhaust filtration through the activated charcoal beds. The discharge of the combined air ejector is vented into the auxiliary building stack downstream of the filters.

The VNPAB exhaust system consists of two filter fans (W-30A&B), two stack fans (W-21A&B), and the associated ductwork, filter housings, and dampers necessary to ensure the required exhaust flow path can be maintained. Each of the two filter fans and each of the two stack fans are powered by independent safety related power supplies with EDG backup. Exhaust stack fan W-21A and exhaust filter fan W-30A are powered from the safety-related Class 1E, 480 V Motor Control Center (MCC) 1B-42. Exhaust stack fan W-21B and exhaust filter fan W-30B are powered from the safety-related Class 1E, 480 V MCC 2B-32. The filter and stack fan control switches are located in the control room on the back of the main control board. One filter fan and one stack fan are normally in operation. A low exhaust flow condition is indicated and alarmed in the control room.

Primary Auxiliary Building Battery and Inverter Room Ventilation system is discussed in FSAR Section 8.7.

9.5.3 SYSTEM EVALUATION

The Auxiliary Building Ventilation System provides sufficient control of building temperatures during normal, abnormal, and accident conditions to maintain equipment within operational temperature limits. This system also filters the exhaust from rooms potentially containing iodine vapor, and rooms potentially containing particulates, during normal and accident conditions to limit offsite releases, and support auxiliary building habitability.

The drumming station supply and exhaust systems are similar to the auxiliary building ventilation system with the exception that the exhaust system has no provision for iodine removal and is discharged via a separate, monitored vent stack.

No credit is given for the VNPAB exhaust system in the control room or offsite dose bounding analysis described in FSAR Chapter 14.3.5, Radiological Consequences of a Loss of Coolant Accident (Reference 1).

Restoration of the VNPAB system within two hours of a LOOP assures adequate cooling for PAB safety related equipment during the worst case design basis accident (Reference 4).

The VNPAB system is classified as non-safety related, however components in the exhaust system required to direct radioactive releases in the PAB to the vent stack are classified as AQ (Augmented Quality). The seismic adequacy of the VNPAB exhaust system has been demonstrated using a methodology that follows the guidelines of Reference 2 and Reference 3. The VNPAB exhaust system design provides redundancy for all active mechanical components and active and passive electrical components needed to provide PAB exhaust flow. The design considers relay failures; failures of contacts to change state; and the shorting of relay, solenoid, or starter coils that could cause a damper to change to an undesirable state or prevent starting of a fan. The failure analysis does not include conductor short circuits or failure of one conductor, cable or device causing a failure of another conductor, cable, or device in the same location or raceway. The VNPAB exhaust system fans are supplied from the safety related Class 1E system by safety related circuit breakers which will isolate a fault on the non-safety related portions of the system and keep it from propagating to the Class 1E system. The fan motors and power cables located in potentially harsh environments are qualified for the expected environmental conditions (Reference 1).

9.5.4 REQUIRED PROCEDURES AND TESTS

Gaseous waste monitoring of the Primary Auxiliary Building ventilation system is performed per the requirements of the Offsite Dose Calculation Manual (ODCM).

The VNPAB exhaust system is included in the scope of the Maintenance Rule (10 CFR 50.65) and the License Renewal (10 CFR 54.37(b)) programs. The W-30A&B filter fan motors and associated power cables, and the power cables to the W-21A&B stack fans are included in the scope of the EQ Program (10 CFR 50.49).

9.5.5 REFERENCES

1. 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.

2. Seismic Qualification Utility Group (SQUG), "Generic Implementation Procedure (GIP) For Seismic Verification of Nuclear Plant Equipment," Revision 2, Corrected February 14, 1992.

3. Electric Power Research Institute Final Report 1014608, "Seismic Evaluation Guidelines for HVAC Duct and Damper Systems: Revision to 1007996," dated December, 2006.

4. NRC Safety Evaluation, "Point Beach Nuclear Plant (PBNP), Units 1 and 2 -Issuance of License Amendments Re: Auxiliary Feedwater System Modification (TAC Nos. ME1081 and ME1082)," dated March 25, 2011.

5. NFPA 805 Fire Protection Program Design Document (FPPDD).

9.6 SERVICE WATER SYSTEM (SW)

9.6.1 DESIGN BASIS

The Service Water (SW) system shall provide sufficient flow to support the heat removal requirements of components required to mitigate the consequences of a Loss of Coolant Accident (LOCA) in one unit, while supporting the normal flow of the unaffected unit. Although SW is required to mitigate other plant accidents as well, a LOCA combined with normal operation of the unaffected unit is the most limiting event for the heat load imposed on the SW system.

The SW system shall provide sufficient flow to the spent fuel pool heat exchangers to provide adequate heat removal of spent fuel decay heat (see Section 9.9, Spent Fuel Pool Cooling and Filtration). The SW system shall provide a long term makeup water source to the suction of the auxiliary feedwater (AF) pumps when the normal makeup source (the CSTs) is not available.

The service water system also has the following augmented quality functions. The SW system shall supply water to safe shutdown equipment and fire suppression in the G-01 and G-02 diesel generator rooms and containment hose reels during plant fires (Reference 5). The SW system provides cooling water for the blowdown evaporator heat exchangers.

The service water system is sized to ensure adequate heat removal based on the highest expected temperatures of cooling water, maximum loading and leakage allowances. Calculations show that adequate service water flow is available at 85°F indicated temperature to transfer the design basis accident heat loads during the post-DBA injection and recirculation phases with three service water pumps in operation. All essential safety related heat exchangers have been demonstrated by analysis to be capable of transferring their design basis heat loads at 85°F (Reference 3).

The pumphouse structure has been designed to remain intact under a tornado wind having a tangential velocity of 300 mph plus a forward progress of 60 mph. The structure is capable of remaining intact for a pressure drop of 1/2 psi. Before this pressure drop is realized, the building would be vented by the failure of the louvers and doors. Interior missile shield walls and exterior walls protecting the service water pumps are constructed of reinforced concrete with a minimum thickness of 12”. The internal missile shield walls have been located to preclude the possibility of damage from a missile passing through a louver or door. Reinforced concrete walls of 12” thickness cannot be penetrated by the design tornado missiles (Reference 8 and Reference 9).

9.6.2 SYSTEM DESIGN AND OPERATION

The service water (SW) system flow diagrams are shown in Figure 9.6-1 through Figure 9.6-7. The service water system has six electric motor driven centrifugal pumps which take a suction from the pump bays in the Circulating Water (CW) pump house. Two service water pumps are connected to separate 480 volt buses (2B-03 and 1B-04), one per bus. The four remaining pumps are connected, two per bus, to two separate 480 volt buses (1B-03 and 2B-04). In the event of a loss of normal electrical power to the safeguard buses, each of the emergency diesel-generator units are sized to supply three service water pumps in addition to the other vital engineered safeguards loads powered from that train for the unit in which the event occurred, as well as, the loads required by the other unit to maintain a hot shutdown condition. Four service water pumps (P-32B, C, E, F) can be supplied power from the X-08 transformer when normal power is unavailable.

The service water pumps supply a header which exits the pumphouse through two below ground pipelines leading to the Class I section of the control building. The two pipelines, called North and South headers, run to the auxiliary building where they rejoin to form the West header. The West header consists of the piping between MOVs SW-2869 and SW-2870. Motor operated valves (SW-2869, 2870, 2890, and 2891) allow isolation of the main loop headers in the event of a piping failure such that the safe shutdown function of the SW system can be retained. The piping failure is considered a passive failure and is not assumed to occur concurrent with a design basis accident. The return lines are manifolded by areas and are discharged to the condenser circulating water discharge in either Unit 1 and/or Unit 2.

The SW system, serving both units, supplies cooling water to equipment in the steam plant, to the containment ventilation coolers and to the reactor auxiliary systems. Non-essential services in each unit receive water from their respective header (North or South).

Supply of service water for essential services is redundant and can be maintained in case of failure of one loop section header (Reference 3 and Reference 4). Table 9.6-1 is a list of the essential service loads supplied by the service water (SW) system. Return service water is directed to the return line of the circulating water (CW) system.

The service water system pumps and motor operated valves are operated from the C01 control panel in the control room. The service water system is normally operated with both the North and South supply header cross connect valves open and the West Header cross connect valves open. Normally, two of the six pumps are capable of carrying the required normal cooling load for the two units. During periods of higher lake temperatures or when RHR cooling is in service, operation of three pumps is normally necessary. Service water pump flowrate is dependent upon the number of pumps running, the system valve lineup and positioning. Typical flowrates for the system in accident conditions vary from about 3,000 gpm to 21,000 gpm. Control room operators can shift SW pumps, split the SW headers, and isolate various SW loads as the plant requires.

The service water pumps are connected to the 480 volt safeguards buses and can be supplied by the Emergency Diesel Generators (DG) in the event of loss of offsite power. Under the conditions of a loss-of-coolant accident (LOCA) and concurrent loss-of-offsite power (LOOP), any three SW pumps are capable of providing the necessary cooling capacity for the essential loads for the affected unit and supply service water for the normal operation of the unaffected unit (Reference 3).

With a Safety Injection (SI) signal present, the containment cooler outlet valves (1/2SW-2907 and 1/2SW-2908) open, non-essential service water load valves close, and all six service water pumps receive start signals on a timed sequence. In the case of an undervoltage condition coincident with an SI signal, bus voltage must be restored before these actions begin. Table 9.6-2 lists the valves that close to isolate non-essential service water loads.

The containment ventilation coolers (HX-15) are supplied in pairs from the service water loop. The redundant motor operated valves in the containment cooler service water discharge lines (1/2SW-2907,2908) will automatically open on a safeguards actuation signal. Each cooler inlet and outlet are provided with a manual shutoff and drain capability. Manual valves allow each cooler to be isolated individually for leak testing. Service water to each cooler is isolated during the performance of the integrated leakage rate test. The containment ventilation cooler SW discharge lines are continuously monitored for radioactivity. A small bypass flow from the return line of each cooler is diverted through a common header to radiation monitor 1/2RE-216. Upon indication of radioactivity in the common monitor, each cooler discharge line could be monitored individually to locate a defective cooler. The defective cooler might then be removed from service with its manual isolation valves.

The containment cooling coils are completely closed inside containment and no leakage is expected from these units. During normal operation the service water system supply and return pressure for the ventilation coolers can be above or below the containment design pressure of 60 psig. Following a loss-of-coolant accident, the service water system supply and return pressure for the ventilation coolers is normally below the containment design pressure of 60 psig. The service water system is considered a closed system inside containment.

The essential loads of the SW system are designed to minimize any sedimentary blockage of the service water side. The automatic initiating valves for essential loads are generally located on the service water discharge side of heat exchangers. Table 9.6-3 shows a listing of the valves that are automatically opened when required.

The diesel generators (G-01 and G-02) employ jacket cooling and shell and tube heat exchangers. The Service Water system provides the source of cooling for the engine heat exchangers (G-01 and G-02 only). In the event of a loss of power to the safeguards buses, service water is not immediately available for cooling G-01 and G-02 until the buses are restored. Adequate heat absorption capacity is provided to operate the diesel generators (G-01 and G-02) until the service water system starts.

The service water system is the safety related water supply for the auxiliary feedwater pumps (1/2 P-29 and 1/2 P-53). Normally closed motor-operated valves (1/2 AF-4006 and 1/2 AF-4067) are provided to allow the suction supply for the AF pumps to be transferred to the SW system. The AF pump suctions are automatically transferred to the service water system as described in Section 10.2 (Reference 1).

The service water system is capable of supplying water to the suction of the non-essential Standby Steam Generator (SSG) pumps (P-38 A&B) via normally closed, manually actuated motor operated valves AF-4009 and AF-4016.

The spent fuel (SF) pool cooling system is not considered an essential load and cooling for this system may be interrupted. Service water will be interrupted during an accident after a SI signal from either unit. It will be necessary to manually restore spent fuel pool cooling following service water isolation.

The service water headers in the auxiliary building primarily supply cooling water to the; four component cooling heat exchangers, containment fan coolers, and the spent fuel pool cooling system. The component cooling heat exchangers are utilized to remove heat from the primary coolant system through the residual heat (RH) removal loop. The residual heat (RH) removal loop is employed during normal shutdown operations, and would also be placed in service following a loss-of-coolant accident for cooling of the recirculation flow from the reactor containment sump.

The service water system is treated to control biological fouling in the system piping and heat exchangers. Sodium hypochlorite, Sodium Bromide, Nalco 73551 (bio-detergent), and Nalco 3DT121 (silt dispersant) have been approved as system additives to prevent the buildup of slime and algae in the system and to minimize zebra mussel colonization. In addition, a chemical called EVAC has been approved for use in periodic treatments to kill any adult zebra mussels which have settled in the system. Other treatments may be considered in the future and will be evaluated prior to implementation. All treatments must be performed within the requirements of our Department of Natural Resources (DNR) Discharge Permit under the Wisconsin Pollutant Discharge Elimination System (WPDES).

9.6.3 SYSTEM EVALUATION

The service water system is designed to prevent a component failure from curtailing normal station operation. The service water loop can be aligned to provide two independent systems. In the event of a major malfunction, it is possible to isolate the portion of the system affected and maintain essential services to the plant. In addition to the header isolation valves, each component also has individual isolation valves to permit removing any piece of equipment from the system.

Service water pumps are normally controlled from main control room panel C-01, (see Section 7.5.4 for description of local controls). The SW pump capacity is sufficient to simultaneously meet the flow requirements of a design basis accident, together with failed closure of one train of the motor operated valves for isolation of nonessential services.

Service water piping beyond the Class-1 structures only supplies non-essential equipment. That piping can be isolated by the safeguards sequence automatically, by remote manual actuation of powered isolation valves, and by local manual valves. Both the powered and manual isolation valves are located within the Class-1 structure.

The service water piping in the Control Room Heating, Ventilation and Air Conditioning Room is Category I (seismic).

Evaluation of the internals of the North Service Water Header Zurn Strainer, SW-2911-BS, and the South Service Water Header Zurn Strainer, SW-2912-BS, determined that the internal components will not fail during or after a seismic event and cause blockage of flow. Operation of the backwash function is not part of the evaluation (Reference 6 and Reference 7).

Almost all of the motor-operated valves in the service water system are supplied with electrical power from the safeguards buses. There are two exceptions; MOV-2818 (service water isolation to cable spreading room air conditioning) is supplied from MCC B-21 which is supplied by 480V AC safeguards bus 2B-04. MCC B-21 is stripped from 2B-04 on a Unit 2 Safety Injection signal or on loss of AC power. MOV-2819 (service water isolation to the control room air conditioning) is powered from MCC B-22 which is supplied by 480V AC non-safeguards bus 2B-02.

The service water system is in operation at all times during plant operation and shutdown, and therefore is in a high state of readiness for any abnormal or emergency plant conditions.

9.6.4 REQUIRED PROCEDURES AND TESTS

The SW system components are tested and inspected in accordance with Technical Specification surveillance criteria and surveillance frequencies by the Surveillance Frequency Control Program (Reference 10). Testing verifies motor-driven pump operability, and operability of all required valves.

The passive portions of the system are monitored in accordance with the Open-Cycle Cooling (Service) Water System Surveillance Program (Section 15.2.14) during the period of extended operation. (NRC SE dated 12/2005, NUREG-1839)

The originally installed service water pumps underwent a hydrostatic test in the vendor shop at a test pressure of one and one-half times the shutoff head of the pump. In addition, the normal capacity vs. head characteristics were determined for each pump. During plant construction, the service water piping was hydrostatically tested in the field at one and one-half times design pressure. The welds in the shop fabricated service water piping were randomly radiographed in accordance with ASME Boiler and Pressure Vessel Code, Section VIII. Repair, replacement, and modification work on the service water system components is completed in accordance with the requirements of 10 CFR 50 Appendix B and ASME Section XI.

9.6.5 REFERENCES

1. NRC Safety Evaluation, "Point Beach Nuclear Plant (PBNP), Units 1 and 2 - Issuance of License Amendments Re: Auxiliary Feedwater System Modification (TAC Nos. ME1081 and ME1082)," dated March 25, 2011.
2. Not Used
3. 10 CFR 50.59/72.48 Screening (SCR) 2013-0024, "Revise TRM 3.7.7, OI 70, TS 33, TS 34, AOP 13A, AOP 8F, FSAR 9.6.1, OI 155, PC 97 Parts 1-8, and 1(2)-SOP-VNCC-001-4 to Allow 85F SW Inlet Temperature and to specify operability limits on low pump bay level for the G01/G02 EDGs and the lower elevation CFCs," dated March 15, 2013.
4. NRC Safety Evaluation, "Point Beach Nuclear Plant, Units 1 and 2 - Issuance of Amendments Re: Service Water System Operability (TAC Nos. MB4630 and MB4631), dated August 29, 2002.
5. NFPA 805 Fire Protection Program Design Document (FPPDD).
6. Screening Evaluation Work Sheet SQ-002126, "North Service Water Header Zurn Strainer, SW-2911-BS," Revision 1, 03/07/03.
7. Screening Evaluation Work Sheet SQ-002127, "South Service Water Header Zurn Strainer, SW-2912-BS," Revision 1, 03/07/03.
8. Bechtel Topical Report B-TOP-3, "Design Criteria for Nuclear Power Plants Against Tornadoes," (Proprietary) dated March 12, 1970.
9. Amirikian, Araham, "Design of Protective Structures, A New Concept of Structural Behavior," Bureau of Yards and Docks, Department of the Navy, P 51, August 1950.
10. NRC Safety Evaluation, "Point Beach Nuclear Plant Units 1 and 2 - Issuance of Amendments Regarding Relocation of Surveillance Frequencies to Licensee Control (TAC NOS. MF4379 and MF4380)," dated July 28, 2015.

Table 9.6-1 ESSENTIAL SERVICE WATER LOADS

PAB Battery Room Coolers
Emergency Diesel Generator Engine Coolant Heat Exchanger (G-01 & G-02)
Component Cooling Water (CC) Heat Exchangers
Containment Ventilation Coolers (HX-15)
Turbine Driven Auxiliary Feedwater Pumps (Pump Suction Supply)
Motor Driven Auxiliary Feedwater Pumps (Pump Suction Supply)
Containment Ventilation Fan Motor Coolers

Table 9.6-2 NON-ESSENTIAL LOAD ISOLATION VALVES

VALVE	DESCRIPTION
SW-2816 AND SW-4479	Service Building Isolation
SW-2817 AND SW-4478	Water Treatment Area Isolation
SW-2927A and SW-2930A	Spent Fuel Pool Isolation
SW-2927B and SW-2930B	Spent Fuel Pool Isolation
SW-LW-61/62	Radwaste System Isolation

Table 9.6-3 ESSENTIAL SW AUTOMATIC VALVES

VALVE DESIGNATION	DESCRIPTION
1/2SW-2907 & 2908	Containment Fan Coolers (OUTLET MOVs)
1/2 AF-4067	1/2 P-53 Motor Driven AFW Pump Suction
1/2 AF-4006	1/2 P-29 Turbine Driven AFW Pump Suction

Figure 9.6-1 UNIT 1 SERVICE WATER SYSTEM

[redacted]

Figure 9.6-2 UNIT 1 SERVICE WATER SYSTEM

[redacted]

Figure 9.6-3 UNIT 1 SERVICE WATER SYSTEM

[redacted]

Figure 9.6-4 UNIT 1 SERVICE WATER SYSTEM

[redacted]

Figure 9.6-5 UNIT 1 SERVICE WATER SYSTEM

[redacted]

Figure 9.6-6 UNIT 2 SERVICE WATER SYSTEM

[redacted]

Figure 9.6-7 UNIT 2 SERVICE WATER SYSTEM

[redacted]

9.7 INSTRUMENT AIR (IA) / SERVICE AIR (SA)

There are two compressed air systems that supply air plant-wide; the instrument air (IA) system and the service air (SA) system. Nitrogen gas bottles or IA accumulator tanks are used as backup pneumatic sources for some plant equipment.

9.7.1 DESIGN BASIS

The IA and SA systems perform the following functions:

Safety Related Functions:

1. Portions of the IA and SA systems form part of the containment pressure boundary.
2. IA system backup accumulators provide the pneumatic motive force for AFW pump minimum recirculation and flow control air operated valves (AOV).
3. The IA system provides the pneumatic motive force for the PORVs when aligned for LTOP protection. (See AQ classification discussion in Section 9.7.3).
4. IA system backup accumulators provide the pneumatic motive force for closure of the main feedwater isolation valves (MFIV).

Augmented Quality Functions:

1. An IA system accumulator provides the pneumatic motive force for the cross-over steam dump valves to provide turbine generator over speed protection.
2. The IA system is credited in the event of a fire and has been evaluated in the at-power and non-power analyses (Reference 1).
3. The IA system provides valve position indication for the associated containment isolation valves (PAM, type B variable).

Non-Safety Related, Non-QA Functions:

1. The IA system supplies dry, oil-free air to pneumatic controllers and control valves required for the normal operation of both units.
2. The SA system provides non-dried, oil-free air to the plant services header for equipment not requiring the dry air provided by the IA system.
3. The SA system can be cross-tied to provide a backup supply to the IA system automatically or manually.
4. The IA system provides pressure indication which can be used to indicate the operation of safety systems and other systems important to safety (PAM, type D variable).
5. The IA system provides the pneumatic motive force to allow remote operation of the steam generator (SG) atmospheric dump valves (ADV) from the control room.

9.7.2 SYSTEM DESIGN AND OPERATION

The instrument air system consists of two water-cooled air compressors (K-2A and K-2B) which take filtered suction from room air and discharge through their associated aftercooler. The air passing through the aftercoolers is cooled, the moisture is removed, and the air stream is then routed to the associated air receiver (T-33B and T-33C). The air compressors and aftercoolers are cooled with water from the service water system. A single IA compressor is capable of supplying the necessary air supply to serve both units. The air receivers act as a reservoir to store the pressurized air for use in the system. The air receivers each have a separate, normally cross-connected, discharge line with in-line filters which feed its respective air dryer unit (Z-31 and Z-39). An electrically operated dryer bypass valve for each dryer unit is energized closed during normal operation. The dryer bypass valve will open on loss of power to the in-service dryer or on low instrument air header pressure.

Both air dryer units are normally in service. Each unit consists of two parallel dryer towers with one lined up to accept air from the receivers and the other off-line for regeneration. The dryer towers are filled with desiccant which absorbs moisture from the air stream. Regeneration of the desiccant is accomplished by heating with electric heaters while maintaining a continuous air purge. Switching dryer towers is accomplished automatically. Abnormal operation of the dryer is annunciated in the control room. The dryer unit is designed to produce instrument air with a dewpoint of -40 degrees F from saturated air at 100 degrees F. Air leaving the drying towers passes through afterfilters which collect any desiccant dust prior to supplying the north and south instrument air headers. The north header supplies air to the auxiliary building, Unit 2 containment, Unit 2 turbine hall, water treatment area, and the north service building. The south header supplies the auxiliary building, Unit 1 containment, Unit 1 turbine hall, and the circulating water pump house.

Instrument air is supplied to instrumentation, valves, dampers, and pneumatic controllers throughout Unit 1 and Unit 2. Branch lines are taken off the main headers which are equipped with manual isolation valves permitting isolation of the branch line in the event of a rupture downstream of the isolation valve. Instrument air to each containment is routed through two parallel paths, each containing a manual isolation valve, an air-operated isolation valve, and a check valve outside containment. The parallel flow paths are cross-connected in containment through a manual isolation valve to supply a common header.

Each instrument air compressor has a three-position control switch (OFF-AUTO-CONSTANT) located in the control room. Normally one instrument air compressor controlling in CONSTANT, which allows the compressor to run constantly and load and unload as system demand requires, is sufficient to supply the system. The second air compressor is normally in AUTO control and will start when pressure drops to a preset pressure. After the standby compressor starts, it will run constantly loading and unloading as necessary until it is manually secured. K-2A and K-2B IA compressors are powered from safety related motor control centers 1B-32 and 2B-42 respectively. Upon loss of power, the compressors will stay deenergized until they are manually restored via the control switch. Additional backup is supplied by the service air system to the inlet of the dryers by two automatic 2-inch pressure controlled cross-connect valves (IA-3014 and IA-3079) or two 3-inch manual valves.

In order to maintain operability on loss of instrument air, some components use nitrogen bottles, regulators, and check valves or air accumulators and check valves to maintain pressure at the component for varying periods of time. Nitrogen fixed gas bottles are provided for instrument air backup; (1) to the pressurizer power operated relief valves for low temperature overpressure protection (LTOP), (2) to the pressurizer spray valves, and (3) to the standby steam generator pump (P-38 A&B) discharge and minimum recirculation valves. For fire scenarios, additional nitrogen supply to the standby steam generator feed pump valves is available using the plant nitrogen storage tank. Instrument air accumulators are provided for the main steam isolation valves, and the crossover steam dump valves.

Each unit's motor driven AFW pump (1/2P-53) has two air accumulators to provide a safety related backup air supply for its minimum recirculation and discharge flow control AOVs. Each unit's turbine driven AFW pump (1/2P-29) has an air accumulator to provide a safety related backup air supply for its recirculation AOV. These accumulators are normally supplied with air from the IA system through an air-amplifier device which steps up the air pressure in the accumulator to a level significantly higher than that provided by the IA system compressors. The SA system provides the motive force for the air-amplifier devices. Regulators are used to provide the proper air pressure to the AOVs. Isolation check valves are provided to isolate the accumulator tanks from the rest of the instrument air system.

Each main feedwater isolation valve (MFIV) has an accumulator to provide a safety related pneumatic supply for the closure function of the valve. These accumulators can be supplied with air from the IA system through an air-amplifier device which steps up the air pressure in the accumulator to a level higher than that provided by the IA system compressors. The IA system provides the motive force for the air-amplifier devices. Regulators are used to provide the proper air pressure to the MFIV. The IA system also provides the motive force for the non-safety related function of opening each of the MFIVs. A high pressure nitrogen supply is also provided to the accumulator and to the portion of the pneumatic system used to open the valve. Isolation check valves are provided to isolate the accumulators from the rest of the instrument air system and from the nitrogen supply (Reference 6, Reference 7).

The service air system consists of two air compressors (K-3A and K-3B), which take filtered suction from room air and discharge through their associated aftercooler. Air leaving the aftercoolers is routed to receivers (T-33A and T-33D) and then to the main service air header which runs throughout the plant and branch lines which supply individual components. The service air containment isolation valves (1/2 SA-17 and 1/2 SA-27) are manual valves which are locked shut during power operation.

Service air compressor K-3A is powered from 480 VAC bus 1-B04 and both the compressor and aftercooler are air cooled. Service air compressor K-3B, intercooler, and aftercooler are cooled with water from the service water (SW) system. Normally one service air compressor is running and the other aligned to start automatically when service air or instrument air pressure drops to a preset level. When running, the compressor will load and unload as system demand requires. The service air system functions include; backup supply to instrument air, supply to service air loads, backup supply to control room emergency breathing air system, and means for supplying compressed air to support containment integrated leak rate testing.

9.7.3 SYSTEM EVALUATION

Plant cool down via operation of the atmospheric steam dump valve (ADV) on the intact steam generator (SG) is credited in the SGTR overfill analysis. The analysis assumes a concurrent loss-of-offsite-power (LOOP) and that the cool down is initiated within 17 minutes following isolation of the ruptured SG. In order to meet the assumed initiation time, the ADV must be capable of remote operation from the control room which requires the availability of IA. On a LOOP, the IA compressors initially load shed from the safety related MCCs and can be restarted from the control room when emergency diesel generator (EDG) loading allows. The use of the non-safety related IA system for operation of the ADV is justified based on the defense-in-depth provided by the following: (Reference 3 and Reference 4).

1. With a LOOP on the affected unit only, the instrument air compressor powered from the other unit would be available.

2. With a LOOP on both units, there is available volume in the IA receiver(s). In the meantime, the IA compressors are loaded on the EDGs by steps in the applicable abnormal operating procedure and alarm response procedure.

3. Local manual operation of the ADV is available if required.

The AFW pump backup air accumulators provide enough air to allow operation of the associated discharge flow control and recirculation AOVs for 24 hours without relying on operator action to manually gag the recirculation valves in the correct position as required by specific operating license conditions. Four hours of backup air is required for pump operability. If instrument air is lost and the safety related backup supply is depleted, the operators will be required to manually throttle the flow control and recirculation valves consistent with decay heat removal requirements (Reference 2, Reference 5).

The MFIV accumulators provide the pneumatic supply for MFIV closure. The associated air amplifiers compensate for system leakage and maintain the operating pressure in the accumulators. Low accumulator pressure is alarmed in the control room. Either the high pressure nitrogen system or IA amplifier are designed to maintain accumulator pressure. A lower pressure backup nitrogen system provides pressure to the MFIV actuators to ensure the MFIVs do not close in the event IA is lost when a MFIV closure signal is not present.

The design basis function of containment isolation for instrument air is provided by two air operated isolation valves (IA 3047 and 3048) in parallel for each unit. The valves are seismic Class I and will automatically shut on a containment isolation signal. The valves are normally held open by instrument air pressure and will fail shut on loss of instrument air. These containment penetrations are also provided with check valves which provide backup for the automatic isolation function.

When aligned for LTOP protection, nitrogen fixed gas bottles provide a backup pneumatic supply for the PORVs. Generic Letter 90-06 characterizes the LTOP function as safety related, but allows non-safety related quality components to be used to perform the function. Those IA system components used to provide support for the LTOP function are classified as Augmented Quality (AQ).

The capability to isolate the instrument air supply to certain air-operated valves inside containment is credited in the event of a fire and has been evaluated in the at-power and non-power analyses (Reference 1).

Air cooled SA compressor K-3A can be powered by emergency diesel generator G-03 or G-04 which do not rely on service water for cooling. K-3A provides a source of backup air to the IA system which is independent of service water and is normally aligned for automatic operation (Reference 3, Reference 8).

9.7.4 REQUIRED PROCEDURES AND TESTS

The inservice inspection requirements are described in the PBNP Inservice Testing Program. Plant procedures provide guidance on the returning of an IA compressor to service after a LOOP.

9.7.5 REFERENCES

1. NFPA 805 Fire Protection Program Design Document (FPPDD).

2. NRC Safety Evaluation, "Point Beach Nuclear Plant (PBNP), Units 1 and 2 - Issuance of License Amendments Re: Auxiliary Feedwater System Modification (TAC Nos. ME1081 and ME1082)," dated March 25, 2011.

3. 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.

4. 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.

5. NRC 2009-0116, "License Amendment Request 261 - Extended Power Uprate Response to Request for Additional Information," dated November 21, 2009.

6. EC 12052 (258480), EPU - Feedwater Isolation Valve Addition - Unit 2.

7. EC 12054 (258482), EPU - Feedwater Isolation Valve Addition - Unit 1.

8. 50.59 Screening SCR 2010-0159-01, EC 13506, "Self-Cooled Air Compressor," dated October 13, 2010.

9.8 CONTROL ROOM VENTILATION SYSTEM (VNCR)

The Control Room Ventilation System (VNCR) is designed to provide heating, ventilation, air conditioning, and radiological habitability for the control and computer rooms, which are both within the Control Room Envelope (CRE). The Control Room Emergency Filtration System (CREFS) is a subset of the VNCR system that is associated with the equipment necessary to ensure the habitability of the control room during challenges from radioactivity, hazardous chemicals, and fire byproducts, such as fire suppression agents and smoke, during both normal and accident conditions. CREFS consists of one emergency air filtration unit, two emergency fans, two recirculation fans, and required ducts, valves, instrumentation, doors, barriers, and dampers necessary to establish the required flowpaths and isolation boundaries that recirculate and filter the air within the CRE. CREFS also includes the doors, walls, floor, roof, penetrations, and barriers that form the CRE boundary that limits the inleakage of unfiltered air. CREFS is an emergency system, parts of which operate during normal operation. (Reference 3, Reference 4)

9.8.1 DESIGN BASIS

The following General Design Criteria are applicable to CREFS as described in Reference 4.

GDC 2 Performance Standards

GDC 11 Control Room

GDC 37 Engineered Safety Features

GDC 38 Reliability and Testability of Engineered Safety Features

GDC 70 Control of Releases of Radioactivity to the Environment

The VNCR system equipment was designed to be capable of maintaining a room temperature of 75°F, with outside air temperatures varying from -15°F to 95°F. Instrumentation and associated circuitry in the control room is generally rated for an ambient temperature range of 40°F to 120°F. The reactor protection instrumentation has a 110°F ambient temperature limitation as discussed in Section 7.2.3.5.

The control room HVAC system was not designed or built as a safeguards system. The basis for this decision was that equipment in the control room would operate for some time without cooling and there would be no danger to personnel in the room (Reference 8). However, current analysis has demonstrated the need for an active cooling source during prolonged design basis conditions to satisfy GDC 11 (Reference 2). Since original construction, various upgrades to the system have been made to improve reliability of the system and make it easier to restore control room HVAC following a loss of off-site power. These upgrades included providing emergency diesel generator backed power for the HVAC supply fans, filter fans and control panel C-67; new higher capacity chillers; and control room envelope boundary upgrades. The chillers, chilled water pumps and other equipment which provide heating, cooling and humidification for the control room are powered from the non-safety related electrical distribution system (Reference 9, Reference 10, Reference 11 and Reference 12).

The Service Water (SW) System provides an alternate compensatory measure using an active cooling source that maintains adequate temperature and is available during loss of offsite power assuming a single emergency train failure. This alignment satisfies GDC 11. (Reference 16)

VNCR is capable of operating in five different modes as described in Section 9.8.2. Mode 5 places the system in the configuration necessary for radiological habitability by providing for control room pressurization to limit inleakage, makeup and recirculation through HEPA and charcoal filters to remove contaminates. Calculations referenced in Section 14.3.5 demonstrate that the system is capable of meeting the dose limits of 10 CFR 50.67. The design factors affecting the systems ability to meet the above dose limits include automatic actuation on a containment isolation signal, high radiation signal from a control room area monitor, or high radiation signal from a noble gas monitor in the control room supply duct; emergency filtration flow rate 4950 cfm ± 10%, with a minimum filtered recirculation flow of 1955 cfm; maintaining a positive pressure within the CRE; meeting CRE unfiltered inleakage limits; and meeting minimum filtration efficiencies for the HEPA and charcoal filters. (Reference 3)

The limiting design basis accident for the control room dose is the large break LOCA. PBNP is analyzed for a loss of offsite power with a LOCA for control room dose calculations (Reference 3).

The VNCR System is credited in the event of a fire and has been evaluated in the at-power and non-power analyses (Reference 17).

9.8.2 SYSTEM DESIGN AND OPERATION

The control room ventilation system is located in the Mechanical room above the Control Room and is controlled from control room panel C-67. The system is designed for 5 modes of operation. Mode 1 is normal operation, Mode 2 is 100% recirculation, Mode 3 is 25% filtered return air / 75% recirculation, Mode 4 is 25% filtered outside air / 75% recirculation, and Mode 5 allows a combination of outside air and return air to pass through the emergency HEPA/charcoal filter unit. Flow paths for these 5 modes are depicted in Figure 9.8-1.

For Mode 1, one of the two normal supply/recirculation fans (W-13B1 or W-13B2) is started. A maximum of 2000 cfm of outside air is provided to the fan suction from an intake penthouse located on the roof of the auxiliary building. The make-up air and the return air from the control and computer rooms passes through roughing filter F-43 and cooling units HX-100 A&B before entering one of the normal recirculation fans. Room thermostats and/or humidistats control operation of the chilled water unit supplying the cooling units. After leaving the normal recirculation fan, filtered and cooled air is supplied to the mechanical room and through separate heating coils, HX-92 and HX-91 A&B, and humidifiers, Z-78 and Z-77, to the computer and control rooms respectively. Room thermostats and humidistats also control the operation of the heating coils and humidifiers. Also operating in Mode 1 are computer room supplementary air conditioning unit W-107A/HX-190A/HX-191A or W-107B/HX-190B/HX-191B and control room washroom exhaust fan W-15.

Mode 2 operation is 100% recirculation of the air and is aligned manually from panel C-67. When this mode is aligned, the outside air damper closes, the washroom exhaust fan is de-energized, the washroom exhaust fan dampers close, and the damper supplying the reactor engineering room opens.

Mode 3 operation employs one of two control room emergency filter fans (W-14A or W-14B) and filtration unit, F-16, which includes a roughing filter, a HEPA filter, and a charcoal filter. This mode is aligned from panel C-67. A portion (approximately 25%) of the recirculated air is directed through filter bank F-16 and the operating filter fan back to the suction of the normal recirculation fan. Operation in this mode also de-energizes the washroom exhaust fan W-15, closes the washroom exhaust fan dampers, and opens the damper supplying the reactor engineering room. This mode of operation can be used for smoke removal in the event of a fire. (Reference 4).

Mode 4 is similar to Mode 3 except return air inlet damper VNCR-4851B to the emergency filter fans is closed and outside air supply damper VNCR-4851A is open. This allows approximately 4950 cfm of make-up air to pass through filter F-16 and the emergency fan to the suction of the normal recirculation fan, ensuring a positive pressure is maintained in the control and computer rooms to limit in-leakage. This mode is aligned manually from panel C-67.

Mode 5 (emergency HEPA/charcoal filtered outside air and HEPA/charcoal filtered return air) operation is similar to Mode 4 except that the return air inlet damper VNCR-4851B to the makeup fans opens. This allows a combination of outside air and return air to pass through the emergency HEPA/charcoal filter unit to the suction of the control room recirculation fan (total filtered flow ≥4455 cfm). The makeup flow rate is sufficient to assure a positive pressure that will prevent excessive unfiltered in-leakage into the control room ventilation boundary. Mode 5 is automatically initiated by a containment isolation signal, by a high radiation signal from the control room monitor RE-101, or by a high radiation signal from the noble gas monitor RE-235 located in the supply duct to the control room. The transfer to Mode 5 operation is completed within 60 sec. from receiving the actuation signal. A filter fan and a recirculation fan will automatically load onto an EDG if offsite power is lost. These fan loads have been included in the emergency diesel generator loading tabulation during a loss of coolant accident (FSAR Table 8.8-1). This mode of operation can also be manually initiated from panel C-67. Redundancy is provided for active components that must reposition from their normal operating position. (Reference 3)

The control room ventilation system contains a backup filtration system that can be manually aligned and operated if necessary when CREFS is not functional. The backup system includes the F-280 filtration unit with pre-filter, HEPA filter, and charcoal filter; W-275 filter fan; and associated ductwork with bubble tight dampers to provide the CRE boundary when the system is not in service. A manual transfer switch allows the use of the starter and controls for the CREFS W-14A fan to be used to power and control the W-275 backup system fan. The W-275 fan motor has electrical characteristics equivalent to the W-14 fan motors such that diesel loading will not be adversely affected when the backup filtration unit is placed into service. The backup filtration system is classified as Augmented Quality (AQ) and is designed for external pressurization and jet impingement loading due to a high energy line break (HELB) in the turbine building to ensure the CRE boundary is not affected. The backup filtration system can provide a minimum of 2,000 cfm of filtered outside air directly into the control room. This flow rate is sufficient to assure a positive pressure within the CRE that will prevent excessive unfiltered in-leakage. (Reference 1, Reference 3, Reference 14)

See Appendix A.5, Section A.5.2 and Section A.5.6.3 for seismic adequacy of the CREFS system.

Other features of the control room ventilation system include the capability to exhaust smoke from the control room, computer room, or cable spreading room through dedicated smoke and heat vent fan, W-13C. The associated dampers for this evolution are interlocked so that only one room can be lined up for smoke and heat removal at a time. This operation precludes smoke damage to the air filters in the recirculation system. The controls for smoke and heat removal are from panel C-67A located on the exterior north wall of the control room. The computer room has supplementary air conditioning units, W-107A/HX-190A/HX-191A and W-107B/HX-190B/HX-191B to assist the normal control room ventilation system in maintaining computer room temperatures below equipment design limits. Filter F-16 has an automatically initiated water suppression system to mitigate a fire in the charcoal bed.

Moisture elements and flow switches at the outlets of humidifiers Z-77 and Z-78 send a signal to stop humidification if duct humidity gets too high or air flow gets too low in order to prevent condensation in the duct work. A flow switch downstream of each emergency fan W-14A, B and each supplementary air conditioning fan W-107A, B and flow switches downstream of normal/ recirculation fans W-13B1, B2 automatically start(s) the standby fan on a low flow condition.

9.8.3 SYSTEM EVALUATION

Note: See Appendix A.1 for a description of the effects of a station blackout (SBO) on control room and computer room ventilation.

The original specification for the control room ventilation system was to maintain a room temperature of 75°F with outside air temperatures varying from -15°F to 95°F with a single train in continuous operation. Continuous room temperatures are normally maintained ≤75°F to provide assurance that personnel and equipment temperature limits can be maintained during a temporary (2 hour or less) loss of the control room ventilation system.

<u>Control Room Equipment Ambient Temperature Design Limits</u>

Instrumentation and associated circuitry in the control room is generally rated for an ambient temperature range of 40°F to 120°F. Following a loss of the control room ventilation system, room heat loads would most likely prevent the room temperature from ever reaching 40°F, however, 120°F could be reached during a prolonged unavailability of the system.

<u>Computer Room Equipment Ambient Temperature Design Limits</u>

The computer room multiplexers (MUX) are the most temperature sensitive components, with an inlet air ambient temperature limit of 95°F. In the event that elevated temperatures in the computer room lead to eventual MUX failures, contingency actions provide for monitoring the minimum required post-accident in-core thermocouple temperatures on dedicated recorder displays located on the ASIP panels. Manual monitoring of in-core thermocouple temperatures at the MUX input terminals is possible using portable M&TE. The SPEC 200 racks have an internal temperature limit of 140°F.

<u>Radiological Dose Limits</u>

An evaluation of CREFS ability to maintain the LOCA radiological dose of control room occupants to within the 5 rem total effective dose equivalent (TEDE) allowable limit of 10 CFR 50.67 is provided in Section 14.3.5.

With CREFs inoperable, the backup filtration system, in combination with the ingestion of potassium iodide (KI), is also capable of maintaining the radiological dose of control room occupants to within the 5 rem TEDE limit if placed into service within 1 hour after the start of an accident. (Reference 1, Reference 3, Reference 7)

Chemical Hazards and Smoke

Low CRE unfiltered air inleakage limits the infusion of toxic chemicals and smoke by-products into the CRE, thereby promoting habitability. Additionally, SCBAs and portable smoke ejection equipment is available and can be used if the CRE boundary is not functional. The VNCR system provides for smoke removal in the event of a fire in the control room.

9.8.4 REQUIRED PROCEDURES AND TESTS

Required procedures and tests are identified in Technical Specification (TS) 3.3.5, “Control Room Emergency Filtration System (CREFS) Actuation Instrumentation,” TS 3.7.9, “Control Room Emergency Filtration System (CREFS),” TS 5.5.10, “Ventilation Filter Testing Program (VFTP),” and Technical Requirements Manual (TRM) 4.10, “Ventilation Filter Testing Program (VFTP),” TS 5.5.18, “Control Room Envelope Habitability Program,” and TRM 4.18, “Control Room Envelope Habitability Program.” TRM 3.7.9, “Control Room Emergency Filtration System (CREFS)” describes the required mitigating actions when CREFS is not functional.

Procedures are in place to prevent either the control room or the computer room from exceeding their temperature limitations under accident conditions, coincident with a LOOP and single train failure (Reference 16). Abnormal operating procedures direct operators to take compensatory measures if control room cooling is lost and cannot be restored. Potential compensatory measures include restoring power to the control room chiller HX-038B, restoring power to the cable spreading room chiller HX-038A and opening cross-tie valves between the cable spreading room and the control room chilled water systems, and aligning service water directly to the control room cooling coils (HX-100A/B) (Reference 15).

Augmented testing of the control room chilled water pumps is performed quarterly under the Inservice Testing Program to ensure pump flow requirements are met (Reference 13).

9.8.5 REFERENCES

1. SCR 2009-0148-08, EC 11690 - Alternate Source Term Implementation and CREFS Upgrade to Support Alternate Source Term License Amendment Request, December 22, 2011.

2. Calculation Number 2005-0054, Control Building GOTHIC Temperature Calculation.

3. 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.

4. 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.

5. Not Used

6. Not Used

7. Calculation CN-CRA-10-43, Point Beach Control Room Dose Sensitivity to Changes in the Modeling of the Control Room Emergency Filtration System Out of Service, Revision 1, dated April 28, 2011.

8. Westinghouse Letter to Wisconsin Electric, Control Room H&V and AC Power Supply, May 18, 1970.

9. SER 96-004, Control Room HVAC Upgrade Modifications, January 22, 1996.

10. SE 2000-0121, Replacement of Control and Computer Room Ventilation System Chiller Unit HX-38B, December 1, 2000.

11. SE 2000-0106, Replacement of Cable Spreading Room Ventilation System Chiller Unit HX-38A, October 18, 2000.

12. SE 2001-0049, Upgrade Control Room Envelope Boundary, August 17, 2001.

13. SCR 2010-0018, Revision to Section 9.8.4 of PBNP FSAR, January 20, 2010.

14. SCR 2010-0234-04, EC 15414 - CREFS Backup Filtration System, December 22, 2011.

15. 50.59 Evaluation 2010-003, EC 15413 "Control Room Alternate Cooling," November 1, 2010.

16. SCR 2012-0026, "FSAR Section 9.8, Control Room Ventilation System (VNCR) Change Concerning Temperature," dated March 12, 2012.

17. NFPA 805 Fire Protection Program Design Document (FPPDD).

Figure 9.8-1 CONTROL ROOM VENTILATION OPERATING MODES (Sheet 1)

[redacted]

Figure 9.8-1 CONTROL ROOM VENTILATION OPERATING MODES (Sheet 2)

[redacted]

Figure 9.8-1 CONTROL ROOM VENTILATION OPERATING MODES (Sheet 3)

[redacted]

Figure 9.8-1 CONTROL ROOM VENTILATION OPERATING MODES (Sheet 4)

[redacted]

Figure 9.8-1 CONTROL ROOM VENTILATION OPERATING MODES (Sheet 5)

[redacted]

9.9 SPENT FUEL COOLING & FILTRATION (SF)

The spent fuel pool cooling system, common to Units 1 and 2, is designed to remove decay heat from fuel assemblies stored in the spent fuel pool after removal from the reactor vessel. A discussion of the sharing of the components of this system between the two units is given in Appendix A.6.

The spent fuel pool cooling system consists of two separate cooling trains, with a common suction and return header, each having an identical heat exchanger and pump. Water from the pool is pumped through one or both heat exchangers for cooling and returned to the pool. When purification is required, a portion of the flow may be diverted through the interconnecting spent fuel pool purification system. Service Water (SW) provides the heat exchange medium for removal of decay heat (Reference 4).

9.9.1 DESIGN BASIS

Fuel and Waste Storage Decay Heat

The following PBNP General Design Criteria (GDC) are applicable to the Spent Fuel Cooling and Filtration System (Reference 8):

Criterion 4: Sharing of Systems
Criterion 67: Fuel and Waste Storage Decay Heat
Criterion 68: Fuel and Waste Storage Radiation Shielding
Criterion 69: Protection Against Radioactivity Release from Spent Fuel and Waste Storage

The refueling water provides reliable and adequate cooling medium for spent fuel transfer and heat removal from the spent fuel pool is provided by the Spent Fuel Pool Cooling System.

The Spent Fuel Pool (SFP) Cooling System is designed to remove the decay heat produced by irradiated fuel assemblies stored in the spent fuel pool. The heat removal capabilities for the cooling system are: (Reference 1, Reference 2, Reference 7, and Reference 8)

1. Capable of maintaining the temperature in the spent fuel pool less than or equal to 120°F during normal refueling operations with one cooling loop in operation ("normal refueling" is a fuel shuffle with only a partial core offloaded into the pool);

2. Capable of maintaining the temperature in the spent fuel pool less than or equal to 120°F following a full core off load with two cooling loops in operation; and

3. Capable of maintaining the temperature in the spent fuel pool less than or equal to 145°F following a full core off load with one cooling loop in operation.

Decay heat load is calculated before each refueling to ensure it is within the Spent Fuel Pool Cooling System capacity (Reference 7, Reference 8).

- Maximum SFP Decay Heat Loads are provided in Reference 9. Heat exchanger capacity vs SW temperature is provided in Reference 11.

The calculated values for the bulk water temperature of the SFP are not safety limits, and the nominal conditions assumed in the analysis are not operational limits.

In the event of complete failure of the cooling system for a long period of time, the fuel pool water inventory can be maintained with borated water from the refueling water storage tank or chemical and volume control system (CVCS), or non-borated water from the de-ionized water, reactor make up water, SW, or fire protection systems (Reference 8).

- Time to boil and make-up requirements are provided in Reference 9.

As discussed in Reference 3, the late 1970's design criteria for the SFP thermal and hydraulic analyses were derived from the NRC position paper "Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," which include (with updates):

a. Decay heat loads are calculated using the methods found in ANSI/ANS-5.1-1979 (Reference 7, Reference 8).

b. Boiling shall not occur in the water within the fuel assembly and the adjacent water/poison boxes (Reference 3).

c. Adequate time exists for an alternate cooling method to be implemented in the event of a complete loss of SFP Cooling System capability. The heat-up rate is to be calculated and the time required for pool boiling to occur will also be found (Reference 3).

SFP piping and the SW piping supplying the SFP heat exchangers are classified Safety Related and Seismic Class I (Reference 8).

All piping and components of the spent fuel cooling system are designed to the applicable codes and standards listed in Table 9.9-1. Austenitic stainless steel piping is used in the spent fuel pool cooling system, for the piping to and from the pool. Piping is arranged so that a failure of any one pipe will not drain the water in the spent fuel pool below the top of the fuel elements.

The Spent Fuel Cooling and Filtration System is credited in the event of a fire and has been evaluated in the at-power and non-power analyses (Reference 12).

9.9.2 SYSTEM DESIGN AND OPERATION

The spent fuel pool cooling pumps take suction through branch lines off a common header from beneath the surface of the north half of the spent fuel pool, pump the water through the tube side of the spent fuel pool cooling heat exchangers, and return it via a common header to the south half of the spent fuel pool. The system piping is arranged so that either pump can supply either heat exchanger. The clarity and purity of the spent fuel pool water are maintained by passing up to the design flow of 60 gallons per minute through a filter and demineralizer. The spent fuel pool cooling heat exchangers are cooled by service water on the shell side. The SFP Cooling system is shown as a portion of Figure 9.9-1, and the service water system associated with the spent fuel pool cooling system is shown in Figure 9.6-4. Service water can be supplied by either the North or West service water supply header, and is discharged to the Unit 1 or Unit 2 circulating water overboard through redundant return headers.

The spent fuel pool cooling system piping and the service water system piping supplying the spent fuel pool heat exchangers are classified Safety-Related, Seismic Class I. Although the branch lines serving the spent fuel pool heat exchangers were extensively modified using primarily ASME Section III Class 3 requirements, system code requirements are established by the original design basis and code of construction (USAS B31.1.0-1967).

The spent fuel cooling system is operated intermittently or continuously whenever there are spent fuel assemblies in the spent fuel pool. Cooling requirements are dependent upon the number of spent fuel assemblies stored in the spent fuel storage racks and the elapsed time that the spent fuel has been in storage. Clarification and purification requirements are a function of various conditions such as atmospheric contamination, fuel rod leakage, and work being performed in the pool or pool area.

The spent fuel pool cooling pumps and heat exchangers are normally operated as independent trains designated as Train "A" (P-12A and HX-13A) and Train "B" (P-12B and HX-13B). After locally starting the pump in the selected train, the heat exchanger inlet valve is positioned to control the pump discharge pressure. Purification of the spent fuel pool water is accomplished by establishing flow through the demineralizer and filter. Temperature is controlled by manually positioning the service water flow control valve(s) (SW-661 and 746) located in the service water return lines.

The fuel pool purification system interfaces with the spent fuel pool cooling system as shown on Figure 9.9-1. The purification system inlet taps off the cross-connect line between the "A" and "B" cooling trains at the discharge of the fuel pool cooling pumps. The purification system return line connects with the cooling system return header. The purification system is not safety related.

The refueling water circulation pump is used to circulate the water in the refueling water storage tank in a cleanup loop through the spent fuel pool demineralizer and filter and back to the refueling water storage tank (RWST). The refueling water circulating pump may also be aligned to take suction from the refueling canal through the drain connection and discharge through the spent fuel pool demineralizer and filter, or through the CVCS primary demineralizers. The return flow from these cleanup loops to the reactor is directed to the suction of the residual heat removal pumps. If the transfer canal is washed after refueling, the water can be flushed through the refueling water circulating pump to the drumming station. The refueling water circulating pump can also take suction from the boric acid blender of Unit 1 in order to increase the boron concentration in the spent fuel pool or take suction from the refueling cavity and discharge to the RWST. The refueling water circulating pump discharge piping is used for reflood of a dry cask storage container using the spent fuel pool as a reflood source.

Spent Fuel Pool Cooling System Components

Spent Fuel Pool Cooling Heat Exchanger

Each fuel pool cooling heat exchanger is a U-tube heat exchanger with service water on the shell side and fuel pool water on the tube side. All surfaces wetted by the borated fuel pool water are of stainless steel or stainless steel clad carbon steel. The tubes are rolled and seal welded to the tube sheet. The shell is made of carbon steel.

Spent Fuel Pool Cooling Pumps

The fuel pool pumps are centrifugal, horizontal pumps with stainless steel casings and a design flow rate of 1,250 gpm at rated head. Mechanical seals are used for shaft sealing. Each fuel pool pump is driven by an electric motor. Both pumps are flanged for convenient removal from the system for maintenance, and casings are provided with drain connections.

Piping and Valves

All fuel pool cooling system piping and fittings are stainless steel with standard wall thickness. Construction is welded throughout, except where flanged joints are provided for a flow measuring orifice, connections to the pumps and heat exchangers, and one set of flanges to accommodate initial pressure testing of the suction piping. All fuel pool cooling piping and fittings are 150 lb pipe class.

Fuel pool cooling system valves are stainless steel, 150 lb rating with trim suitable for borated water.

Spent Fuel Pool Filter

The spent fuel filter removes particulate material from the spent fuel pool water. The filter cartridge is synthetic fiber and the vessel shell is stainless steel.

Spent Fuel Pool Demineralizer

The demineralizer is sized to pass approximately 60 gallons per minute to provide adequate purification of the fuel pool water for unrestricted access to the working area, and to maintain optical clarity.

Refueling Water Circulating Pump

The refueling water circulating pump is used primarily to circulate water in a loop between the refueling water storage tank and the spent fuel pool demineralizer and filter. All wetted surfaces of the pump are austenitic stainless steel. The pump is operated manually from a local station.

Spent Fuel Pool Skimmer

A skimmer pump and strainer are provided for surface skimming of the spent fuel pool water. Flow from this pump is returned to the spent fuel pool.

9.9.3 SYSTEM EVALUATION

The spent fuel pool cooling system is provided with two pumps and two heat exchangers. The electrical components of the two trains are supplied with power from separate vital buses. The pumps and heat exchangers are provided with cross connecting piping so that either pump may be used with either heat exchanger to maximize system availability and reliability. The spent fuel pool cooling system operates intermittently or continuously whenever there are spent fuel assemblies in the fuel pool dependent upon heat removal requirements.

Manual flow control valves located in the heat exchanger service water return header may be throttled to control spent fuel pool water heat exchanger outlet temperatures. The following parameters are monitored or alarmed to determine the need for cooling system operation:

1. Fuel Pool Temperature (High temperature alarmed in control room)
2. Fuel Pool Level (High/Low level alarmed in control room)
3. Spent fuel pool cooling system flow (locally)

4. Spent fuel pool temperature (locally)

5. Heat exchanger outlet temperature (locally)

6. Heat exchanger service water inlet and outlet temperature (locally)

The normal operating pressure of the service water system is higher than the normal operating pressure of the spent fuel pool cooling system. In the event of a heat exchanger tube break, differential pressure will normally result in leakage from the service water system to the spent fuel pool cooling system. Under certain conditions, for example during refueling when higher service water flowrates to the spent fuel pool heat exchangers are required, service water pressure may fall below spent fuel pool cooling system pressure. Under these conditions, a heat exchanger tube break will result in leakage from the spent fuel pool cooling system into the service water system. A spent fuel pool heat exchanger tube rupture is considered improbable based upon the low operating pressures, the seismic installation of the heat exchanger, and the heat exchanger design specifications. If a tube break were to occur, indication of the break would be provided by process radiation monitoring equipment in the downstream service water piping, which monitors the service water system for released radioactivity.

The probability of inadvertently draining the water from the cooling loop of the spent fuel pool is exceedingly low. In the unlikely event of the cooling loop of the spent fuel pool being drained, the spent fuel storage pool itself cannot be drained and no spent fuel is uncovered since the spent fuel pool cooling suction and return connections terminate or contain a siphon breaker that would limit water drawdown to a level approximately 21 feet 11 inches above the active fuel. (Reference 6)

Whenever a leaking fuel assembly is transferred from the fuel transfer canal to the spent fuel storage pool, a small quantity of fission products may enter the spent fuel pool water. A small purification loop consisting of filtration and ion-exchange is provided for removing these fission products and other contaminants from the water.

A fuel pool high temperature alarm, located in the control room, will notify the operator of a fuel pool cooling malfunction. In the event of a failure of the operating pump, the standby pump may be started. In the event of loss of service water flow through the on-line heat exchanger due to a malfunction of a service water component in that train, the operating pump may be cross-connected with the standby heat exchanger. In the event of complete failure of the cooling system for a long period of time, the fuel pool water inventory can be maintained with borated water from the RWST or CV system, or non-borated water from the de-ionized water (DI), RMW, SW, or FP systems.

Assuming a loss of SFP cooling in the worst case conditions of a full core offload and an initial SFP temperature of 145ºF, the time-to-boil is approximately 7 hrs. This is sufficient time for plant personnel to take corrective actions to establish a means of spent fuel pool cooling. A makeup water supply of 50 gpm is adequate to maintain SFP level at the evaluated heat loads (Reference 8, Reference 9).

9.9.4 REQUIRED PROCEDURES AND TESTS

The active components of the spent fuel pool cooling system are in either continuous or intermittent use during normal plant operation. Periodic visual inspections and preventive maintenance can be conducted as necessary without interruption of cooling system operation. The inservice testing requirements are described in the PBNP Inservice Testing Program and the IST Background Document.

The decay heat load is calculated prior to each refueling based on decay time, power history, and SFP inventory from previous outages. The calculated heat load is compared to both the bounding decay heat load and the ability of the SFP cooling system based on the expected SW temperature at the time of fuel transfer to ensure the outage load is within the capability of the system (Reference 8).

Corrective actions to address a loss of spent fuel pool cooling are procedurally controlled and include: (Reference 5)

1. Restoring spent fuel pool cooling water flow (e.g., starting a standby pump)
2. Restoring service water flow (e.g., cross connecting trains)
3. Establishing alternative cooling by maximizing ambient losses (e.g., with SFP ventilation system)
4. Maintaining water level with the use of the borated or unborated water sources
5. Making necessary repairs to restore spent fuel pool cooling

9.9.5 REFERENCES

1. NRC Safety Evaluation Report dated April 4, 1979, “Safety Evaluation Relating to the Modification of the Spent Fuel Storage Pool.”
2. Wisconsin Electric Letter to NRC, “Spent Fuel Storage Expansion Modification to Change Request No. 54,” dated September 29, 1978. Attachment A is the “Spent Fuel Storage Modification Description,” Revision 2, dated September 29, 1978. Attachment B is Wachter Report, “Design and Analysis of High Density Spent Fuel Storage Racks for Point Beach Nuclear Plant,” Revision 2, dated September 29, 1978.
3. Wachter Report, “Thermal and Hydraulic Analysis Report, Spent Fuel Storage Pool,” WEP-T-12, dated May 23, 1978.
4. Wisconsin Electric Letter to NRC, “Docket Nos. 50-266 and 50-301 Amendment No. 24 to Final Facility Description and Safety Analysis Report Point Beach Nuclear Plant, Units 1 and 2”, dated June 30, 1978.
5. WE Letter to NRC, VPNPD-96-094, “Response to Resolution of Spent Fuel Storage Pool Safety Concerns”, dated November 13, 1996.
6. PBNP Calculation 2005-0037, “Spent Fuel Pool Anti-siphon Provisions,” dated December 2, 2005.

7. FPL Energy Point Beach, LLC letter NRC 2009-0030, "License Amendment Request 261, Extended Power Uprate," dated April 7, 2009.

8. NRC Safety Evaluation, Issuance of Amendments Regarding Extended Power Uprate, dated May 3, 2011.

9. Calculation 129187-M-0014, Evaluation of the Spent Fuel Pool Cooling System for EPU Operation, Rev 0, dated December 10, 2008.

10. Calculation N-93-048, "Time to SFP Boiling Following a Loss of SFP Cooling," Rev 4, dated January 24, 2007.

11. Calculation PGT-2003-1495, "Evaluation of the Point Beach Nuclear Plant Spent Fuel Pool Heat Exchangers' Thermal Performance as a Function of Service Water Temperature for Single Train Operation." Revision 3, dated October 31, 2011.

12. NFPA 805 Fire Protection Program Design Document (FPPDD).

Table 9.9-1 SPENT FUEL POOL COOLING SYSTEM COMPONENT DATA*
Sheet 1 of 4

System cooling capacity, 2 pumps/2 heat exchangers, BTU/hr	32.316×10^6
Spent fuel pool heat exchanger	
Quantity	2
Type	Shell and U-tube, horizontal
Nominal design heat transfer per heat exchanger, BTU/hr	16.158×10^6
Shell side (service water)**	
Design inlet temperature, ºF	65
Design flow rate, gpm	1250
Design pressure, psig	100
Operating pressure, psig	40
Design temperature, ºF	100
Material	Carbon steel
Tube side (Spent fuel pool water)**	
Design inlet temperature, ºF	120
Design flow rate, gpm	1250
Design pressure, psig	150
Operating pressure, psig	25
Design temperature, ºF	200
Material	Stainless steel
Spent Fuel Pool Pump Data	
Quantity	2
Type	Horizontal centrifugal
Design flow rate, gpm	1250
Discharge pressure, psig	26
Motor horsepower	25
Design pressure, psig	150
Design temperature, ºF	200
Material	Stainless steel

* Note: The SF System is shared by Unit 1 and Unit 2. Note that the information in this table represents the design characteristics of the listed components based on particular conditions assumed during the specification/ procurement phase, and in most cases are derived from the vendor's data sheets. These parameters should not be construed as operating or design limits.

** This information is derived from the Stone & Webster Tubular Heat Exchanger Data Sheet, dated March 15, 1977.

Table 9.9-1 (continued) SPENT FUEL POOL COOLING SYSTEM COMPONENT DATA
Sheet 2 of 4

Spent Fuel Storage Pool	
Pool volume, ft^3	48283 (Reference 10)
Boron concentration, ppm boron	2100 to 4000
Spent Fuel Pool Filter	
Quantity	1
Type	Replaceable cartridge
Internal design pressure of housing, psig	200
Design temperature, ºF	250
Design flow rate, gpm	60
Maximum differential pressure across filter element at rated flow (clean cartridge), psi	5
Spent Fuel Pool Demineralizer	
Quantity	1
Type	Flushable
Design pressure, psig	200
Design temperature, ºF	250
Design flow rate, gpm	60
Resin volume, $ft.^3$	20
Vessel volume, $ft.^3$	27
Spent Fuel Pool Skimmer	
Quantity	1
Design flow rate gpm	100
Vertical fluctuation range:	
Floating, inch	4
Manual adjustment, feet	2
Spent Fuel Pool Skimmer Strainer	
Quantity	1
Type	Basket
Design flow rate, gpm	100
Design pressure, psig	50
Design temperature, ºF	200
Maximum differential pressure across the strainer element at rated flow, clean, psi	1
Perforation, inch	1/8

Table 9.9-1 (continued) SPENT FUEL POOL COOLING SYSTEM COMPONENT DATA
Sheet 3 of 4

Spent Fuel Pool Skimmer Pump	
Quantity	1
Type	Horizontal centrifugal
Design flow rate, gpm	100
Total developed head, ft H_2O	50
Design pressure, psig	150
Design temperature, ºF	200
Material	Stainless steel
Refueling Water Circulating Pump	
Quantity	1
Type	Horizontal centrifugal
Design flow rate, gpm	100
Total developed head, ft H_2O	150
Design pressure, psig	150
Design temperature, ºF	200
Spent Fuel Pool Cooling Loop Piping and Valves	
Design pressure, psig	150
Design temperature, ºF	200
Spent Fuel Pool Skimmer Loop Piping and Valves	
Design pressure, psig	150
Design temperature, ºF	200
Refueling Water Purification Loop Piping and Valves	
Design pressure, psig	150
Design temperature, ºF	200

Table 9.9-1 (continued) SPENT FUEL POOL COOLING SYSTEM COMPONENT DATA
Sheet 4 of 4

Code Requirements

Spent fuel pool filter	ASME III**, Class C
Spent fuel pool heat exchanger	ASME III, Class 3, 1974 Edition, Summer 1975 Addenda
Spent fuel pool demineralizer	ASME III, Class C
Spent fuel pool loop piping and valves	USAS B31.1.0, 1967 Edition
Spent fuel pool pump motor	NEMA MG-1 IEEE 334-1974
Spent fuel pool pump	ASME III, Class 3, 1974 Edition, Summer 1975 Addenda
Service water lines serving spent fuel pool cooling system	USAS B31.1.0, 1967 Edition

** ASME III - American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III

Figure 9.9-1 UNIT 1 SPENT FUEL POOL COOLING SYSTEM

[redacted]

9.10 FIRE PROTECTION SYSTEM (FP)

The design philosophy and specifics of the fire protection program are contained in the FPPDD, "NFPA 805 Fire Protection Program Design Document" (Reference 6), and NP 1.9.14, "PBNP Fire Protection Plan" (Reference 7), as described below.

9.10.1 FIRE PREOTECTION

The fire protection program is based on the NRC requirements, Nuclear Electric Insurance Limited (NEIL) Property Loss Prevention Standards and related industry standards. With regard to NRC criteria, the fire protection program meets 10 CFR 50.48(c), which endorses, with exceptions, the National Fire Protection Association's (NFPA) 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants - 2001 Edition," (Reference 2). Point Beach Nuclear Plant Units 1 and 2 (PBNP) has further used the guidance of NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CPR 50.48(c)," (Reference 4) as endorsed by Regulatory Guide 1.205, "Risk-Informed, Performance Fire Protection for Existing Light-Water Nuclear Power Plants," (Reference 3) (Reference 5).

Adoption of NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition in accordance with 10 CFR 50.48(c) serves as the method of satisfying 10 CFR 50.48(a) and General Design Criterion (GDC) 3.

NFPA 805 does not supercede the requirements of GDC 3 (see Table 1.3-1), 10 CFR 50.48(a) or 10 CFR 50.48(f). Those regulatory requirements continue to apply. However, under NFPA 805, the means by which GDC 3 or 10 CFR 50.48(a) requirements are met may be different than under 10 CFR 50.48(b). NFPA 805 identifies fire protection systems and features required to meet Chapter 1 performance criteria through the methodology in Chapter 4 of NFPA 805. Also, under NFPA 805, the 10 CFR 50.48(a)(2)(iii) requirement to limit fire damage to SSCs important to safety so that the capability to safely shut down the plant is satisfied by meeting the performance criteria in Section 1.5.1 of NFPA 805.

A Safety Evaluation was issued on September 8, 2016, by the NRC, that transitioned the existing fire protection program to a risk-informed, performance-based program based on NFPA 805, in accordance with 10 CFR 50.48(c). (Reference 1)

9.10.1.1 DESIGN BASIS SUMMARY

9.10.1.1.1 Defense-In-Depth

The fire protection program is focused on protecting the safety of the public, the environment, and plant personnel from a plant fire, and its potential effect on safe reactor operations. The fire protection program is based on the concept of defense-in-depth. Defense-in-depth shall be achieved when an adequate balance of each of the following elements is provided:

1. Preventing fires from starting,
2. Rapidly detecting, controlling, and extinguishing those fires that do occur, thereby limiting fire damage,

3. Providing an adequate level of fire protection for SSCs important to safety, so that a fire that is not promptly extinguished will not prevent essential plant safety functions from being performed.

9.10.1.1.2 NFPA 805 Performance Criteria

The design basis for the fire protection program is based on the following nuclear safety and radiological release performance criteria contained in Section 1 of NFPA 805:

1. Nuclear Safety Performance Criteria: Fire protection features shall be capable of providing reasonable assurance that, in the event of a fire, the plant is not placed in an unrecoverable condition. To demonstrate this, the following performance criteria shall be met.

   a. Reactivity Control: shall be capable of inserting negative reactivity to achieve and maintain subcritical conditions. Negative reactivity insertion shall occur rapidly enough such that fuel design limits are not exceeded.

   b. Inventory and Pressure Control: with fuel in the reactor vessel, head on and tensioned, inventory and pressure control shall be capable of controlling coolant level such that subcooling is maintained and fuel clad damage as a result of a fire is prevented for a PWR.

   c. Decay Heat Removal: shall be capable of removing sufficient heat from the reactor core or spent fuel such that fuel is maintained in a safe and stable condition.

   d. Vital Auxiliaries: shall be capable of providing the necessary auxiliary support equipment and systems to assure that the systems required under (a), (b), (c), and (e) are capable of performing their required nuclear safety function.

   e. Process Monitoring: shall be capable of providing the necessary indication to assure the criteria addressed in (a) through (d) have been achieved and are being maintained.

2. "Radioactive Release Performance Criteria: radiation release to any unrestricted area due to the direct effects of fire suppression activities (but not involving fuel damage) shall be as low as reasonably achievable and shall not exceed applicable 10 CFR, Part 20, Limits.

Chapter 2 of NFPA 805 establishes the process for demonstrating compliance with NFPA 805.

Chapter 3 of NFPA 805 contains the fundamental elements of the fire protection program and specifies the minimum design requirements for fire protection systems and features.

Chapter 4 of NFPA 805 establishes the methodology to determine the fire protection systems and features required to achieve the nuclear safety performance criteria outlined above. The methodology shall be permitted to be either deterministic or performance-based. Deterministic requirements shall be "deemed to satisfy" the performance criteria, defense-in-depth, and safety margin and require no further engineering analysis. Once a determination has been made that a fire protection system or feature is required to achieve the nuclear safety performance criteria of Section 1.5 of NFPA 805, its design and qualification shall meet the applicable requirement of Chapter 3 of NFPA 805.

9.10.1.1.3 Code of Record

The codes and standards used for the design and installation of credited fire protection systems are listed in FPPDD, NFPA 805 Fire Protection Program Design Document (Reference 6).

9.10.1.2 SYSTEM DESCRIPTION

9.10.1.2.1 Required Systems

Nuclear Safety Capability Systems, Equipment, and Cables

Section 2.4.2 of NFPA 805 defines the methodology for performing the nuclear safety capability assessment. The systems, equipment, and cables required for at-power and non-power analyses comprising the nuclear safety capability assessment are contained in FPTE 2016-003, PBN NFPA 805 Nuclear Safety Capability Assessment (Reference 8), and FPTE 2016-004, Non-Power Operation Modes Transition Review (Reference 9), respectively.

Fire Protection Systems and Features

Chapter 3 of NFPA 805 contains the fundamental elements of the fire protection program and specifies the minimum design requirements for fire protection systems and features. Compliance with Chapter 3 is documented in FPPDD, NFPA 805 Fire Protection Program Design Document.

Chapter 4 of NFPA 805 establishes the methodology and criteria to determine the fire protection systems and features required to achieve the nuclear safety performance criteria of Section 1.5 of NFPA 805. These fire protection systems and features shall meet the applicable requirements of NFPA 805 Chapter 3. These fire protection systems and features are documented in FPPDD, NFPA 805 Fire Protection Program Design Document (Reference 6).

Radioactive Release

Structures, systems, and components relied upon to meet the radioactive release criteria are documented in FPPDD, NFPA 805 Fire Protection Program Design Document.

9.10.1.2.2 Definition of “Power Block” Structures

Where used in NFPA 805 Chapter 3 the terms “Power Block” and “Plant” refer to structures that have equipment required for nuclear plant operations. For the purposes of establishing the structures included in the fire protection program in accordance with 10 CFR 50.48(c) and NFPA 805, the plant structures listed in the FPPDD, NFPA 805 Fire Protection Program Design Document (Reference 6) are considered to be part of the ‘power block.’

9.10.1.3 SAFETY EVALUATION

The FPPDD, NFPA 805 Fire Protection Program Design Document (Reference 6) documents the achievement of the nuclear safety and radioactive release performance criteria of NFPA 805 as required by 10 CFR 50.48(c). This document fulfills the requirements of Section 2.7.1.2 “Fire Protection Program Design Basis Document” of NFPA 805. The document contains the following:

1. Identification of significant fire hazards in the fire area. This is based on NFPA 805 approach to analyze the plant from an ignition source and fuel package perspective.

2. Summary of the Nuclear Safety Capability Assessment (at power and non-power) compliance strategies.

   a. Deterministic compliance strategies

   b. Performance-based compliance strategies (including defense-in-depth and safety margin)

3. Summary of the Non-Power Operations Modes compliance strategies.

4. Summary of the Radioactive Release compliance strategies.

5. Summary of the Fire Probabilistic Risk Assessments.

6. Key analysis assumptions to be included in the NFPA 805 monitoring program.

9.10.1.4 FIRE PROTECTION PROGRAM DOCUMENTATION, CONFIGURATION CONTROL AND QUALITY ASSURANCE

In accordance with Chapter 3 of NFPA 805 a fire protection plan documented in NP 1.9.14, PBNP Fire Protection Plan (Reference 7), defines the management policy, program direction and defines the responsibilities of those individuals responsible for the plan's implementation.

The PBNP Fire Protection Plan:

1. Designates the senior management position with immediate authority and responsibility for the fire protection program.

2. Designates a position responsible for the daily administration and coordination of the fire protection program and its implementation.

3. Defines the fire protection interfaces with other organizations and assigns responsibilities for the coordination of activities.

4. Identifies the appropriate authority having jurisdiction for the various areas of the fire protection program.

5. Identifies the procedures established for the implementation of the fire protection program, including the post-transition change process and the fire protection monitoring program.

6. Identifies the qualifications required for various fire protection program personnel.

7. Identifies the quality requirements of Chapter 2 of NFPA 805.

Detailed compliance with the programmatic requirements of Chapters 2 and 3 of NFPA 805 are contained in the PBN Fire Protection Plan.

9.10.2 REFERENCES

1. NRC Safety Evaluation, “Point Beach Nuclear Plant, Units 1 and 2 - Issuance of Amendments Regarding Transition to a Risk-Informed, Performance-Based Fire Protection Program in Accordance with 10 CFR 50.48(c) (CAC Nos. MF2372 and MF2373),” dated September 8, 2016.

2. National Fire Protection Association Standards, NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition.

3. Regulatory Guide 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," Revision 1, dated December 2009.

4. NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48(c)," Revision 2, dated April 2008.

5. FAQ 12-0062, "Updated Final Safety Analysis Report (UFSAR) Standard Level of Detail," Revision 1, dated May 21, 2012.

6. NFPA 805 Fire Protection Program Design Document (FPPDD).

7. NP 1.9.14, "PBNP Fire Protection Plan."

8. FPTE 2016-003, "PBN NFPA 805 Nuclear Safety Capability Assessment."

9. FPTE 2016-004, "Non-Power Operation Modes Transition Review.

9.11 SAMPLING SYSTEM (SC)

This system provides samples for laboratory analysis to evaluate reactor coolant, and other reactor auxiliary systems chemistry during normal operation, and to evaluate the reactor coolant system chemistry during post-accident conditions. Each unit has a similar sampling system and no installed equipment is shared between units except the drains and vents to the waste disposal system. The description contained herein is equally applicable to either unit. A description of the containment atmosphere sampling system is provided in Section 6.5.

9.11.1 DESIGN BASIS

The sample system is designed to provide a means for obtaining a post accident sample as required by NUREG-0737. It is possible to obtain and analyze a sample without radiation exposures to any individual exceeding the criteria of GDC-19 (10 CFR 50 Appendix A) (Reference 1 and Reference 2).

The hot leg sample line is normally open and is continuously monitored by RE-109, Failed Fuel Monitor, except during periods of sampling or during periods of low pressure cold shutdown operations. This provides a remote method of evaluating for failed fuel. Sampling system discharge flows are limited under normal and anticipated fault conditions (malfunctions or failure) to preclude any fission product releases beyond the limits of 10 CFR 20.

The steam generator sample line is normally open with a small flow continuously monitored by RE-219, Steam Generator Blowdown Liquid Monitor. This flow is normally discharged to the service water system, and flow is automatically isolated on a high radiation signal from RE-219 (Reference 3).

Safety-related isolation functions of the sampling system include: (1) automatic isolation of sample lines penetrating containment (valves SC-951, 953, 955, 966A, 966B, 966C, and MS-2083 and 2084) on a containment isolation signal to prevent the release of radioactivity to the environment, and (2) automatic isolation of steam generator sample lines (valves MS-2083, and 2084) during a steam generator tube rupture event to isolate the ruptured steam generator and terminate the release.

9.11.2 SYSTEM DESIGN AND OPERATION

The system is capable of obtaining reactor coolant samples during reactor operation, during cooldown when the system pressure is low and the residual heat removal loop is in operation, and during post-accident conditions. Access is not required to the containment. Sampling of other process water, such as tanks in the waste disposal system, is accomplished locally. Equipment for sampling secondary and nonradioactive fluids is separated from the equipment provided for reactor coolant samples. Leakage and drainage resulting from the sampling operations are collected and drained to tanks located in the waste disposal system.

Two types of samples are obtained by the system: high temperature, high pressure reactor coolant system and steam generator blowdown samples which originate inside the reactor containment, and low temperature, low pressure samples from the chemical and volume control and residual heat removal systems.

High Pressure, High Temperature Samples

A sample connection is provided from each of the following:

1. The pressurizer steam space
2. The pressurizer liquid space
3. One primary coolant hot leg
4. Blowdown from each steam generator

Low Pressure, Low Temperature Samples

A sample connection is provided from each of the following:

1. The mixed bed demineralizer inlet header
2. The mixed bed demineralizer outlet header
3. The residual heat removal system, just downstream of the heat exchangers
4. The residual heat removal system, upstream of the heat exchangers
5. The volume control tank gas space
6. Charging pumps discharge header (high pressure, low temperature)

The high pressure, high temperature samples and the residual heat removal system samples leaving the sample heat exchangers are held to a temperature at or below approximately 130^{o}F to minimize the generation of radioactive aerosols.

The sampling system, shown in Figure 9.11-1, provides the representative samples for laboratory analysis. Analysis results provide guidance in the operation of the reactor coolant, residual heat removal, steam and power conversion, and chemical and volume control systems. Analyses show both chemical and radiochemical conditions. Typical information obtained includes reactor coolant boron and chloride concentrations, fission product radioactivity level, hydrogen, oxygen, and fission gas content, corrosion product concentration, and chemical additive concentration.

The information is used in regulating boron concentration adjustments, evaluating fuel element integrity and CVCS demineralizer performance, and regulating additions of corrosion controlling chemicals to the systems. The sampling system is designed to be operated manually, on an intermittent basis. Samples can be withdrawn under conditions ranging from full power to cold shutdown.

Reactor coolant liquid lines, which are normally inaccessible and require frequent sampling, are sampled by means of permanently installed tubing leading to the sampling room. Sampling system equipment is located inside the auxiliary building with most of it in the sampling room. The delay coil and sample lines with remotely operated valves are located inside the reactor containment.

Reactor coolant pressurizer steam, pressurizer liquid, and hot leg liquid samples originating inside the reactor containment flow through separate sample lines to the sampling room. Each of these connections to the reactor coolant system has a remote operated isolation valve (SC-951, 953, and 955 respectively), located close to the sample source. The samples pass through the reactor containment and a second remote operated isolation valve (SC-966A, B, and C respectively), to the auxiliary building, and into the sampling room, where they are cooled (pressurizer steam samples condensed and cooled) in the sample heat exchangers. The sample stream pressure is reduced by a manual throttling valve located downstream of each sample pressure vessel. The sample stream is purged to the volume control tank in the chemical and volume control system until sufficient purge volume has passed to permit collection of a representative sample. If the volume control tank is unavailable or isolated, the sample purge may be accomplished to the sample sink or to the waste disposal system. After sufficient purging, the sample pressure vessel is isolated and then disconnected for laboratory analysis of the contents.

Alternatively, liquid samples may be collected by bypassing the sample pressure vessels. If the volume control tank is unavailable or isolated, the sample purge may be accomplished to the sample sink or to the waste disposal system. After sufficient purge volume has passed to permit collection of a representative sample, either a portion of the sample flow is diverted to the sample sink where the sample is collected or the sample is collected from the purge stream.

The reactor coolant sample originating from the residual heat removal system has a remote operated, normally closed isolation valve (SC-959) located close to the sample source. The sample line from this source is connected into the sample line coming from the primary system hot leg at a point upstream of the sample heat exchanger. Samples from this source can be collected either in the sample pressure vessel or at the sample sink as with hot leg samples.

Required post-accident sampling can be accomplished from the primary system hot leg or the residual heat removal system. The sampling stations are located at accessible locations on the outside wall of the Unit 1 and Unit 2 sample rooms. Sampling is accomplished with a sample vessel, constructed of stainless steel and shielded with approximately 2-3/4 inches of lead, which is connected to the sampling station with compression fittings. The valving of the sample lines and the sample vessel allows recirculation with the sample vessel installed, ensures that sample flow is forced through the vessel when the sample is collected, and provides double valve protection against leakage when the vessel is removed. The sample vessel is transported to and from the sample station on a standard industrial four-wheel cart modified with special provisions for lifting and holding the sample vessel. The post-accident reactor coolant sample lines are purged and recirculated prior to obtaining a sample and are flushed with demineralized water as required subsequent to obtaining a sample. The isolation valve arrangement and the valve operating sequence minimize the possibility of sample loss (Reference 2 and Reference 4).

Samples originating at the chemical and volume control system letdown line at the mixed bed demineralizer inlet and outlet pass through the purge line to the volume control tank. If the volume control tank is unavailable or isolated or the pressure of the sample is low such that an adequate flow rate cannot be established, the sample purge may be accomplished to the sample sink or to the waste disposal system. Samples are obtained by diverting a portion of the flow to the sample sink where liquid and gas samples are obtained or the sample is collected from the purge stream.

The charging pump sample line, originating from the header on the discharge side of the pumps, is connected into the sample line coming from the mixed bed demineralizer inlet and outlet. Liquid samples from the charging pump discharge header pass directly through the purge line to the volume control tank. If the volume control tank is unavailable, the sample purge may be accomplished to the sample sink or waste disposal system. Samples are obtained by diverting a portion of the flow to the sample sink or the sample is collected from the purge stream.

The sample sink, which is contained in the laboratory bench as a part of the sampling hood, contains a drain line to the waste disposal system. Local instrumentation is provided to permit manual control of sampling operations and to ensure that the samples are at suitable temperatures and pressures before diverting flow to the sample sink.

Samples of the steam generator liquid are obtained from the blowdown lines of each steam generator by separate sample lines. These lines are equipped with a remote operated isolation valve (MS-2083 and 2084) and manual isolation valve (MS-316 and 317) in each line immediately outside the containment. The remote operated valve is automatically closed upon the receipt of a signal from the blowdown sample radiation monitor or the containment isolation system.

The sample lines are routed to the sample room where the liquid is cooled. Each individual sample is then split into three routes: one goes to the sample sink to provide periodic samples for chemical analysis as required or preferred, a second goes to radiation monitor RE-219, and a third line handles a continuous flow for a constant reading of conductivity, pH, and sodium. This third line also provides a sample for routine lab analyses or other in-line monitors at the secondary sample panel in the turbine hall.

Components

A summary of principal component data is given in Table 9.11-2.

Sample Heat Exchangers

Five sample heat exchangers reduce the temperature of samples from the pressurizer steam space (HX-14A), the pressurizer liquid space (HX-14B), each steam generator (HX-59A&B) and the reactor coolant (HX-14C) to approximately 130^{o}F before samples reach the sample vessels and sample sink. The tube side of the heat exchangers is austenitic stainless steel, the shell side is carbon steel. The inlet and outlet tube sides have socket-weld joints for connections to the high pressure sample lines. Connections to the component cooling water lines are socket-weld joints. The samples flow through the tube side and component cooling water circulates through the shell side.

Delay Coil

The reactor coolant hot leg sample line contains a delay coil, consisting of coiled tubing, which has sufficient length to provide at least a 40 second sample transit time within the containment and an additional 20 second transit time from the reactor containment to the sampling hood.

This allows for decay of the short-lived N-16 isotope to a level that permits normal access to the sampling room.

Sample Pressure Vessels

The high pressure coolant sample trains, the residual heat removal sample train, and the volume control tank gas space sample train each contain provisions for the installation of sample pressure vessels which are used to obtain liquid or gas samples. The hot leg and the residual heat removal system sample lines have a single sample pressure vessel in common. Integral isolation valves are furnished with the vessel and couplings are connected to nipples extending from the valves on each end with compression fittings. The vessels, valves and couplings are austenitic stainless steel.

Sample Sink

The sample sink is located in a hooded enclosure which is equipped with an exhaust ventilator that discharges to the auxiliary building ventilation system. The work area around the sink and the enclosure is large enough for sample collection and storage for radiation monitoring equipment. The sink perimeter has a raised edge to contain any spilled liquid.

Piping and Fittings

All liquid and gas sample lines are austenitic stainless steel tubing and are designed for high pressure service. Socket-welded joints are used in the portions of the sampling lines which experience severe thermal transients. Lines are so located as to protect them from accidental damage during routine operation and maintenance.

Valves

Remotely operated stop valves are used to isolate all sample lines leaving containment and the residual heat removal system. Manual stop valves are provided for component isolation and flow path control at all normally accessible sampling system locations. Manual throttle valves are provided to adjust the sample flow rate as indicated on Figure 9.11-1. All valves in the system are constructed of austenitic stainless steel or equivalent corrosion resistant material.

Remotely operated isolation valves are provided inside and outside the reactor containment on all reactor coolant sample lines leaving the containment. Each valve is provided with and will trip closed upon receipt of a containment isolation signal. The steam generator blowdown sample lines leaving containment have a remotely operated isolation valve outside containment. These valves will trip closed upon receipt of a containment isolation signal or a steam generator blowdown high radiation signal from RE-219.

9.11.3 SYSTEM EVALUATION

Leakage Provisions

Leakage of radioactive reactor coolant from this system within the containment is evaporated to the containment atmosphere and removed by the cooling coils of the containment air recirculation and cooling system or collected in floor drains, which is then directed to containment Sump A. Leakage of radioactive material from the most likely places outside the containment is collected by placing the entire sampling station under a hood provided with a connection to the auxiliary building ventilation system. Liquid leakage from the valves in the hood is drained to the waste disposal system.

Incident Control

The system operates on an intermittent basis, and under administrative manual control.

Malfunction Analysis

To evaluate system safety, the failures or malfunctions are assumed concurrent with a loss-of-coolant accident, and the consequences analyzed. The malfunctions analyzed include failure of the coolant sample isolation valves inside containment to isolate and a sample line break inside containment. The radiological consequences of both of these malfunctions are mitigated by the automatic closure of the isolation valves outside containment on a containment isolation signal.

Codes and Standards

System component code requirements are given in Table 9.11-1.

9.11.4 REQUIRED PROCEDURES AND TESTS

The inservice inspection requirements are described in the PBNP Inservice Testing Program.

9.11.5 REFERENCES

1. NRC Safety Evaluation dated December 22, 1982.
2. WE letter to NRC, “NUREG-0737 Item II.B.3 Post Accident Sampling System,” dated September 30, 1982.
3. WE Safety Evaluation 91-051.
4. PBNP Emergency Plan Implementing Procedures.
5. NUREG 0737, Item II.B.3, “Post-Accident Sampling Capability.”

Table 9.11-1 SAMPLING SYSTEM CODE REQUIREMENTS

Sample heat exchanger	ASME VIII, (no stamp required)
Sample pressure vessels	ASME III*, Class C
Piping and valves	USAS B31.1**

* ASME III - American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

** USAS B31.1 - Code for Pressure Piping and special nuclear cases where applicable.

Table 9.11-2 SAMPLING SYSTEM COMPONENTS
Sheet 1 of 2

Sample Heat Exchanger

General

Number	5 per unit
Type	Shell and coiled-tube
Design heat transfer rate (duty for 652.7°F sat. steam to 127°F liquid), each, BTU/hr	2.14×10^5

Shell

Design pressure, psig	150
Design temperature, °F	350
Total Component cooling water flow to the 5 heat exchangers (minimum), gpm	75
Operating cooling water temperature, in (maximum), °F	105
Material	Carbon steel

Tubes

Tube diameter O.D., in.	3/8
Design pressure, psig	2485
Design temperature, °F	680
Sample flow, normal, each, lb/hr.	209
Maximum allowable pressure loss, each 209 lb/hr, psi	10
Operating sample temperature, in (maximum), °F	652.7
Operating sample temperature, out (maximum), °F	127
Material	Austenitic stainless steel

Table 9.11-2 (Continued)

Sheet 2 of 2

Sample Pressure Vessels

Number, total	5 per unit
Approx. Volume, pressurizer steam sample, 2 supplied, ml	85
Approx. Volume, pressurizer liquid sample, 2 supplied, ml	85
Approx. Volume, reactor coolant hot leg sample, 2 supplied, ml	85
Approx. Volume, volume control tank sample, 2 supplied, ml	85
Approx. Volume, high radiation sample, 2 supplied, ml	19
Design pressure, psig	2485
Design temperature, °F	680

Piping

Liquid and gas sample line internal diameter, in.	0.245
Design pressure, psig	2485
Design temperature, °F	680

Figure 9.11-1 UNIT 1 SAMPLING SYSTEM

[redacted]